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

P.O. BOX 270
HARTFORD, CONNECTICUT 06101
(203) 666-6911

March 27, 1979

Docket No. 50-336

Director of Nuclear Reactor Regulation
Attn: Mr. R. Reid, Chief
Operating Reactors Branch #4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) R. Reid letter to W. G. Counsil dated March 14, 1979.
(2) W. G. Counsil letter to R. Reid dated January 27, 1979.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Additional Information on Reload Application

In Reference (1), Northeast Nuclear Energy Company (NNECO) was requested to respond to NRC Staff questions concerning the application addressing Cycle 3 operation. Accordingly, Attachment 1 is provided to address each of the items detailed in Reference (1). It is noted that draft responses have been telecopied to members of your Staff to expedite your review.

Concerning the requirements of 10CFR170, NNECO has determined that no additional fee is required as described in Reference (2).

We trust this information is responsive to your request.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Vice President

Attachment

7904090 178

DOCKET NO. 50-336

ATTACHMENT 1

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

ADDITIONAL INFORMATION ON RELOAD APPLICATION

MARCH, 1979

MILLSTONE UNIT NO. 2 CYCLE 3 RELOAD
RESPONSES TO FIRST-ROUND QUESTIONS

Question 1.1 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?

Response: The following table summarizes clad collapse data for Batches B, C, D, and E fuel.

Batch	Cycles In-Core	Actual or Expected Operation - EFPH	Predicted Time to Clad Collapse - EFPH
B	1, 2, 3	26790	> 30000
C	1, 2, 3	26790	> 30000
D	2, 3, 4	24050	> 29000
E	3, 4, 5	25230	> 28600

Question 1.2 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.

Response: Figures 5-3 through 5-6 were generated using the fine mesh PDQ program in a quarter-core geometry calculated with reflective boundary conditions. In these quarter-core calculations, the axis locations contain only half assemblies and the corresponding material inventories that are associated with each half. This was done so that a full core loading map generated from a quarter-core map has the correct isotopic inventories in each assembly. Since the inventory in a half assembly located along the bottom row may be different than the remaining half of the same assembly located along the right-hand column, the resulting power distributions may also differ. This "assembly splitting" technique has been shown to be conservative (i.e., yielding higher peaking factors) when compared to power distributions calculated in full core geometry. One-pin peaking factors calculated with the assembly splitting technique are more realistic than those obtained with assembly averaging because the averaging method does not account for large fissile gradients across the assembly and may result in slightly lower one-pin peaking factors.

Question 1.3 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.

Response: The following parameters were calculated with the ROCS computer code:

- Fuel Temperature Coefficients
- Moderator Temperature Coefficients
- Inverse Boron Worths
- Critical Boron Concentrations
- CEA drop distortion factors and reactivity worths
- Reactivity Scram Worths and Allowances
- Reactivity worth of regulating CEA banks
- Changes in 3-D core power distributions that result from inlet temperatures maldistributions (asymmetric steam generator transient)

None of these parameters require detailed knowledge of pin peaking factors and in most cases are calculated more accurately by ROCS because of its ability to account for 3-D effects.

The usual C-E practice in reload safety analyses is to select reference input physics parameters which "envelope" the values expected for present and future core loading patterns. The physics analysis task then becomes one of demonstrating that the reference values are conservative for a specific cycle. Depending upon the proximity of a value calculated by ROCS to that used in the safety analysis a determination is made whether to perform "backup" calculations with PDQ. A good illustration of how this determination is made is seen in the method used for selecting the worst CEA drop parameters.

The most severe CEA drop is the one which produces the maximum power distortion and the minimum reactivity insertion (and hence moderator temperature reduction).

The general method for determining the worst CEA drop is described as follows:

1. A set of conservative "pre-drop" power distributions are selected which span the operating conditions expected during the cycle for the range of previous cycle endpoints chosen. This includes rodded and unrodded distributions at the beginning and end of the cycle.

2. The candidates for the worst CEA drop are selected. A substantial number of candidates - especially single CEA's - are eliminated a priori on the basis of judgment and calculations performed for over 14 different operating cycles of the 217 assembly C-E core.

3. The CEA drop event is then calculated with ROCS for all of the power distributions and all of the CEA's identified above.

4. The worst distortion factor and worth are compared to the values which have been or which will be used in the transient safety analysis.

5. If there is a question whether the uncertainty in the ROCS calculation could cause the ROCS results to be worse than the safety envelopes, the PDQ calculations are performed as necessary to resolve the question.

The results of the specific ROCS calculations performed for the CEA drop are summarized in the table below. The location of bank 7 and the type 10 and 11 CEA's are shown in figure 5-2 of the submittal.

Millstone II Cycle 3 CEA Drop Survey
ROCS Power Distortion Factor and Worth (% increase/worth %Δρ)

Time in Cycle 3	Cycle 2 Termination	Unrodded Core Type 10	Unrodded Core Type 11	Bank 7 inserted Type 10	Bank 7 inserted Type 11
BOC	Early	12.3/.142	12.7/.108	13.2/.178	11.3/.118
BOC	Late	10.8/.132	13.1/.114	11.8/.169	11.9/.125
EOC	Late	12.5/.164	13.2/.130	13.3/.190	11.5/.126

The most severe distortion factor shown here is 13.3%. However, this distortion occurs for a CEA drop in which the worth is substantially higher than for other CEA's. Past analyses indicate that a .04%$\Delta\rho$ worth increase is equivalent to a 1% distortion factor increase. Use of this relationship yields the conclusion that the most severe case is the drop of the type 11 at BOC into the unrodded core with late Cycle 2 shutdown. This case is underlined in the table above.

An envelope distortion factor of 16% and an envelope worth of .08%$\Delta\rho$ were used in the transient safety analysis. ROCS generally predicts CEA drop distortion factors within 2% and CEA worths within .02%$\Delta\rho$ of PDQ. If ROCS underpredicts a distortion factor relative to PDQ, it generally also underpredicts the worth.

Because of the above observation and the margin between the worst case ROCS results and the conservative envelope values of distortion factor and worth used in the safety analysis, only one PDQ calculation was performed. The PDQ case chosen was for the type 11 dual CEA inserted into the unrodded core at BOC (late cycle 2 shutdown). This PDQ calculation showed a distortion of 14.8% and a worth of .126%$\Delta\rho$. Although the PDQ distortion factor is higher than ROCS (14.8% vs. 13.1%), the worth is also higher (.126%$\Delta\rho$ vs .114%). Using the relationship discussed above relating distortion factor and worth, normalizing the PDQ worth to the ROCS worth yields an adjusted PDQ distortion factor of 14.4%. The difference between the ROCS factor and the adjusted PDQ factor is 1.3%, which is within the two percent variation expected between PDQ and ROCS calculations. Moreover, the PDQ results are well within the safety envelopes.

Because of this agreement between PDQ and ROCS and C-E's general experience with this type of calculation, it was judged that no other CEA drop would have parameters more severe than the 16% distortion factor and .08%$\Delta\rho$ used in the transient safety analysis. In particular, our experience has shown that if any other dropped CEA is worse than the type 11, it is worse by an amount which is small when compared to the reserve margin associated with the conservative safety envelopes.

Question 1.4

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?

Response

Two cases were analyzed for the Loss of Feedwater Flow Event. The first case was to determine what combination of parameters and control system operation modes would maximize the RCS pressure and cause the greatest decrease in DNBR. For this case, it is conservative to assume that the Steam Dump and Bypass System and the Pressurizer Relief Valves are inoperable. Tables 7.1.6-1 and 7.1.6-2 pertain to this case. A second case analyzed was to ensure that at least ten minutes exist before the operator must initiate auxiliary feedwater. For this case, it is conservatively assumed that the Pressurizer Relief Valves and Steam Dump and Bypass System are operable, since this would maximize the steam flow out of the steam generators and thus decrease the water inventory in the steam generator.

For clarity, the text in section 7.1.6 should be changed to read:

The initial conditions listed in table 7.1.6-1, with the exception of the Steam Dump and Bypass System and the Pressurizer Relief Valves, were used to analyze the event to ensure that at least ten minutes exist before the operator must initiate auxiliary feedwater.

Question 1.5

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

Response

Section 7.1.8 conforms with the format established in previous reload license submittals. No pages have been left out. The RCS Depressurization Event was analyzed to determine a bias term input to the TM/LP trip. The action of the RPS prevents SAFDL's on DNBR and linear heat rate from being exceeded. The bias term for the RCS Depressurization Event was compared to the bias term from the CEA Withdrawal Event and found to be less limiting. The analysis assumptions and key parameters were selected to maximize the rate of RCS pressure drop, since this leads to the most conservative bias term for the event.

Question 1.6

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.

Response

Analyses have shown that the Doppler multiplier has no significant impact on the Loss of Flow Analysis. The average fuel temperature change is 3-5°F from the time the event is initiated to the time of minimum DNBR. Within this time span, there is first a power and fuel temperature rise and then a decrease. Therefore, a Doppler multiplier of 0.85 would be conservative during the period of fuel temperature increase, while a Doppler multiplier of 1.15 would be conservative during the period of fuel temperature decrease. Under these conditions a Doppler multiplier of 1.00 would more closely represent the average transient conditions. Neverless, the overall effect of the Doppler defect contribution is third order, and for this reason it was reduced from 1.15 to 1.00 in the Cycle 3 analysis.

Question 1.7

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.

Response

In Section 8.2 of CENPD-199, "C-E Setpoint Methodology", the events in the category of Malfunctions to One Steam Generator were discussed in detail;

the Loss of Load was demonstrated to be by far the most limiting in terms of approaching SAFDL's and margin requirements. Consequently, only the loss of load to one steam generator event, was analyzed for the reload licensing submittal.

Question 1.8

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?

Response

Yes, the difference between the integrated heat flux and the integrated core power is due only to the internal heat of the fuel extracted during the cooldown. There are no other sources of heat.

Question 1.9

For Section 7.3.3, what is the minimum DNBR reached?

Response

The main purpose of the Steam Generator Tube Rupture analysis is to determine the maximum site boundary dose. This is achieved by delaying reactor trip on low pressurizer pressure as long as possible in order to maximize the concentration of radioactivity in the steam generators. For the steam generator tube rupture event analyzed, the RCS experiences a slow pressure drop; and therefore the Thermal Margin/Low Pressure trip actuates a reactor trip to prevent exceeding SAFDL's. The TM/LP trip will occur earlier than the low pressurizer pressure trip condition, and insures a minimum DNBR greater than 1.19. Therefore, the DNBR calculation is not performed, because it is the trip system which prevents the DNBR from dropping below 1.19.

Question 1.10

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.

Response

The minimum CE-1 DNBR for the Seized Rotor is 0.97. The method for predicting fuel pin failure is discussed in detail in CENPD-183, "C-E Methods for Loss of Flow Analysis".

Question

1.11 For Tech. Spec. Figure 2.1-1, it is our understanding that a 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 Tech. Spec. simply as an example?

1.12 For Tech. Spec. 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.

Answers to NRC Questions 11 & 12

Both questions 1.11 and 1.12 are addressed to Figure 2.1-1: Thermal Margin Safety Limit Curves. Figure 2.1-1 shows the loci of points of thermal power, reactor coolant system pressure and maximum cold leg temperature with four reactor coolant pumps operating for which the minimum DNBR is no less than 1.19 with the family of axial shapes and corresponding radial peaks shown in Figure B2.1-1. These loci are an example valid only for the power distributions shown in Figure B2.1-1, they bear no relationship to the TM-LP limits given by Figures 2.2-3 and 2.2-4.

It is the TM-LP limits in combination with the Limiting Conditions for Operation which assure that the Anticipated Operational Occurrences listed in Table 7-1 of the submittal do not result in a DNBR less than 1.19.

The TM-LP limits given by Figures 2.2-3 and 2.2-4 in general do not result in straight lines of temperature vs. power because they are synthesized using the radial peak vs. power variation assured by the PDIL (specification 3.1.3.6) and the variable high power level trip (Table 2.2-1).[1] Since this radial peak vs. power variation is non-linear the TM-LP limits are, in general, also non-linear.

Figure 2.2-1 shows straight lines because the radial peaks given in Figure B2.1-1 are assumed not to vary with power.

---

[1] Reference CENPD-199-P

Question 1.13

In Section 7.1.7, please provide a description of the Feedwater Malfunction Event. 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 parameters 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.

Response

The Feedwater Malfunction Event causes excess cooling of the reactor coolant system and the steam generators due to a decrease in feedwater temperature or an increase in feedwater flow. A decrease in feedwater temperature may be caused by the loss of one of the feedwater heaters or accidental starting of the auxiliary feedwater system. An increase in feedwater flow may be caused by complete opening of a feedwater control valve, overspeed of the feedwater pumps with feedwater valve control in manual, or inadvertent start of the second feedwater pump at low power. No Sequence of Events or table of plant parameters was presented for this DBE, since it was not necessary to reanalyze this event. In response to this question, the attached table lists the parameters and corresponding initial conditions assumed for the reference cycle Excess Load analysis and those used to conclude that the Feedwater Malfunction Event is less severe. Since operating at stretch power will reduce the potential mismatch between steam demand and the cooldown associated with the maximum possible feedwater supplied to the steam generators for the full power case, the statement in the FSAR that the Excess Load Event is more adverse than Excess Heat Removal due to Feedwater Malfunction Event is still valid for Cycle 3 at 2700 MWt as stated in the license submittal.

COMPARISON OF KEY PARAMETERS ASSUMED AS INITIAL CONDITIONS FOR EXCESS LOAD AND THE FEEDWATER MALFUNCTION EVENTS.

PARAMETER	UNITS	FSAR (EXCESS LOAD)	CYCLE 3 (FEEDWATER MALFUNCTION)
Initial Core Power Level	MWt	0-102% of 2560	0-102% of 2700
Core Inlet Coolant Temperature	^{o}F	544	551
Core Coolant Flow	$x10^{6}$ lb/hr	122	133.7
Reactor Coolant System Pressure	PSIA	2250	2200
Moderator Temperature Coefficient	$x10^{-4} \frac{\Delta\rho}{^{o}F}$	-2.5	-2.5
Doppler Coefficient Multiplier		0.85	0.85
Steam Generator Pressure	PSIA	844	860

Question 1.14

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 test, 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.

Response

If the acceptance criteria are exceeded during startup testing, the plant power will be held constant while the plan of action for resolving discrepancies between measured and predicted parameters is implemented. The only exception to this is if Technical Specifications dictate a power reduction. Two examples are (1) if a CEA bank worth measurement exceeds the acceptance criteria during low power tests, power will be held below 5% until the discrepancy is resolved and (2) if the F_R^I limit is exceeded at full power, power will be reduced in accordance with Technical Specifications.

Question 1.15

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.

Response

The review criteria for power distribution verification tests is $\pm$ 10%. This criteria allows confirmation of core design and allows for any differences between measured and predicted power distributions caused by differences in power history, burnup, CEA position, xenon, and samarium concentrations. Should the 10% criteria be exceeded, the situation will be reviewed for acceptability in view of plant characteristics at the time of these measurements.