DMB016

December 8, 1986

Docket No. 50-336

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Mr. John F. Opeka, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

The Commission has issued the enclosed Amendment No. 113 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your applications dated October 20, October 24, and October 27, 1986.

The amendment changes the Technical Specifications (TS) to provide for: (1) revised temperature/pressure limits in TS 3/4.4.9, "Pressure/Temperature Limits" and TS Figure 3.4-2, "Reactor Coolant System Pressure Temperature Limitations for 12 Full Power Years," (2) a change to the surveillance frequency for determining reactor coolant system (RCS) flow rate in TS 4.2.6, "DNB Margin," and (3) changes to several TS's associated with RCS flow and reactor power peaking limits. The changes to the TS support Cycle 8 operation of Millstone Unit 2 based on a reduced Reactor Coolant System flow rate of 340,000 gpm from the previous cycle's 350,000 gpm. Extended cycle operation beyond the projected end of cycle (EOC) 8 is, however, based on a previous assumption of 350,000 gpm RCS flow rate. Accordingly, should you desire to operate Millstone Unit 2 beyond the projected EOC 8 please provide a supplemental evaluation and proposed TS, as needed, at least 90 days prior to the projected EOC 8.

The NRC staff has also reviewed revised loss-of-coolant accident (LOCA) calculations which assume a reduced RCS flow. These calculations are based upon the LOCA model described in WCAP-10054, Addendum 1 (NOTRUMP) which was submitted by letter dated August 10, 1984. Our review and approval of the NOTRUMP model resolves TMI Action Item II.K.3.30, "Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K" and our review and approval of the NOTRUMP calculations resolves TMI Action Item II.K.3.31, "Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46."

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

Sincerely,

/S/

David H. Jaffe, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 113 to DPR-65
2. Safety Evaluation

cc w/enclosures:
See next page

PBD-8:	PBD-8:	FOB	OGC	PBD-8:
PKreutzer	DJaffe:jch		Bachmann	AThadani
12/2/86	12/2/86	/ /86	12/3/86	12/8/86

12/3/86

Mr. John F. Opeka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 2

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. DPR-65

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

   A. The applications for amendments by Northeast Nuclear Energy Company, et al. (the licensee), dated October 20, October 24, and October 27, 1986 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 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 Appendix A, as revised through Amendment No. 113, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A C Thadani

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 8, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 113

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 areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Page	Insert Page
2-2	2-2
2-4	2-4
3/4 2-8(a)	3/4 2-8(a)
3/4 2-9	3/4 2-9
3/4 2-13	3/4 2-13
3/4 2-14	3/4 2-14
3/4 4-17	3/4 4-17
3/4 4-19	3/4 4-19
B3/4 4-6	B3/4 4-6

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.

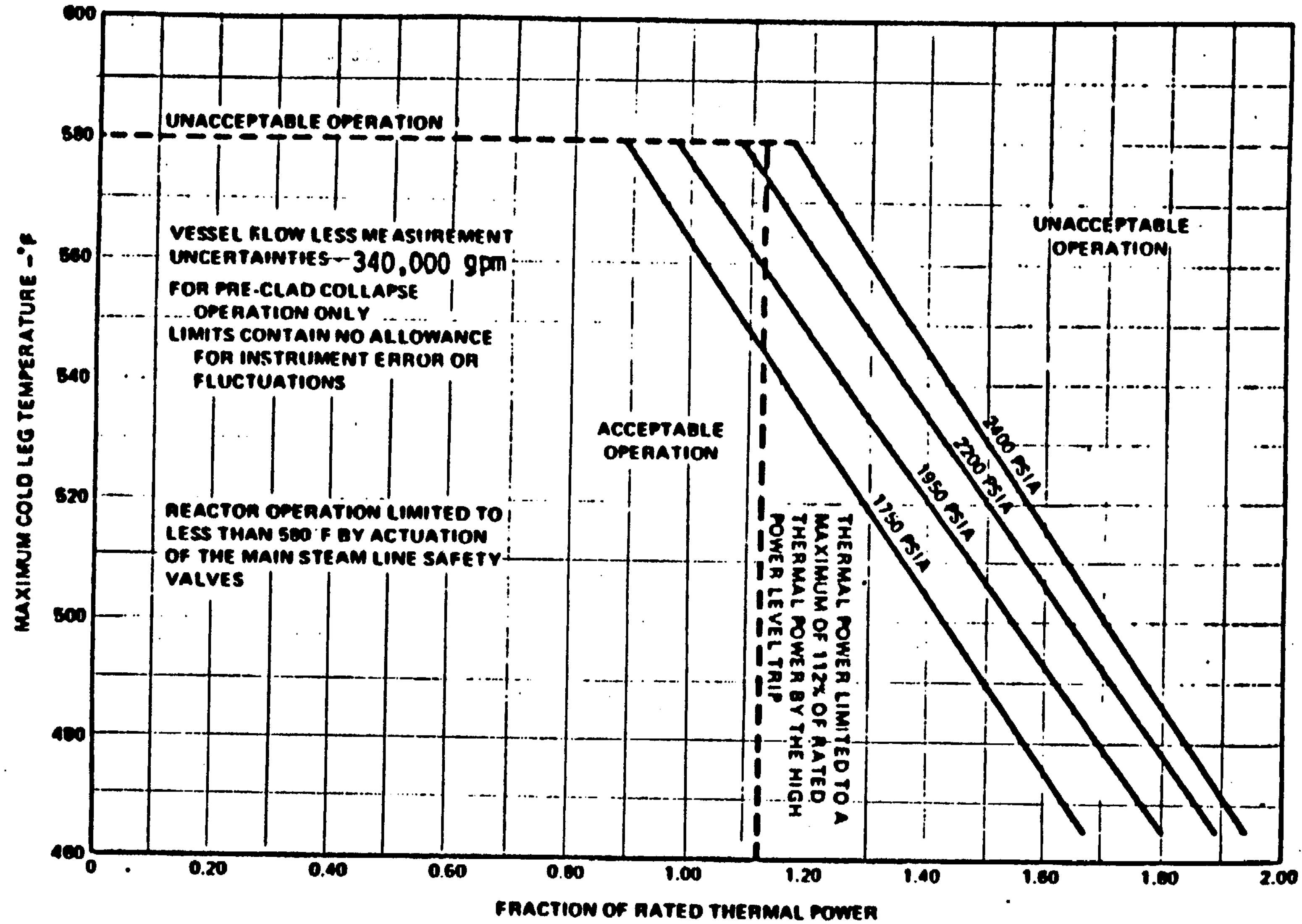


FIGURE 2.1-1 Reactor Core Thermal Margin Safety Limit – Four Reactor Coolant Pumps Operating

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

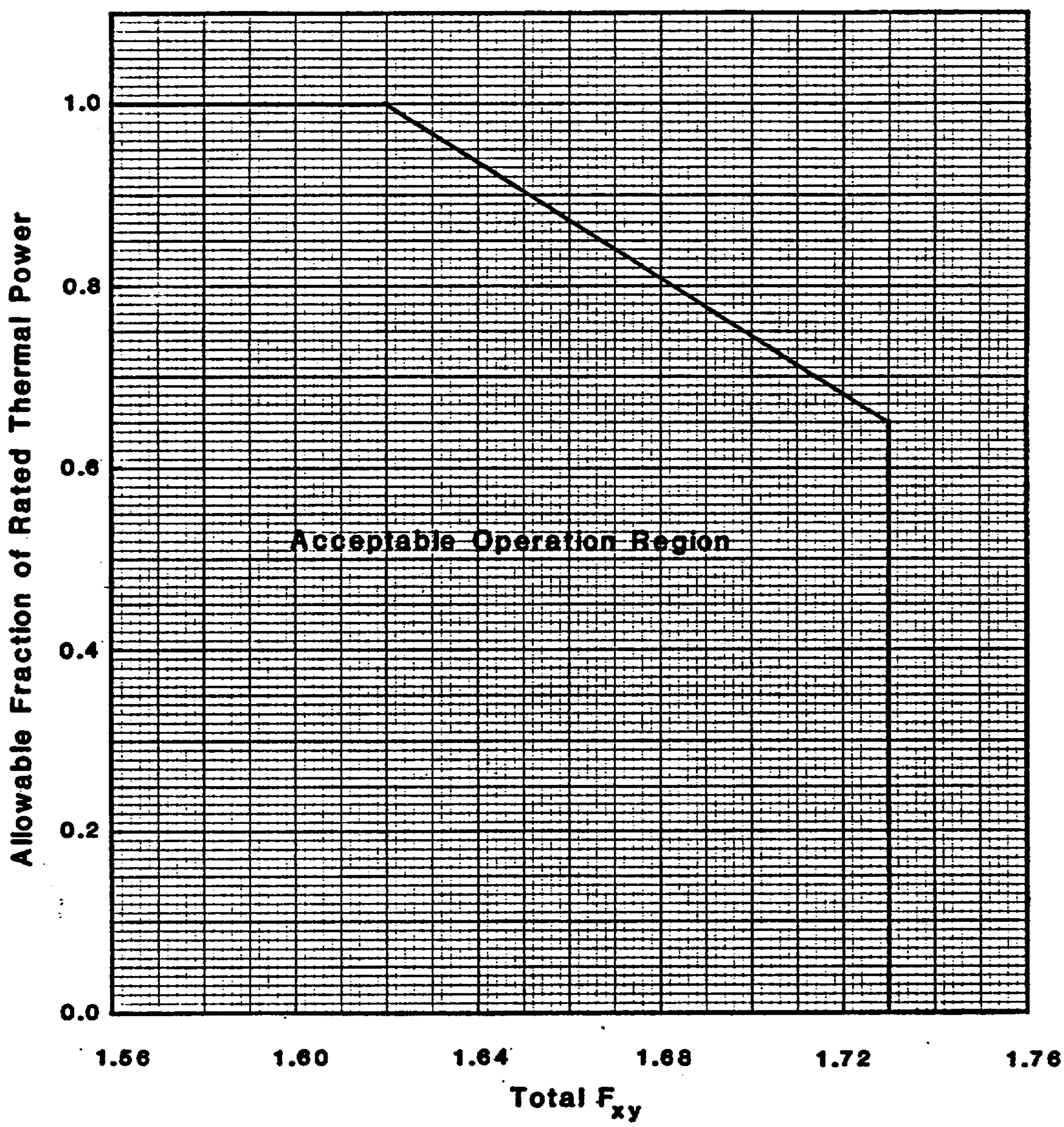
TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level-High		
Four Reactor Coolant Pumps Operating	≤ 9.6% above THERMAL POWER, with a minimum setpoint of ≤ 14.6% of RATED THERMAL POWER, and a maximum of ≤ 106.6% of RATED THERMAL POWER.	≤ 9.7% above THERMAL POWER, with a minimum of ≤ 14.7% of RATED THERMAL POWER, and a maximum of ≤ 106.7% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 91.7% of reactor coolant flow with 4 pumps operating*.	≥ 90.1% of reactor coolant flow with 4 pumps operating*.
4. Reactor Coolant Pump Speed - Low	≥ 830 rpm	≥ 823 rpm
5. Pressurizer Pressure - High	≤ 2400 psia	≤ 2408 psia
6. Containment Pressure - High	≤ 4.75 psig	≤ 5.23 psig
7. Steam Generator Pressure - Low (2) (5)	≥ 500 psia	≥ 492 psia
8. Steam Generator Water Level - Low (5)	≥ 36.0% Water Level - each steam generator	≥ 35.2% Water Level - each steam generator
9. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2 (4).

* Design Reactor Coolant flow with 4 pumps operating is 340,000 gpm.

FIGURE 3.2-3a Total Radial Peaking Factor vs Allowable Fraction of Rated Thermal Power



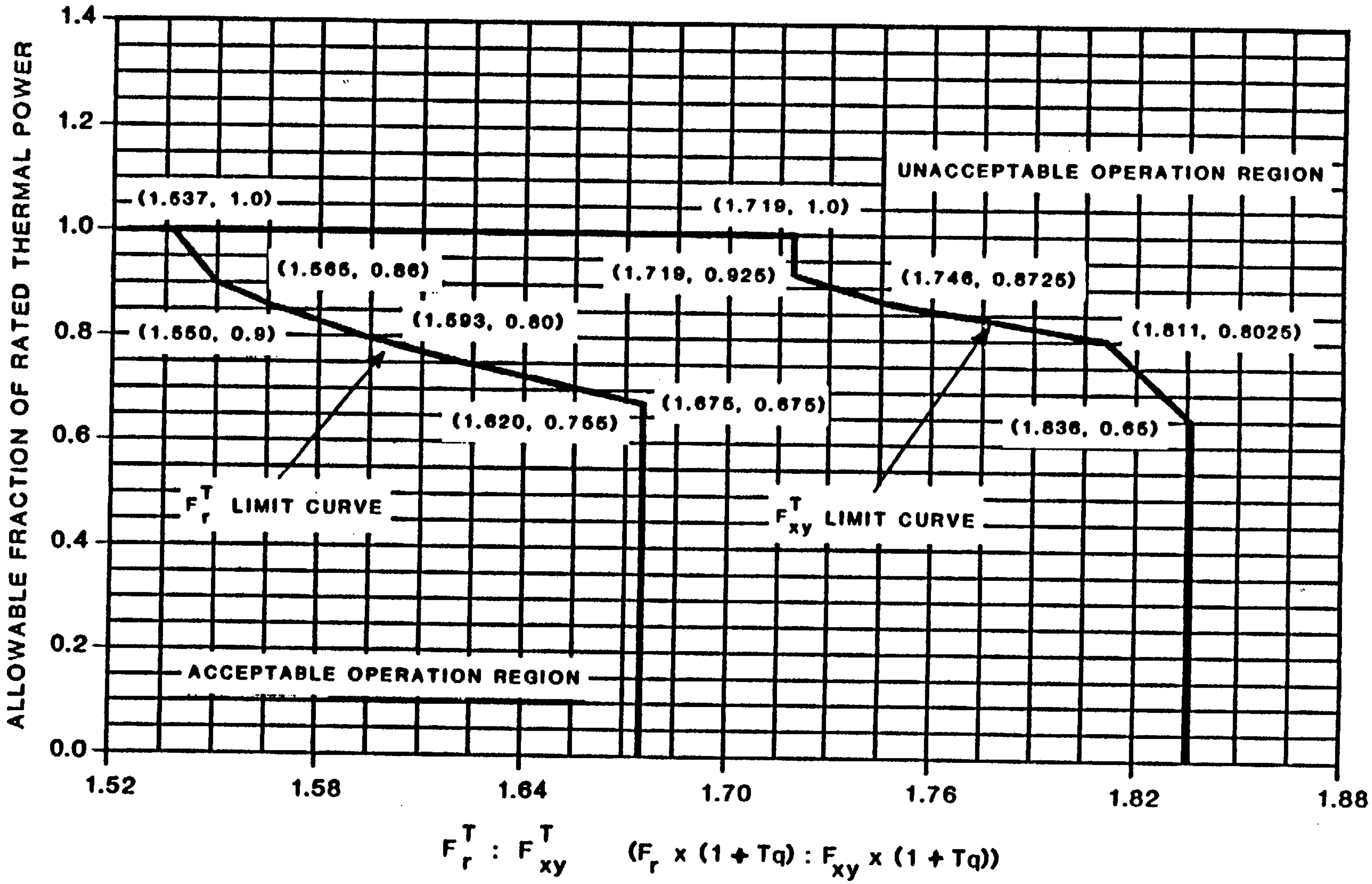


FIGURE 3.2-3b Total Radial Peaking Factor vs. Allowable Fraction of RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$ shall be limited to $\leq$ 1.537.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.537, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3b and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ and F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.020.

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump Combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

*See Special Text Exception 3.10.2.

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.02 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days when the Channel High Deviation Alarm is OPERABLE,

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

DNB MARGIN

LIMITING CONDITION FOR OPERATION

3.2.6 The DNB margin shall be preserved by maintaining the cold leg temperature, pressurizer pressure, reactor coolant flow rate, and AXIAL SHAPE INDEX within the limits specified in Table 3.2-1 and Figure 3.2-4.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its specified limits, restore the parameter to within its above specified limits within 2 hours or reduce THERMAL POWER to ≤ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.6.1 The cold leg temperature, pressurizer pressure, and AXIAL SHAPE INDEX shall be determined to be within the limits of Table 3.2-1 and Figure 3.2-4 at least once per 12 hours. The reactor coolant flow rate shall be determined to be within the limit of Table 3.2-1 at least once per 31 days.

4.2.6.2 The provisions of Specification 4.0.4 are not applicable.

TABLE 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 549°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 340,000 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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 20°F in any one hour period with T_{avg} at or below 110°F, 30°F in any one hour period with T_{avg} at or below 140°F and above 110°F, and 50°F in any one hour period with T_{avg} above 140°F.

b. A maximum cooldown of 80°F in any one hour period with T_{avg} above 300°F and a maximum cooldown of 30°F in any one hour period with T_{avg} at or below 300°F and above 200°F, and 20°F in any one hour period with T_{avg} at or below 200°F.

c. A maximum temperature change of 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

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

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation 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 or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to making the reactor critical.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-3. The results of these examinations shall be used to update Figure 3.4-2.

FIGURE 3.4-2

MILLSTONE UNIT 2

REACTOR COOLANT SYSTEM

PRESSURE TEMPERATURE LIMITATIONS FOR 12 FULL POWER YEARS

(P_{error} = 24psi, T_{error} = 6°F)

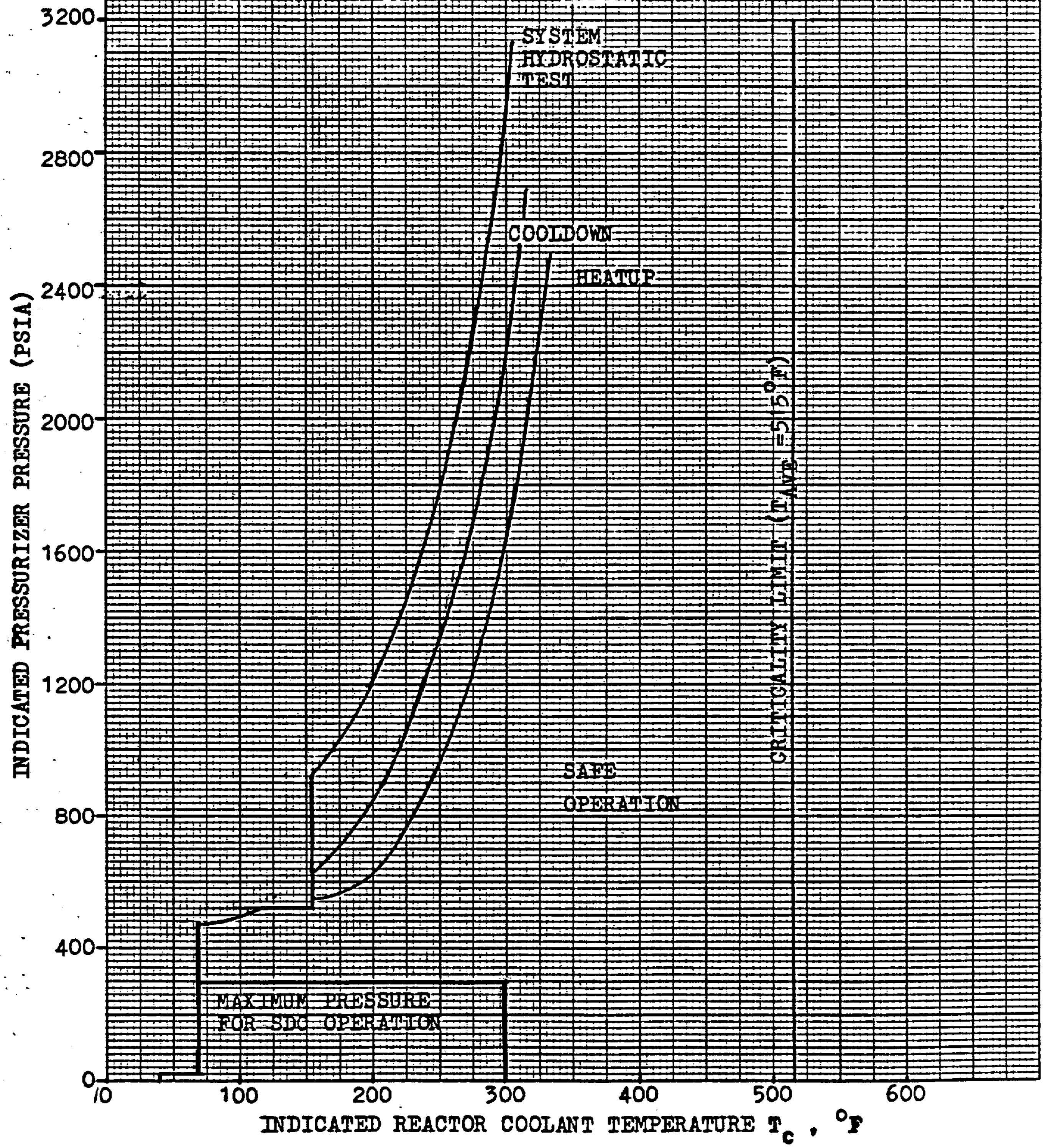


TABLE 4.4-3

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

CAPSULE	SCHEDULE (EFPY)
W-97	3.0
W-104	10.0
W-284	17.0
W-263	24.0
W-277	32.0
W-83	Spare
W-97 (Flux Monitor)	10.0

Reducing T_{avg} to < 515°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with iodine spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.0 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup and cooldown limit curves (Figure 3.4-2) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to the maximums described in Section 3.4.9.1. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table 4.6-1 of the Final Safety Analysis Report. Reactor operation and resultant fast neutron irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using the methods described in SECY-82-465, "NRC Staff Evaluation of Pressurized Thermal Shock," November 1982. Because it is more conservative, this method was used rather than the proposed Revision 2 to Regulatory Guide 1.99.

The heatup and cooldown limit curves shown on Figure 3.4-2 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to the RT_{NDT} + 100°F

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

INTRODUCTION

By applications for license amendments dated October 20, October 24, and October 27, 1986, Northeast Nuclear Energy Company, et al. (the licensee or NNECo), requested changes to the Technical Specification (TS's) for Millstone Unit No. 2. The proposed changes to the TS provide for: (1) revised temperature/pressure limits in TS 3/4.4.9, "Pressure/Temperature Limits" and TS Figure 3.4.2, "Reactor Coolant System Pressure Temperature Limitations for 12 Full Power Years," (2) a change to the surveillance frequency for determining reactor coolant system (RCS) flow rate in TS 4.2.6, "DNB Margin," and (3) changes to several TS associated with RCS flow and reactor power peaking limits.

The proposed changes to the TS associated with RCS flow rate and reactor power peaking limits were necessitated by the expectation that steam generator tube repair (plugging and/or sleeving) would result in a decrease in the RCS flow rate. The licensee had previously submitted revised loss-of-coolant accident (LOCA) calculations which assume a reduced RCS flow rate. The LOCA calculations on the model on which they are based are evaluated herein.

The October 27, 1986 application (Ref. 1) contains proposed TS which support Cycle 8 operation. The safety analyses are described in the Cycle 8 Reload Safety Evaluation (RSE) report (Ref. 2).

The Millstone Unit 2 Cycle 8 safety analysis is based on the Basic Safety Report (BSR) (Ref. 3). The BSR serves as the reference fuel assembly and safety analysis report for the use of Westinghouse fuel at Millstone Unit 2, a Combustion Engineering designed PWR. References 4, 5, and 6 document the staff's review and acceptance of the BSR. In addition to the BSR, the analysis and evaluation of Cycle 8 was performed using the methodology of Reference 7. This methodology was approved in Reference 8.

The Cycle 8 safety analysis has been performed for a reactor coolant system (RCS) flow rate of 340,000 gpm. This flow rate is based on an assumed plugging of 1500 tubes in each steam generator for a total of 17.6 % of all of the tubes. Plugging such a number of steam generator tubes affects not only the steam generator heat transfer area but also the RCS flow rate and, to a lesser extent, the RCS volume. This RCS flow rate is applicable to the projected end-of-cycle (EOC). The Cycle 8 safety analysis also includes extended cycle operation beyond the projected full-power EOC by means of power and temperature reductions. This extended cycle operation, however, is based on a RCS flow rate of 350,000 gpm. NNECO would have to perform additional LOCA analyses to support extended coastdown operation beyond EOC 8 at the reduced RCS flow rate of 340,000 gpm.

2.0 DESCRIPTION OF THE CYCLE 8 CORE

The Millstone Unit 2 reactor core consists of 217 fuel assemblies, each of which is a 14 x 14 array containing 176 fuel rods and 5 control rod guide tubes made of Zircaloy. The fuel rods consist of slightly enriched uranium dioxide, sintered, ceramic pellets placed in Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. The fuel assemblies each have top and bottom stainless steel nozzles. The fuel assembly structure includes 9 Inconel grids for 212 of the fuel assemblies, while 5 fuel assemblies have Zircaloy grids. Each control rod guide tube occupies 4 lattice positions in the 14 x 14 array.

The fuel management scheme is based on the so-called "out-in" strategy with loading pattern and fresh fuel enrichments chosen to provide a Cycle 8 burnup of 9,400 MWD/MTU. All of the fuel for Cycle 8 has been manufactured by Westinghouse. The loading pattern is as follows: Sixteen fresh fuel assemblies, designated Region K1, are distributed in the interior of the core with eight fuel assemblies arranged in an interior ring about the center location and eight fuel assemblies in an outer ring near the core periphery. These fuel assemblies are zone enriched, each containing 60 fuel rods at 2.6 weight percent (w/o) uranium-235 placed around the control rod guide tubes and 116 fuel rods at 2.9 w/o uranium-235 occupying the remaining fuel lattice positions. Forty-eight fresh fuel assemblies, designated Region K2, are placed in each of the core periphery locations. These fuel assemblies are zone enriched, each containing 60 fuel rods at 2.9 w/o uranium-235 placed around the guide tube locations and 116 fuel rods at 3.3 w/o uranium-235 occupying the remaining fuel lattice positions. One Region F fuel assembly, removed from the core at the end of Cycle 5, will be reinserted at the center core location for Cycle 8. Four Region H1 fuel assemblies, removed from the core at the end of Cycle 6, will be reinserted into the core in addition to 20 Region H1 fuel assemblies from Cycle 7. Cycle 8 also includes the following fuel regions from Cycle 7 scatter-loaded in the interior of the core: 12 Region G2 fuel assemblies, 44 Region H2 fuel assemblies, 24 Region J1 fuel assemblies, and 48 Region J2 fuel assemblies. The one Region F2 and 4 of the Region G2 fuel assemblies have been reassembled using Combustion Engineering (CE) fuel assembly skeletons (i.e., bottom and top nozzles, and other structural members). Eighteen fuel rods have been replaced with stainless steel rods in eight fuel assemblies with the number of replaced fuel rods per assembly varying from one to five.

The safety analysis for Cycle 8 used the design parameters shown in Table 1.

These parameters are the same as for the previous Cycle 7 core except for the RCS flow rate and assumed steam generator tube plugging level. NNECO has also considered operation beyond the projected end of Cycle 8. The safety analysis for this extended mode of operation was based on the following assumptions:

1. Cycle 8 is to be extended beyond EOC 8 by a maximum of 1000 MWd/MTU.

2. Cycle 8 full power operation is to be extended as much as possible by core inlet temperature reduction.

3. After the full power operation limit is reached, power operation is to be extended by a combination of power and core inlet temperature reductions.

4. The hot zero power (HZP) coolant temperature is a constant 532°F.

5. Peak linear heat generation rate is less than or equal to 15.6 kW/ft.

6. Extended operation is based on a minimum guaranteed flow of 350,000 gpm.

Since the extended Cycle 8 operation safety analysis is based on a higher RCS flow rate than that assumed in the Cycle 8 safety analysis, such extended Cycle 8 operation must be addressed by NNEC in a future submittal.

Table 1
Cycle 8 Design Parameters

Core Power (MWT)	2700
System Pressure (psia)	2250
Reactor Coolant Flow (gpm)	340,000
Core Inlet Temperature (°F)	549
Average Linear Power Density (kW/ft)	6.067
Steam Generator Tube Plugging (%)	17.6
Cycle 7 burnup (MWd/MTU)	10,700 (-0,+1000)

3.0 EVALUATION OF THE FUEL SYSTEM DESIGN

The fuel system design for Millstone Unit 2 Cycle 8 is the same as that approved (Ref. 6) for previous fuel cycles with Westinghouse fuel. The Cycle 8 core contains only Westinghouse manufactured fuel rods. The fresh fuel assemblies for Cycle 8 are the same mechanically as the previous Cycle 7 fuel except that the fuel rod backfill pressure has been reduced. The rod internal pressure in the Westinghouse fuel will not exceed the primary system coolant pressure during Cycle 8 (Ref. 3). Since approved methods and methodology have been used in the fuel system design, the staff concludes that it is acceptable.

4.0 EVALUATION OF THE NUCLEAR DESIGN

The nuclear design procedures and models used for the analysis of the Millstone Unit 2 Cycle 8 reload core (Refs. 1 and 2) are the same as those used for Cycle 7. These are documented in the Millstone Unit 2 BSR (Ref. 3) and have been approved (Ref. 4) for the analysis of the Millstone Unit 2 core using Westinghouse reload fuel beginning with Cycle 4. In addition, Reference 7 documents the methodology used by Westinghouse for performing this as well as other reloads. This methodology was approved in Reference 8.

The nuclear design analysis for Cycle 8 specifically included the zone-enriched fuel assemblies, the 18 stainless steel rods in reconstituted fuel assemblies and the loading pattern of the various fuel batches, previously described, in order to determine maximum linear heat generation rates achievable in normal operation, control rod worths for the shutdown margin evaluation, and the kinetics parameters for use in the transient and accident evaluation. These calculations were performed with approved methods and are, therefore, acceptable.

In Table 2 of Reference 2, the kinetics parameters for the Cycle 8 core are given and compared to current limits. They are all within current limits except for a minor difference in the range of the delayed neutron fraction and for a difference in the maximum differential rod worth at HZP. The maximum differential rod worth is nearly 3 times greater than the current limit. The effect of these differences in the kinetic parameters are included in the Cycle 8 safety analysis.

The control rod worths and shutdown margin requirements at the most limiting time (EOC) for the Cycle 8 nuclear design are presented in Table 3 of Reference 2. At EOC 8, the reactivity worth with all control rods inserted, assuming that the highest worth control rod is stuck out of the core, is 6.15%. This reactivity worth also assumed a reduction in worth of 10% for uncertainty. The reactivity worth required for shutdown, including the contribution required to accommodate the reactivity effects of the steamline break event at EOC 8, is 6.12%. Therefore, sufficient control rod worth is available to accommodate the reactivity effects of the steamline break event at the worst time in core life allowing for the most reactive control rod stuck in the fully withdrawn position and allowing for calculational uncertainties. For extended power operation, Reference 2 states that sufficient shutdown margin exists from the beginning (EOC 8) to the end of the period of extended power operation. The staff has reviewed the calculated control rod worths and uncertainties in these worths based upon comparison of calculations with experiments presented in the BSR (Ref. 3) and other Westinghouse reports. On the basis of this review, the staff concludes that NNECO's assessment of reactivity control is suitably conservative and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability assuming the most reactive control rod is stuck in the fully withdrawn position.

5.0 EVALUATION OF THE THERMAL-HYDRAULIC DESIGN

Millstone Unit 2 Cycle 8 utilized the BSR (Ref. 3) thermal-hydraulic design methods which were approved by the staff in Reference 6. The BSR was also used as the basis for Cycles 4 through 7 operation. As was done for Cycle 7, the stainless steel rods in the reconstituted fuel assemblies were treated as heated rods in the DNB analysis. This is conservative since it results in higher subchannel enthalpy predictions. The Cycle 7 analysis takes a partial credit of 3.0% of the net conservatism which exists between convoluting and summing the uncertainties of measured plant power parameters in terms of power. This partial credit was applied in previous cycles and is discussed in more detail in a staff SER for Cycle 4 (Ref. 9).

The DNB analysis for Cycle 8 was performed for a minimum RCS flow rate of 340,000 gpm and a radial peaking factor, F_r, of 1.537. A reduction in flow rate from 370,000 gpm to 362,500 gpm and a conservative reduction in F_r from 1.63 to 1.597 was previously implemented during Cycle 5 operation. A further reduction in flow rate from 362,500 gpm to 350,000 gpm and a conservative reduction in F_r from 1.597 to 1.565 was implemented for Cycles 6 and 7 operation. As previously indicated by the power and flow sensitivities reported for Cycle 4 (Ref. 10, page 5), a flow reduction can be offset by a power (or F_r) reduction in a 2:1 ratio to maintain a constant DNBR. Thus the reduction in flow rate has been more than offset by the reduction in the radial peaking factor and this has been confirmed in the Cycle 8 analyses and the staff concludes, therefore, that the reduction of flow rate for Cycle 8 is acceptably offset by a reduction in the radial peaking factor.

6.0 EVALUATION OF THE ACCIDENT AND TRANSIENT ANALYSES

The licensee examined each transient and accident analysis presented in the BSR, and updated in safety analyses for subsequent cycles of operation, to ensure that the calculated consequences still meet applicable criteria. The reloaded core parameters that may affect the outcome of a transient or accident are typically the core kinetic parameters, the shutdown margin, control element assembly (CEA) worths, and core peaking factors. In addition, the impact of the assumption of 17.6 percent steam generator tube plugging on previous safety analyses had to be determined. The plugging of steam generator tubes leads to a reduced steam generator heat transfer area, a slight reduction in RCS volume, and a reduction in the RCS flow rate.

The licensee evaluated the various Cycle 8 reload core parameters against current limits. For the kinetic parameters, the maximum delayed neutron fraction and the least negative Doppler power coefficient above 30 percent power exceed the current limits. These small changes were determined not to invalidate previous analyses. Shutdown margin requirements were determined to be the same as for Cycle 7. The CEA worths were determined to satisfy shutdown margin requirements. However, the increase in the Cycle 8 maximum differential control rod worth necessitated the reanalysis of the CEA withdrawal from a subcritical configuration. This reanalysis is discussed below. All Cycle 8 core peaking factors were determined to be within the reference cycle limits.

The licensee evaluated the following transients and accidents for Cycle 8:

1. CEA withdrawal from a subcritical condition
2. CEA withdrawal at power
3. RCS depressurization
4. Loss of flow
5. CEA ejection
6. Malfunction of one steam generator
7. Steamline break
8. Uncontrolled boron dilution
9. Excess heat removal due to feedwater malfunction
10. Startup of an inactive reactor coolant pump
11. Excessive load event
12. Loss of load and/or turbine trip
13. Loss of normal feedwater
14. CEA drop
15. Single reactor coolant pump seized rotor

The staff's review and evaluation of each of the listed events is presented below.

The CEA withdrawal from a subcritical condition transient leads to a power excursion. This power excursion is terminated, after a fast power rise, by the negative Doppler reactivity coefficient. The power excursion results in a heatup of the coolant. The coolant temperature rise is small since the power excursion is very rapid with an immediate reactor trip. The nuclear power excursion is dominated by the Doppler reactivity coefficient. The analysis was performed using approved methods and the principal assumptions employed in the BSR analysis for this event. The reactivity insertion rate was 76.7×10^{-5} delta k/k/sec and a variable high power (VHP) trip setpoint of 25 percent was used. The results show that the peak nuclear power was 358 percent of nominal, the peak heat flux was 46.5 percent of nominal, the peak average fuel temperature was 1625°F, and the peak clad inner temperature was 744°F. All applicable criteria are met and, in particular, the DNB design basis.

The CEA withdrawal at power transient is a reactivity insertion at power caused by a continuous, uncontrolled CEA withdrawal because of equipment malfunctions or operator error. The transient results in an increase in the core heat flux and an increase in the reactor coolant temperature. The analysis was performed using an approved method to determine the effect of the Cycle 8 core physics parameters and reduced RCS flow rate on the minimum DNBR. Both maximum and minimum reactivity feedback cases were considered. The variable high power and thermal margin/low pressure trips were assumed to be operable in the analysis. In the analysis, the reactor tripped by the high power level (112 percent of full power) trip function. The DNBR limit was not violated and, consequently, the applicable criterion for this event is met.

The RCS depressurization transient is caused by the opening of both pressurizer relief valves while the reactor is at power. The transient causes the RCS pressure to decrease, the RCS temperature to decrease, and the pressurizer level to increase. This transient is terminated by the thermal margin/low pressure trip function. The analysis, with an approved method, indicates that the thermal margin/low pressure trip provides adequate protection against DNB. The DNBR limit was not violated and, consequently, the applicable criterion for this event is met.

The loss of flow transient may occur because of the loss of electrical power to the reactor coolant pumps. The transient causes the reactor coolant temperature and pressure to increase. This transient is terminated by the reactor coolant pump speed sensor and the low reactor coolant flow sensor. The analysis, performed with approved methods, indicates that the reactor trips on low reactor coolant flow rate. The maximum fuel average temperature in the hot channel was 1999°F and the maximum nuclear power was 102.8 percent of nominal power. The DNBR limit is not violated and, consequently, the applicable criterion for this event is met.

The CEA ejection accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the CEA ejection. This accident leads to a reactivity insertion which could lead to localized core damage. The core power rise is limited by Doppler reactivity feedback and the accident is terminated by a reactor trip on a high power level signal. For HZP, the variable high power trip with a 25 percent power setpoint is assumed to be effective while, for hot full power (HFP), the variable high power trip with a setpoint of 112 percent power setpoint is assumed to be effective. The analysis, performed with approved methods, indicates that, for both the HZP and HFP cases, the average fuel pellet enthalpy does not exceed the NRC criterion of 280 calories per gram. Based on the results obtained by the licensee, the staff concludes that the applicable criterion for this accident is met.

The malfunction of (loss of load to) one steam generator transient is assumed to occur due to the closing of the main steam line isolation valve to one steam generator. This transient causes the temperature and pressure in the isolated steam generator to rise until the safety valves lift. The unisolated steam generator attempts to compensate for the load imbalance which leads to a RCS inlet temperature imbalance. The colder water from the unisolated steam generator causes a positive reactivity insertion, due to the negative moderator temperature coefficient. The resultant power increase would trip the reactor due to high power level.

This high power trip was not assumed to be available in the analysis but rather the trip was assumed to occur on low steam generator water level. This is a conservative assumption since it delays a reactor trip. The analysis, performed with approved methods, indicates that the isolated steam generator temperature increases to 566°F and the pressure increases to 1020 psia, the unisolated steam generator temperature decreases to 506°F while its pressure decreases to 497 psia, and the core nuclear power increases to 130.8 percent of nominal full power. The analysis indicates that the DNBR limit is not violated and, therefore, the applicable criterion for this event is met.

A steamline break accident may result from a stuck open safety or relief valve or a ruptured steamline. Such an accident would result in a rapid depressurization of the steam generators which causes a primary system cooldown. This RCS cooldown, in conjunction with a negative moderator temperature coefficient, causes a positive reactivity insertion and a possible return to power. The licensee determined that the effect of a reduced RCS flow rate, reduced RCS coolant inventory, and reduced heat transfer coefficient, due to the steam generator tube plugging, resulted in a less severe cooldown of the primary coolant system than in previous evaluations performed for higher RCS flow rates. In addition, Cycle 8 peaking factors were used with the existing accident statepoint. The licensee concludes from this evaluation that the DNB design basis is met for this accident. The staff concurs with the licensee's assessment of the steam line break accident for Cycle 8, including the effect of the steam generator tube plugging.

An uncontrolled boron dilution transient results from the addition of pure water to the RCS by the Chemical and Volume Control System (CVCS). Manual operations must be performed to cause an uncontrolled boron dilution event. An uncontrolled boron dilution transient results in a positive reactivity addition to the core. This transient was evaluated by the licensee to account for the approximately 5 percent reduction in RCS coolant volume due to the steam generator tube plugging. This reduction in volume affects only those events for which the analysis assumes that the RCS is filled. The staff's criteria require a minimum time allowance of 30 minutes for operator intervention to terminate the transient during the refueling mode and 15 minutes during any other mode of operation. The licensee has demonstrated by the reanalysis provided in Reference 2 that the criteria have been met. The limiting boron dilution transient occurs for the cold shutdown mode of operation for which 19 minutes is the calculated time to a loss of shutdown margin. The staff concludes, therefore, that NNECO meets the criteria for this transient for Cycle 8.

The excess heat removal due to feedwater malfunction transient was not analyzed since it is bounded by the steam line break accident, which is the limiting cooldown event. The startup of an inactive reactor coolant pump transient was not analyzed since Millstone Unit 2 operation with less than 4 pumps is not permitted. The excessive load event was not analyzed since it is bounded by the steam line break accident, which is the limiting cooldown event.

A loss of load and/or turbine trip transient leads to a mismatch between the power generated in the reactor primary system and the power extracted by the secondary system. This power mismatch represents an undercooling event which results in a core temperature and pressure increase. No credit is taken in the BSR analysis for reactor trips other than trips on pressurizer high pressure. The effect of the steam generator tube plugging results in a reduction in both the RCS flow rate and volume. The reduced RCS flow rate and volume would cause the reactor to trip earlier on high pressurizer pressure and result in a lower total energy input into the coolant than for the BSR analysis. The licensee states that DNBR limits will not be violated and that the conclusions of the BSR for this transient remain valid. The staff concurs with the licensees's assessment of this transient.

A loss of normal feedwater transient leads to a mismatch between the power generated in the reactor primary system and the power extracted from the secondary system. The reduced RCS flow rate and volume due to steam generator tube plugging would result in a more rapid heatup of the primary system as compared to the analysis presented in the BSR. This would increase the volume of water in the pressurizer. The licensee performed sensitivity studies to estimate the expected increase in pressurizer volume. The licensee determined that the reduced RCS flow rate and volume would increase the water level in the pressurizer during the transient by a negligible amount. The licensee concludes that the conclusions in the BSR for this transient remain valid. The staff concurs with this assessment.

The CEA drop transient results in a mismatch between the power produced in the primary system and the power extracted by the secondary system. If the reactor does not trip, it may return to power with, perhaps, a power overshoot. This return to power is a consequence of the positive reactivity insertion caused by a negative moderator temperature coefficient. Subsequent to the BSR analysis, the licensee reanalyzed this event for a reduction in RCS flow rate from 370,000 gpm to 362,500 gpm. The results indicated that DNBR did not decrease below its initial value and that the BSR analysis was more limiting. The licensee concludes that, for a reduction in RCS flow rate to 340,000 gpm, the BSR analysis remains more limiting. The licensee stated that, to provide additional confirmation of this conclusion, an analysis was performed of the existing statepoint for this transient using Cycle 8 peaking factors. This limited reanalysis confirmed that the DNB design basis was met. The staff concurs, therefore, with the licensee's assessment of this transient.

A single reactor coolant pump (RCP) seized rotor accident is initiated by an instantaneous seizure of a RCP rotor at full power. The reduced coolant flow due to steam generator tube plugging does not affect the time to DNB since DNB, in previous analyses, is conservatively assumed to occur at the beginning of the transient. The flow coastdown for this accident is so rapid that the reactor trip low flow setpoint would be reached at nearly the same time as in the most recent analyses. The licensee states that the peak pressure will not increase above the previous value of 2778 psia which is well below the pressure at which vessel stress limits are exceeded.

The licensee states that peak temperatures or pressures are reached in considerably less than one loop transport time constant and, therefore, the effect of reduced RCS flow rate is not significant. The licensee recalculated the current statepoint using Cycle 8 peaking factors and RCS flow rate and confirmed that the number of fuel rods experiencing DNB remained less than the current limit value of 3 percent. The staff concurs with the licensee's assessment of this accident for Cycle 8 operation.

Based on its evaluation of the results presented by the licensee in Reference 2, the staff concludes that the licensing basis analysis conclusions, as presented in the BSR, remain valid for Cycle 8 operation for a RCS flow rate of 340,000 gpm.

7.0 LOSS OF COOLANT ACCIDENT (LOCA) EVALUATION

By letter dated August 29, 1986 (Ref. 11), Northeast Nuclear Energy Company (NNECo) provided revised LOCA analyses for Millstone Unit 2. The purpose of these analyses was to support the Millstone Unit 2 Cycle 8 reload, as well as satisfy NUREG-0737 Item II.K.3.31. Supplemental information supporting these analyses was provided in References 12 and 13.

This safety evaluation addresses the LOCA analyses performed for the Millstone Unit 2 Cycle 8 reload. Additionally, the report evaluates the Millstone Unit 2 compliance with the requirements of NUREG-0737 Item II.K.3.31.

7.1 Large Break LOCA Analysis

A large break LOCA analysis was provided in Reference 11 to support the Cycle 8 reload. The previous analysis for Millstone Unit 2 assumed 15.3 percent of the steam generator tubes were plugged and a reactor coolant system (RCS) flow rate of 350,000 gpm. The revised analysis is based upon 23.4 percent tube plugging and a reduced RCS flow of 335,000 gpm. These values bound the values used for the remainder of the Cycle 8 safety analyses (Ref. 1) which assumed 17.6 percent tube plugging and a RCS flow rate of 340,000 gpm.

The analysis was performed for the worst case large break LOCA, a double-ended cold leg guillotine break with a discharge coefficient of 0.6. Additionally, the analysis included the effects of removal of the thermal shield, adding additional steel in containment and updating of the safety injection flows. The analysis was based upon the approved Westinghouse 1981 evaluation model which includes the modification in response to NUREG-0630 (Ref. 15).

The results of the worst case large break LOCA evaluation were a peak cladding temperature of 2142°F, maximum local metal-water reaction of 6.17%, and a total core metal-water reaction of less than 0.3%. These all satisfy the requirements of 10 CFR 50.46 which requires peak cladding temperature of less than 2200°F, maximum local metal-water reaction of less than 17%, and a total core metal-water reaction of less than 1%.

During the staff's review of the licensee's calculations, the staff requested the licensee to evaluate whether the evaluation model utilized for the Millstone Unit 2 analysis contained the modelling error for the control rod thimbles for which Westinghouse notified the staff on June 2, 1986. The licensee responded (Ref. 12) that the old method of modelling the control rod thimbles existed in the Reference 11 analysis. The licensee evaluated the impact of the modelling change and reported that the impact would increase peak cladding temperatures by less than 20°F. Thus, the licensee concluded that even with correction of the error, Millstone Unit 2 still complied with the requirements of 10 CFR 50.46.

The staff generically evaluated the control rod thimble modelling error in Reference 16 and concluded that a new ECCS analysis is not required for licensees that utilized the 1981 version of the Westinghouse ECCS evaluation model. This conclusion was based upon the fact that the error was small and would not result in the peak cladding temperatures for any current analyses exceeding 2200°F. Accordingly, the staff finds the ECCS evaluation model utilized for Millstone 2 Cycle 8 acceptable.

Since the evaluation model utilized complies with Appendix K to 10 CFR Part 50 and the results of the worst case large break LOCA meets the requirements of 10 CFR 50.46, the staff finds the large break LOCA analyses acceptable.

Additionally, since the assumptions used for the large break LOCA analysis bounds the values used for the Millstone Unit 2 Cycle 8 safety analyses and the supporting plant Technical Specifications, the staff finds Cycle 8 operation to be acceptable.

7.2 Small Break LOCA Analysis

A revised small-break LOCA evaluation was also performed to support Cycle 8 operation for Millstone Unit 2. The worst case small-break, a 4-inch cold leg pump discharge break, was reanalyzed assuming that 23.4 percent of the steam generator tubes were plugged and a minimum RCS flow rate of 335,000 gpm. These analyses were performed using the small-break LOCA ECCS evaluation model described in Reference 17. The results of the calculation yielded a peak cladding temperature of 2135°F, a local metal-water reaction of 10.7%, and whole-core metal-water reaction of less than 0.3%.

The staff has previously reviewed the small-break LOCA evaluation model and found it in compliance with Appendix K to 10 CFR Part 50 (Ref. 18). The staff also concluded in Reference 18 that this model satisfied the requirements of NUREG-0737 Item II.K.3.30.

During its review, the staff expressed concern that the 4-inch line break may not be the limiting break for Millstone Unit 2. Within Reference 17, the results of a spectrum of breaks, which were analyzed for Millstone Unit 2, were reported. These included 3-, 4-, and 6-inch cold leg breaks. It appeared that a break size between the 3- and 4-inch line breaks, which would not result in accumulator injection, could yield a higher peak cladding temperature. Supplemental information was provided in References 3 and 10 which demonstrated that the 4-inch line break was the limiting break for Millstone Unit 2. The staff reviewed the additional information and concluded in Reference 18 that the 4-inch line break was indeed the limiting break size for Millstone Unit 2.

Thus, the staff finds that an approved ECCS evaluation model was utilized for the small-break LOCA analyses. Additionally, the consequences for the limiting break satisfy the requirements of 10 CFR 50.46. Thus, the staff finds that the small-break analysis for Millstone Unit 2 is in compliance with 10 CFR 50.46.

The small-break LOCA analysis performed for Millstone Unit 2 was also performed to satisfy the requirements of NUREG-0737 Item II.K.3.31. This item required licensees to demonstrate compliance with 10 CFR 50.46 using NRC-approved models which satisfied NUREG-0737 Item II.K.3.30. As noted above, the staff has concluded that the small-break LOCA evaluation model used satisfied II.K.3.30, and the plant specific calculations meet the requirements of 10 CFR 50.46. Therefore, the staff finds that the small-break LOCA analysis for Millstone Unit 2 fulfills the requirements of NUREG-0737 Item II.K.3.31.

8.0 PRESSURE/TEMPERATURE LIMITATION

The October 20, 1986 application requested changes to the TS associated with pressure/temperature limitations. The proposed TS change involves new reactor vessel pressure-temperature limits for heatup, cooldown and the inservice hydrostatic test to be valid for 12 effective full power years (EFPY).

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50. Pressure-temperature limits are dependent upon the initial reference temperature (RT_{NDT}) for the controlling (limiting) materials in the beltline and closure flange regions of the reactor vessel and the increase of RT_{NDT} resulting from neutron irradiation damage to the controlling beltine materials. USNRC Standard Review Plan Section 5.3.2, NUREG-0800, Revision 1, July 1981, is used to evaluate the acceptability of pressure-temperature limits for reactor vessels. Appendix H, 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," supplements Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements," by requiring a program from which fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

The most recent surveillance capsule report "Evaluation of Irradiated Capsule W-97," Reactor Vessel Materials Irradiation Surveillance Program, April 1982 (TR-N-MCM-008) was submitted on January 4, 1984. This report was followed by additional responses from the licensee dated February 3, 1984 and March 16, 1984. Combustion Engineering (CE) performed the evaluation of the capsule for Millstone, Unit 2.

The increase in RT_{NDT} resulting from neutron irradiation damage is estimated using the methods documented in draft Regulatory Guide 1.99, Revision 2, "Radiation Damage to Reactor Vessel Materials." Although this regulatory guide is a draft, its methodology is considered by the staff to be the most up-to-date method for predicting neutron irradiation damage. This method of predicting neutron irradiation damage to materials is dependent upon the amount of neutron fluence received by the material and the amount of copper and nickel in the material. The beltline material with the greatest adjusted RT_{NDT} at the end of license (EOL) is the controlling beltline material. For the Millstone 2 reactor vessel the controlling material is base metal. At the EOL, plate number C505-2 has the greatest adjusted RT_{NDT}.

The initial RT_{NDT} for plate number C505-2 is 25°F. From Regulatory Guide 1.99, Revision 2, margin for the adjusted RT_{NDT} is 2 times the square root of the sum of σ_I^2 and σ_Δ^2 where σI is the standard deviation for the initial reference temperature and $\sigma\Delta$ is the standard deviation for the change in temperature. Because 25°F is a measured value, σ_I is zero. The value of σ_Δ is 17°F. The resulting margin is 34°F. The copper content and nickel content by weight for plate number C505-2 are 0.13% and 0.64%, respectively. Based on this copper and nickel content for the plate, the chemistry factor is 91.

The reactor vessel inside radius is 86 inches and the outside radius is 94.625 inches which yields a reactor vessel wall thickness of 8.625 inches. Flaws are postulated on the inside surface and the outside surface of the vessel. Distance is measured from the inner radius of the vessel outward. The flaws on the inside surface and outside surface are referred to by location as 1/4 thickness (1/4t) and 3/4 thickness (3/4t), respectively.

The inside surface fluence at 32 effective full power years (EFPY) is predicted to be 5.0×10^{19} neutrons per square centimeter (n/cm^2) at an energy greater than 1MeV (E>1MeV). This is based on an initial fluence rate of 9.6×10^{17} n/cm^2 per EFPY (E>1MeV) at 2700 MW_{th} core thermal power. The removal of the thermal shield after 5 EFPY caused this initial fluence rate to increase by 74%. Stated another way, from 0 to 5 EFPY the rate of change of surface with the thermal shield in place was 0.096×10^{19} n/cm^2 (E>1MeV) per EFPY. After removal of the thermal shield the rate of change of surface fluence increased to 0.167×10^{19} n/cm^2 (E>1MeV) per EFPY. At 5 EFPY the accumulated surface fluence was 0.48×10^{19} n/cm^2 (E>1MeV). Thus, the inside surface fluence of 1.6×10^{19} n/cm^2 (E>1MeV) used with the controlling material corresponds to 11.7 EFPY. The licensee has rounded this value to 12 EFPY.

Analysis of capsule W-97 yielded an end of license (32 EFPY) fluence at the vessel-clad interface of 2.96×10^{19} n/cm^2 (E>1MeV). The lead factor for the capsule to vessel inner diameter surface was determined as 1.36. Since the end of license neutron fluence from the capsule W-97 dosimetry is less than the 5×10^{19} n/cm^2 (E>1MeV) at end of license proposed by the licensee to calculate the reactor vessel pressure-temperature limits, the value of 5×10^{19} n/cm^2 (E>1MeV) will estimate conservatively the EOL neutron fluence for reactor pressure vessel plate number C505-2.

Table 2 compares values of RT_{NDT} measured from surveillance material in Millstone 2 to the values predicted using Regulatory Guide 1.99, Revision 2. The values predicted by the regulatory guide exceed the values measured from the surveillance material. The materials contained in the surveillance capsule are representative of the materials used to fabricate the beltline of the Millstone 2 reactor vessel. Since the predicted values for ΔRT_{NDT} using Regulatory Guide 1.99, Revision 2, exceed the measured values from Surveillance materials that represent the Millstone 2 reactor vessel beltline, the method in Regulatory Guide 1.99, Revision 2, should conservatively predict the ΔRT_{NDT} for the Millstone 2 reactor vessel beltline controlling material.

The attentuation ratio for the surface fluence was estimated using the "dpa equivalent" exponential decay model with an exponent of -0.24x, where x is depth in inches, i.e., $f = f_o e^{-0.24x}$. At 11.7 EFPY for the controlling plate this yields a fluence of 9.5×10^{18} n/cm² (E>1MeV) at 1/4t and a fluence of 3.4×10^{18} n/cm² (E>1MeV) at 3/4t. The staff agrees with the licensee with regard to these values.

The licensee used the embrittlement prediction method of SECY-82-465 with a two standard deviation allowance for shift error (48°F). No error allowance was made for the initial value since it was measured. The limiting adjusted RT_{NDT} at 11.7 EFPY was determined to be 152°F at 1/4t and 133°F at 3/4t. The prediction method in draft Regulatory Guide 1.99 Revision 2, results in adjusted reference temperatures of 149°F at 1/4t and 123°F at 3/4t. Since the SECY-82-465 method was consistent with the original plant design basis and yielded more conservative values in this instance, the licensee uses it as the basis for the change. This approach is acceptable to the staff provided every adjusted reference temperature computed by the prediction method of SECY-82-465 is compared to the value of the adjusted reference temperature computed by the prediction method of Regulatory Guide 1.99 Revision 2; and the more conservative value (greater numerical value) at each location is used. Instead of SECY-82-465 the licensee should use Regulatory Guide 1.99, Revision 2 in its entirity as the basis for computing reactor vessel pressure-temperature limits.

The criterion ($K_{IR} \geq 1.5K_{Im} + K_{It}$) from Section III, ASME Code, Article G-2000, Vessels, was used to determine the pressure-temperature limit during an inservice hydrostatic test. The flaw on the inside surface (1/4t) is controlling. The pressure-temperature limits computed by the staff and the licensee are essentially the same (differ by less than a nominal 2%) for the inservice hydrostatic test. The curve for pressure-temperature limit at 11.7 EFPY for inservice hydrostatic testing is acceptable to the staff.

The criterion ($K_{IR} \geq 2K_{Im} + K_{It}$) from Section III, ASME Code, Article G-2000, Vessels was used to determine the pressure-temperature limit during various heatup and cooldown rates of the reactor vessel. For heatup, depending on the heatup rate and pressure, either the flaw on the inside surface (1/4t) or the outside surface (3/4t) is controlling. For cooldown, the flaw on the inside surface (1/4t) is controlling.

TABLE 2
Comparison of Measured ΔRT_{NDT} and Predicted ΔRT_{NDT}

Capsule/Surveillance Material	Neutron Fluence (x 10^{19} n/cm²)	ΔRT_{NDT} Measured From Surveillance Material (°F)	ΔRT_{NDT} Predicted By Regulatory Guide 1.99 Revision 2 (°F)
Capsule W-97			
Base Metal (Transverse)	0.378	96	107
Base Metal (Longitudinal)	0.378	70	107
Weld (Inside)	0.378	76	160
Weld (Outside)	0.378	37	160

The heatup curve, proposed TS Figure 3.4-2, is a composite curve for reactor vessel pressure temperature limits. The heatup curve was calculated by starting at 70°F, then heating at a rate of 20°F per hour to 110°F, then 30°F per hour from 110°F to 140°F, and finally 50°F per hour to 550°F. This heatup sequence appears in proposed TS 3.4.9.1(a). The pressure-temperature limit curve computed by the staff and the licensee are essentially the same (differing by less than a nominal 2%) for heatup. The curve for pressure-temperature limit at 11.7 EFPY for heatup is, therefore, acceptable to the staff.

The cooldown curve, also shown in proposed Figure TS 3.4-2, is a composite curve for the reactor vessel pressure-temperature limits. The cooldown curve was calculated by starting at 550°F, then cooling at a rate of 80°F per hour to 300°F, 30°F per hour from 300°F to 200°F, and finally 20°F per hour from 200°F to 70°F. This cooldown sequence appears in proposed TS 3.4.9.1 (b). The pressure-temperature limit curve computed by the licensee bounds the curve computed by the staff for cooldown. The curve for pressure-temperature limit at 11.7 EFPY for cooldown is, therefore, acceptable to the staff.

Pressure-temperature limits for heatup and cooldown during core operations are obtained by adding 40°F to the temperature values in proposed TS Figure 3.4-2. Thus, a minimum critically temperature of 515°F is acceptable to the staff. Values of 156°F for the lowest service temperature and 70°F for the minimum boltup temperature are determined by considerations in the reactor coolant system other than the reactor pressure vessel. These temperature values are therefore acceptable to the staff. Also maximum service pressure of 520 psia is acceptable to the staff.

9.0 TECHNICAL SPECIFICATIONS

The TS addressed herein were submitted with the applications for license amendments dated October 20, October 24, and October 27, 1986.

9.1 Pressure/Temperature Limits

As indicated in Section 8.0, herein, the proposed heatup and cooldown curves in proposed TS Figure 3.4-2, and the proposed heatup and cooldown rates in proposed TS 3.4.9.1(a) and 3.4.9.1(b), are acceptable.

The licensee has also proposed a change to the action statement of TS 3.4.9.2 which provides remedial action to be taken when the pressure/temperature limits for the pressure vessel are exceeded. The existing TS had required an evaluation of "... fracture toughness properties" to determine acceptability for continued operation. The licensee has proposed that overall "structural integrity" should be evaluated. Since an evaluation of structural integrity would include an evaluation of fracture toughness and other factors, the licensee's proposed wording would require a more wide-ranging evaluation of the reactor pressure vessel in the event that pressure/temperature limits are violated. Accordingly, the licensee's proposed change is acceptable.

9.2 Reduced Reactor Coolant Flow Rate

This proposed change affects TS Figure 2.2-1 and TS Tables 2.2-1 and 3.2-1. It involves lowering the required RCS flow rate from 350,000 gpm to 340,000 gpm. This new lower flow rate is etablished to correspond to plugging 17.6 percent of the tubes in each steam generator. The staff finds this change acceptable because it is offset by a reduction in the radial peaking factor (F_r) and its effect has been included in the transient and accident analyses for Cycle 8 operation.

The October 24, 1986 application for licensee amendment also proposes a change to the RCS flow surveillance requirements of TS 4.2.5.2. At the present time, RCS flow must be determined every 12 hours. The licensee proposes that the surveillance interval be increased to require RCS flow measurement every 31 days.

The measurement of RCS flow, together with other measurements, is important to assure that the core thermal margins are sufficient. In this regard, the departure from nucleate boiling (DNB) ratio is an important indicator of the reactor core thermal margin. Significant changes in DNB ratio due to RCS flow changes could result from two sources. The first type of flow-related DNB change could result from the loss of one or more reactor coolant pumps. This change would be dramatic and would result in the automatic shutdown of the reactor by the reactor protection system (RPS). The second type of flow-related DNB change could result from the deposition of corrosion products (crud) in the core. Experience has shown that crud buildup, should it occur, is a long term problem that is manifested over several months and thus would be observed over several of the proposed surveillance intervals.

Based upon the above, we conclude that the proposed change to TS 4.2.5.2 is acceptable.

9.3 Radial Peaking Factor (F_r)

This proposed change affects TS Figure 3.2-3b and TS 3.2.3. The licensee's Cycle 4 reload safety analysis has shown that the DNB analysis penalty which results from a reduction in primary RCS flow rate can be offset by a reduction in allowable F_r in a 2:1 ratio of RCS flow rate to F_r. The proposed reduction in F_r more than offsets the RCS flow rate reduction. In addition, the Cycle 8 reload safety analysis confirms the acceptability of the transients and accidents with respect to the licensing basis criteria. Accordingly, these proposed changes to the TS are acceptable.

10.0 SUMMARY

The staff has reviewed the fuel system design, nuclear design, thermal-hydraulic design and the transient and accident analyses presented in the Millstone Unit 2 Cycle 8 reload report. The staff conludes that the proposed reload and associated modified Techncial Specifications are acceptable for Cycle 8 operation.

The modified Technical Specifications for Cycle 8 are based on an RCS flow rate of 340,000 gpm. Since the proposed extended cycle operation is based on a minimum RCS flow rate of 350,000 gpm, such extended cycle operation should not be undertaken prior to NRC review and approval.

With regard to the LOCA calculations, the large- and small-break analyses performed in support to Millstone Unit 2 Cycle 8 satisfies the requirements of 10 CFR 50.46 and are acceptable. Additionally, Millstone Unit 2 has satisfied the requirements of NUREG-0737 Item II.K.3.31. Reference 18 documents the satisfaction of NUREG-0737 Item II.K.3.30.

The staff has used the method of calculating pressure-temperature limits described in USNRC Standard Review Plan Section 5.3.2, NUREG-0800, Revision 1, July 1981, to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage to the beltline material was estimated using the method documented in draft Regulatory Guide 1.99, Revision 2. The amount of copper and nickel in the controlling material, plate number C505-2, is 0.13% and 0.64% by weight, respectively. For the vessel inner surface, an end of license neutron fluence of 5×10^{19} n/cm^2 (E>1MeV) and a fluence of 1.6×10^{19} n/cm^2 (E>1MeV) at 11.7 EFPY is acceptable. The proposed pressure-temperature limits for inservice hydrostatic testing, heatup and cooldown meet the safety margins of Appendix G, 10 CFR Part 50 and may be incorporated into the technical specifications for the plant. Figure 3.4-2 is valid for 11.7 EFPY.

11.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

12.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 8, 1986

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13.0 REFERENCES

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4. Letter from R. A. Clark (NRC) to W. G. Counsil (NNECO), dated June 22, 1981 (documents the staff's physics review of the BSR).

5. Letter from R. A. Clark (NRC) to W. G. Counsil (NNECO), January 12, 1982 (documents the staff's transient analysis review of the BSR).

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9. NRC memorandum from L. S. Rubenstein to T. M. Novak, dated October 2, 1980 on "SE Input for Millstone 2 Cycle 4 Reload."

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15. "Westinghouse ECCS Evaluation Model - 1981 Version." WCAP-9220-P-A Revision 1, February 1982.

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19. Letter, E. P. Rahe, Jr. (Westinghouse) to J. Lyons (NRC), "Response to the NRC Request for Additional Information on Topical Report WCAP-10054, Addendum 1," October 23, 1986.