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FEB 2 1976

Docket No. 50-336

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. 7 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2. The amendment consists of changes to the Technical Specifications in response to your applications dated October 27, and December 11, 1975.

This amendment modifies the Technical Specifications to extend the implementation dates for liquid and gaseous release limits and to correct typographical errors and to add information of a clarifying nature.

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

Please note that we have discontinued the use of separate identifying numbers for changes to technical specifications. Sequential amendment numbers will be continued as in the past.

Sincerely,

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

Enclosures:
1. Amendment No. 7
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

OFFICE ➤	RL:ORB-3	RL:ORB-3	OELD	RL:ORB-3		
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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-65

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

   A. The applications 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 October 27 and December 11, 1975, comply 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; and

   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.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C (2) of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised".

3. The license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for Donald M. Elliott

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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 2, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 7

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Appendix A

Replace page 2-1, 2-2, B2-1, B2-2, B2-3, B2-4, 2-5, 2-6, 3/4 2-13, 3/4 2-14, 3/4 5-5, 3/4 5-6, 3/4 6-25, 3/4 6-26 with the attached revised pages. No change has been made on pages 2-1, 2-6, B2-2, B2-4, 3/4 2-13, 3/4 5-6 and 3/4 6-25.

Appendix B

Replace page vii with the attached revised page.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

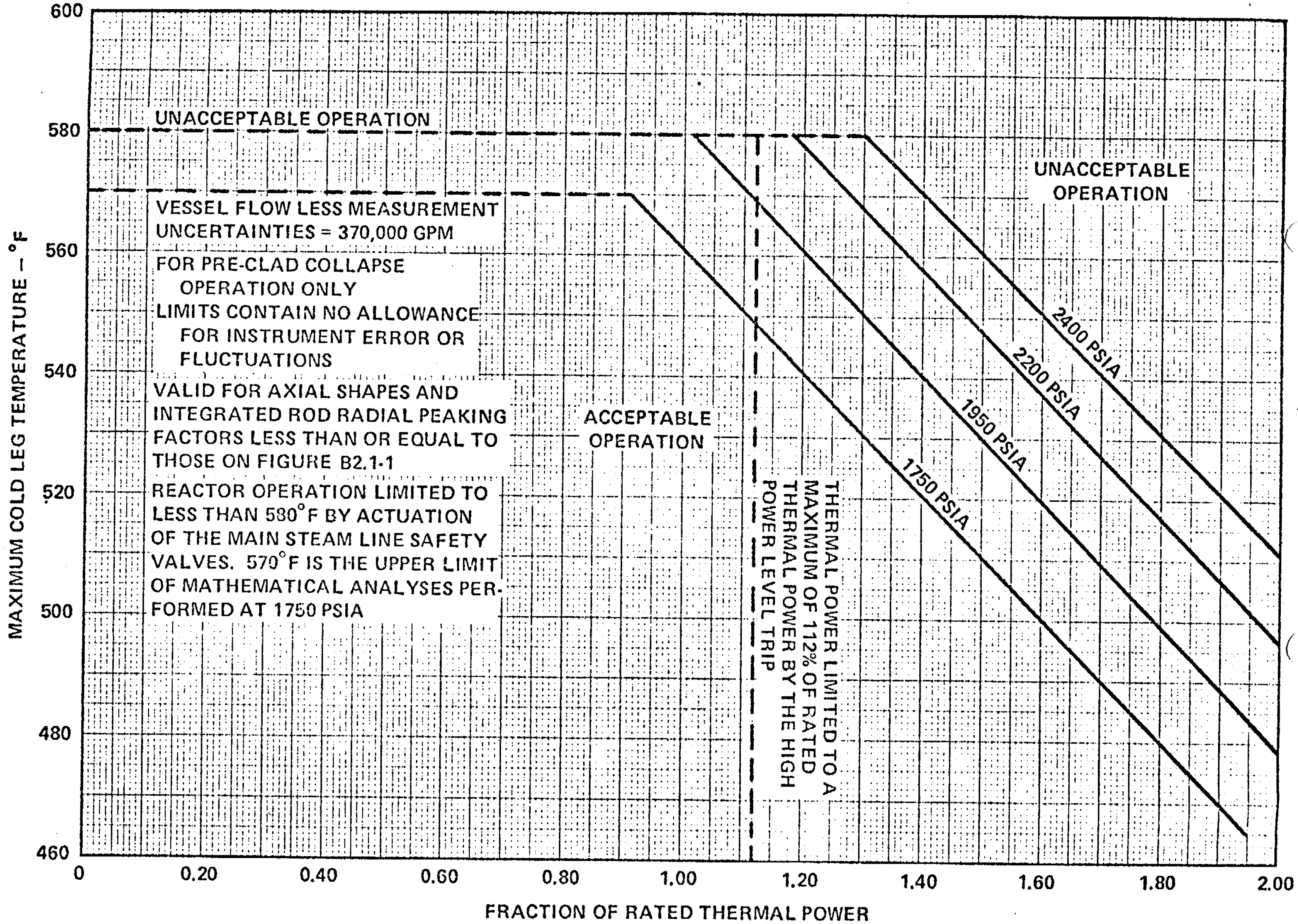

Figure 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 1100 psig	$\geq$ 1100 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
12. Steam Generator Water Level - High (5)	$\leq$ 85.40%	$\leq$ 85.40%

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

(5) Each of four channels actuates on the higher of two signals from two downcomer level differential pressure transmitters on each steam generator.

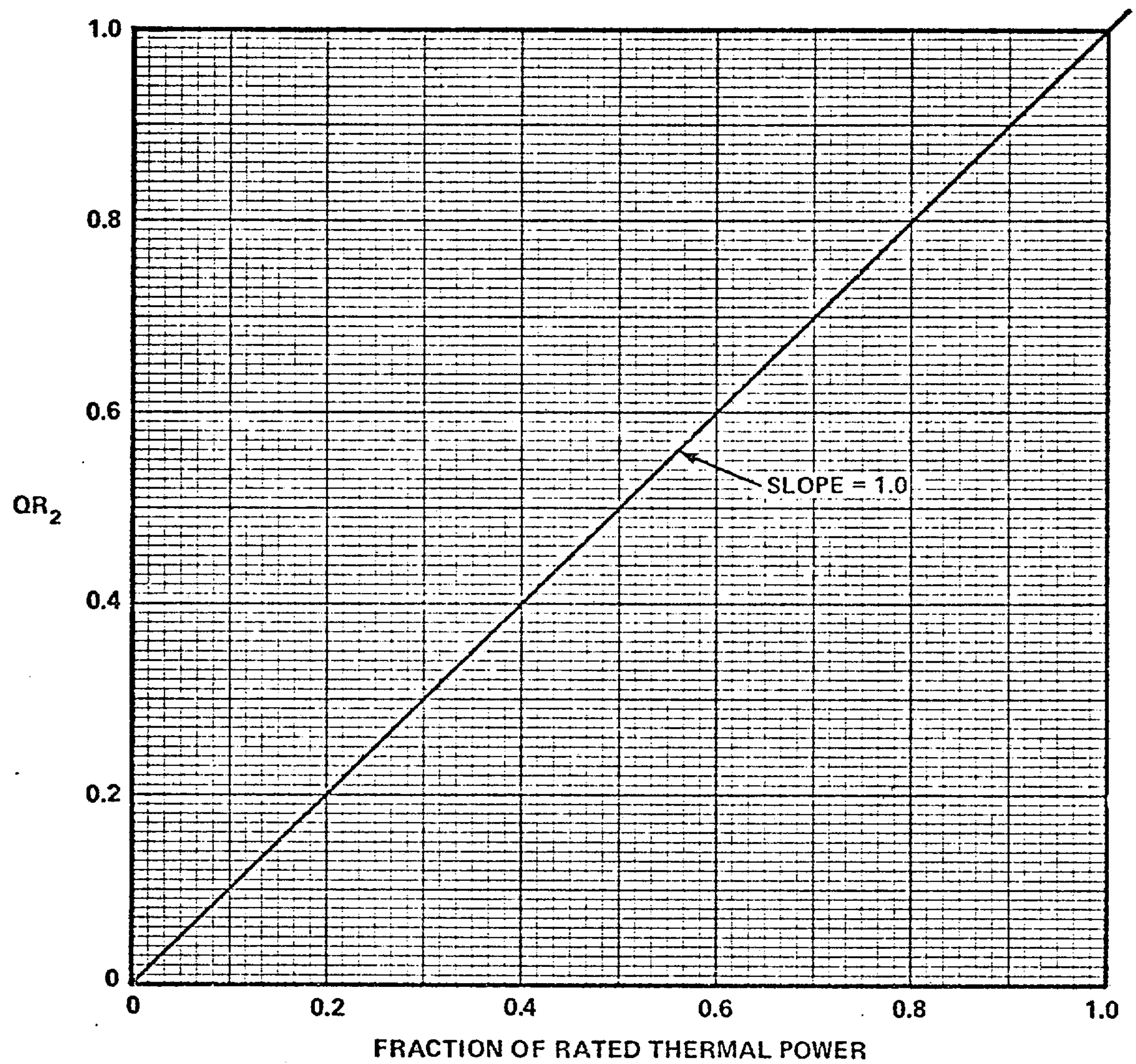

FIGURE 2.2-1

Local Power Density — High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as as appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.30 for the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F (570°F for 1750 psia). The dashed line at 580°F coolant inlet temperatures is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. The 570°F limit represents the maximum temperature used in the mathematical analyses used to generate the 1750 psia isobar. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL

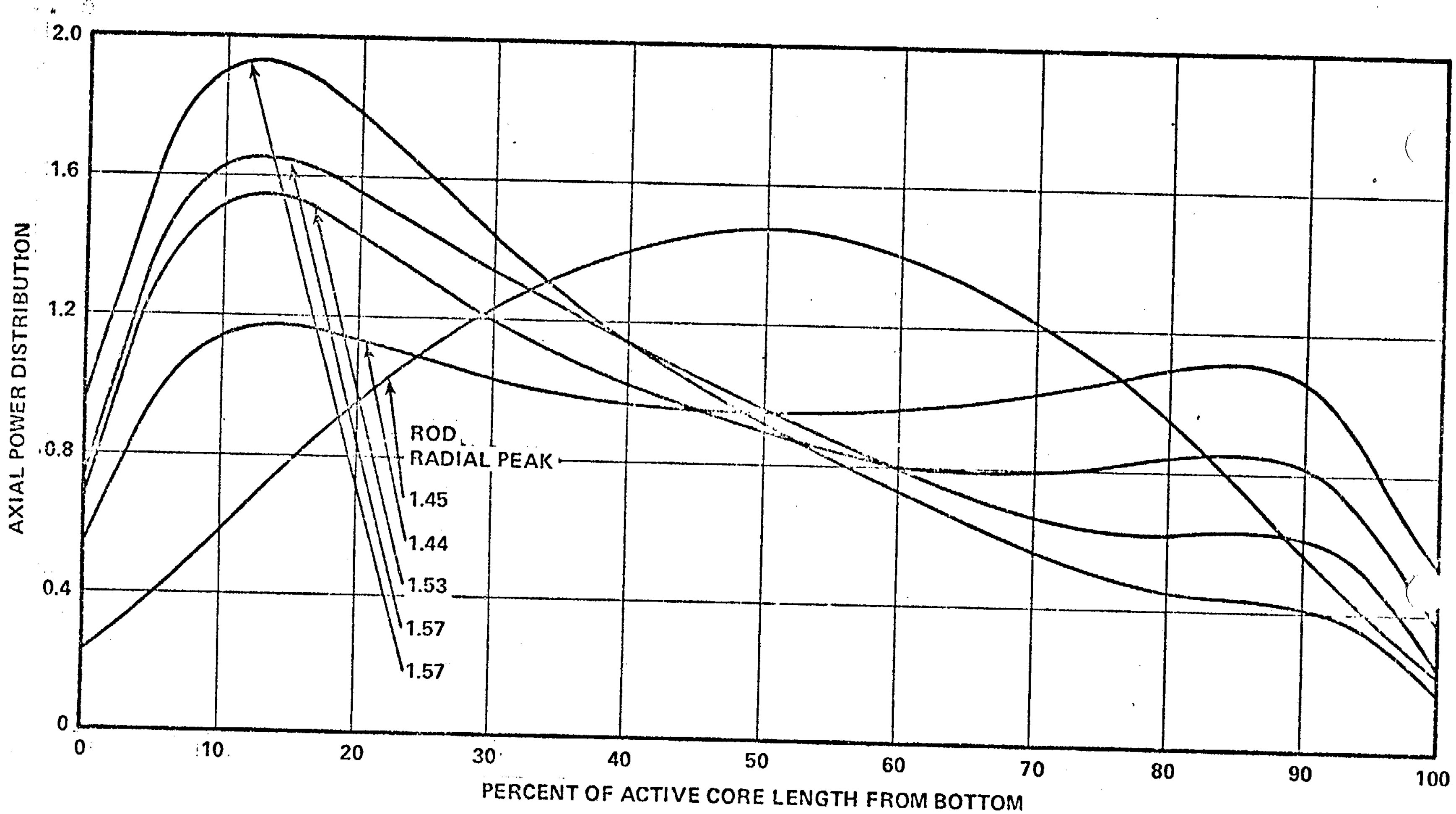

FIGURE B2.1-1

Axial Power Distributions for Thermal Margin Safety Limits

POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.30 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SET POINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.88% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL power decreases. The trip setpoint has a maximum value of 107% of RATED THERMAL POWER and a minimum setpoint of 15% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the accident analyses.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit

TABLE 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 544°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 370,000 gpm
AXIAL SHAPE INDEX	Figure 3.2-4

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

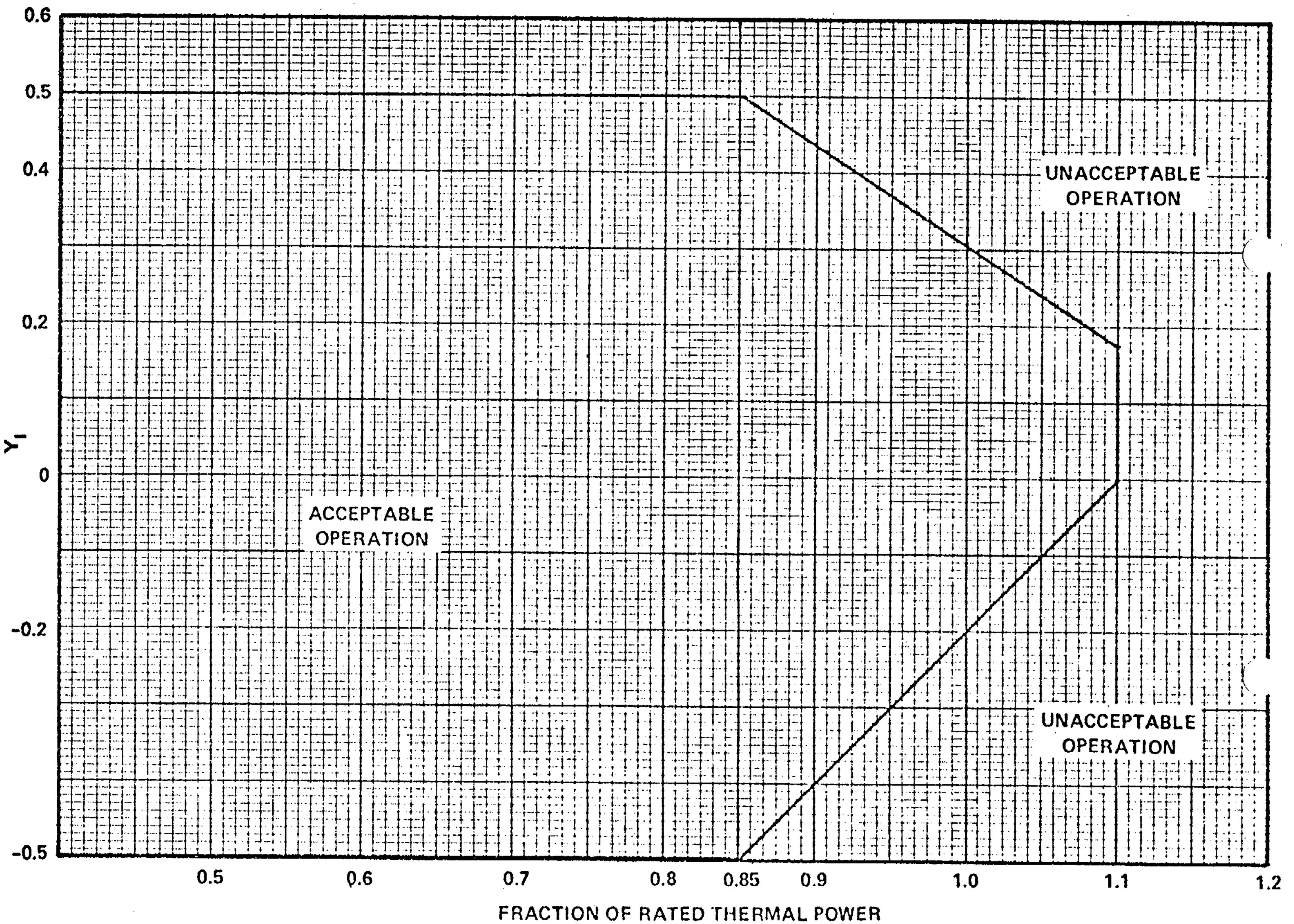

FIGURE 3.2-4

AXIAL SHAPE INDEX Operating Limits with 4 Reactor Coolant Pumps Operating

SURVEILLANCE REQUIREMENTS (Continued)

8. Verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open
2-SI-659	SRAS Recirc.	Open*
2-SI-660	SRAS Recirc.	Open*
2-CH-434	Thermal Bypass	Closed**

b. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

c. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psia.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 65 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 0.35 $\pm$ 0.05 lbs of TSP from a TSP storage basket is submerged, without agitation, in 50 $\pm$ 5 gallons of 180 $\pm$ 10°F borated water from the RWST, the pH of the mixed solution is raised to $\geq$ 6 within 4 hours.

*To be closed prior to recirculation following LOCA.
**2-CH-434, a manual valve, shall be locked closed.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by cycling each power operated valve in the subsystem flow path not testable during plant operation through one complete cycle of full travel.

3/4.6.5 SECONDARY CONTAINMENT

ENCLOSURE BUILDING FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two separate and independent enclosure building filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one enclosure building filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each enclosure building filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 10 hours with the heaters on.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 12 months or after every 720 hours of system operation and (1) after each complete or partial replacement of a HEPA filter or charcoal adsorber bank, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following painting, fire or chemcial release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 9000 cfm $\pm$ 10%.

3. Subjecting the carbon contained in at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers to a laboratory carbon sample analysis and verifying removal efficiency of $\geq$ 90% for radioactive methyl iodide at an air flow velocity of 0.67 ft/sec $\pm$ 20% with an inlet methyl iodide concentration of 0.05 to 0.15 mg/m^3, $\geq$ 95% relative humidity, and $\geq$ 125°F; other test conditions shall be in accordance with USAEC RDT Standard M-16-1T, June 1972. The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying once entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 9000 cfm $\pm$ 10% during system operation.

Schedule for the Implementation of the Environmental Technical Specifications

The following is an implementation schedule for those items in the Environmental Technical Specifications where backfits in equipment of instrumentation is involved.

Section	Item	Date
2.1	Thermal	August 1, 1975
2.3	Chemical	August 1, 1975
2.4	Radioactive Effluents (Unit 1)	
2.4.1	Liquids	February 1, 1976
2.4.2	Gaseous	April 1, 1976

For Unit 2 the following specifications in Section 2.4 are scheduled to be operational on the dates given.

2.4	Radioactive Effluents (Unit 2)	
2.4.1.2.G	Liquids	November 1, 1975
2.4.1.3.B	Liquids	November 1, 1975
Page 2.4-23, Table 2.4-4	Gaseous	November 1, 1975
2.4.2.3.A	Gaseous	January 1, 1976

All sections shall be implemented within 30 days of the final issuance of the Environmental Technical Specifications.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-336

Introduction

By application for license amendment dated October 27, 1975, Northeast Nuclear Energy Company (NNECO) requested a change to the Millstone Unit 2 Environmental Technical Specifications (ETS). The proposed change would extend the implementation dates for those portions of the Millstone Unit 2 ETS which address liquid and gaseous release limits for Millstone Unit 1. In addition, on December 11, 1975 NNECO submitted two applications for license amendment for Millstone Unit 2 which would correct typographical errors and add information of a clarifying nature to the Radiological Technical Specifications.

Discussion and Evaluation

I. Change to Appendix B - Environmental Technical Specifications

The application for license amendment dated October 27, 1975 requested that the implementation date for Millstone Unit 1 liquid release specifications (Section 2.4.1) be changed from November 1, 1975 to February 1, 1976 and the gaseous release specifications (Section 2.4.2) be changed from January 1, 1976 to April 1, 1976. NNECO requested that the above implementation dates for Millstone Unit 1 be changed for both the Millstone Unit 2 ETS and the then pending ETS for Millstone Unit 1. The licensee stated that requested delays were necessitated by unforeseen system design, equipment delivery, and construction problems associated with the augmented radwaste system for Millstone Unit 1.

On December 19, 1975, we issued Amendment No. 12 to the Millstone Unit 1 Provisional Operating License which incorporated the ETS into the license as Appendix B. This amendment included the revised implementations dates. In order to maintain consistency between Millstone Units 1 and 2 ETS, the implementation dates for Millstone Unit 2 ETS Sections 2.4.1 and 2.4.2, as they apply to Millstone Unit 1, should be changed. The environmental impact of this action, which was considered prior to the issuance of Amendment 12 to the Millstone Unit 1 Provisional Operating License, is negligible. Accordingly, the proposed revised implementation dates for Sections

2.4.1 and 2.4.2 of the Millstone Unit 2 ETS, as they apply to Millstone Unit 1, are acceptable.

II. Changes to Appendix A - Radiological Technical Specifications

The application for license amendment, dated December 11, 1975, contained requests for changes to the Radiological Technical Specifications for Millstone Unit 2 each of which was of a clarifying or corrective nature. A description and our evaluation of the proposed changes is contained in the following subsections:

1. "Modify Figure 3.2-4, page 3/4 2-14, by deleting the notations Q_1 and Q_2 and the dashed vertical lines at 85 and 100 percent power."

   Figure 3.2-4 is a plot of Axial Shape Index as a function of Rated Thermal Power. The two vertical broken lines, appearing at .85 and 1.0 Thermal Power and labeled Q_1 and Q_2, respectively, are not referenced elsewhere in the Technical Specifications and have no significance with regard to Technical Specification 3.2.5 or 4.2.5 which reference Figure 3.2-4. Therefore, deletion of the broken lines and their associated labels Q_1 and Q_2 has no safety significance and is acceptable.

2. "On page 3/4 5-5 paragraph 4.5.2.a.8, change valve number CH 434 to read 2-CH-434** and add the following foot note to the bottom of the page:

   **2-CH-434, a manual valve, is to be locked closed."

   The proposed change in valve notation would make paragraph 4.5.2.a.8 consistent with the information presented in Figure 9.2-2 (P&ID - Chemical and Volume Control System) of the Final Safety Analysis Report for Millstone Unit 2. Addition of the proposed footnote is appropriate since paragraph 4.5.2.a.8 states "Verifying that the following valves are in the indicated position with power to the valve operator removed"; valve 2-CH-434 is manually operated and thus has no valve operator. We find this change acceptable.

3. "On page 3/4 6-26, Section 4.6.5.1.b, change ...HEPA filter or charcoal adsoerber... to read ... HEPA filter or charcoal adsorber..."

   This change corrects a typographical error; it has no safety significance and is acceptable.

4. "Change item 10 of Table 2.2-1, page 2-5, to read: Loss of Turbine--Hydraulic Fluid Pressure-Low (3)".

   This change corrects the omission of "Table Notation (3)" and makes Table 2.2-1 consistent with the notation used in Table 3.3-1 of the existing Technical Specifications. This change has no safety significance and is acceptable.

5. "Modify Figure 2.1-1, Reactor Core Thermal Margin Safety Limit, as follows:

   (a) Add a heavy dashed line across the 580°F line from zero power to the 2400 psia isobar.

   (b) Add a heavy dashed line across the 570°line from zero power to the 1750 psia isobar.

   (c) Add a heavy dash line vertically at 112 percent power from 460°F to 580°F.

   (d) Move the words "Acceptable Operation" to the left of the dashed vertical line at 112 percent power.

   (e) Add the following note vertically along the dashed 112 percent power line:

   Thermal Power Limited to a Maximum of 112% of Rated Thermal Power by the High Power Level Trip.

   (f) Add the following note horizontally along the Fraction of Rated Thermal Power coordinate:

   Reactor Operation Limited to Less than 580°F By Actuation of the Main Steam Line Safety Valve. 570°F is the Upper Limit of Mathematical Analyses Performed at 1750 psia.

   (g) Change in associated Technical Specification Bases to reflect the above changes."

   Figure 2.1-1 represents the Reactor Core Thermal Margin Safety Limit. The proposed changes do not alter any of these operating limits but supply additional information on related reactor operational limitations. This proposed change has no safety significance and is acceptable.

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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: February 2, 1976

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

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 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 2, located in the Town of Waterford, Connecticut. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications to extend the implementation dates for liquid and gaseous release limits and to correct typographical errors and to add information of a clarifying nature.

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 is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or 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 October 27, and December 11, 1975, (2) Amendment No. 7 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut 06385.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 2nd day of February, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald M. Elliott

Donald M. Elliott, Acting Chief
Operating Reactors Branch #3
Division of Reactor Licensing

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

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

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

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 statement, negative declaration or 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 October 27, and December 11, 1975, (2) Amendment No. 7 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut 06385.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 2nd day of February, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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