UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 14, 1979

Docket No. 50-336

Mr. W. G. Counsil, Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101

Dear Mr. Counsil:

In the process of reviewing your Cycle 3 reload/stretch power request dated February 12, 1979, we find that additional information as detailed in the enclosure is needed to complete our review. The additional information being requested was previously sent to Mr. R. Kacich of your staff by telecopy on March 2, 1979.

In order to meet the agreed upon schedule for this review, please provide the additional information by at least March 28, 1979.

Sincerely,

R. W. Reid

R. W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure: As stated

cc: See next page

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REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

CYCLE 3 RELOAD/STRETCH POWER APPLICATION

1. In Section 4.1, it is stated that no clad collapse is expected in cycle 3. What will the maximum exposure of the B assemblies be during cycle 3 and at what exposure is clad collapse predicted?

2. In section 3 it is stated that the cycle 3 core will be 90° rotationally symmetric. If Figures 5-3, 5-4, 5-5 and 5-6 represent a 90° rotationally symmetric core then one would expect the powers in the bottom row of assemblies to be the same as those in the right hand column of assemblies. Please explain why this is not the case.

3. In Section 5.2, you state that at this time ROCS is accepted for scoping calculations, but not for safety calculations. It is not clear from section 5.2.3.1 which calculations are performed with ROCS and which calculations are performed using fine mesh 2D PDQ. Please supply a list of all (if any) safety related calculations that were performed with ROCS.

   In answering this question, please address the following concern. This is a concern for many situations, but for concreteness we will discuss the dropped CEA event. In our conception, ROCS would be used to determine which dropped CEA would produce the most adverse effect. Having determined the proper CEA, the dropped rod safety analysis would be performed using fine mesh PDQ. Our concern is that if the whole analysis were done using PDQ, a different CEA might have been chosen which would produce more adverse effects than the CEA chosen by the ROCS analysis. This is a serious concern if the ROCS analysis shows that several different CEA's would produce effects almost as adverse as the worst CEA. There is very little concern if the ROCS analysis predicts one CEA to be far and away the worst CEA to drop. Thus what we would like for each such application of ROCS is a statement of your confidence in the ROCS scoping calculation and the reason for your confidence.

4. In Section 7.1.6, Table 7.1.6.1 states that the Steam Dump and Bypass System and the Pressurizer Relief Valves are assumed to be Inoperative. However, in the text these systems are assumed Operative. Which is correct?

5. In Section 7.1.8, there is only one page of discussion here and no tables or figures. Have some pages been left out?

   In Table 7.2.1-1, the cycle 2 Doppler Multiplier is 1.15 and the cycle 3 Doppler Multiplier is 1.00. Please justify this change.

7. In Section 7.2.5, it is stated that Loss of Load is the limiting transient, and thus only this transient is reported.

   Were all four events analyzed and only this one event reported, or was only the Loss of Load event analyzed? If the Loss of Load was the only transient analyzed, please give the justification for assuming that this is the limiting transient.

8. In Figure 7.3.2-9 the core power is about 1/2% for approximately 10 seconds, and zero otherwise. In Figure 7.3.2-10 the core heat flux is about 2% for approximately 2 minutes. Thus the integrated heat flux is much greater than the integrated core power. Is this difference due simply to the internal heat of the fuel extracted during the cooldown, or are there other sources of heat?

9. For Section 7.3.3, what is the minimum DNBR reached?

10. In Section 7.3.4, it is stated that the minimum DNBR is less than 1.19, but the minimum value is not stated. Please state the minimum DNBR in the transient. If available, please state the minimum DNBR for the worst 1% of fuel pins, worst 2% of fuel pins etc. Also if available please provide a graph of DNBR vs time.

11. For Technical Specification (TS), Figure 2.1-1, it is our understanding that the family of Thermal Margin Limit Lines such as Figure 2.1-1 are used in computing the TM-LP setpoints. Is this single figure included in the TS simply as an example?

12. For TS Figure 2.1-1, it is our understanding that with the current CE TM-LP methodology, the curves in Figure 2.1-1 would in general not be straight lines. Please explain why they are straight lines in the BG&E request.

13. Please provide a description of the Feedwater Malfunction Event in Section 7.1.7. This description should include the following:

    a. Cause of the malfunction
    b. Nature of the malfunction (Is the main feedwater only affected, or is the auxiliary feedwater affected as well?)
    c. Sequence of events
    d. Case analyzed should correspond to worst time in life and worst initial conditions. Plant parameter values should be listed. This list should include the power mismatch caused by the feedwater malfunction.

e. The parameters in (d) above should be compared with those for the reference analysis Excess Load Event which is cited as being more limiting than the Feedwater Malfunction Event.

14. In your startup test program, Section 9, you state a plan of action if a measured parameter differs from the predicted value by more than the acceptable criteria. However, it is not clear what the state of the plant is during this time. For example, if the test is a low power tests, would the plant be kept below 5% power until action was completed or would the plant be allowed to escalate in power? If the answer varies for different tests or conditions, please explain the variations.

15. The purpose of the startup test program is to provide assurance that the core conforms to the design. The means by which this is done are at the discretion of the licensee, but these means must be technically justifiable. One possible approach would be to divide the test criteria into two categories, review and acceptance. Review criteria would be sufficiently narrow as to highlight any deviation which may indicate that the core is incorrectly loaded and that the assumptions made in the safety analysis are not valid. Procedures to be followed if review criteria are not met should not be keyed to shutting down the plant but to indicate further review or analysis to assure safe operation for the length of the cycle. The broader acceptance criteria would be keyed to assuring that the response of the plant to accidents and transients is in accordance with design.

    Please provide review criteria for the power distribution verification tests. The stated acceptance criteria are adequate.