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March 22, 1993

Docket No. 50-245
B14403

Re: 10CFR50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Proposed Revision to Technical Specifications
Pressure-Temperature Limit Curves

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its Operating License, No. DPR-21, by incorporating the changes identified in Attachment 1 into the Technical Specifications of Millstone Unit No. 1.

Description of Proposed Changes

The Millstone Unit No. 1 Technical Specifications contain limitations on allowable reactor coolant system pressures and temperatures. The proposed changes revise the pressure-temperature limits for the reactor vessel. Specifically, technical specification Limiting Condition for Operation 3.6B, Surveillance Requirement 4.6.B, Figures 3.6.1, 3.6.2, and 3.6.3 along with tne corresponding bases for Section 3.6 are revised. The current limitations are valid to 16 Effective Full Power Years (EFPY). Millstone Unit No. 1 could achieve 16 EFPY as soon as June 12, 1993, assuming continued full power operation. The proposed changes in the heatup and cooldown curves are based on a General Electric Company Report on Vessel Surveillance Materials Testing results which was transmitted to the NRC in NNECO letter dated December 30, 1992.[1] Section 4 of this report deals with peak reactor pressure vessel fluence evaluation, Section 5 evaluated the Charpy V-notch impact testing including impact on upper shelf energies (USE), and Section 6 evaluated tensile testing.

(1) J. F. Opeka letter to U.S. Nuclear Regulatory Commission, "Reactor Vessel Material Surveillance Capsule," dated December 30, 1992.

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Discussion

Revised limits have been calculated in order to continue operation beyond 16 EFPY. These new limits reflect the predicted radiation-induced embrittlement of the reactor vessel through 32 EFPY. Pressure-temperature limits are required by 10CFR50 Appendix G, "Fracture Toughness Requirements," to provide an adequate margin of safety during any condition of normal operation, including anticipated operational occurrences. These limits depend on the metallurgical properties of the reactor vessel materials. The vessel beltline region material properties change over the lifetime of the vessel due to the effects of neutron irradiation. The amount of neutron irradiation to which these materials are exposed determines the shift in the material's reference temperature for nil ductility transition (RT_{NDT}). The shift in this value can be measured from the results of tests of reactor vessel surveillance specimens and the end-of-life values can be predicted from the calculational methodology of Regulatory Guide 1.99, Revision 2. The results of the test are included in Attachment 2 to this letter. Attachment 2 is similar to our December 30, 1992, letter, except this report includes the fluence/lead factor calculations and the pressure-temperature limit results.

The pressure-temperature limits must, therefore, be modified periodically to reflect the vessel's exposure to irradiation. This ensures that operating conditions in the vessel will be maintained within acceptable limits.

The design bases for the plant includes protection against brittle fracture. These are described in the Technical Specification Bases Section 3.6. Allowance for radiation embrittlement is included in accordance with guidance provided by the nuclear steam system supplier vendor.

The proposed operating limits are based on calculational methods contained in ASME Code, Section XI and 10CFR50, Appendix G, January 1992. At the time that the Millstone Unit No. 1 reactor vessel was fabricated (1965), the fracture toughness requirements were not as comprehensive as the current ASME Code, Section XI requirements. However, Paragraph III.A of 10CFR50, Appendix G states that an approved method may be used to demonstrate equivalence of pre-1972 Code fracture toughness data with post-1972 Code requirements. Toughness property correlations were derived for the vessel materials in order to use the available data to give a conservative estimate of RT_{NDT}, consistent with the intent of 10CFR50, Appendix G criteria. These toughness correlations vary, depending upon the specific material analyzed, and were derived from the results of industry research. The operating limits are proposed for operation through 23.5×10^6 megawatt days thermal (MWD_{th}) at which time the neutron fluence at the 1/4T (one-fourth the thickness of the vessel wall, measured from the inside) location in the reactor vessel wall will be about 1.1×10^{18} n/cm^2. Regulatory Guide 1.99, Revision 2, was used as the basis for calculating the t/4 values given the surface values. The changes proposed under this license amendment request are intended to incorporate the materials testing results contained from the most recent surveillance capsule, and establish curves which are valid through 32 EFPY.

Millstone Unit No. 1 is aware of the ability to delete the surveillance schedules from the technical specifications as allowed by Generic Letter 91-01. At this time, however, NNECO has no plans to pursue this initiative.

Safety Assessment

Appendix H of 10CFR50 and ASTM E185 provide the methodology used to establish and maintain the reactor vessel neutron radiation damage surveillance monitoring program. The Millstone Unit No. 1 program included three surveillance capsules mounted in the inside surface of the reactor vessel at the 120°, 210°, and 300° azimuth locations. The first (i.e., 210°) capsule was withdrawn after 8.96 EFPY, while the second (i.e., 300°) capsule was withdrawn after 14.8 EFPY during the 1991 refueling outage. ASTM E185 requires that the third capsule be removed at the end of the plant life.

The specimens in the 300° capsule were tested and the results were compared to the values calculated in accordance with Regulatory Guide 1.99, Revision 2. The parameters used in the calculation were 6.6×10^{17} n/cm^2 for the fluence, 0.21 percent copper and 0.49 percent nickel for the plate, and 0.20 percent copper and 1.05 percent nickel for the weld material. The measured and calculated decrease in USE is provided in Attachment 3, Table I, while the measured and calculated increase in RT_{NDT} is provided in Attachment 3, Table II. These results indicate that the measured values for both the decrease in USE and the increase in RT_{NDT} are within the estimation range of Regulatory Guide 1.99, Revision 2, except for the decrease in USE of the weld material for the 300° capsule. For the weld, the decrease in USE was measured to be 21 percent which is slightly greater than the calculated decrease of 18 percent. However, this is not expected to impact the safe operation of the vessel through the end of its current design life, since the end-of-life USE is expected to remain about 50 ft-lbs (i.e., 53.7 ft-lbs).

The USE of the weld for 32 EFPY was calculated as required by Regulatory Guide 1.99, Revision 2, Paragraph C.2.2, which requires that the surveillance test results be factored into the calculation if the calculated values do not bound the measured values. For the Millstone Unit No. 1 weld, this was accomplished by plotting the weld surveillance test results on Figure 2 of the Regulatory Guide and comparing the "actual" Copper content to the "effective" copper content. This approach indicated that the surveillance weld fell on the Regulatory Guide line for 0.25 percent copper while the actual copper content of the weld is only 0.20 percent copper. This indicates that although the weld only contains 0.20 percent copper, it behaves as if it contained 0.25 percent copper resulting in greater-than-expected decrease in USE for the same fluence. To account for this additional decrease in USE, the copper content of the beltline welds was obtained by adding 0.05 percent copper to the actual measured copper content. The USE at 32 EFPY was then calculated as described in paragraph C.1.2 of the Regulatory Guide using the modified copper content for the welds.

The surveillance capsule test results were incorporated into the calculation of the pressure-temperature limit curves as required by 10CFR50, Appendix G and provided in the revised technical specification heatup and cooldown curves. These curves were derived in accordance with the methodology described in Appendix G to Section XI of the ASME Code. The allowable pressure-temperature limits were calculated for (1) hydrostatic and leak tests, (2) non-nuclear heatup/cooldown and low-level physics tests, and (3) core critical operation. These curves were developed considering the structural requirements of three separate vessel regions: the closure flange region, the core beltline region, and the remainder of the vessel, or nonbeltline region. The beltline region includes the low-intermediate and low shell course plates and associated welds and the recirculation inlet nozzles. The nonbeltline region includes the remainder of the shell course plates and welds, top and bottom heads, and all other nozzles, particularly the feedwater and the control rod drive (CRD) inlet nozzles.

The nonbeltline region curves were established by adding the highest RT_{NDT} for the nonbeltline discontinuities to the pressure (P) versus ($T-RT_{NDT}$) curves developed for the most limiting BWR/6 components which are the CRD penetration and feedwater nozzles. The highest RT_{NDT} for Millstone Unit No. 1 is 40°F obtained from the nozzle NDT requirements in the vessel purchase specification. Although Millstone Unit No. 1 is a BWR/3, the results of the BWR/6 nozzle analyses were used for Millstone Unit No. 1, since the BWR/3 nozzle design and operating conditions were found to be similar to those of the BWR/6.

The curves for the beltline region were determined by calculating the allowable stress intensity (K_{Im}) for the limiting material properties based on an expected fast neutron fluence (i.e., E > 1.0 Mev) of 1.0 X 10^{18} n/cm^2 at the t/4 location of low intermediate shell plate No. G2002-5 for 32 EFPY. This fluence combined with a 0.19 percent copper and 0.51 percent nickel content for plate G2002-5 results in an increase of RT_{NDT} of 111.2°F and an adjusted RT_{NDT} of 137.2°F. The allowable K_{Im} is then obtained from Figure G-2210-1 of the ASME Section XI, Appendix G for the adjusted RT_{NDT}. Once the allowable K_{Im} is obtained, the thermal contribution to the stress intensity (K_{It}) is calculated for a cooldown and heatup rate of 100° F/HR using the following formula:

$$K_{It} = M_t + \Delta T_w$$

where M_t is the thermal correction factor obtained from Figure G-2214-2 of ASME Section XI, Appendix G, and

ΔT_w is the through wall temperature differential during heatup/cooldown.

The allowable heatup and cooldown pressure (p_a) is then calculated as follows:

$$K_{Im} = \frac{K_{Ia} - K_{It}}{2}$$

$$\sigma_m = \frac{K_{Im}}{M_m}$$

$$p_a = \frac{\sigma_m t}{R}$$

where M_m is the membrane correction factor obtained from Figure G-2214-1 of ASME Section XI, Appendix G, t is the vessel thickness; and R is the vessel mean radius. The allowable heatup and cooldown pressure is calculated as described above for the following locations:

(1) t/4 flaw in the limiting beltline region,
(2) t/4 flaw in the limiting nozzle (i.e., feedwater nozzle),
(3) 0.24" deep flaw in the closure head flange region, and
(4) a t/4 flaw in all other locations remote from discontinuities.

The final curve is then obtained by drawing a curve which bounds all of the above locations.

The hydrostatic pressure test curve was also calculated using the above methodology, with the exception that the factor of safety of 1.5 is used on pressure rather than the factor of safety of 2.0 used for normal operating conditions. The contribution of the thermal stress to the total stress intensity was assumed to be negligible during the system hydrostatic test since the allowable heatup and cooldown rate is limited to less than 10°F/HR. Since the pressure test curves become progressively more restrictive over time, separate curves have been included for 18, 21, 24, 28, and 32 EFPY to minimize the inherent conservatism.

The curves for the hydrostatic pressure and leak tests and for the nonnuclear heatup/cooldown include a separate curve for the vessel lower head region. The reason for including a separate curve is that during certain operating modes, stratification in the vessel can occur, resulting in significantly colder water accumulating in the lower head while the remainder of the vessel remains at higher temperatures. These conditions occur predominantly when the recirculation pumps are either off or operating at low speed and injection through the control rod drives is used to pressurize the vessel. This operating mode results in cooldown rate and/or temperature at the lower head which, in some cases, exceeds the beltline region requirements. Since the neutron fluence at the beltline region causes the beltline pressure/temperature requirements to be significantly more restrictive than the bottom head which is not subjected to any significant fluence, the separate curves provide additional operator flexibility when vessel stratification is introduced during the above conditions.

In addition to the heatup, cooldown, and hydrostatic pressure test curves calculated as discussed above, 10CFR50 Appendix G specifies other requirements relating to core criticality, minimum pressurization temperature, and minimum vessel head stud tensioning temperature. The following is a discussion of these requirements and how they have been incorporated into the proposed pressure-temperature limit curves presented here:

(a) 10CFR50, Appendix G, Section IV.A.1 requires that the vessel maintain a minimum of 50 ft-lbs throughout its design life. This requirement is satisfied for the Millstone Unit No. 1 vessel since the USE for the limiting material will remain above 50 ft-lbs through the remainder of its design life.

(b) 10CFR50, Appendix G, Section IV.A.2 requires that when the core is not critical and the pressure exceeds 20 percent of the preservice hydrostatic test pressure, the temperature of the closure flange region exceeds the region's most limiting RT_{NDT} by at least 120°F for normal operation and 90°F for the hydrostatic pressure and leak tests. This requirement is incorporated into the curves through the vertical line at 116°F which extends between 312 psig and approximately 550 psig.

(c) 10CFR50, Appendix G, Section IV.A.3 requires that when the core is critical, the temperature of the reactor vessel must not be lower than 40°F above the temperature calculated in (b) above and the requirements of ASME Appendix G. This requirement is included in the pressure/temperature limit curves provided for the heatup and cooldown curves during core operation.

(d) 10CFR50, Appendix G, Section IV.A.4 requires that if there is no fuel in the reactor vessel during hydrostatic pressure or leak tests, the minimum temperature must exceed the most limiting RT_{NDT} by at least 60°F. This requirement is not explicitly included in the pressure/temperature limit curves, since the most limiting RT_{NDT} plus 60°F is 100°F and the minimum test temperature is approximately 200°F and, therefore, it bounds this requirement.

(e) ASME Section XI, Appendix G, G-2222(c) recommends that when the flange region is stressed by the full bolt preload and a pressure not exceeding

20 percent of the preservice hydro, the metal temperature in the highly stressed regions must exceed the region's most limiting RT_{NDT}. This recommendation is incorporated into the curves by allowing vessel operation only when the flange temperature exceeds 86°F. This is shown as a vertical line at 86°F which was obtained by taking the most limiting flange region RT_{NDT} of 26°F and adding an additional margin of 60°F. The 60°F margin was added for additional conservatism since the stress in the flange region resulting from bolt preload is high compared to the remainder of the vessel and because the original ASME Code of construction required this additional margin.

The pressure-temperature limitations, discussed above, are adjusted to account for instrument uncertainties and static head resulting from differences in system elevation. The instrument uncertainties, which have been previously evaluated, resulted in a combined temperature and pressure correction of +4.31°F and -12.5 psi. These corrections were incorporated into the calculated allowable pressures and temperatures.

Significant Hazards Consideration

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a significant hazards consideration because the changes do not:

1. Involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.

   The proposed curves will not result in any plant operational or hardware modifications. They will only decrease the allowable pressure vs. temperature during vessel heatup/cooldown and pressure tests to account for the anticipated end-of-life vessel embrittlement.

   The revision to the heatup and cooldown curves will ensure that the plant is maintained in a safe condition. NNECO performed a four-step process whereby NNECO established a surveillance plan according to 10CFR50 Appendix H. This required periodic removal of surveillance capsules from the reactor vessel. Secondly, NNECO performed Charpy impact tests, tensile test, and neutron flux measurements. These tests provide data for the actual neutron irradiation damage to the reactor vessel in terms of RT_{NDT} and USE. NNECO then calculated the adjusted RT_{NDT} for a postulated crack in the vessel using Regulatory Guide 1.99, Revision 2 guidance. Finally, NNECO compared the actual RT_{NDT} shift to the predicted RT_{NDT} shift. This process identified the condition of the Millstone Unit No. 1 reactor vessel and prompted the revised curves. The parameters identified in Regulatory Guide 1.99 Revision 2 have been addressed with acceptable results. Therefore, the probability of occurrence or consequence of an accident previously analyzed has not been increased.

2. Create the possibility of a new or different kind of accident previously evaluated.

   The proposed curves will not result in any plant operational changes. They will only decrease the allowable pressure vs. temperature during vessel heatup/cooldown and pressure tests.

   The intent of the pressure temperature limits is to prevent brittle fracture of the reactor vessel. By evaluating the surveillance capsule specimens, NNECO is able to establish new limits for Millstone Unit No. 1 to operate within. The adherence to the pressure temperature curves will ensure that no new or different kinds of accidents are created.

3. Involve a significant reduction in a margin of safety.

   The pressure-temperature limit curves were calculated in accordance with the requirements of 10CFR50, Appendix G which in turn requires compliance with the ASME Code Section XI, Appendix G prescribed methodology and associated margins of safety. This methodology and margin of safety is applicable to both the current and the new proposed curves.

   The adherence to these curves will ensure that the plant is maintained in a safe condition. These curves have been developed so that the reactor coolant pressure boundary is maintained with sufficient margin to ensure that, when stressed under operating, maintenance, testing and postulated accident conditions that the boundary behaves in a nonbrittle manner, and that the probability of rapidly propagating fracture is minimized. In addition, these analyses have been performed to ensure that the fracture toughness of the reactor vessel materials caused by neutron radiation is maintained within the required range.

The changes proposed under this license amendment request are intended to incorporate the materials testing results obtained from the 300 degree surveillance capsule which has been recently evaluated. These proposed curves are more restrictive than the existing curves. Based on this, it is concluded that the proposed core critical heatup/cooldown, nonnuclear heatup/cooldown and hydro test pressure-temperature limitations are safe and do not constitute a significant hazards consideration. The limits were calculated in accordance with the requirements of 10CFR50, Appendix G using the methodology provided in ASME Section XI, Appendix G. The limits provide protection against nonductile failure of the reactor vessel through 32 EFPY.

Moreover, the Commission has provided guidance concerning the application of the standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51FR7751) of amendments that are considered not likely to involve a significant hazards consideration. The changes proposed herein most closely resemble example (ii), a change that constitutes an additional limitation, restriction or control not presently included in the technical specifications. The proposed heatup/cooldown curves are more restrictive than the existing curves. The basis of the new curves is the same as the basis of the current curves, merely updated to reflect an interval of time later in service life of the reactor pressure vessel.

NNECO has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released off site, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NNECO concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement.

The Millstone Unit 1 Nuclear Review Board has reviewed and approved the proposed change and has concurred with the above determination.

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of this proposed amendment.

Regarding our proposed schedule for this amendment, we respectfully request issuance by June 12, 1993.

We trust you will find this information satisfactory and remain available to discuss this with you at your convenience.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

cc: See Page 10

cc: T. T. Martin, Region I Administrator
J. W. Andersen, NRC Acting Project Manager, Millstone Unit No. 1
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin McCarthy, Director, Radiation Control Unit, Department of Environmental Protection, Hartford, CT 06116

Subscribed and sworn to before me

this 22 day of March, 1993

Lorraine J. D'Amico
Notary Public

Date Commission Expires: 3/31/93