March 6, 1990

Docket No. 50-423

Mr. Edward J. Mroczka
Senior Vice President
Nuclear Engineering and Operations
Connecticut Yankee Atomic Power Company
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

SUBJECT: MILLSTONE UNIT 3 - ISSUANCE OF AMENDMENT (TAC NO. 75233)

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. NPF-49 for Millstone Nuclear Power Station, Unit No. 3, in response to your application dated October 20, 1989.

The amendment modifies Technical Specification (TS) Table 4.4-5, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule" to provide a revised in-vessel material capsule withdrawal program and revised capsule lead factors.

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

Sincerely,

/s/

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.48 to NPF-49
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDI-4:	PM:PDI-4	:PD:PDI-4	:BC:EMTB	:OGC	:	:
NAME	:SNorris:	DJaffe:lm	: JStolz	CYCHENG	:[redacted]	:	:
DATE	:2/15/90:	2/15/90	:2/16/90	:2/15/90	:2/22/90	:	:

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OGC concurrence based on wont issue until on or after 2-24-90

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

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

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

DOCKET NO. 50-423

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. NPF-49

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 October 20, 1989 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. NPF-49 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 48, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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: March 6, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILTIY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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	Insert
xiv	xiv
3/4 4-36	3/4 4-36
B3/4 4-8	B3/4 4-8
B3/4 4-11	B3/4 4-11
B3/4 4-12	B3/4 4-12
B3/4 4-13	B3/4 4-13
B3/4 4-14	B3/4 4-14
B3/4 4-15	B3/4 4-15
B3/4 4-16	--

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3/4.1.2 BORATION SYSTEMS	B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES	B 3/4 1-3
3/4.2 POWER DISTRIBUTION LIMITS	B 3/4 2-1
3/4.2.1 AXIAL FLUX DIFFERENCE	B 3/4 2-1
3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2
FIGURE B 3/4.2-1a TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR FOUR LOOP OPERATION	B 3/4 2-3
FIGURE B 3/4.2-1b TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER FOR THREE LOOP OPERATION	B 3/4 2-4
3/4.2.4 QUADRANT POWER TILT RATIO	B 3/4 2-6
3/4.2.5 DNB PARAMETERS	B 3/4 2-7
3/4.3 INSTRUMENTATION	
3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	B 3/4 3-1
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3/4.3.4 TURBINE OVERSPEED PROTECTION	B 3/4 3-7
3/4.4 REACTOR COOLANT SYSTEM	
3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2 SAFETY VALVES	B 3/4 4-2
3/4.4.3 PRESSURIZER	B 3/4 4-2
3/4.4.4 RELIEF VALVES	B 3/4 4-2
3/4.4.5 STEAM GENERATORS	B 3/4 4-3
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4
3/4.4.7 CHEMISTRY	B 3/4 4-5
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-7

BASES

SECTION	PAGE
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FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E*1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE..................................	B 3/4 4-10
3/4.4.10 STRUCTURAL INTEGRITY..................................	B 3/4 4-15
3/4.4.11 REACTOR COOLANT SYSTEM VENTS............................	B 3/4 4-15
3/4.5 EMERGENCY CORE COOLING SYSTEMS	
3/4.5.1 ACCUMULATORS..	B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS...............................	B 3/4 5-1
3/4.5.4 REFUELING WATER STORAGE TANK............................	B 3/4 5-2
3/4.6 CONTAINMENT SYSTEMS	
3/4.6.1 PRIMARY CONTAINMENT.......................................	B 3/4 6-1
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS......................	B 3/4 6-2
3/4.6.3 CONTAINMENT ISOLATION VALVES..............................	B 3/4 6-3
3/4.6.4 COMBUSTIBLE GAS CONTROL...................................	B 3/4 6-3
3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM....................	B 3/4 6-3
3/4.6.6 SECONDARY CONTAINMENT.....................................	B 3/4 6-4
3/4.7 PLANT SYSTEMS	
3/4.7.1 TURBINE CYCLE...	B 3/4 7-1
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION...........	B 3/4 7-3
3/4.7.3 REACTOR PLANT COMPONENT COOLING WATER SYSTEM..............	B 3/4 7-3
3/4.7.4 SERVICE WATER SYSTEM......................................	B 3/4 7-3
3/4.7.5 ULTIMATE HEAT SINK..	B 3/4 7-3
3/4.7.6 FLOOD PROTECTION..	B 3/4 7-4
3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM..................	B 3/4 7-4
3/4.7.8 CONTROL ROOM ENVELOPE PRESSURIZATION SYSTEM...............	B 3/4 7-4
3/4.7.9 AUXILIARY BUILDING FILTER SYSTEM..........................	B 3/4 7-4
3/4.7.10 SNUBBERS..	B 3/4 7-5

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLATE METAL
COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT %.
PHOSPHORUS CONTENT : 0 010WT %.
RT_{NDT} INITIAL : 60°F
RT_{NDT} AFTER 10 EFPY : 1/4T, 122°F
3/4T, 101°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

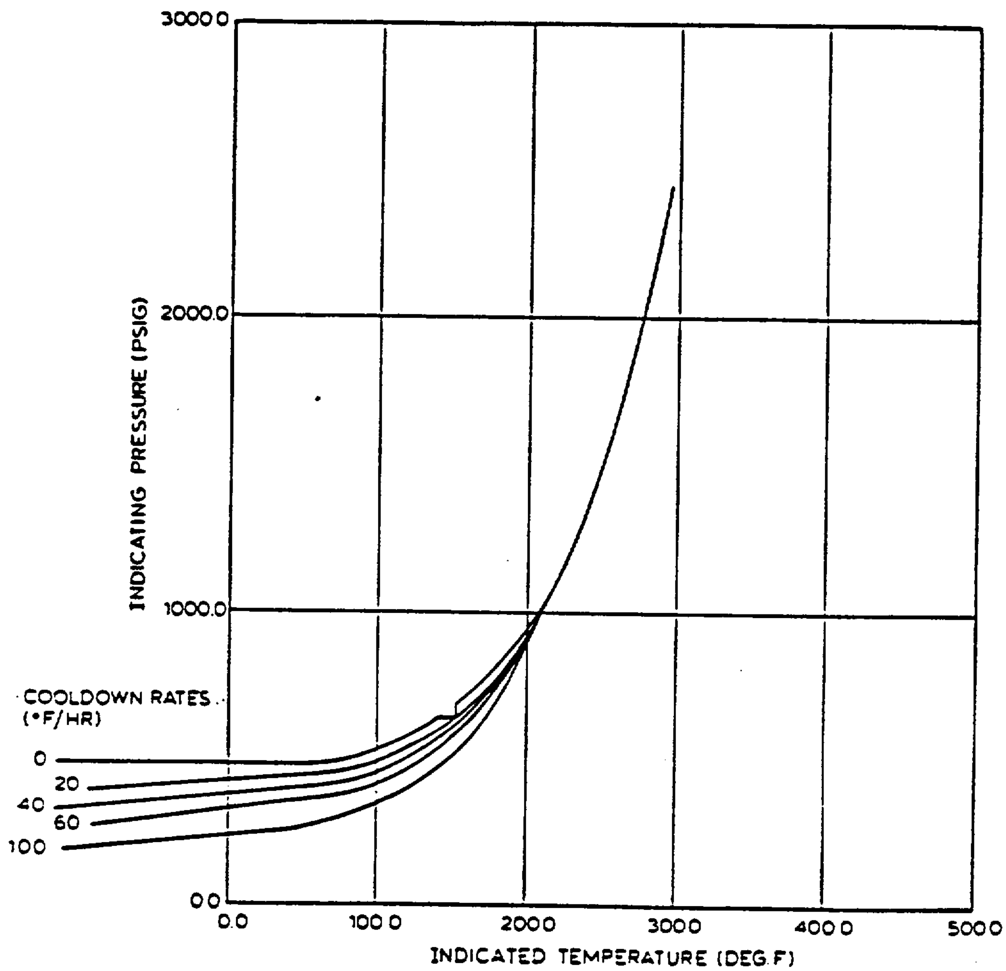

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	3.98(a)	First Refueling (1.3 EFPY actual)
Y	241°	3.74	9
V	61°	3.74	16
W	121.5°	4.01	STANDBY
X	238.5°	4.01	STANDBY
Z	301.5°	4.01	STANDBY

a) Plant specific evaluation

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor 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 reactor coolant will be detected in sufficient time to take corrective action. 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

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. Also, the 10 CFR 50, Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states the minimum metal temperature of the closure flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20% of the preservice hydrostatic test pressure (636 psia). The minimum temperature of the closure flange and vessel flange regions is 150°F since the limiting RT_{NDT} is 30°F (See Table B 3/4.4-1). The heatup curve shown in Figure 3.4-2 is not impacted by the 10 CFR 50 rule. However, the cooldown curve shown in Figure 3.4-3 is impacted by the rule.

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater

PRESSURE/TEMPERATURE LIMITS (Continued)

capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel were performed in accordance with the 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 10 effective full power years (EFPY) of service life. The 10 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant neutron irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 10 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in the following paragraphs.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

PRESSURE/TEMPERATURE LIMITS (Continued)

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RTNDT. Thus, the governing equation for the heatup-cooldown analysis is:

$$C K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location

PRESSURE/TEMPERATURE LIMITS (Continued)

for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

COLD OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 5.4 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 less than or equal to 350°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 less than or equal to 50° above the RCS cold temperatures, or (2) the start of a charging pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Protection System (COPS) is derived by analysis which models the performance of the COPS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one safety injection pump and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50° above primary temperature.

The Maximum Allowed PORV Setpoint for the COPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

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 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 80 Edition and Addenda through Winter except where specific written relief has been granted pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 48

TO FACILITY OPERATING LICENSE NO. NPF-49

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

INTRODUCTION

By application for license amendment dated October 20, 1989, Northeast Nuclear Energy Company, et al. (the licensee), requested changes to Millstone Unit 3 Technical Specifications (TS).

The proposed amendment would change Millstone Unit 3 Technical Specifications (TS) Table 4.4-5, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule" to provide a revised in-vessel material capsule withdrawal program and revised capsule lead factors.

DISCUSSION AND EVALUATION

The Reactor Coolant System pressure/temperature limit curves for plant heatup, cooldown, and inservice leak and hydrostatic pressure testing operations are provided in the Technical Specifications. These curves define limits to ensure the prevention of nonductile failures of materials incorporated within the reactor coolant system (RCS). The allowable pressure/temperature for specified heatup and cooldown rates are calculated in accordance with Appendix G of Section III of the ASME Boiler and Pressure Vessel Code and 10 CFR 50, Appendix G. The heatup and cooldown limit curves are calculated using the most limiting value of the RT_{NDT} (reference nil-ductility transition temperature) inherent in the reactor vessel material. The initial value of RT_{NDT} is determined from material tests made at the time of the vessel fabrication. During the service life of the reactor vessel, the RT_{NDT} increases above the initial value because of neutron irradiation. The amount of change (delta RT_{NDT}) depends upon the neutron fluence and material chemical composition. The transition temperature shift is determined from fluence measurements, calculations, and trend curves based on tests of irradiated specimens that predict the effects of neutron irradiation. The irradiated specimens are actual (or archive) reactor vessel material specimens and are positioned around the reactor vessel to provide surveillance of the irradiation levels to which the reactor vessel is subject. The specimens are maintained in an inert environment within a corrosion-resistant capsule to prevent deterioration of the surface of the specimens during radiation exposure.

Associated with each surveillance capsule location is a lead factor, the ratio of the instantaneous neutron flux density at the location of the specimens in a surveillance capsule to the maximum calculated neutron flux density at the inside surface of the reactor vessel wall. The lead factor is thus used to extrapolate the surveillance measurement from the specimens to the reactor vessel wall, thereby the material property changes of the reactor vessel are monitored through its life. The in-vessel capsule irradiation program is described in Section 5.3.1.6 of the Millstone Unit 3 Final Safety Analysis Report. Each surveillance capsule is also subject to a withdrawal schedule, per TS 4.4.9.1.2, as specified in TS Table 4.4-5. The specimens within the withdrawn capsule are subjected to various inspections and tests to determine the delta RT_{NDT} and any needed changes in the heat-up and cooldown limit curves. The number of capsules to be withdrawn over the life of the reactor pressure vessel is required by Appendix H to 10 CFR Part 50 to meet the requirements of ASTM E185.

The licensee has proposed a change to the number of surveillance capsules to be withdrawn (and the associated withdrawal schedule) and the lead factors as specified in TS Table 4.4-5. The proposed changes result from analysis of the first capsule which was withdrawn during the first refueling outage. At the present time, TS Table 4.4-5 describes a capsule program containing four capsules. The first capsule was withdrawn during the first refueling outage and subsequent capsules are to be withdrawn at 5, 9 and 15 effective full power years (EFPY). The requirements of ASTM E185 allow a program to contain only three capsules if the end-of-life (EOL) RT_{NDT} is less than 100°F. Based upon the evaluation of the first capsule to be removed, the licensee has projected that the EOL RT_{NDT} will be less than 100°F and has proposed a change to TS Table 4.4-5. The revised capsule program would have three capsules. The first capsule would be withdrawn during the first refueling outage (already accomplished) and subsequent capsules at 9 and 16 EFPY. Changes to the TS Bases have also been proposed.

We concur with the licensee's evaluation that the EOL RT_{NDT} projection of less than 100°F permits the irradiation capsule program to be reduced from four to three capsules. The proposed irradiation capsule program conforms to ASTM E185 and, in this regard, meets the requirements of Appendix H to 10 CFR Part 50. Accordingly, the proposed changes to TS Table 4.4-5 are acceptable.

ENVIRONMENTAL CONSIDERATION

This 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. We have 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 staff 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.

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 (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.

Dated: March 6, 1990

Principal Contributor: D. Jaffe

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AMENDMENT NO.48 TO FACILITY OPERATING LICENSE NO. NPF-49

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