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The Northeast Utilities System

OCT 30 1998

Docket No. 50-336
B17512

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

Millstone Nuclear Power Station, Unit No. 2
Response to The Request For Additional Information Relating to The Review of
Millstone Unit No. 2 Steam Line Break Analysis (TAC NO. MA3410)

This letter provides Northeast Nuclear Energy Company's (NNECO) response to the request for additional information relating to the review of Millstone Unit No. 2 Steam Line Break Analysis.

In a letter dated August 12 1998,[1] NNECO proposed to amend Operating License DPR-65 by incorporating changes into the Millstone Unit No. 2 Technical Specifications. The proposed changes update the list of documents, describing the analytical methods used to determine the core operating limits, specified in Technical Specification 6.9.1.8b. The plant-specific analysis used by NNECO for the Steam Line Break (SLB) utilizes the revised Siemens Power Corporation Methodology, which is currently being reviewed by the NRC staff. In a letter dated October 22, 1998,[2] the NRC requested additional information regarding the Technical Specification Amendment request which contained two questions relating to the Millstone Unit No. 2 SLB analysis. Attachment 1 provides NNECO's response to these two questions.

There are no regulatory commitments contained within this letter.

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(1) M. L. Bowling to the Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Changes to Technical Specifications, Updating List of Documents Describing the Analytical Methods Specified in Technical Specification 6.9.18b," dated August 12, 1998.

(2) D. G. McDonald to M. L. Bowling, "Request For Additional Information Regarding Technical Specification Amendment Request - Millstone Nuclear Power Station, Unit No. 2," dated October 22, 1998.

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Should you have any questions regarding this submittal, please contact Mr. Ravi G. Joshi at (860) 440-2080.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

M L Bowling

Martin L. Bowling, Jr.
Recovery Officer - Technical Services

Attachment

cc: H. J. Miller, Region I Administrator
D. G. McDonald, Jr., NRC Senior Project Manager, Millstone Unit No. 2
D. P. Beaulieu, Senior Resident Inspector, Millstone Unit No. 2
E. V. Imbro, Director, Millstone ICAVP Inspections
S. Dembek, NRC Project Manager, Millstone Unit No. 1

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Attachment 1

Millstone Nuclear Power Station, Unit No. 2
Response to The Request For Additional Information Relating to The Review of
Millstone Unit No. 2 Steam Line Break Analysis

October 1998

Request for Additional Information for Review of the
Millstone Unit No. 2 Steam Line Break Analysis

Question 1:

The values assumed in the steam line break (SLB) analysis are listed in Table 3.1 (EMF-98-036) for thermal-hydraulic parameters and in Table 3.3 for different trip setpoints and delay time for actions of various safety systems. The values are different from the current FSAR values listed in FSAR Tables 14.1.5-3 and 14.1.5-4. The break size for the limiting case is also different from the FSAR value.

Please identify all the changes to the input parameters that are important to the results of the SLB analysis and provide the technical bases to demonstrate the acceptance of each change.

Response:

Section 14.1.5 of the Millstone Unit 2 FSAR has been completely revised to reflect the new steam line break analysis. The revisions to FSAR Section 14.1.5 have been determined to constitute an Unreviewed Safety Question with respect to 10CFR50.59 and has been submitted to the Nuclear Regulatory Commission (NRC) for review in conjunction with a Technical Specification change for the Control Room Ventilation System. The revised FSAR section 14.1.5 is included in Attachment 5 of NNECO's letter dated September 28, 1998[1] to the NRC. The revised FSAR section 14.1.5 identifies the changes to the input parameters of the SLB analysis and provide the technical bases to demonstrate the acceptance of each change.

Question 2:

The results of the analysis indicate that 0.5 percent of the fuel in the core will fail during an SLB event while the FSAR SLB calculations show no fuel failure.

Please provide technical bases to show the acceptance of the proposed SLB analysis with fuel failure. In addition, provide an example calculation to show that the fuel failure will be limited to 0.5 percent of the fuel in the core due to fuel centerline melt.

Response:

The revised FSAR Section 14.1.5, which is included in Attachment 5 of NNECO's letter dated September 28, 1998 to the NRC, provides the basis for acceptance of the proposed steam line break analysis with fuel failure. A subsection has been added addressing the offsite dose and control room dose consequences.

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(1) M. L. Bowling to The Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Proposed Revision to Technical Specifications, Control Room Ventilation System," dated September 28, 1998.

The following example is provided by Siemens Power Corporation. This example provides a sample calculation for determination of the extent of fuel failure.

Subject: Example Linear Heat Generation rate (LHGR) Fuel Failure Calculation

The LHGR centerline melt limit represents the core-wide maximum allowable LHGR on a UO_2 rod to preclude centerline melt on either a UO_2 or gadolinia rod. The fuel centerline melt LHGR limit for Millstone Unit 2 is 21.0 kW/ft. The peak LHGR is calculated as follows:

$$LHGR_{peak} = LHGR_{avg} \times F_q \times F_e$$

Where:

$LHGR_{avg}$	=	Maximum post-scram core average LHGR (based on core power) from ANF-RELAP
F_q	=	Nuclear heat flux hot channel factor from a "snap-shot" steady-state XTGPWR run performed at the time of maximum post-scram reactor power
F_e	=	Engineering uncertainty factor (F_e is typically set to 1.03)

The limiting LHGR for the Main Steam Line Break (MSLB) analysis, Hot Full Power (HFP) with offsite power available, is:

$$LHGR_{peak} = 0.869 \times 27.116 \times 1.03 = 24.27 \frac{kW}{ft}$$

If the $LHGR_{peak}$ value is below the centerline melt limit, no fuel failures are predicted to occur. If the calculated $LHGR_{peak}$ is above the centerline melt limit, a fuel failure assessment is performed. This calculation indicates that there are fuel failures. To determine the number of assemblies predicted to fail, the F_q associated with the 21 kW/ft must be calculated:

$$F_q" = \frac{LHGR_{limit}}{LHGR_{avg} \times F_e}$$

$$F_q" = \frac{21.0}{(0.869 \cdot 1.03)} = 23.46$$

The steady-state XTGPWR run also calculates the maximum F_q for each quarter assembly in the core. The F_q map provided by XTGPWR is reviewed to determine the number of quarter assemblies that exceed the F_q" limit. If the peak F_q in an assembly is determined to exceed the F_q" limit, the entire quarter assembly is conservatively predicted to fail. The HFP MSLB with offsite power available case conservatively predicts that 4 quarter assemblies, or 1 full assembly, may fail. The number of failed assemblies, reported as a percentage of the total core, is 0.46% (1 failed assembly / 217 total assemblies).