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 40°F in any one hour period with T_{avg} at or below 200°F, 50°F in any one hour period with T_{avg} at or below 300°F and above 200°F, and 100°F in any one hour period with T_{avg} above 300°F.

b. A maximum cooldown of 100°F in any one hour period with T_{avg} above 300°F and a maximum cooldown of 20°F in any one hour period with T_{avg} below 300°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 fracture toughness properties 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.

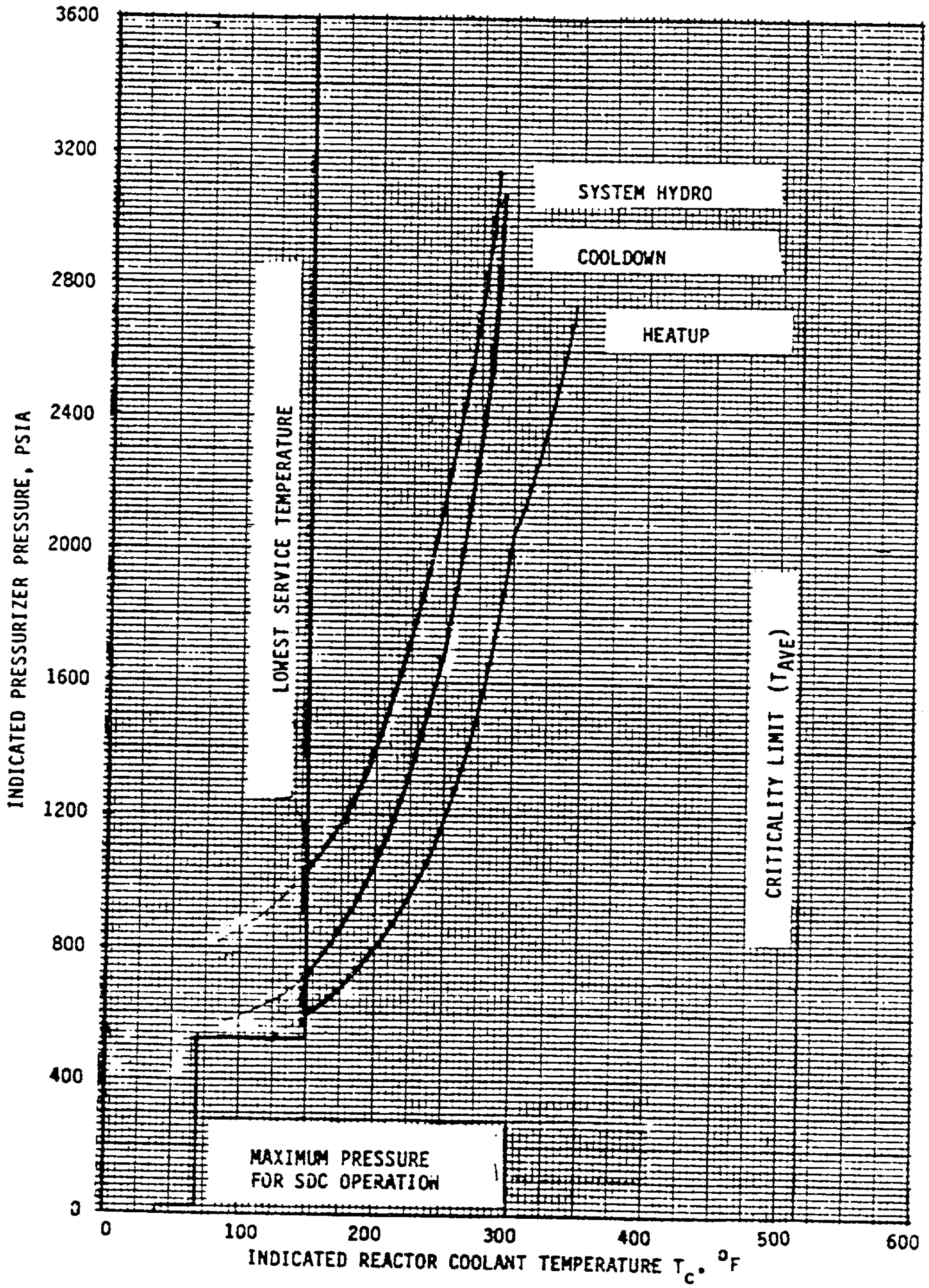

Figure 3.4-2
Reactor Coolant System Pressure Temperature Limitations for 7 Full Power Years

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 100°F per hour. 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 (E>1 Mev) 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.

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

DCS MS-016

APR 10 1984

Docket No. 50-336

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Mr. W. G. Counsil, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit 2, in response to your application dated January 4, 1984.

This amendment modifies the technical specifications to revise the pressure-temperature limits and the maximum rate of heatup for the reactor coolant system. In addition, the reactor vessel material irradiation specimen withdrawal schedule is revised and the technical specification bases have been updated to conform with recent changes in 10 CFR 50 Appendix G.

We request that, as a confirmatory item, you provide separate baseline Charpy V-notch curves for each surveillance weld material. Since weld metal is not predicted to be limiting for operation in the effective period of the proposed curve, this request does not affect our current findings in the enclosed Safety Evaluation. We request that you supply this information within one year of the date of this letter.

The information requested in this letter affects fewer than 10 respondents; therefore OMB clearance is not required under P.L. 96-511.

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

Sincerely,

Original signed by

Kenneth L. Heitner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 94 to DPR-65
2. Safety Evaluation

cc w/enclosures:
See next page

~~ORB#3:DL~~	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
~~PMKreutzer~~	KHeitner/pn	JRMiller	Bachmann	GCLainas
~~3/ /84~~	3/21/84	3/ /84	3/29/84	4/4/84
	KH		CMB	
			Amendment + FR Notice	

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PDR ADOCK 05000336
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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. 94
License No. DPR-65

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee) dated January 4, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 94 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 10, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Remove and replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove	Insert
3/4 4-17	3/4 4-17
3/4 4-19a	3/4 4-19
3/4 4-19	-
3/4 4-19c	-
3/4 4-20	3/4 4-20
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	-
B 3/4 4-8	-
B 3/4 4-9	-
B 3/4 4-10	-
B 3/4 4-11	B 3/4 4-7
B 3/4 4-12	-

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

Included in this evaluation is consideration of flange protection in accordance with 10 CFR 50, Appendix G. The requirement makes the minimum temperature RT_{NDT} plus 90°F for hydrostatic test and RT_{NDT} plus 120°F for normal operation when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure.

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

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

The OPERABILITY of two PORVs or an RCS vent opening of greater than 1.3 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are ≤ 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator ≤ 43°F (31°F when measured by a surface contact instrument) above the coolant temperature in the reactor vessel or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ISSUANCE OF AMENDMENT NO. 94 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-336

INTRODUCTION

In a letter from W. G. Counsil to J. R. Miller dated January 4, 1984, the Northeast Nuclear Energy Company (NNECO) requested a revision to the Millstone Nuclear Power Station, Unit No. 2 (Millstone-2) Technical Specifications (TS). The licensee's proposed TS changes were to Section 3.4.4.9, Pressure/Temperature Limits and to Table 4.4-3 Reactor Vessel Material Irradiation Surveillance Schedule.

EVALUATION

The Millstone-2 pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50, are dependent upon the initial reference temperture, RT_{NDT}, for the limiting materials in the beltline and closure flange regions of the reactor vessel, the water/metal temperature profile through the reactor vessel pressure boundary wall during heatup and cooldown, and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline materials.

The Millstone-2 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each reactor vessel material. The initial RT_{NDT} is calculated based on test results performed on transversely oriented Charpy V-notch and drop weight specimens. The licensee has drop weight tested some of the closure flange and beltline region materials, but has Charpy V-notch tested these materials with longitudinally oriented specimens rather than transversely oriented specimens. This test data is reported in Table 4.6-1 of the Millstone-2 FSAR and a letter from D. C. Switzer to G. Lear dated December 9, 1977. The staff's review of this data indicates that the limiting closure flange region material is Vessel Flange Code No. C-500 and the limiting beltline materials are Plate Code No. C-505-2, and Weld Seam No. 9-203. Weld Seam No. 9-203 contains material fabricated using Linde 0091 flux batches 3998 and 3999 and weld wire heat numbers 90136 and 10137. However, for seven (7) effective full-power years (EFPY), which is the effective period of the proposed pressure-temperature limits, the limiting beltline material is Plate Code No. C-505-2. Later in the plant's life, the weld will become limiting, because its rate of increase in RT_{NDT} resulting from neutron irradiation damage is predicted to be greater than that of the plate material.

When drop weight tests have been performed, the staff uses the criteria in Section 1.1(3)b of Branch Technical Position-MTEB 5-2, "Fracture Toughness Requirements" to determine the initial RT_{NDT} of a material. Based on this criteria, the initial RT_{NDT} for Vessel Flange Code No. C-500 and Beltline Plate Code No. C-505-2 would be +10°F and +25°F, respectively. These initial RT_{NDT} values were reported by the licensee in Table 4.6-1 of the Millstone-2 FSAR and in Table 3.2-1 of the licensee's report, "Thermal Shield Damage Recovery Program Final Report," December 1983. The criteria in Branch Technical Position-MTEB 5-2 is currently under review by the staff. If it is judged nonconservative, we will request the licensee to reevaluate their pressure-temperature limits.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the methodology documented in Commission Report SECY-82-465, "Pressurized Thermal Shock." The staff's current method of determining the increase in RT_{NDT} resulting from neutron irradiation damage is documented in Regulatory Guide 1.99, Rev. 2, December 22, 1983. The amount of neutron irradiation damage, which is predicted using Regulatory Guide 1.99, Rev. 2 methodology, depends upon the amount of neutron fluence, and the amount of copper and nickel in the material.

In Table I, we have compared the mean plus two standard deviation increase in RT_{NDT} predicted by the Regulatory Guide 1.99, Rev. 2 method with that measured from the Millstone-2 reactor vessel beltline surveillance program. The test results from the Millstone-2 reactor vessel beltline surveillance program is reported in Licensee Report TR-N-MCM-008, "Evaluation of Irradiated Capsule W-97," dated April 1982. The prediction method in Regulatory Guide 1.99, Rev. 2 provides conservative estimates for the effect of neutron irradiation of the Millstone-2 reactor vessel beltline materials, because the increase in RT_{NDT} predicted by the Regulatory Guide 1.99, Rev. 2 method exceeds that from the surveillance material.

We have estimated the neutron fluence to be received by the reactor vessel beltline materials by extrapolating to 7 EFPY the neutron fluence estimates in Table 3.1-2 of Licensee Report, "Thermal Shield Damage Recovery Program Final Report." The neutron fluence estimates identified in this report have been reviewed by the staff.

The amount of time that pressure-temperature limits are effective depends upon the amount of material embrittlement at the 1/4 T and 3/4 T vessel locations. The measurement used by the staff to indicate the amount of material embrittlement is the adjusted reference temperature (ART). The ART is the sum of the initial RT_{NDT}, increase in RT_{NDT} caused by neutron irradiation damage, and the amount of margin required to obtain a conservative upper bound for neutron embrittlement for the limiting material. The licensee calculated the ART for the limiting Millstone-2 beltline material using the methodology in Commission Report SECY-82-465. The licensee indicated that at 7 EFPY, the ART for the limiting Millstone-2 beltline material at the 1/4 T and 3/4 T locations will be 139°F and 123°F, respectively. The staff has used the method documented in Regulatory Guide 1.99, Rev. 2 to estimate

Table I

Comparison of Regulatory Guide 1.99, Rev. 2
Predicted Increase in RT_{NDT} and the Observed Increase in RT_{NDT}
Reported for the MNPS-2 Surveillance Material

Material	Increase in Reference Temperature	
	Mean Plus Two Standard Deviations Using the Regulatory Guide 1.99, Rev. 2 Method (°F)	Observed from MNPS-2 Surveillance Material Tests (°F)
Plate C-506-1	107	96
Weld Flux Lot 3998	155	76
Weld Flux Lot 3999	127	48

the ART for the limiting Millstone-2 beltline material. This method predicts the ART after 7 EFPY of operation will be 133°F and 109°F for the limiting Millstone-2 beltline material at the 1/4 T and 3/4 T locations, respectively. We believe these estimates are conservative, because the increase in RT_{NDT} observed by the surveillance material is less than that predicted by the Regulatory Guide 1.99, Rev. 2 method (Table I).

To calculate the water/metal temperature profile during a heat up or a cool-down, the licensee used a finite element computer code, which was modeled based on one dimensional heat transfer and infinite conductivity at the fluid/clad interface. We have evaluated the metal/water temperature profile which results from the model. We conclude that the model produces a conservative metal/temperature profile which may be used in calculating heat-up and cool-down curves.

CONCLUSIONS

Based on our analysis, which indicates the licensee has used conservative values for (a) the initial RT_{NDT} of the limiting closure flange and beltline materials, (b) the ART of the limiting beltline material, and (c) the water/metal temperatures profile, we conclude that the licensee's proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50, for a period of time corresponding to 7 EFPY and may be incorporated into the Millstone-2 TS.

The licensee's proposed surveillance capsule withdrawal schedule must meet the requirements of Appendix H, 10 CFR 50. Appendix H, 10 CFR 50 requires that the surveillance capsule withdrawal schedule meet the intent of ASTM E-185-82 and provide material test data throughout the life of the plant. The staff has reviewed the licensee's proposed surveillance capsule withdrawal schedule and concludes that it meets the requirements of Appendix H, 10 CFR 50.

ADDITIONAL INFORMATION

During the staff's review of the Licensee Report, "Evaluation of Irradiation Capsule W-97," it was noticed that the baseline unirradiated Charpy V-notch weld test data was generated from two (2) sets of materials. One weld material was fabricated using Linde 0091 flux batch 3998 and wire heat number 90136 and the other weld material was fabricated using Linde 0091 flux batch 3999 and wire heat number 10137. Since these are materials which have been prepared using different heats of wire and batches of flux, their initial properties will be different. The licensee has not considered these differences in evaluating the effect of neutron irradiation on the surveillance weld material. We request that the licensee provide separate baseline Charpy V-notch curves for each of these materials so that the staff may evaluate the effect of neutron irradiation on each weld metal.

Since at 7 EFPY the weld material is not limiting, the effect of neutron irradiation on the surveillance capsule weld material will not affect the staff's conclusion concerning the licensee's proposed pressure temperature limits. However, later in the plant's life, the weld material will become limiting, because its predicted rate of neutron irradiation damage is greater than that of the plate material. At that time, the effect of neutron irradiation on the capsule weld material will become important since the results of these tests will be used to determine the margins required for safe operation of the Millstone-2 reactor vessel.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: April 10, 1984

Principal Contributor:
B. J. Elliot, MTEB