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August 26, 1988

Docket No. 50-336
B13005
Re. 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Description of Cycle 10 Analysis Package

INTRODUCTION:

In a letter dated June 3, 1988,(1) Northeast Nuclear Energy Company (NNECO) submitted to the NRC Staff a proposed schedule for various reports and analysis supportive of Millstone Unit No. 2's Cycle 10 reload. The Cycle 10 startup is currently scheduled for March 1989.

In support of the proposed schedule, NNECO hereby submits a description of Cycle 10 analysis package which includes preliminary results to date and instances where we differ from generic topicals.

BACKGROUND:

Cycle 10 at Millstone Unit No. 2 will be the first cycle to use fuel designed and fabricated by Advanced Nuclear Fuels, Inc. (ANF). ANF will also be providing the plant's safety analysis beginning with Cycle 10. The safety analysis and the technical specifications for the plant will require a substantial number of modifications due to the following changes:

- The cycle will be increased from a current typical length of about 350 full power days to approximately 420 full power days.

- Cycle 10 will use gadolinium as a burnable poison. Recent cycles have not used any burnable poisons.

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(1) E. J. Mroczka letter to U.S. Nuclear Regulatory Commission, Document Control Desk, "Cycle 10 Reload License Amendment Schedule," dated June 3, 1988.

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- The fuel is designed to accommodate a much higher burnup. This will allow the loading of a small number of higher enriched fuel assemblies.

Neither the analysis nor the fuel design is finalized at this time. The information contained herein is subject to change. It is being provided to give the most up-to-date information available. The final information will be provided with the Cycle 10 license amendment request currently scheduled to be submitted no later than November 15, 1988.

FUEL DESIGN:

The generic mechanical design analysis for ANF 14 x 14 fuel assemblies was previously submitted to the NRC Staff in November 1983.(2) The Millstone Unit No. 2 fuel design is similar to the generic fuel design and will be described in detail in a letter to be submitted by September 1, 1988. The major highlights of this submittal are discussed below.

The Millstone Unit No. 2 fuel assemblies are 14 x 14 arrays containing 176 fuel rods in a cage structure of 5 guide tubes and 9 grid spacers. Both the guide tubes and the fuel rod cladding are made of a Zircaloy-4 for low neutron absorption and high corrosion resistance. Eight of the nine spacers in each assembly are made of a Zircaloy-4 structure with Inconel-718 springs. The ninth spacer, located at the bottom of the fuel assembly, is made with Inconel-718 and uses a high thermal performance spacer design which has been adapted for assembly debris resistance. The fuel assembly tie plates are stainless steel castings with Inconel holddown springs. The fuel assembly upper tie plate is mechanically locked to the guide tubes and may be easily removed to allow inspection to irradiated fuel rods as described in the Reference 2 report. The assemblies are designed for a peak assembly burnup of 52,500 MWD/MTU.

PLANNED CHANGES TO TECHNICAL SPECIFICATIONS:

There are several significant changes to the Millstone Unit No. 2 technical specifications planned for Cycle 10. The currently expected changes are summarized below:

- Linear Heat Generation Rate (Specification 3.2.1) - The maximum allowable Linear Heat Generation Rate (LHGR) is being reduced from its current value of 15.6 kw/ft to 15.1 kw/ft. However, two of the currently required uncertainty factors included in the calculation

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(2) XN-NF-09(P)(A), "Generic Mechanical Design Report - EXXON Nuclear 14 x 14 Fuel Assemblies for Combustion Engineering Reactors," dated November, 1983.

of this value will no longer be required for ANF fuel. These two uncertainty factors are the flux peak augmentation factor (an axially varying correction) and the axial fuel densification and thermal expansion uncertainty factor (a constant value of 1.01). The net effect of the reduction in the maximum LHGR and the removal of the two uncertainty factors is a small increase in the actual allowable LHGR for ANF fuel.

- Total Integrated Radial Peaking Factor, F_R^T (Specification 3.2.3) - This value is being increased from its current value of 1.537 to 1.61. A related change is to the allowable power versus F_R^T given in Technical Specification Figure 3-2-3b. These changes are possible because the previously NRC approved ANF methodology for calculating the Departure from Nucleate Boiling Ratio (DNBR) allows substantially increased margin when compared to the Westinghouse DNBR methodology that is currently being applied to Millstone Unit No. 2.

- Total Planar Radial Peaking Factor, F_{xy}^T (Specification 3.2.2) - This value is being removed from the plant's Technical Specifications. This deletion is possible because ANF's 3-D methodology does not require it.

- Moderator Temperature Coefficient (Specification 3.1.1.4) - Both the most positive and most negative Moderator Temperature Coefficients (MTCs) are being changed. The most positive allowable MTC with power less than or equal to 70% will change from 5 pcm/°F to 7 pcm/°F. The most negative allowable MTC at rated thermal power will change from -24 pcm/° to -28 pcm/°F. Both of these changes are necessitated by the increased cycle length. The most positive MTC for power greater than 70% power is not expected to change for Cycle 10.

- Shutdown Margin (Specification 3.1.1.1) - The shutdown margin required for Modes 1 through 4 is being changed from 2900 pcm to 3600 pcm. This change is necessitated by the impact of the more negative MTC on the steamline break transient. The increase in the shutdown margin can be accommodated because the change in the fuel management strategy to a Low Leakage Core results in higher calculated rod worths and hence larger shutdown margins.

SAFETY ANALYSIS CHANGES:

The above described change planned for the Millstone Unit No. 2 technical specifications require a large amount of the plant's safety analysis to be reanalyzed. The list of analyses currently slated for reanalysis includes:

SRP Event	Name
15.1.3	Increase in Steam Flow
15.1.5	Steam System Piping Failures Inside and Outside of Containment
15.2.1	Loss of External Load
15.2.4	Closure of the Main Steam Isolation Valves (MSIVs)
15.2.7	Loss of Normal Feedwater Flow
15.3.1	Loss of Forced Reactor Coolant Flow
15.3.3	Reactor Coolant Pump Rotor Seizure
15.4.1	Uncontrolled Control Rod/Bank Withdrawal from a Subcritical or Low Power Condition
15.4.2	Uncontrolled Control Rod/Bank Withdrawal at Power
15.4.3(1)	Dropped Control Rod/Bank
15.4.3(5)	Single Control Rod Withdrawal
15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant
15.4.8	Spectrum of Control Rod Ejection Accidents
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve
15.6.5	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary (Both the Large Break LOCA and the Small Break LOCA accidents require reanalysis)

The analysis of these events is still ongoing. The results of these analyses are planned to be included with the license amendment request scheduled for submittal on November 15, 1988. The only exception to this is the Small Break LOCA analysis which is scheduled for submittal on October 21, 1988.

The analysis is generally performed in accordance with the standard ANF methodology. There are two reports which have been submitted and are awaiting final NRC approval that will be referenced by the Cycle 10 analysis. These are:

Steam Line Break Methodology, XN-NF-84-93, Supplement 1

Plant Transient Methodology of CE PWRs, XN-NF-84-73, Appendix B

The Cycle 10 analyses have been performed in accordance with NRC approved or submitted ANF methodology with the following exception. ANF has not previously performed an analysis of the single MSIV closure event, 15.2.4, for either a Combustion Engineering or Westinghouse plant. Thus, ANF has not submitted a methodology for this event to the NRC. ANF intends to analyze the single MSIV closure using the RELAP5/MOD2 model developed and used in the steam line break analysis for Millstone Unit No. 2. The steam line break model will be used because it includes the split core model required to treat the asymmetric conditions produced by a single MSIV closure. The intent of the ANF analysis will be to show that this event is bounded by another event.

All of the analysis discussed above will assume an RCS flow rate of 340,000 gpm, which is the current minimum flow allowed by the plant's Technical Specifications. It is currently expected that, following completion, the analysis will be reevaluated at a lower RCS flow rate. The technical specification change requested supported by this reduced flow analysis is expected to be submitted in February 1989, if necessary to support Cycle 10 startup, as a supplement to the reload Cycle 10 license amendment change request.

<u>CYCLE 10 FUEL MANAGEMENT</u>:

The reload design for the fresh fuel to be inserted for Cycle 10 consists of 60 dual enrichment assemblies. The average enrichment for rods containing no burnable absorbers is 3.30 w/o U-235 (3.00 around the guide and instrument tubes and 3.45 w/o elsewhere). For rods containing 1.0 w/o gadolinium, the UO_2 is enriched to 2.85 w/o U-235. For rods containing 6.0 w/o gadolinium, the UO_2 is enriched to 2.10 w/o U-235. The 60 assemblies in the reload can be broken down as follows:

- 20 assemblies containing no burnable poison rods
- 8 assemblies containing 8 rods with 1.0 w/o Gd_2O_3
- 32 assemblies containing 12 rods with 6.0 w/o Gd_2O_3 and 4 rods with 1.0 w/o Gd_2I_3

The total batch burnable absorber requirement for the Cycle 10 fuel is thus 576 Gd_2O_3 rods. In addition to the fresh fuel, there are thirteen reinsert assemblies planned for use in the Cycle 10 design.

The projected Cycle 10 burnup is 13,061 MWD/MTU assuming a Cycle 9 end of cycle burnup of 9700 MWD/MTU. The Cycle 10 burnup corresponds to approximately 420 effective full power days.

Please contact us if you have any questions regarding this information.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

E. J. Mroczka
Senior Vice President

cc: W. T. Russell, Region I Administrator
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