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June 11, 1993

Docket No. 50-336
B14445

Re: 10CFR50.90

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

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
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-65, by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2.

Background

10CFR50 Appendix G requires that the fracture toughness of the reactor coolant pressure boundary ferritic materials be maintained at such a level as to minimize the probability of nonductile failure of the pressure boundary during normal plant operation. These requirements are incorporated in the plant safety technical specifications in the form of pressure/temperature (P/T) limit curves for normal plant heatup, cooldown, and hydrostatic pressure tests. Since these limit curves are directly related to the vessel fracture toughness, and the fracture toughness decreases with increased neutron fluence, these curves must be periodically updated to reflect these changes. The purpose of this proposed change is to replace the existing limit curves, which will expire at 12 effective full power years (EFPY), with new curves which are valid through 20 EFPY, and to incorporate the material test results from the W-104 surveillance capsule evaluation. The W-104 surveillance capsule evaluation report was submitted to the NRC Staff in a letter dated November 27, 1991.[1]

The Millstone Unit No. 2 Technical Specifications provide heatup, cooldown, and hydrostatic test P/T limit curves for the reactor coolant system (RCS).

(1) J. F. Opeka letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station Unit No. 2 Reactor Vessel Material Irradiation Surveillance Capsule W-104," dated November 27, 1991.

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These curves are calculated in accordance with the requirements of 10CFR50 and ASME Section XI, Appendix G, for the following loading conditions:

1. Normal Operation, including heatup and cooldown.

2. Inservice leak and hydrostatic tests.

3. Reactor core operation.

A review of the major components of the reactor coolant pressure boundary indicated that the controlling locations were the reactor vessel closure head, reactor vessel outlet nozzle, and the reactor vessel beltline region. Since the closure head region is significantly stressed at relatively low-temperatures (due to mechanical loads resulting from bolt preload), this region typically controls the P/T limits early in the plant life. Later in the plant life, the beltline region of the reactor vessel begins to control the P/T limits due to the decrease in fracture toughness as a result of irradiation degradation.

During heatup, the higher temperatures at this inside surface of the vessel result in tensile stresses at the outer surface of the vessel wall, and in compressive stresses at the inside surface. Under these conditions, the 3t/4 flaw (i.e., a flaw with a depth of 0.25 times the vessel wall thickness open to the outer surface) is the limiting flaw, even though it is subjected to less radiation damage than the t/4 flaw. The allowable pressure curve during heatup is then obtained by calculating the crack-driving potential resulting from the thermal stresses, and subtracting it from the material crack growth resistance, yielding the allowable pressure crack-driving potential and, therefore, the allowable pressure. A factor of one is imposed on the thermal loads, while a factor of two is imposed on the pressure loads. These calculated allowable P/T limits are then corrected for the instrumentation uncertainties and for the differences in the RCS static head.

The ability of the material in the beltline region to resist crack propagation (K_{IR}) is obtained by calculating the limiting adjusted Reference Temperature, Nil Ductility Transition (RT_{NDT}) and then using the K_{IR} curve provided in ASME Section XI, Appendix G, to obtain the material fracture toughness. The RT_{NDT} is the reference temperature at which the material begins the transition from brittle to ductile behavior. This parameter is measured through drop weight tests and Charpy impact tests where a minimum of 50 ft/lbs and 35 mils of lateral extension is obtained in the material's weak direction. Analytically, the adjusted RT_{NDT} is obtained by calculating the increase due to neutron exposure and adding it to the unirradiated RT_{NDT} plus a margin to account for the measurement and calculation uncertainty, as required by Regulatory Guide 1.99, Rev. 2. Since these values were compared to the surveillance test results and found to be more limiting than the measured RT_{NDT} values, no adjustments were required. The limiting upper-shelf energy (USE) was also measured and calculated using the methodology described in Regulatory Guide 1.99, Rev. 2. Since both methods indicated that the USE will remain above 50 ft/lbs through the remaining life of the vessel, the material fracture toughness in the upper shelf is found to be sufficient to preclude ductile crack extension as required by 10CFR50 Appendix G.

The P/T limit curves for 20 EFPY were calculated for the limiting beltline material; i.e., base metal heat #C5843-2. At 20 EFPY this material is expected to exhibit an RT_{NDT} of 145 degrees F for the t/4 location and 118 degrees F at the 3t/4 location. The 20 EFPY period was chosen because the fluence for capsule W-104 (i.e., 10 EFPY) which contained material representative of the beltline region, was equivalent to the fluence at the vessel inside surface at 15 EFPY and equivalent to the fluence at the vessel t/4 location at 24 EFPY; i.e., 8.84×10^{18} n/cm^2. Based on this, it was concluded that capsule W-104 provides adequate basis for developing P/T limits through 20 EFPY since controlling locations are the t/4 and 3t/4 locations.

Description of Proposed Changes

The Millstone Unit No. 2 Technical Specifications contain limitations on allowable (RCS) pressures and temperatures. The proposed changes revise the P/T limits for the reactor vessel. Specifically, Figure 3.4-2, "Millstone Unit 2 Reactor Coolant System Pressure-Temperature Limitations for 12 Full Power Years," on page 3/4 4-19, is being revised to reflect the change in the curves and the title changed to "Millstone Unit 2 Reactor Coolant System Pressure Temperature Limitations for 20 EFPY." Also, page B 3/4 4-6, Reactor Coolant System, "Bases" is being revised to reflect the deletion of the wording contained in paragraph 2 of "...SECY-82-465, 'NRC Staff Evaluation of Pressurized Thermal Shock,' November 1982. Because it is more conservative, this method was used rather than the proposed...." The revised sentence will read, "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 purpose for this deletion is that Regulatory Guide 1.99, Revision 2, is the basis for the proposed curves. In paragraph 4 of the same page, "ASTM E185-73" is being replaced with "10CFR50 Appendix H." The basis for this change is that 10CFR50 Appendix H is the Regulatory document which provides the requirements for Reactor Vessel Surveillance programs.

Safety Assessment

A comparison of the existing P/T curves (i.e., for 12 EFPY) to the proposed curves (i.e., for 20 EFPY) indicates that they are similar, even though the vessel wall will be exposed to neutron radiation for a significantly longer period of time. The reasons for this finding are as follows:

1. The RT_{NDT} for 12 EFPY was obtained by taking the bounding value from the Guthrie and the draft Regulatory Guide 1.99, Rev. 2, estimation methods. The Guthrie method was found to be more limiting for the material and neutron environment in the Millstone Unit No. 2 reactor vessel. The RT_{NDT} for 20 EFPY was obtained using the methodology recommended in Regulatory Guide 1.99, Revision 2, only.

2. The projected maximum fluence between 7 EFPY and 12 EFPY was obtained by assuming the same core-loading patterns as those used between startup and 7 EFPY. Since Millstone Unit No. 2 changed to a low-leakage core at approximately 10 EFPY, the surface fluence at the limiting location for 20 EFPY is expected to be less than the previously calculated surface fluence at the limiting location for 12 EFPY, based on the recently evaluated W-104 capsuled; i.e., $1.32 \times 10^{19} n/cm^2$ versus $1.60 \times 10^{19} n/cm^2$.

Low-Temperature Overpressure Protection

NUREG-0800 requires that a low-temperature overpressure protection (LTOP) system be enabled at low temperatures (i.e., relative to the material RT_{NDT}) to ensure that inadvertent system transients do not result in catastrophic nonductile failure of the RCS. This system at Millstone Unit No. 2 consists of two power-operated relief valves (PORVs), and associated relief piping, with a setpoint of 450 psi. This relief pressure, when compensated for valve overshoot and maximum system pressure accumulation, results in a maximum system pressure of 465 psi. This assumes that only one PORV is actuated during the mass addition transient.

The heatup and cooldown rates used in developing the attached curves were chosen in a manner as to ensure that the pressure-relieving capabilities of the existing LTOP system remain adequate through 20 EFPY. Although the lowest allowable pressure during plant cooldown is slightly less than the maximum system pressure during the postulated LTOP transient (i.e., 449 psia versus 465 psia), the LTOP valve setpoint is found to be acceptable through 20 EFPY. This conclusion is based on the criteria proposed in the ASME Code case N-514 developed for LTOP conditions. This code case, endorsed by the NRC, states the following:

1. The maximum allowable pressure during an LTOP event shall not exceed 110 percent of the P/T limit curves developed in accordance with the ASME Section III/XI, Appendix G, criteria; i.e., 494 psi.

2. The LTOP enable temperature shall exceed the most limiting RCS RT_{NDT} plus 50 degrees F, or 200 degrees F, whichever is greater. Since the most limiting RT_{NDT}+50 is 208.1 degrees F and the Millstone Unit No. 2 enable temperature is 275 degrees F, this requirement is satisfied.

When the capabilities of the existing LTOP system are compared to the proposed P/T limit curves, it is concluded that both of the above requirements will be satisfied through 20 EFPY without any changes in the LTOP pressure-relief valve setpoints.

Impact of Postulated Failures

The failure to correctly calculate the P/T limit curves is not expected to directly result in nonductile failure of the RCS; i.e., through-wall extension of a previously undetected crack. The margins of safety against nonductile failure of the RCS are ensured through the requirements of 10CFR50.61, which states that failure of the RCS under worst-case pressurized thermal shock (PTS) events is highly unlikely as long as the maximum RT_{NDT} does not exceed 270 degrees F anywhere in the RCS. The 270 degree F requirement is not expected to be exceeded during the current design license of the RCS.

Failure to comply with the Appendix G requirements could significantly decrease the plant margins of safety, and it is, therefore, considered a serious challenge to the structural integrity of the RCS. For this reason, the following actions have been implemented to ensure that the Appendix G requirements are not exceeded during normal plant operations:

1. The P/T calculations and associated fluence evaluations were performed under the requirements of the Babcock & Wilcox Quality Assurance Program, which included an independent review of all design inputs and calculations.

2. The proposed curves were compared to the existing curves to minimize the probability of systematic calculational errors.

3. Defense-in-depth during low-temperature operation of the RCS is provided through the LTOP system which has been verified to remain adequate through 20 EFPY for the assumed core loading patterns.

4. The format of the curves and heatup/cooldown rate requirements were maintained to minimize the probability of operator error by minimizing the changes in the operating procedures/requirements.

In summary, it is concluded that failure to comply with the Appendix G requirements could significantly decrease the margins of safety against nonductile failure of the vessel/RCS. Therefore, the P/T limit curves were developed and implemented under a rigorous Quality Assurance Program to ensure compliance with the 10CFR50 Appendix G requirements against nonductile failure of the RCS. In addition, the vessel neutron irradiation damage estimation has been validated through the Millstone Unit No. 2 surveillance program, including the recent evaluation of surveillance capsule W-104. This evaluation also demonstrated that the USE for the limiting vessel materials will remain above the 10CFR50 Appendix G requirement of 50 ft/lbs through the remainder of the vessel design life.

The heatup and cooldown curves are intended to provide limitations on the RCS pressure during plant heatup, cooldown, and system hydrostatic and leak tests to ensure that nonductile failure of the RCS does not occur. These limits are implemented during plant operation through administrative operator requirements (i.e., Plant Technical Specifications) and through plant hardware (i.e., LTOP and disabling of certain ECCS pumps) which provide defense-in-depth. These curves must be periodically updated as discussed above, to incorporate the effects of neutron exposure on the fracture resistance of the reactor vessel. Since the purpose of updating the limit curves is to update the changes in vessel properties, the proposed heatup, cooldown, and hydrostatic test curves for 20 EFPY are safe, since they were calculated in accordance with the ASME Section XI, Appendix G, requirements and do not require any plant hardware or operational changes.

Significant Hazards Consideration

NNECO has reviewed the proposed changes in accordance with 10CFR50.90 and has concluded that the changes are safe and do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

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

   The proposed curves will not result in any plant operational or hardware modifications. They are adjusted to incorporate the results of the testing program on surveillance capsule W-104 which was removed from Millstone Unit No. 2 vessel after 9 EFPY. The proposed change upgrades the P/T limits to account for the neutron irradiation damage and it incorporates the recently developed LTOP criteria recommended by the ASME Code which specifies a maximum LTOP pressure of 110 percent of the Appendix G pressure. The previous criteria required that the LTOP pressure be maintained below the Appendix G allowable pressure. This change is found to be acceptable since it will continue to preclude nonductile failure of the RCS while providing operator flexibility and minimizing the frequency of challenges to the LTOP system. The parameters identified in Regulatory Guide 1.99, Revision 2, have been addressed and have showed acceptable results. Therefore, the probability of occurrence or consequences of an accident previously analyzed have not been increased.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   The proposed curves will not result in any plant operational changes. The P/T limit curves were developed and implemented under a rigorous Quality Assurance Program to preclude nonductile failure of the RCS. In addition, the vessel neutron irradiation damage estimation has been validated through the Millstone Unit No. 2 surveillance program, including the evaluation of surveillance capsule W-104. This evaluation also demonstrated that the USE for the limiting vessel materials will remain above the 10CFR50, Appendix G requirement of 50 ft-lbs, through the remainder of the vessel design life. The adherence to the P/T curves will ensure that no new or different kinds of accidents are created.

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

   The margins of safety against nonductile failure of the RCS are ensured through the requirements of 10CFR50.61, which states that failure of the RCS under worst case pressurized thermal shock events is highly unlikely as long as the maximum RT_{NDT} does not exceed 270°F anywhere in the RCS. The 270°F requirement is not expected to be exceeded during the current design license of the RCS.

   The adherence of 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 Commission has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (March 6, 1986, 51 FR 7751) of amendments that are considered not likely to involve an SHC. The change proposed herein is not enveloped by a specific example. As described above, the proposed change does not constitute an SHC in that the change does not involve a significant increase in the probability of occurrence or consequence of an accident previously analyzed, does not create the possibility of a new or different kind of accident, nor involve a reduction in a margin of safety.

Environmental Consideration

NNECO has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not increase the type 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)(a) for categorical exclusion from the requirements for an environmental impact statement.

Based upon the information contained in this submittal and the environmental assessment for Millstone Unit No. 2, there are no significant radiological or nonradiological impacts associated with the proposed action, and thus the proposed license amendment will not have a significant effect on the quality of the human environment.

The Millstone Unit No. 2 Plant Operations Review Committee and Nuclear Review Board have reviewed the proposed changes and have concurred with the above determination.

To allow for the incorporation of these curves into the technical specifications before Millstone Unit No. 2 reaches 12 EFPY, we request issuance in December 1993, with the amendment effective within 30 days of issuance.

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

If you have questions regarding this amendment, please contact our Licensing Engineer.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: J. F. Opeka
Executive Vice President

BY: W D Rby
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Subscribed and sworn to before me

this 11th day of June, 1993

Ruth J. Dietrich

Date Commission Expires: 3/31/95