Northeast
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September 29, 1995

Docket No. 50-336
B15371

Re: 10CFR50.90

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

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Pressurizer and Main Steam Safety Valves

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its Operating License, DPR-65, for Millstone Unit No. 2 by incorporating the attached Technical Specifications revision to Sections 3.4.2.1, 3.4.2.2, 3.7.1.1, and Table 4.7-1.

The proposed license amendment combines three separate changes to the Millstone Unit No. 2 Technical Specifications which pertain to safety valves. The first proposed modification will expand the as-found tolerance of the lift setting pressure for the pressurizer and the main steam safety valves from the current value of ±1 percent to ±3 percent. Clarifications have also been added specifying that the lift setting pressure shall be determined at normal operating conditions and shall be set within ±1 percent of the required lift setting. The second portion of this modification will eliminate the need to verify the main steam safety valve orifice size. The third modification will modify the main steam safety valve action statement to reflect that if a main steam safety valve is inoperable and compensating action cannot be taken that the plant must be brought to hot shutdown (Mode 4) in 12 hours instead of cold shutdown (Mode 5) in 30 hours.

Attachment 1 to this letter provides a safety assessment for the proposed changes. Attachment 2 is the determination of no significant hazards considerations. Attachment 3 is a copy of the marked-up version of the appropriate sections of the current Technical Specifications. Attachment 4 contains the retyped Technical Specification sections.

NNECO has reviewed the proposed Technical Specification changes in accordance with 10CFR50.92 and concludes that the changes do not involve a significant hazards consideration. NNECO has also reviewed the proposed license amendment against the criteria of

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10CFR51.22 for environmental considerations and concludes that the changes do not increase the types and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Thus, NNECO concludes that the proposal satisfies 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement.

The Millstone Unit No. 2 Nuclear Safety Assessment Board concurs with the above determinations. In accordance with 10CFR50.91(b), NNECO is providing the State of Connecticut with a copy of this proposed license amendment.

Regarding the proposed schedule for this amendment, we request issuance at your earliest convenience and implementation within 60 days of issuance. It should be noted that the NRC has approved a similar change for Millstone Unit No. 3.[1]

There are no commitments contained within this letter. If there are any questions regarding this submittal, please contact Mr. Mario Robles at (203) 440-2073.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: J. F. Opeka
Executive Vice President

BY: ______________________
D. B. Miller, Jr.
Senior Vice President - Millstone

cc: See Page 3

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(1) V. L. Rooney to J. F. Opeka, "Issuance of Amendment (TAC No. M90492)," dated March 17, 1995.

cc: T. T. Martin, Region I Administrator
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin T.A. McCarthy, Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Subscribed and sworn to before me

this 29th day of September, 1995

Gerard P. van Noordennen

Date Commission Expires: 12/31/97

Docket No. 50-336
B15371

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Pressurizer and Main Steam Safety Valves
Safety Assessment of Proposed Changes

September 1995

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Pressurizer and Main Steam Safety Valves

Description of Proposed Change

The proposed license amendment will modify Limiting Condition for Operations 3.4.2.1 and 3.4.2.2 by modifying the as-found tolerance for the pressurizer code safety valves from ±1 percent to ±3 percent. Notes have also been added indicating the lift setting pressure shall be determined at normal operating conditions and shall be set within ±1 percent of the lift setting. The action statement for Technical Specification 3.7.1.1 has been modified so that if the limiting condition for operation cannot be met, the plant must be brought to hot shutdown (Mode 4) within 12 hours instead of cold shutdown (Mode 5) within 30 hours. Surveillance Requirement 4.7.1.1 and Table 4.7-1 have been modified to eliminate the need to verify the orifice size of each main steam safety valve. Table 4.7-1 has also been modified to correct the as-found tolerance for the main steam safety valve from ±1 percent to ±3 percent. Notes have also been added to the table which specify that the lift setting should be determined at normal operating conditions and should be set at ±1 percent of the lift setting.

Safety Assessment

Pressurizer Safety Valve Tolerance Limit

Currently, the pressurizer safety valve setpoint surveillances are performed in accordance with the ASME code. Past history(1) has shown that while the as-left condition for the pressurizer safety valve was within the ±1 percent tolerance limit, the as-found setpoint from the next surveillance often had drifted outside the ±1 percent tolerance limit. With the proposed change, the as-left condition will remain the same. Thus, there is no change in the way that the pressurizer safety valves will be maintained. However, the as-found tolerance limit is being relaxed to ±3 percent to reflect the history of larger drift in the setpoint. This maintains conformance to the ASME code while allowing more margin for setpoint drift.

The pressurizer safety valves are credited for mitigation of RCS overpressurization events. The limiting RCS overpressurization

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(1) The most recent surveillance results were reported in S. E. Scace letter to the U.S. Nuclear Regulatory Commission, "License Event Report 92-013-00," dated August 12, 1992.

event is the loss of electrical load. The loss of electrical load was reanalyzed with a +3 percent tolerance from the nominal setpoints for the pressurizer safety valves and the main steam safety valves. The analysis shows that RCS pressure remains below 110 percent of design. Thus, there is no reduction in margin of safety even with a ±3 percent tolerance on these safety valve setpoints.

The RCS pressure used for accidents where minimum DNBR is a concern bounds the -3 percent lower bound on the pressurizer safety valve nominal setpoint. Thus, the allowance of a ±3 percent tolerance has no impact on minimum DNBR for the limiting DNBR transients.

Since there is no change in the maintenance of the safety valve, there is no impact on the probability of failure of the pressurizer safety valve. In addition, even with the larger tolerance, the safety valve setpoint will provide margin to normal operation, high pressurizer pressure trip setpoint, and the power operated relief valve (PORV) setpoint. This action will minimize the challenges to the safety valves. Thus, there is no impact on the probability of an inadvertent opening of a pressurizer safety valve.

Thus, it is concluded that with the increase in tolerance in the as-found setpoint for the pressurizer code safety valve, the margin of safety for RCS overpressurization and DNBR will be maintained and there is no increase in consequences or probability of any design basis accident.

Main Steam Safety Valve Shutdown Requirement

Technical Specification 3.7.1.1 requires all main steam safety valves (MSSVs) to be operable in Modes 1, 2, and 3. If the MSSVs are not operable, the action statement specifies that operation in Modes 1, 2, and 3 may continue provided that either the inoperable valves are restored to operable status or the high power level trip setpoint be reduced per Table 3.7-1 (which allows up to three of the eight valves on any one steam generator to be inoperable). Otherwise, the plant is required to be in Mode 3 within the next 6 hours and in Mode 5, cold shutdown (RCS average temperature less than or equal to 200°F), within the following 30 hours.

The proposed modification is to require that the plant be in hot shutdown (Mode 4) within the following 12 hours instead of cold shutdown (Mode 5) in 30 hours. The basis for the proposed change is that the LCO does not require the MSSVs to be operable in Mode 4 (RCS average temperature less than 300°F but greater than 200°F). Thus, the action statement is being changed to be consistent with the LCO. This change is also consistent with NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants."

In hot shutdown (Mode 4) reactor decay heat is either removed by the main steam system or the shutdown cooling system. If the shutdown cooling system is operating, then the steam generators are not being utilized for secondary decay heat removal, and the MSSVs are not relied on for steam generator pressure control. If the steam generators are removing decay heat, then the atmospheric dump valves or the turbine bypass valves are being used to control steam generator pressure. With the steam generators removing decay heat and the RCS average temperature less than 300°F, the steam generator would have to be saturated at a temperature less than 300°F. This corresponds to a steam generator pressure of less than 70 psia. With this low steam generator pressure, there is a large margin to overpressurization. In addition, with the RCS cold leg temperature ≤ 275°F, the cold overpressurization system is placed in service by either lowering the PORV setpoint to 450 psig or venting the RCS (LCO 3.4.9.3). As such, the main steam safety valves are not required to be operable in Mode 4. Since the main steam safety valves are not required for Mode 4, it is appropriate to cooldown to Mode 4 in the event that the main steam safety valves cannot be made operable within the required time frame.

<u>Main Steam Safety Valve Orifice Size</u>

The main steam safety valve orifice size represents the smallest inside diameter of the safety valve nozzle, an internal part of the valve. The orifice size is not adjustable and can only be changed by replacement of the nozzle. Replacement of the nozzle requires removal and disassembly of the safety valve. Further, only one size nozzle is available for these safety valves. Since there is no adjustment possible to the orifice size, and changes to the orifice requires a modification of the valve that would be covered under the design change process, the specification of the orifice size in the Technical Specification is unnecessary. Removing the specification will have no impact on the plant configuration or operation. Thus, the safety analysis will be unaffected by the change.

<u>Main Steam Safety Valve Tolerance</u>

Similar to the pressurizer safety valves, the main steam safety valve setpoint surveillances are performed in conformance to the ASME code. Past history[2] has shown that while the as-left condition for the main steam safety valve was within the ±1 percent

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(2) The most recent surveillance results were reported in D. B. Miller, Jr. letter to the U.S. Nuclear Regulatory Commission, "License Event Report 94-030-00," dated October 28, 1994 and LER 92-010-00," dated June 29, 1992.

tolerance limit, the as-found setpoint from the next surveillance often had drifted outside the ±1 percent tolerance limit. With the proposed change, the as-left condition will remain the same. Thus, there is no change in the way that the main steam safety valves will be maintained. However, the as-found tolerance limit is being relaxed to ±3 percent to reflect the history of larger drift in the setpoint. This maintains conformance with the ASME code while providing additional margin for setpoint drift.

The main steam safety valves are credited for mitigation of steam generator overpressurization events. The limiting steam generator overpressurization event is the inadvertent closure of one main steam isolation valve. This accident was reanalyzed with a 3 percent increase in the nominal setpoint for the main steam safety valves. The analysis shows that steam generator pressure remains below 110 percent of design. Thus, there is no reduction in margin of safety even with a ±3 percent tolerance on the main steam safety setpoint. In addition, the loss of load was also reanalyzed with a ±3 percent tolerance, and it reconfirmed that the inadvertent closure of one main steam isolation valve remains the limiting steam generator overpressure transient.

The steam generator safety valve setpoint can also affect the calculation of offsite doses associated with a steam generator tube rupture. A lower opening pressure for the safety valves will increase the secondary side releases during the shutdown and cooldown in the mitigation of the tube rupture. The steam generator tube rupture has been reanalyzed assuming a safety setpoint 3 percent below the nominal setpoint. In addition, the offsite doses have been extended to one hour to provide more margin for operator action to terminate the releases from the affected steam generator. The results are summarized in the following table.

		Dose (Rem)			
Iodine Spiking Criteria		Current Analysis		Lower Lift Pressure Analysis	
		EAB	LPZ	EAB	LPZ
Spike Caused by Accident	Thyroid	0.153	0.0165	0.1594	0.01671
	Whole Body	0.084	0.0264	0.1455	0.04485
Pre-Accident Iodine Spike	Thyroid	0.78	0.081	0.8125	0.08455
	Whole Body	0.084	0.0264	0.1455	0.04485

From this table it is shown that the combined effect of extended releases and a ±3 percent tolerance on the main steam safety valve setpoint has a small effect on the calculated offsite doses. The results are well below the Standard Review Plan acceptance criteria. Thus, the proposed change does not significantly increase the consequences associated with the steam generator tube rupture.

Since there is no change in the maintenance of the safety valve, there is no impact on the probability of failure of the main steam safety valve. Thus, there is no impact on the probability of an inadvertent opening of a main steam safety valve.

Thus, it is concluded that with the increase in tolerance in the as-found setpoint for the main steam safety valve, the margin of safety for steam generator overpressurization will be maintained, and there is no significant increase in consequences or probability of any design basis accident.

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Proposed Revision to Technical Specifications
Pressurizer and Main Steam Safety Valves
Determination of No Significant Hazards Consideration

September 1995

Millstone Nuclear Power Station, Unit No. 2
Proposed Revision to Technical Specifications
Pressurizer and Main Safety Valves
Determination of No Significant Hazards Consideration

Significant Hazards Consideration Determination

In accordance with 10CFR50.92, NNECO has reviewed the proposed changes and has concluded that they 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 evaluated.

   The change in the as-found pressurizer safety valve tolerance will not increase the probability of occurrence of any of the design basis accidents. Even with the larger tolerance, the setpoint will provide margin to normal operation, the reactor setpoint, and PORV setpoint. This minimizes the challenges to safety valves and assures that there is no increase in the probability of an inadvertent opening of a pressurizer safety valve. Similarly, even with the increase in allowed as-found tolerance for the main steam safety valves, the setpoints will still provide margin to normal operation. Thus, there is no impact on the probability of an inadvertent opening of a steam generator safety valve.

   The loss of load event and the inadvertent closure of one main steam isolation valve have been reanalyzed to show that even with a ±3 percent tolerance for the pressurizer safety valves and the main steam safety valves, that both the peak RCS pressure and the peak steam generator pressure remain below 110 percent of design. Thus, even with the larger as-found tolerances, the margin of safety for RCS and steam generator overpressurization is maintained.

   The steam generator tube rupture has been reanalyzed to take into account the ±3 percent as-found tolerance and to extend the margin for operator action to one hour. A comparison of the calculated doses shows that with the new assumptions, there would be a very small increase in calculated doses. The increased calculated doses, however, remain well below the Standard Review Plan acceptance criteria.

The proposed change in the shutdown mode does not impact the probability or consequences of an accident previously evaluated. The proposed change makes the action required for inoperable main steam safety valves consistent with the modes that the technical specification is applicable and would not modify the assumptions made in any accident previously analyzed.

The change to delete the main steam safety valve orifice size from technical specifications has no impact on any design basis accident analysis.

Based upon these evaluations, it is concluded that the proposed changes do not significantly increase the probability or consequences of any design basis accident.

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

   The proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

   The proposed changes do not change the as-left setpoints. The change in as-found tolerances for the safety valves is being made to reflect the results of past surveillances that indicate that the setpoints can drift more than the current criteria. However, there is no change in the plant configuration or in as-left setpoints.

   The proposed change which requires the plant to go to Mode 4 in 12 hours instead of Mode 5 in 30 hours if the action statement is not met, is consistent with the applicable modes of the technical specification (i.e., the technical specification is not applicable in Mode 4). No new or different kind of accident from those previously analyzed can be postulated as a result of this proposed change.

   Thus, the changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

   As discussed above, the loss of load event and the inadvertent closure of one main steam isolation valve have been reanalyzed to show that even with a ±3 percent tolerance for the pressurizer safety valves and the main steam safety valves, that both the peak RCS pressure and the peak steam generator pressure remain below 110 percent of design. Thus, even with the larger as-found tolerances, the margin of safety for RCS and steam generator overpressurization is maintained. In addition, the steam generator tube rupture has been reanalyzed with a ±3 percent tolerance on the steam generator safety valves and the results show an insignificant increase in the calculated doses.

The proposed change also directs the operator to bring the plant to hot shutdown instead of cold shutdown to be consistent with the applicable modes of the technical specification. There is no impact on the assumptions made or the results of any accident previously analyzed.

Therefore, it is concluded that the changes do not involve a significant reduction in the margin of safety.

Moreover, the commission has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (51 FR 7751, March 6, 1986) of amendments that are considered not likely to involve an SHC. Although the proposed changes are not enveloped by a specific example, the discussions above clarify that the changes to the safety valves are not an SHC.