April 14, 1978

Docket No. 50-336

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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. 37 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 21, 1978, as supported by letters dated September 28, 1977, January 11, February 1 and 15, March 3, 9, and 15, and April 4 and 6, 1978.

This amendment permits operation with up to 500 tubes per steam generator plugged, incorporates a supplementary steam generator inservice inspection program associated with dented tubes, and reduces allowable primary-to-secondary leakage limits to 0.5 gpm through one steam generator.

We have concluded, in the enclosed Safety Evaluation, that the previously performed steam generator repairs and inspections are acceptable.

This action does not constitute authorization to restart for Cycle 2 operation. That action is being handled separately.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original Signed By

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

notified John Streeter of I&E and Rick Laird and Ed Farrell of NNECO. 1800 4/14/78 MConner

Enclosures:
1. Amendment No. 37 to License No. DPR-65
2. Safety Evaluation
3. Notice

Cont. 1
GD

cc w/enclosures: See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB#4:DOR	ORB#4:DOR	ORB#4:DOR	STSG:DOR	OELD	C-ORB#4:DOR	AD-E&P:DOR
SURNAME	RIngram *	*MFairtile:dn	MConner*	JMcGough*	J. R. GRAY	RWReid *	BGrimes*
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cc w/enclosures: See next page

BGrimes
4/14/78

OFFICE ➤	ORB#4:DOR	ORB#4:DOR	STSG:DOR	OELD	C-ORB#4:DOR	ORB#4:DOR
SURNAME ➤	RIngram	MFairtile	JMcGough		RReid	EConner
DATE ➤	4/13/78	4/13/78	4/13/78	4/ /78	4/14/78	4/13/78

cc: William H. Cuddy, Esquire
Day, Berry & Howard
Counselors At Law
One Constitution Plaza
Hartford, Connecticut 06103

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N.W.
Washington, D. C. 20005

Mr. Albert L. Partridge, First Selectman
Town of Waterford
Hall of Records - 200 Boston Post Road
Waterford, Connecticut 06385

Northeast Nuclear Energy Company
ATTN: Superintendent
Millstone Plant
P. O. Box 128
Waterford, Connecticut 06385

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, N. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region I Office
ATTN: EIS COORDINATOR
John F. Kennedy Federal Building
Boston, Massachusetts 02203

Waterford Public Library
Rope Ferry Road, Route 156
Waterford, Connecticut 06385

cc w/enclosures & incoming dtd:
3/21/78, as supplemented 9/28/77, 1/11, 2/1 & 15, 3/3, 3/9 & 3/15, and 4/4 & 4/6
Connecticut Energy Agency
ATTN: Assistant Director, Research
and Policy Development
Department of Planning and Energy
Policy
20 Grand Street
Hartford, Connecticut 06106

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. 37
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 21, 1978, as supported, 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. 37 , are hereby incorporated in the license. The licensees 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

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 14, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 37

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 4-5
3/4 4-6
3/4 4-7
3/4 4-7a
3/4 4-7b
3/4 4-7c
3/4 4-7d
3/4 4-7e (added)
3/4 4-7f (added)
3/4 4-7g (added)
3/4 4-7h (added)
3/4 4-9
B 3/4 4-2
B 3/4 4-2a

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following Augmented Inservice Inspection Program and Supplementary Inservice Inspection Program.

4.4.5.1 Augmented Inservice Inspection Program

4.4.5.1.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-5.

4.4.5.1.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-6. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.1.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.1.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (>20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.1.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-6) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspection include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-6 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.1.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-6 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.1.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg and cold leg sides) completely past the first tube bend to the center top of the tube.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-6.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.1.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4.4.5.2 Supplementary Inservice Inspection Program

4.4.5.2.1 Steam Generator Sample Selection - Both steam generators shall be inspected during the Scheduled Inspection. At least one steam generator shall be inspected during the Unscheduled Inspection. (See Specification 4.4.5.2.4 for definitions of Scheduled Inspection and Unscheduled Inspection).

4.4.5.2.2 Steam Generator Tube Sample Selection and Testing - The steam generator tube, minimum sample size; sample location; test result classification; and action requirements shall be as specified in Tables 4.4-7 and 4.4-8. Percentages in these tables refer to unplugged tubes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.2.3 Frequency - The supplementary surveillance requirements described in this specification are to be implemented during the next Scheduled Inspection; and at each Unscheduled Inspection, as defined herein, that occurs prior to the Scheduled Inspection. An Unscheduled Inspection will not be required due to primary-to-secondary leakage exceeding the limits of Specification 3.4.6.2.c during initial operation of Cycle 2 if the leakage is a result of sample removal operations or an identified oversight in the repair operations. In the event of the above, take the corrective action(s) required to reduce that leakage to within the limits of Specification 3.4.6.2.c.

4.4.5.2.4 Definitions

Tube Deformity: Tube distortion occurring as a result of the primary and, or, secondary effects of denting.

Denting: Constriction of the Inconel tubing that occurs at tube/tube support plate junctions, as a result of magnetite buildup in these regions.

Tube Plugging Criteria: Tube characteristics used to determine which tubes to plug, specifically tubes which:

a) Do not accept the ECT probe;

b) Are surrounded by tubes identified in a) above;

c) Lie along an apparent continuous series of cracks as indicated by ECT results.

Tube Support Plate Tubes: Tubes that pass through a partial tube support plate.

Accessible Tube: A tube that is accessible to remote, "finger-walker" type of devices used for conducting eddy current examinations.

Egg-Crate Tubes: Tubes that pass through "egg-crate" supports but do not pass through tube support plates.

Scheduled Inspection: The Inservice Inspection that implements the requirements of the eddy current testing program described in Specification 4.4.5.1. As indicated in Specification 4.4.5.1, this Inspection shall be performed not less than 12 or more than 24 months after the May 1977 Outage Inspection.

SURVEILLANCE REQUIREMENTS (Continued)

Unscheduled Inspection: An Inservice Inspection that is performed concerning exceeding specified primary-to-secondary leakage limits through the steam generators pursuant to Specifications 3.4.6.2 and 4.4.5.1.3.c.

Tube Testing: Inspection of the steam generator tube from the point of entry (hot leg and cold leg sides) to the elevation necessary to obtain the results specified in Table 4.4-7 or Table 4.4-8.

Exposed Peripheral Tubes: The tubes corresponding to circumferential tubes located in Rows 1-90.

TABLE 4.4-5

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Two
First Inservice Inspection	One
Second & Subsequent Inservice Inspections	One[1]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C–2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S. G.</td><td>C–1</td><td>None</td></tr>
<tr><td>C–2</td><td>Plug defective tubes</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–3</td><td rowspan="3">Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G.

Prompt notification to NRC pursuant to specification 6.9.1</td><td>All other S. G.s are C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S. G.s C–2 but no additional S. G. are C–3</td><td>Perform action for C–2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S. G. is C–3</td><td>Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1</td><td>N/A</td><td>N/A</td></tr>
</table>

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

TABLE 4.4-7

STEAM GENERATOR TUBE TESTING - SCHEDULED INSPECTION

SAMPLE SIZE/LOCATION	RESULT	ACTION REQUIREMENT
99% of accessible, tube-support-plate tubes	Blockage of 0.540 inch diameter probe at support-plate elevation	Plug tube
	Nonblockage of 0.540 inch diameter probe at support plate elevation	Plug tubes as required by Tube-Plugging Criteria
3% of tube-support-plate tubes	Dent indication at: (a) tube sheet elevation; (b) egg-crate elevation; (c) tube-support-plate elevation	Determine degree of progression, if any
3% of eggcrate tubes	Dent indication at: (a) egg-crate elevation; (b) tube-sheet elevation	Determine degree of progression, if any
50% of accessible, exposed, peripheral tubes (Steam Generator No. 1 only)	Tube defect(s) between tube sheet and first eggcrate elevation	(a) Plug defective tubes (b) Test similar sample of Steam Generator No. 2 and implement similar plugging actions
	No tube defects between tube sheet and first eggcrate elevation	None

TABLE 4.4-8

STEAM GENERATOR TUBE TESTING - UNSCHEDULED INSPECTION

SAMPLE SIZE/LOCATION	RESULT	ACTION REQUIREMENT
99% of accessible tubes in four outermost peripheral rows* around tube-support-plate peripheries	Blockage of 0.540 inch diameter probe at support-plate elevation	Plug tube
	Nonblockage of 0.540 inch diameter probe at support plate elevation	Plug tube as required by Tube-plugging Criteria
3% of tube-support plate tubes	Dent indication at: (a) tube-sheet elevation; (b) egg-crate elevation; (c) tube-support plate elevation	Determine degree of progression, if any

*Some of the tubes in these four peripheral rows are plugged; therefore, less than four complete rows are to be tested.

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.

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through both steam generators and 0.5 GPM through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in COLD SHUTDOWN within 36 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity at least once per 12 hours.

b. Monitoring the containment sump inventory at least once per 12 hours,

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady State Limit within 24 hours or be in COLD SHUTDOWN within the next 36 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in COLD SHUTDOWN within 36 hours.

MODES 5 and 6

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psia, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-1.

3/4.4.1 REACTOR COOLANT LOOPS

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. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

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

3/4.4.2 and 3/4.4.3 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.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

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, in combination with a Supplementary Inservice Inspection Program. 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 evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. Stress corrosion cracking could also be initiated from the primary side, if sufficiently large tube strains were introduced as a result of the primary and secondary effects of denting. The Supplementary Inspection Program assures that tubes that have developed excessive strains will be identified and removed from service, on a preventive basis, before cracking would develop, in accordance with the Tube Plugging Criteria described in Specification 4.4.5.2. Futhermore, the potential causative factors for developing excessive tube strain have been eliminated, or greatly reduced by (a) the steam generator repairs that were implemented during the November 1977 Outage, (b) condenser integrity resulting from the condenser retubing implemented in the May 1977 Outage, and (c) the phasing in of the Full Flow Condensate Polishing System during Cycle 2.

BASES

The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM, per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes, and certain deformed tubes, will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Mechanical tube defects caused by loose parts are unlikely, based on experimental data addressing loose parts effects. To provide conclusive assurance of the validity of this statement, and to demonstrate that hypothesized degradation does not occur, suspect tubes are to be inspected during the "Scheduled Inspection" as defined in Specification 4.4.5.2.4.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 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 application dated March 21, 1978, as supported by letters dated September 28, 1977, January 11, February 1 and 15, March 3, 9, and 15, and April 4 and 6, 1978, Northeast Nuclear Energy Company (NNECO or the licensee) proposed changes to the Millstone Nuclear Power Station, Unit No. 2 Technical Specifications. These proposed changes would permit operation with up to 500 tubes per steam generator plugged, incorporate a supplementary steam generator inservice inspection program associated with dented tubes, and reduce allowable primary-to-secondary leakage limits to 0.5 gpm through one steam generator.

Evaluation of Steam Generator Operation with Plugged Tubes

Plugging of steam generator tubes results in two changes to the response of the reactor coolant system. These are: (1) a decrease in primary coolant flow rate due to a reduction in the flow area; and (2) a decrease in the heat transfer area in the steam generator.

The licensee has evaluated the effects of steam generator tube plugging on the loss-of-coolant accident (LOCA) and on the other postulated accidents and anticipated operational occurrences (AOO) analyzed previously for cycle 2 operation. NNECO letters dated February 15 and March 3, 1978, give the results of the analysis of the accidents and transients considering the effects of steam generator tube plugging for cycle 2. NNECO letter dated September 28, 1977, gives the results of the analysis of the postulated AOO's and accidents for cycle 2 with no plugging of steam generator tubes. For the plugged case, the licensee assumed that approximately 6% of the steam generator tubes in each steam generator are plugged (500 tubes per steam generator). The licensee stated that this is conservative, since the actual number is less than 5%.

In evaluating the effects of steam generator tube plugging, the licensee reanalyzed some AOO's where it was not clear that the previous cycle 2 analysis was bounding. Table 1 gives a summary of the evaluation of each AOO and accident and the results of the reanalysis if one was done. Note that the licensee recalculated only one case for the LOCA. The assumption was made that the worst break would not change as a result of steam generator tube plugging. Based on calculations done by NRC staff consultants and previous experience with this problem, we agree with this assumption.

The peak cladding temperature with 6% of the steam generator tubes plugged is lower than with no steam generator tubes plugged. The explanation for this is as follows. The curve of peak cladding temperature as a function of time has two peaks. The first peak is the highest and is due to the stored energy in the fuel. The second broader peak is lower than the first. The second peak is a function of the reflood rate of the core. With 6% of the steam generator tubes plugged, the resistance to flow through the steam generators is larger than with no plugging. Therefore, during the reversed flow period of the blowdown, the flow rate in the reversed direction is larger with plugging than without. This removes more stored energy from the fuel and this results in a lower "first peak" peak cladding temperature. However, due to the increased flow resistance through the steam generators due to tube plugging, the reflood peak has increased. At some amount of tube plugging, greater than covered by this evaluation, the second peak would become larger than the first and the peak cladding temperature would increase again.

Conclusion on Additional Plugging

The licensee has evaluated the effects of plugging up to 500 of the tubes in each steam generator on the postulated anticipated operational occurrences and accidents for Millstone Unit No. 2. We have reviewed the licensee's evaluation and agree with his findings. We conclude that the effect of tube plugging on each event is acceptable and within the bounds of the original analyses.

TABLE 1: EVALUATION OF AOO's AND ACCIDENTS INCLUDING 6% STEAM GENERATOR TUBE PLUGGING

EVENT	REANALYZED	COMMENTS
CEA Withdrawal	Yes	Slight change in set points which are accommodated by conservatism in set point calculations. 0.6 psi increase in bias term.
Full Length CEA	Yes	No change.
Boron Dilution	No	Secondary system does not affect this transient.
RCS Depressurization	No	Rate of depressurization is sufficiently rapid that secondary effects are not important.
Loss of Coolant Flow	Yes	Reanalysis indicates flow coastdown unaffected by plugging. Previously determined required overpower margins are acceptable because of conservative flow values assumed in the original calculation.
CEA Drop	No	Essentially no change in required overpower margins.
CEA Ejection	No	Accident is fast enough that secondary effects are not important.
Seized Rotor	No	This accident is less limiting than a four-pump Loss-of-Coolant Flow.
Loss of Load	No	Less severe than CEA withdrawal.
Excess Load	Yes	Plugging tubes results in a slightly more adverse power transient. Licensee calculations show that the change can be accommodated by conservatism in set point calculations not previously taken credit for.

TABLE 1 (CONTINUED)

EVENT	REANALYZED	COMMENTS
Steam Line Break	No	Plugging steam generator tubes reduces steam generator pressure. This reduces the cooldown rate, making this accident less severe.
Steam Generator Tube Rupture	No	There is a larger pressure differential due to reduced steam generator pressure. Therefore, following a steam generator tube rupture, pressurizer pressure would decrease more rapidly as a result of the rupture, tripping the reactor sooner. The change in time of trip and resulting release, while in the conservative direction, would probably be small.

Discussion and Evaluation of Steam Generator Repairs and Inspection

The Millstone Unit No. 2 steam generator inspection during November 1977 through February 1978 consisted of the following:

1. Eddy current testing all tubes passing through the two top partial support plates in steam generators Nos. 1 and 2.

2. Visual examination of all support plates in both steam generators.

3. Eddy current testing of approximately eight hundred (800) tubes supported by eggcrates only, in both steam generators.

4. Probe for denting of 3,000 tubes at tube/tubesheet intersections.

5. Removing a section of support plate containing at least three (3) intact dented tube/plate intersections, and conducting detailed metallurgical and analytical studies.

6. Removing a tube passing through the tube sheet sludge pile and six (6) eggcrate supports, and conducting detailed metallurgical and analytical studies.

7. Removing and analyzing tube sheet sludge from the hole created by item 5 above.

Approximately 2,200 tubes which pass through the two top support plates were eddy current tested (ECT) in each steam generator with a 0.540" diameter probe at 400 H_z. Essentially all the tubes inspected in both steam generators had dent indications at each support plate. The majority of tubes not passing the 0.540" ECT probe are in regions of the support plate periphery adjacent to the lugs supporting the plates. Steam generator No. 1 had 150 tubes which did not allow passage of the 0.540" probe and steam generator No. 2 had 174 tubes. Dent indications ($\approx$1.0 mil) at tube/eggcrate intersections were observed in approximately 70% of the tubes inspected in both steam generators. Steam generator No. 1 had dent indications ($\approx$1.2 mil) at the tube/tubesheet intersections in 23% of the tubes inspected and steam generator No. 2 had dents at 14% of the tubes inspected.

Tube denting has caused the two top partial support plates in both steam generators to expand and create "hard spots" at supporting lugs on the tube bundle shroud. The stresses induced by the expanding support plates have caused cracking in the ligaments between the tube holes and circulation flow holes in corner areas of the uppermost support plate along the outer band of tubes adjacent to the rim of solid metal at the outer periphery of the plates. Shear stresses have caused cracking along the inner boundary of the solid rim section and shifting at the corner areas of the plate. This produced a shearing action on the tubes and deformed the tube wall of approximately 21 outer peripherial tubes located in the corner areas of the plate.

Steam Generator Repairs and Corrective Actions

Approximately 80% of the plate constraint is attributable to the lugs supporting the plates. Analysis has shown that compressive and shear stresses associated with plate constraint will cause further cracking and shifting of the partial support plates. This condition will cause additional peripherial tubes to deform and increase the rate of denting of tubes in the central portion of the plate. Finite element analysis indicated that stresses in the plate adjacent to the rim would be reduced by removing the lugs and a portion of the peripherial plate rim adjacent to the tube bundle shroud. NNECO has performed the following steam generator modifications and improvements to the secondary water system to minimize the progression of plate cracking and shifting, and further tube damage:

1. Removed all lugs at each support plate and a portion of the peripheral solid rim in the uppermost plate to reduce "hard spots" and minimize the possibility of further cracking and shifting of the plates in each steam generator.

2. Preventively plugged all peripheral tubes adjacent to the solid rim which have the greatest potential for failure including additional tubes near the periphery in the corner regions of both support plates.

3. Plugged all tubes not passing the 0.540" ECT probe and those surrounding.

4. Avoided or minimized the progression of denting by avoiding or minimizing unfavorable chemistry conditions.

5. Excluded seawater ingress by means of assuring condenser tube integrity and a full flow condensate polishing system, which is fully installed and will be available for use shortly after the start of Cycle 2.

Plugging Criteria

In addition to the above tube plugging pattern in items 2 and 3, the licensee has performed the following preventive plugging based on the projected critical tube hoop strain predicted by the finite element analysis on the tube support plate.

1. Plugged all additional tubes in the two partial support plates which would not pass the 0.540" ECT probe.

2. Plugged any tubes that were damaged during the rim and support lug removal operation.

3. Plugged all tubes that lie along an apparent continuous series of ligament cracks in the plates.

The plugging criteria adopted by the licensee will result in 290 tubes plugged in steam generator No. 1 and 352 in steam generator No. 2.

To ensure that the above repairs and corrective actions will not affect the safe operation of the Millstone Unit 2 steam generators, the licensee has performed the following structural analyses:

1. Pipe Break Accident Analysis

2. Plugging Strength Analysis

3. Tube Support Plate Analysis

4. Loose Parts Evaluation

5. Tube Bending Analysis

Evaluation of Steam Generator Repairs and Inspections

We have reviewed the information submitted by the licensee and our evaluation is as follows:

1. Modifications to the partial support plates have significantly reduced the compressive and shear loads to minimize any further potential for cracking and shifting of the plates. Analysis

of stresses on the tube bundle shroud resulting from support plate expansion or continuing magnetite growth in the proposed period of operation indicate that ASME code allowables are met. Therefore, the shroud during normal operating and accident conditions will not be affected by continued support plate expansion. LOCA, main steam line break (MSLB), and safe shutdown earthquake (SSE) analyses indicated that the partial support plate is not necessary to ensure tube integrity. The tubes are primarily supported by the eggcrate and top support bar structures, and loss of lateral support from the partial support plates will be insignificant. The licensee performed these modifications under the provisions of 10CFR Part 50.59 and stated he prepared a safety analysis and determined that no unreviewed safety consideration existed. Thus, prior NRC approval of these modifications was not required.

2. With respect to the effect of continued magnetite growth that causes the support plate expansion and thus denting, the preventive plugging program that was implemented in accordance with the criteria discussed previously is adequate. In this regard, the NRC staff also considered the following additional supportive reasons:

   a. Removal of support plate lugs and cutting of the peripherial solid rim has removed the "hard spot" regions to significantly reduce the potential for small hard spot strain increases.

   b. All of the tubes in hard spot regions and those adjacent to the peripherial solid rim have been plugged. Thus, no unplugged tubes are strained beyond 14%.

   c. No primary-to-secondary tube leakage has been experienced at Millstone Unit No. 2, and the hoop strain in unplugged dented tubes is predicted to be less than 14% which is believed to be below that required for incipient tube cracking.

d. Even though some non-through-wall cracks may develop during the cycle 2 operation and may crack through during normal operation or postulated accidents, the proposed leakage rate of 0.50 gpm per steam generator would assure that no individual crack would reach such proportions that it would be unstable. This consideration is consistent with the rationale upon which the preventive plugging limits were set for wastage or caustic cracking type of degradation in other nuclear power facilities.

e. Even if a LOCA or a MSLB were to occur during the proposed period of operation and some tubes were in a state of incipient failure, the consequences of such an event would not be severe because the dented region is constrained by the support plate which would prevent any rapid tube failure.

3. The fact that most peripheral row tubes at the corners of the partial support plates are plugged lessens the concern over the possible loss of lateral support and shearings of tubes due to the so-called plate shifting effect.

4. Visual inspection of the licensee's tube plug-weld and plugging strength analysis establishes the compliance with the 1977 Section III Edition of the ASME Boiler and Pressure Vessel Code to assure that the plugging procedure will not result in loose plugs during normal operation or accident conditions.

5. We were concerned that the cracked condition of the support plates could result in the generation of small loose pieces of the support plate during plant operation. Experimental and analytical work addressing the effect of a relatively large loose part (approximately four inches long by three-quarter inches in diameter) was described by NNECO. In tests simulating up to a 40-year exposure, no Inconel 600 tube degradation occurred. Based on a NNECO review of the extensive ECT, analytical studies, and steam generator internals removed from the Millstone Unit No. 2 steam generators, we have concluded that no significant sections of a support plate would become separated as "loose parts" during cycle 2. If loose pieces are postulated to exist at the tube bundle entrance (the region where damage would occur if it were to occur), they would be smaller than a 1/2 ounce piece, 3/8" x 1/4" x 1", if generated from the interior of the bundle. Pieces from the circumferential edge are discounted since loose pieces were removed during the rim removal operations. Pieces larger than the one described would be trapped within the tube bundle.

While local flow velocities and tube impact forces cannot be accurately calculated, the prime motive force, recirculating flow from the downcomer, is calculated as 480,000 lb/ft^2-hr., for Millstone Unit No. 2. Based on these data and similar Westinghouse studies, we conclude that the potential for tube damage due to loose parts in the Millstone Unit No. 2 steam generators is not significant.

As further assurance that loose parts do not represent a threat to tube integrity, eddy current and visual inspections are to be performed at the next scheduled inspection. The eddy current inspection is to be performed on "susceptible" tubes as part of the Supplementary Inservice Inspection Program. The visual examination will consist of searching for loose parts in the tube sheet area.

6. Removal of the lugs and a portion of the peripherial solid rim from the upper partial support plate resulted in a gap of approximately 3/4" between the plate and shroud. The 3/4" gap could cause some tube deflection during normal operation or postulated accident conditions. Tube bending analysis indicates compliance with the ASME Code fatigue curve and the Code alternating stress allowable.

7. Future progression of tube denting is expected to be limited because condenser in-leakage of sea water has been minimized by retubing of the main condensers during the May 1977 outage. In the event condenser leaks should occur, contamination of the steam generator feedwater will be controlled by the full flow condensate polishers installed in the secondary coolant system. This system is fully installed and will be available for use shortly after the beginning of Cycle 2.

Conclusion on Repairs and Inspections

We conclude that the previously performed steam generator repairs and inspections are acceptable.

We further conclude, based on our review of the licensee's repairs, inspections and analyses and the installation of secondary system full flow condensate polishers, that the Millstone Unit No. 2 steam generators will retain their integrity through cycle 2 operation.

Discussion of Leakage and Supplementary Inspection Programs

Section 3.4.6.2.c of the current Technical Specifications for Millstone Unit No. 2 allow a primary-to-secondary leakage of 1 gpm from one steam generator. NNECO proposes to reduce this limit to 0.50 gpm.

Significant tube denting has been detected in the Millstone Unit No. 2 steam generators and the resultant hoop strain could initiate primary side cracking. The 0.5 gpm leakage rate would allow early detection of through-wall cracks. Leakage exceeding the limit will require plant shutdown and an unscheduled inspection, to plug leaking tubes, and nonleaking severely dented tubes.

The NRC Standard Technical Specifications for steam generator surveillance requirements were previously issued for Millstone Unit No. 2 by License Amendment No. 18, dated February 10, 1977. NNECO has proposed to add a supplementary inservice inspection program to address tube denting. These changes specify additional sampling greater than the current requirements for both scheduled and unscheduled inspections. The frequency for a scheduled inspection will be not less than 12 months nor more that 24 months after the May 1977 outage. All unscheduled inspections will be performed when the primary-to-secondary leakage rate exceeds 0.50 gpm. Additional sampling requirements will include: (1) 99% of accessible tubes passing through the top two support plates, (2) 3% of tubes passing through eggcrate supports, and (3) 50% of accessible peripheral tubes. Additional plugging criteria for dented tubes specify: (1) all tubes that do not pass a 0.540" ECT probe and those tubes surrounding, and (2) all tubes that lie along an apparent continuous series of ligament cracks in the top two tube support plates.

Evaluation of Leakage and Supplementary Inspection Programs

The primary-to-secondary leakage rate of 0.50 gpm can be detected by radiation monitoring of the steam generator blowdown to assure early identification of through-wall cracking of dented tubes. A limit of 0.50 gpm will ensure that no individual crack will reach such proportions that it may become unstable during normal or accident loading conditions. Analysis has shown that cracks having a leakage rate less than 0.50 gpm will have an adequate margin of safety. Furthermore, the dented region is constrained by the support plate which would also control crack stability and prevent tube failure (bursting) during postulated accident conditions.

Wastage-type defects are not anticipated with the presently used all volatile treatment chemistry for the secondary water treatment. However, if such degradation should occur it would be found in the scheduled inservice inspection discussed below. All tubes exceeding the current 40% plugging limit criteria would then be taken out of service.

Future progression of tube denting or other types of tube degradation is expected to be limited because condenser in-leakage of seawater has been minimized by retubing of the main condensers during the May 1977 outage. In the event condenser leaks should occur, contamination of the steam generator feedwater will be controlled by the full flow condensate polishers installed in the secondary coolant system. This system is fully installed and will be available for use shortly after the beginning of Cycle 2.

The proposed supplementary steam generator surveillance requirements adequately cover the monitoring of tube denting progression. The supplementary inspection program and preventative plugging criteria for dented tubes will ensure that any tubes with excessive hoop straining will be identified and removed from service before through-wall cracks could develop. The steam generator modification performed during January and February 1978 have reduced the potential for unplugged dented tubes developing excessive straining that could initiate primary side cracking.

Conclusion on Leakage and Supplementary Inspections

We conclude that the implementation of a 0.50 gpm leak limit for primary-to-secondary leakage through the steam generator tubes and the addition of the supplementary inservice inspection program related to denting would provide the additional assurance needed for future operation of the Millstone Unit No. 2 steam generators.

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) 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.

Dated: April 14, 1978

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. 37 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 its date of issuance.

This amendment permits operation with up to 500 tubes per steam generator plugged, incorporates a supplementary steam generator inservice inspection program associated with dented tubes, and reduces allowable primary-to-secondary leakage limits to 0.5 gpm through one steam generator.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 10, 1978 (43 FR 5908). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. Operation with sleeved guide tubes for the Control Element Assemblies was also the subject of the above Notice. This action is being handled separately.

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 21, 1978, as supported by letters dated September 28, 1977, January 11, February 1 and 15, March 3, 9, and 15, and April 4 and 6, 1978, (2) Amendment No. 37 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, Route 156, 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 Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors