NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
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July 18, 1985

Docket No. 50-423
A04958

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: (1) B. J. Youngblood to J. F. Opeka, Request for Additional Information, Questions 440.72 through 440.77.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
Transmittal of Responses to Questions 440.72 through 440.77

Enclosed are Northeast Nuclear Energy Company's responses to NRC Reactor Systems Branch questions concerning three-loop operation of Millstone Unit No. 3 contained in Reference (1). These responses should fully resolve the Staff's concern regarding these questions.

If there are any questions, please contact our licensing representative directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

By NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka

J. F. Opeka
Senior Vice President

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STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me J. F. Opeka who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

RESPONSES TO THE NRC (RSB) QUESTIONS
CONCERNING N-1 (3-LOOP) OPERATION

440.72 In Section 15.1.5 you state that the main steam line rupture analysis has been performed by assuming the most limiting single failure of a stuck RCCA with or without offsite power, and assuming the most limiting single failure in the engineered safety features. Identify the single failure and explain why it is the most limiting single failure.

Response:

The limiting single failure assumed in the Engineered Safety Feature systems is the failure of one ECCS train. This is the limiting single failure because it reduces ECCS flow and delays the injection of boron into the core. This allows for a higher power level after the return to criticality.

440.73 You state in Section 15.1.5.2 that the steam line rupture analysis assumes the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. Discuss the design or procedural provisions incorporated to preclude boron stratification when boron solution is injected into the inactive loop.

Response:

If safety injection (SI) is actuated during the course of N-1 Loop operation due to a postulated small break LOCA or a main steam line break, it is expected that boron in the SI flow supplied to the isolated loop cold leg will be well mixed with ambient primary coolant by the time the mixture enters the core region. The isolated loop in the N-1 configuration behaves hydraulically like the stagnant loop that was studied for Pressurized Thermal Shock (PTS) concerns. Experimental thermal fluid mixing data obtained

from the testing of a one-fifth scale model of a typical PWR cold leg and downcomer, Reference 1, clearly demonstrates that good mixing of simulated SI and ambient fluid within the model is obtained at the midplane of the model's downcomer. Further, analytical studies performed using advanced hydrodynamic codes that have been benchmarked against the one-fifth scale model test data have shown that the mixing process continues as the cooler mixture of the SI and ambient primary coolant flows into the reactor vessel lower plenum, Reference 2. These data support the conclusion that, should SI be actuated during N-1 Loop operation, the SI flow supplied to the isolated loop will become well mixed with ambient primary coolant prior to that flow entering the core region. If the SI flow mixes well with the ambient primary coolant, it follows that the boron carried with the SI will also become well mixed with the primary coolant prior to that flow entering the core region.

The experimental and analytical results noted in the preceding paragraph assumed no contribution to the mixing process due to the flow from adjacent active loops, and therefore display a minimum amount of mixing that may be expected due to SI in an isolated loop. The action of flow from the active loops, even at natural circulation flow rates, will promote and further improve the mixing of SI and ambient primary coolant prior to that flow's entry into the core region.

References

1. Report NP-2035, P. H. Rothe and M. W. Fanning, "Thermal Mixing in a Model Cold Leg and Downcomer at Low Flow Rates, Electric Power Research Institute, March, 1983.

2. Presentation to NRC, L. L. Eyler and D. S. Trent, "PTS Mixing Simulations Using the TEMPEST Code," Bethesda, Maryland, December 2, 1982.

440.74 You state in Section 15.2.7.2 that the method of analysis for loss of normal feedwater flow assumes the most limiting single failure in the auxiliary feedwater system, and that only one active steam generator would receive feedwater supply for N-1 Loop operation. It appears that operator's action would be required to divert the feedwater flow to the active SG. Discuss the design or procedural provisions incorporated to preclude feedwater being introduced inadvertently to the inactive SG and overfilling it to the main steam line.

Response:

The N-1 Loss of Normal Feedwater (LONF) analysis assumes that the inactive steam generator is isolated from the main feedwater and auxiliary feedwater systems. The one motor driven AFW pump that is assumed to start is the pump that would feed the isolated steam generator and one other steam generator. Millstone Unit No. 3 three Loop operating procedure requires isolation of main feedwater and auxiliary feedwater to the inactive steam generator. Because the one steam generator is isolated, the operating motor driven AFW pump feeds only one steam generator.

440.75 You indicate in Section 15.2.8.2 that the auxiliary feedwater system is assumed to supply 375 gpm to two intact steam generators for N-1 Loop operation, while Table 15.2.1 suggests that auxiliary feedwater would be delivered to three intact steam generators for both N and N-1 Loop operations. Please clarify these statements.

Response:

For the 3-loop (N-1) case, the auxiliary feedwater system is assumed to supply 375 gpm to two intact steam generators.

440.76 You state in Section 15.2.8.2 that the reactor returns to criticality briefly following a feedline break with offsite power available because of the cooldown caused by steam generator blowdown. It appears that the reactor would also return to criticality following a main steam line break with offsite power available. Discuss the design or procedural precautions incorporated to preclude overcooling the reactor by introducing excessive auxiliary feedwater to the steam generators.

Response:

The procedural precautions are incorporated in the EOP to preclude overcooling the reactor by introducing exessive auxiliary feedwater to the steam generator. The reactor does return to criticality following the main steam line break, (MSLB) but the peak power is less than nominal and the minimum DNBR is greater than 1.30. The MSLB cooldown ($100^{o}F$) is more severe than the Main Feedline Break cooldown, ($50^{o}F$) as can be seen in the coolant temperatures shown on Figures 15.1-16A and 15.2-15A.

440.77 Westinghouse has recently informed the staff that the rod withdrawal event during Mode 3 operations (hot shutdown), when analyzed with only one RCP operating consistent with current technical specifications for Westinghouse plants, may violate the DNB criterion and thereby not conform to GDC-10. We have also determined that the technical specifications for mode 4 operation, which allows only one RCP to be in operation may also not supported by the existing analysis. With regard to your request for approval for N-1 Loop operation, please confirm that the Millstone 3 technical specifications ensure operation of the plant in modes 3 and 4 consistent with the safety analyses which support operation in these modes, in particular with respect to the number of RCP's required to be operational. Address both the current technical specifications as well as any impact N-1 Loop operation might have.

Response:

Millstone 3 Technical Specification 3.4.1.2 is consistent with the guidance provided in NUREG-0452 Rev. 5 (Draft). Action statement 3.4.1.2A requires that the reactor system trip breakers be opened within 1 hour if less than two reactor coolant pumps are in operation. Opening the trip breakers will remove power from the control rod drive mechanisms and is sufficient to prevent a single failure induced control rod drive withdrawal event under these conditions.

The rod withdrawal event analyzed in Mode 3 assumed that two reactor coolant pumps were operating. The analysis was performed for the N Loop configuration: all loops unisolated, 2 loops with forward flow, and 2 inactive loops with reverse flow. A separate analysis was not performed for the N-1 Loop configuration: 1 isolated loop, 2 active loops with forward flow, and 1 inactive Loop with reverse flow. The N Loop analysis is bounding since the N Loop analysis core flow is less than the N-1 Loop analysis core flow.

Westinghouse does not analyze the rod withdrawal from a subcritical event in mode 4. Millstone 3 Technical Specifications (3.4.1.3) will be revised to incorporate the requirement to have two reactor coolant pumps operating in mode 4 when the trip breakers are closed.