NORTHEAST UTILITIES

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Docket No. 50-336
B10960

Director of Nuclear Reactor Regulation
Attn: Mr. James R. Miller, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Steam Generator Tube Rupture Reassessment
for Cycle 6 Operation

On April 13, 1983(1), Northeast Nuclear Energy Company (NNECO) provided to the NRC Staff the Reload Safety Analysis in support of Cycle 6 operation. Since that time, NNECO has identified several fuel assemblies containing leaking fuel pins as well as other fuel related damage which necessitated a revision of the Cycle 6 loading pattern and has informed the Staff of these problems(2),(3), (4).

NNECO has also removed the thermal shield from the core barrel since the support of the shield had degraded to an unsatisfactory condition.

In response to the degradation experienced, NNECO has assessed the impact of a new loading pattern and removal of the thermal shield on those analyzed incidents which comprise the licensing basis for Cycle 6 operation. The Staff was informed of these reassessments via two submittals. The first submittal was docketed November 2, 1983 and presented a summary of our reassessments of the Small and Large Break Loss of Coolant Accidents(5). The second submittal

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(1) W. G. Counsil letter to R. A. Clark, dated April 13, 1983.

(2) W. G. Counsil letter to R. A. Clark, dated June 22, 1983.

(3) W. G. Counsil letter to J. R. Miller, dated September 12, 1983.

(4) W. G. Counsil letter to J. R. Miller, dated November 4, 1983.

(5) W. G. Counsil letter to J. R. Miller, dated November 2, 1983.

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consisted of our November 17, 1983 letter forwarding the supplement to the Reload Safety Analyses[6]. In summary, these documents confirm that the Cycle 6 reload redesign and the impact of thermal shield removal does not result in exceeding previously acceptable safety limits for all incidents comprising the licensing basis for Millstone Unit No. 2.

Our November 17, 1983[6] letter also identified the Steam Generator Tube Rupture (SGTR) event as being in the process of review. Since that time, NNECO has completed this review and a summary of the reevaluation of the SGTR event at Millstone Unit No. 2 is presented as follows.

Impact of Increased Core Flow Rate and
Altered Core Loading Pattern on the SGTR Event

An evaluation of the impact of removing the thermal shield and an altered core loading pattern has been performed for the Steam Generator Tube Rupture (SGTR) event. Additionally, approximately 700 fewer steam generator tubes have been plugged than were assumed in the SGTR analysis.

The effects of removing the thermal shield and plugging fewer steam generator tubes lead to a primary flow rate higher than the proposed minimum technical specifications value of 350,000 gpm for Cycle 6 operation. A low reactor coolant system flow rate is conservative for the SGTR analysis because it delays reactor trip which causes an increase in primary to secondary side leakage. As a result, the SGTR analysis submitted on April 13, 1983[1] is bounding with respect to thermal shield removal and 700 fewer steam generator tubes being plugged.

Since the new loading pattern will not result in any revisions to the technical specifications proposed for Cycle 6 operation[1], and since these limits were chosen conservatively for the SGTR analysis, the reload redesign will not invalidate the results of the Cycle 6 Steam Generator Tube Rupture Analysis.

In summary, the Cycle 6 SGTR will not be made less conservative by the changes discussed above.

Impact of Safety Valve Performance
Characteristics on the SGTR Event

An additional item referenced in the November 17, 1983 letter reported that NNECO was evaluating information received regarding steam generator safety valve performance characteristics and its impact on the SGTR event.

For the Staff's information Millstone Unit No. 2 utilizes 16 safety valves for over pressure protection on the shell side of the steam generators which discharge to atmosphere. Eight of these safety valves are mounted on each of the main steam lines outside the containment upstream of the steam line isolation valves. In addition, two spare valves are kept onsite. Design parameters and additional details for the main steam safety valves are provided in Section 4.3.2 of the Millstone Unit No. 2 FSAR.

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(6) W. G. Counsil letter to J. R. Miller dated November 17, 1983.

On September 12, 1983 Northeast Nuclear Energy Company received notification from Dresser Industries[7] indicating that the Dresser 3707RA Main Steam Safety Valves may not be capable of reseating at a pressure 5% below the opening pressure. Dresser stated that laboratory tests indicate that safety valves similar to those at Millstone Unit No. 2 may reseat at a pressure 7% to 12% below the opening pressure. This effect is commonly referred to as safety valve blowdown. Additionally, Dresser's November 18, 1983[8] letter documented a telecon between ourselves and Dresser Industries describing lifting characteristics of these valves as tested in an independent laboratory.

As Millstone Unit No. 2 is the only Northeast Utilities operating facility with Dresser 3707RA main steam safety valves, NNECO has reviewed all licensing basis events that are impacted by the increased safety valve blowdown. Of these analyses the only event that is impacted by the Dresser information is the SGTR event. In the SGTR event the steam generator safety valves are one of the more significant pathways by which radioactive material could be discharged to the environment. Therefore, NNECO has performed a reanalysis of the SGTR event to estimate the radiological impact of the safety valves reseating at a lower pressure. This reanalysis was performed with RETRAN02-MOD002.

The conclusions of this study indicate that the radiological doses will be increased as a result of the lower reseat pressure. However, the releases are expected to be significantly below the 10CFR100 criteria. NNECO will submit the results of a more complete radiological assessment of the SGTR event for Millstone Unit No. 2 shortly after startup for Cycle 6 operation.

In preparation for performing this evaluation, it was discovered that the safety valve reseat pressure should have been modeled more exactly in the analysis submitted on April 13, 1983. Instead of modeling reseating at a pressure 5% below the opening pressure, the valves were modeled to reseat at the opening pressure. Therefore, this evaluation not only examines the impact of safety valve reseat pressure possibly being less than the design value, but also examines the impact that more exact modeling of the safety valves has on the conclusions reached in the Reference 1 analysis. This reanalysis also assumes conservative safety valve lift setpoints by including a minus 1 percent drift as allowed by Technical Specifications.

A sensitivity study was performed as part of the Reference 1 submittal. This study indicated that the operator's failure to trip the reactor coolant pumps following a low pressurizer pressure signal at 1600 psia results in more severe radiological consequences than if the pumps were tripped according to the emergency operating procedures. Since the effect of modeling the safety valve reseat pressure is significant, this sensitivity study was redone for this submittal to determine if the conclusions of Reference 1 remain valid.

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(7) F. P. Bolger letter to E. W. Darling, dated September 6, 1983.

(8) D. Butler letter to T. Mulder, dated November 18, 1983.

Table 1 summarizes the results for the cases performed in this evaluation and compares them to the results submitted as Case 3 in Reference 1. As shown in Table 1, the impact of modeling the safety valves as designed was investigated in Case 4B. This case resulted in an increase in steam releases to the environment via the steam generator safety valves and atmospheric dump valves of 32% above the results reported in Reference 1. Although this is a significant increase, it is expected that the radiological consequences of this additional release will be well below the criteria specified in 10CFR100.

The impact of the safety valves closing at a pressure 12% below the opening pressure (the maximum blowdown reported by Reference 7) was investigated in Case 5B. As shown in Table 1, this resulted in an increase in steam releases via the safety valves and atmospheric dump valves of 149% above the results of Case 4B and 229% above the results reported in Reference 1. Although this is a significant increase in releases compared to either Case 3 or 4B, it is not expected to violate the criteria specified in 10CFR100 because the 2 hour Exclusion Area Boundary Thyroid dose reported in Reference 1 was so low (0.239 Rem).

In Cases 3, 4B, and 5B, the operator was assumed not to trip the reactor coolant pumps even though the emergency operating procedures require him to do so when the pressurizer pressure decreases below 1600 psia. This assumption was made because a sensitivity study, reported in Reference 1, indicated that this assumption was conservative. Case 5C was run to determine if this assumption remains valid when the safety valves remain open for longer times due to the lower reseat pressure. As shown in Table 1, the steam releases for Case 5C via the safety valves and atmospheric dump valves are predicted to be 53% lower than when the pumps were not tripped (Case 5B). Therefore, the conclusion reached in Reference 1 regarding reactor coolant pump operation remains valid. The reason for this result is that the increased primary coolant flow rate, that occurs when the pumps are running, improves the heat transfer coefficient inside the steam generator U-tubes. This increases the rate of heat transfer from the primary side coolant to the secondary side coolant. As a result of the increased heat load, the steam generator pressure decreases more slowly and the safety valves reseat later when the pumps are not tripped. The sequence of events, shown in Table 2, verifies that the results shown in Table 1 and the above argument are consistent.

In summary, Case 5B is the most conservative case. Results of this analysis indicate increased steam releases of 229% above the results reported in Reference 1. Although this is a significant increase over those flow rates reported in Case 3 of Reference 1, it is expected that the radiological consequences due to these releases should be bounded by a factor of 4 applied to the Reference (1) thyroid dose results. Hence the dose should still be less than 1 Rem and well within 10CFR100 criteria.

NNECO hereby verifies that the 10CFR50.59 determination made in our May 16, 1983 letter[9] referencing the SGTR event remains applicable. This determination was provided to the Staff in accordance with our Reference (1) application for revisions to the Millstone Unit No. 2 Technical Specifications for Cycle 6 operation.

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(9) W. G. Counsil to R. A. Clark, dated May 16, 1983.

At present, NNECO has not QA verified the above reported reanalysis of the Millstone Unit No. 2 SGTR event. QA verification will be completed prior to startup for Cycle 6 operation; NNECO will promptly notify the Staff upon completion of the verification. NNECO is currently analyzing the radiological consequences of the SGTI reanalyis. Northeast Nuclear Energy Company will document a summary of radiological results shortly after startup for Cycle 6 operation.

On November 11, 1983,(10) the Staff provided correspondence on the subject of our Cycle 6 reload submittal and NNECO's qualifications to perform licensing evaluations. This letter identified the Staff's plans to conduct an inspection of our quality assurance program for performing licensing evaluations. At that time, the Staff also identified plans to assess our qualifications for performing thermal-hydraulic transient and accident evaluations. To facilitate this inspection NNECO intends to have available for the Staff's review the evaluations discussed above.

We trust you will find the information contained herein satisfactory.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

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(10) J. R. Miller letter to W. G. Counsil, dated November 11, 1983.

TABLE 1 SUMMARY OF RESULTS

CASE	RESEAT PRESSURE (PSIA)	PUMPS TRIPPED BY OPERATOR?	INTEGRATED SAFETY VALVE FLOW (lbm)	INTEGRATED SAFETY VALVE AND ATM. DUMP VALVE FLOW (lbm)
3 (Ref. 1)	1000 (0%)[b]	NO	4,500	15,695
4B[a]	940 (5%)	NO	10,280	20,774
5B[a]	871 (12%)	NO	45,202	51,714
5C[a]	871 (12%)	YES	18,294	24,495

a. Cases 4B, 5B and 5C were performed assuming a 1% downward drift in the safety valve opening setpoint pressure as allowed by Technical Specifications.

b. Percentages refer to percent below opening pressure for valve reseat.

TABLE 2 SEQUENCE OF EVENTS FOR SGTR CASES (TIME IN SECONDS)

EVENT	CASE 3	CASE 4B	CASE 5B	CASE 5C
Tube Rupture Occurs	0	0	0	0
Low Pressure Trip Condition (1728 psia)	833.2	833.2	833.2	833.2
CEAs Begin Dropping Into Core	834.1	834.1	834.1	834.1
Bypass Valves Begin Opening	836.0	836.0	836.0	836.0
Atmospheric Dump Valves Begin Opening	836.0	836.0	836.0	836.0
SG Safety Valve Lifts	836.0	836.0	836.0	836.0
Maximum SG Pressure - Time (psia)	840.0 (1006)	840.0 (1002)	840.0 (1002)	840.0 (1002)
Pressurizer Empties	840.0	840.0	840.0	840.0
Operator Trips Reactor Coolant Pumps	NA	NA	NA	849.0
SG Safety Valves Reseat	844.0	852.0	960.0	872.0
Bypass Valves Close	Cycling	Cycling	908 Reopen 998	Cycling
Atmospheric Dump Valves Close	872.0	936.0	856.0	852.0
Auxiliary Feedwater Flow Begins	1147.0	1156.0	1140.0	1144.0
Pressurizer Begins Refilling	1126.0	1106.0	1082.0	1320.0
Emergency Operator Actions Assumed to Begin	1800.0	1800.0	1800.0	1800.0