NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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Docket No. 50-336

B10981

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

References: (1) W. G. Counsil letter to J. R. Miller, dated December 12, 1983.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Thermal Shield Recovery Program

Northeast Nuclear Energy Company (NNECO) provided the NRC Staff in Reference (1) the final report detailing the thermal shield damage recovery program for Millstone Unit No. 2. At the time NNECO docketed the Reference (1) report, NNECO had not submitted its final determination with respect to the criteria delineated in 10 CFR 50.59(a)(2).

Pursuant to 10 CFR 50.59, NNECO has completed the review of the plant design change to remove the thermal shield and repair the core support barrel as described in Reference (1). It has been concluded that no unreviewed safety question results from the modifications at Millstone Unit No. 2. This is consistent with the preliminary findings reported in Reference (1).

NNECO is also providing additional information regarding the results of the fracture mechanics evaluation of the flaw tolerance for the core support barrel. This information is provided as revised Section 7.3.5 to be incorporated into the final reports provided to the Millstone Unit No. 2 Project Manager on December 12, 1983. In addition, Section 7.2.1 and Tables 7.2-2 and 7.2-3 have been revised and updated to include values not provided in previous transmittals.

Regarding the potential effects of the thermal shield removal on reactor internals vibration, the following information is provided. As discussed in Section 8.2 of Reference (1), the removal of the thermal shield from the reactor core support barrel will result in a net decrease in reactor vessel pressure drop which in turn will result in a small increase in reactor coolant system flow rate. NNECO estimates this increase in flow will be less than one percent (1%) of reactor coolant system flow rate. The increase in reactor coolant system flow rate due to this modification offsets, to a small degree, the decrease in system flow experienced as a result of plugging and sleeving steam generator tubes.

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NNECO has currently plugged over 1600 steam generator tubes (~9% of the total number of steam generator tubes) and sleeved approximately 2000 tubes. These changes have resulted in a decrease of approximately 3.6% in the reactor coolant system flow rate as measured during initial startup testing at Millstone Unit No. 2. The increase in flow expected from removing the thermal shield is approximately one percent for a net decrease of approximately 2.6% in reactor coolant flow since initial plant startup.

The core inlet flow distributions have been evaluated for plants similar in design to Millstone Unit No. 2 which do not have thermal shields. The reactor internals design, namely the flow skirt, lower support structure bottom plate and core support plate, effectively flattens the reactor coolant flow as it exits the downcomer annulus insuring a uniform core inlet flow distribution.

A review of reactor internals vibration monitoring of plants without a thermal shield whose reactor coolant systems are geometrically identical to Millstone Unit No. 2 and whose measured reactor coolant flow rates were greater than that expected at Millstone Unit No. 2 has been completed. The review did not identify any abnormal reactor internals vibration.

NNECO concludes that the removal of the thermal shield from the Millstone Unit No. 2 core support barrel is not expected to result in reactor internals vibration from the resulting change in flow characteristics, either flow rate or distribution. Furthermore, the thermal shield removal will most likely remove a suspected source of internals vibration as was discussed in Chapter 6 of Reference (1).

We trust you find this information satisfactory.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

7.2 CORE SUPPORT BARREL STRESS ANALYSIS

A comprehensive stress analysis of the core barrel in its undamaged condition with the thermal shield removed has been performed. Effects of the cracks in the core barrel are found to be local and are accounted for in Section 7.3.4 and 7.3.5. In this section, stresses for the core barrel at the lug locations for various loading conditions are provided.

7.2.1 Stress Field in Core Barrel at Lug Elevation

	σ_x	σ_y	τ_{xy}
Static	+ 407		11
Pressure		- 1,284	
Seismic	± 852		273
Thermal OD[1]	+34,000	+34,000	
ID	- 8,000	- 8,000	
OD[2]	+17,000	+17,000	
ID	-17,000	-17,000	
Plant Fluctuations	± 2,345	± 2,345	
Pump Induced Pulsation	± 1,763	± 1,233	109
LOCA	±28,000	± 5,092	

σ_x = Longitudinal Stress (PSI)
σ_y = Hoop Stress (PSI)
τ_{xy} = Shear Stress (PSI)

Notes:

1. This stress is based upon actual thermal gradient and is a peak value at the surface.

2. This stress is based upon a linearized thermal gradient

propogation would occur within the slot and thus terminate at the crack arrestor holes.

7.3.5 Fracture Mechanics Evaluation

An evaluation of flaw tolerance was conducted by Westinghouse Electric Corporation for the Millstone 2 core support barrel. The result of their calculation indicates that fatigue crack growth is limiting in determining the flaw tolerance for this structure.

In the evaluation of operational transients it became apparent that they will not govern the flaw tolerance of the core support barrel as their number of occurrences is small in comparison to other loadings. The key consideration, is the pump excitation, because the number of expected occurrences is approximately 2×10^{11} cycles in the design life. These cycles are dealt with by assuring that the range of applied stress intensity factor due to these loadings falls below the threshold for fatigue crack growth for the 304 stainless steel. Study of the literature indicates this threshold is 6 Ksi $\sqrt{\text{in}}$, which results in an allowable flaw size of 8 inches in length. This means that a flaw in this size range could exceed the threshold and therefore be subject to extensive fatigue crack growth, and therefore, is beyond the limit of allowable size.

Consideration of fatigue crack growth due to normal operational cycles will reduce the allowable below this 8 inch length requiring the determination of what size flaw could grow to 8 inches in the design life. A fatigue crack growth analysis of the region showed that a flaw 3.6 inches in length would grow to 8 inches in length in the design life due to normal operating loads.

When assessing crack extension due to normal operating cycles, the governing loading case was the thermal stress due to the temperature difference across the core support barrel. This temperature profile includes fluid temperature differences and internal heat resulting from gamma energy deposition. This analysis treated the thermal bending stresses as membrane tension which results in a very conservative estimation of fatigue crack growth.

The conservatively calculated allowable flaw size of 3.6 inches is considerably larger than the flaw detection capability as described in Section 4. For those lug areas which were not machined off, an area inaccessible for inspection exists below the lug. A potentially undetected flaw in this area would be significantly smaller than the allowable flaw size providing margin between inspection capabilities and flaws which could challenge the integrity of the structure.

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TABLE 7.2-2

Normal Operating Static Hydraulic Loads On The Core Support Barrel

Type of Load	Loading Value	Loading Condition
Axial Uplift Load	569,000 lbs.	(See Table 7.2-3 Condition No. 2
Radial Pressure Differential Across CSB Wall at Thermal Shield Lug Elevation	22.4 psi (Directed Radially Inwards)	Condition No. 1

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TABLE 7.2-3

Normal Operating Conditions

For

Calculating Hydraulic Loads

Parameter	Condition No. 1 For Maximum Loading	Condition No. 2 For Minimum Loading
Inlet Temp	500°F	548°F
System Flow Rate	422,000 gpm	377,000 gpm
Power Level	Zero Power	2700 MWT