January 27, 1994

Docket No. 50-336

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Mr. John F. Opeka
Executive Vice President, Nuclear
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M86801)

The Commission has issued the enclosed Amendment No.170 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your application dated June 11, 1993, supplemented by letter dated November 15, 1993.

The amendment revises the pressure/temperature (P/T) limits for the reactor vessel. Specifically, Figure 3.4-2, "Millstone Unit 2 Reactor Coolant System Presure-Temperature Limitations for 12 Full Power Years," on page 3/4 4-19, is revised to reflect the change in the curves and the title change to "Millstone Unit 2 Reactor Coolant System Pressure-Temperature Limitations for 20 EFPY."

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Guy S. Vissing, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 170to DPR-65
2. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 27, 1994

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Connecticut Yankee Atomic Power Company
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See next page

Mr. John F. Opeka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit 2

cc:

Gerald Garfield, Esquire
Day, Berry and Howard
Counselors at Law
City Place
Hartford, Connecticut 06103-3499

J. M. Solymossy, Director
Nuclear Quality & Assessment Services
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Kevin T. A. McCarthy, Director
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, Connecticut 06106-5127

Allan Johanson, Assistant Director
Office of Policy and Development
Policy Development & Planning Division
80 Washington Street
Hartford, Connecticut 06106

S. E. Scace, Vice President
Nuclear Operations Services
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

G. H. Bouchard, Nuclear Unit Director
Millstone Unit No. 2
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

Nicholas S. Reynolds
Winston & Strawn
11400 L Street, NW
Washington, DC 20005-3502

R. M. Kacich, Director
Nuclear Planning, Licensing & Budgeting
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

J. P. Stetz, Vice President
Haddam Neck Plant
Connecticut Yankee Atomic Power Company
362 Injun Hollow Road
East Hampton, Connecticut 06424-3099

Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

First Selectmen
Town of Waterford
Hall of Records
200 Boston Post Road
Waterford, Connecticut 06385

P. D. Swetland, Resident Inspector
Millstone Nuclear Power Station
c/o U.S. Nuclear Regulatory Commission
Post Office Box 513
Niantic, Connecticut 06357

Charles Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering
Nuclear Power
12300 Twinbrook Pkwy, Suite 330
Rockville, Maryland 20852

Donald B. Miller, Jr.
Senior Vice President
Millstone Station
Northeast Nuclear Energy Company
Post Office Box 128
Waterford, Connecticut 06385

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 170
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 June 11, 1993, supplemented by letter dated November 15, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.170 , 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 to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Boyle
for John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following page of the Appendix A Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
3/4 4-19	3/4 4-19
B 3/4 4-6	B 3/4 4-6

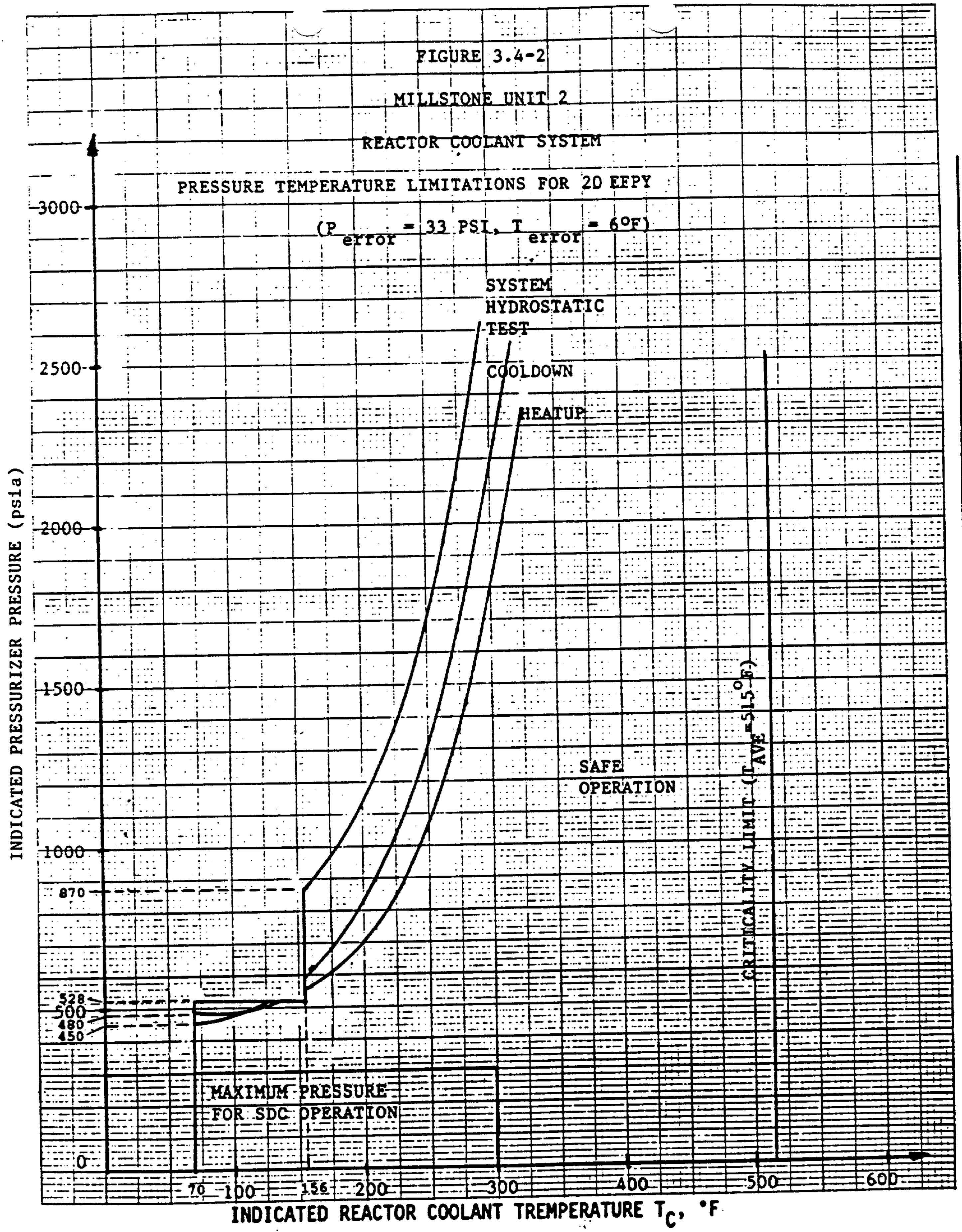
FIGURE 3.4-2
MILLSTONE UNIT 2
REACTOR COOLANT SYSTEM
PRESSURE TEMPERATURE LIMITATIONS FOR 20 EFPY
(P_{error} = 33 PSI, T_{error} = 6°F)
SYSTEM HYDROSTATIC TEST
COOLDOWN
HEATUP
SAFE OPERATION
CRITICALITY LIMIT (T_{AVE} = 515°F)
MAXIMUM PRESSURE FOR SDC OPERATION
INDICATED PRESSURIZER PRESSURE (psia)
3000
2500
2000
1500
1000
870
528
500
480
450
0
70
100
156
200
300
400
500
600
INDICATED REACTOR COOLANT TREMPERATURE T_C, °F

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 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 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 10CFR50 Appendix H, 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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated June 11, 1993, supplemented by letter dated November 15, 1993, Northeast Nuclear Energy Company (NNECO/the licensee) submitted proposed Technical Specification (TS) changes to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2 (Millstone 2). The proposed amendment revises the pressure/temperature (P/T) limits for the reactor vessel. Specifically, Figure 3.4-2, "Millstone Unit 2 Reactor Coolant System Presure-Temperature Limitations for 12 Full Power Years," on page 3/4 4-19, is revised to reflect the change in the curves and the title change to "Millstone Unit 2 Reactor Coolant System Pressure-Temperature Limitations for 20 EFPY" [effective full power years]. The P-T limit curves are revised to reflect the increase in the nil-ductility reference temperature of reactor vessel beltline materials.

The November 15, 1993, submittal provided information that did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

The licensee's arguments were based in part on the results of surveillance capsule W-104, the report was submitted to the staff on November 27, 1991 (Ref. 2).

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the American Society for Testing and Materials (ASTM) Standards and the American Society of Mechanical Engineers (ASME) Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions for operation in the TS for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to monitor embrittlement of reactor vessel materials. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and be removed from the reactor vessel periodically for testing. The capsules should contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

3.0 EVALUATION FOR FAST NEUTRON FLUENCE FOR 20 EFPY

Millstone Unit 2 has shifted to a low leakage core loading pattern in Cycle 10. The new loading pattern contains one new assembly in the perimeter, which shifted the azimuthal peak fluence location from the 27^0 to 1^0 on the quadrant. The azimuthal shift resulted in a lower estimated peak value at the new peak location. Surveillance capsule W-104 verified the calculated estimate.

The analytical method was based on the two dimensional DOT 4.3 finite differencing code. The (r,θ) solution used a 47 group cross section set based on ENDF/B-IV. The cross section scattering approximation was P_3 and the geometrical quadrature was S_8. The source and the source spectrum were based on a combination of U-235 and Pu-239 spectra to account for the use of depleted assemblies in the periphery. A pin-by-pin power distribution was used for the outer assemblies. The pin power distribution was derived from the assembly average power at Millstone and the pin power shape distribution from St. Lucie Unit 2, which is a sister plant to Millstone and has had identical loading patterns to Cycle 10. The analytical method as described above is acceptable.

The licensee stated that up to Cycle 15 the Cycle 10 loading will be used. However, beyond Cycle 15, the licensee stated that it will either continue the Cycle 10 loading pattern or change to an all burned assembly scheme in the periphery, which will further reduce the fluence to the vessel and it will be conservative.

From the preceding discussion we conclude that: (1) the low leakage pattern introduced in Cycle 10 will reduce the peak vessel irradiation, (2) the analytical method used for the estimation of the fluence to 20 effective full power years of operation is acceptable because it complies with staff requirements and because it is conservative, and (3) the Cycle 10 loading pattern will be used to 15 effective full power years of operation and possibly to 20 effective full power years. If, however, it is changed after 15 effective full power years it will become more conservative. From the above we conclude that the Millstone 2 fluence estimate to the pressure vessel for 20 effective full power years of operation is acceptable.

3.1 Evaluation for P/T limits

The staff calculated the ART for each beltline material in the Millstone 2 reactor vessel in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest (limiting) ART at 20 EFPY was the intermediate shell plate C-505-2 (heat number C5843-2) having 0.13% copper (Cu), 0.64% nickel (Ni), and an initial RT_{ndt} of 25°F.

The staff calculated the ART to be 143°F at the 1/4T location (T = reactor vessel beltline thickness) and 113°F for the 3/4T location. The staff used a neutron fluence of 9.32E18 n/cm^2 at 1/4T and 2.79E18 n/cm^2 at 3/4T. The staff used the method in Regulatory Position C.2, Surveillance Data Available, in RG 1.99, because the surveillance data are credible.

For the same limiting material, the licensee calculated an ART of 145°F and 118°F at the 1/4T and 3/4 locations, respectively, using Regulatory Position C.1, Surveillance Data Not Available, in RG 1.99.

RG 1.99 allows use of either methods to calculate the ARTs. RG 1.99 states that if Position C.2 gives a higher value of ARTs than that given by using Position C.1, the ARTs from Position C.2 should be used. If Position C.2 gives lower ARTs, either may be used. The licensee's ARTs are more conservative than the staff's ARTs and, therefore, acceptable to use.

Substituting the ARTs of 145°F and 118°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Paragraph IV.A.1 of Appendix G requires that reactor vessel beltline materials maintain a Charpy upper shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lbs unless it can be demonstrated that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code. The conformance of upper shelf energy to paragraph IV.A.1 will be determined pending the staff evaluation of the licensee's response to Generic Letter 92-01.

The licensee has removed surveillance capsules W-97 and W-104 from Millstone 2 and has performed required tests. The test results from capsules W-97 and W-104 were published in reports by Combustion Engineering and Babcock & Wilcox, respectively (Ref. 1 and 2). The conformance of the surveillance program to Appendix H to 10 CFR Part 50 also will be determined under the staff evaluation of the licensee's response to Generic Letter 92-01.

The staff has performed an independent analysis of the P/T limits to verify the licensee's proposed limits. The staff concludes that the proposed P/T limits for heatup, cooldown, hydrostatic tests, and criticality are valid through 20 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50 and Generic Letter 88-11. Hence, the proposed P/T limits may be incorporated in the Millstone 2 Technical Specifications.

The conformance of the upper shelf energy and reactor vessel material surveillance program to Appendices G and H will be determined pending the staff resolution of Generic Letter 92-01 in 1994.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 39054). 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.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. S. T. Byrne, "Post-Irradiation Evaluation of Reactor Vessel Surveillance Capsule W-97," TR-N-MCM-008, Combustion-Engineering, April 1982

2. Letter from J. F. Opeka of Northeast Utilities to USNRC, dated November 27, 1991 "Analysis of Capsule W-104 Northeast Nuclear Energy Company Millstone Nuclear Power Station, Unit No. 2," B&W Nuclear Service Company, BAW-2142, November 1991

Principal Contributors: J. Tsao
L. Lois

Date: January 27, 1994