NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO: Mr. G. Lear	FROM: Niagara Mohawk Power Corp Syracuse, N.Y. 13202 Gerald K. Rhode	DOCKET NUMBER 50-220 FILE NUMBER DATE OF DOCUMENT 3-14-77 DATE RECEIVED 3-21-77
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DESCRIPTION Ltr trans the following: 1P

ENCLOSURE Addl info concerning the reload application for Cycle 5 operation.....13P

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ACKNOWLEDGED

PLANT NAME: Nine Mile Pt Unit 1

DHL

FOR ACTION/INFORMATION

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NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

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REGULATORY DOCKET FILE COPY

RECEIVED MAR 21 1977 U.S. NUCLEAR REGULATORY COMMISSION

March 14, 1977

Director of Nuclear Reactor Regulation
Attn: Mr. George Lear, Chief
Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220

Dear Mr. Lear:

Your letter dated March 4, 1977 requested additional information concerning the reload application for Cycle 5 operation of Nine Mile Point Unit 1. The enclosed information addresses itself to the attachment to your letter.

Sincerely,

NIAGARA MOHAWK POWER CORPORATION

GERALD K. RHODE
Vice President - Engineering

RECEIVED MAR 21 1977 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

NLR/sz

Enclosure

7708220340

RESPONSES TO MARCH 4, 1977

NUCLEAR REGULATORY COMMISSION QUESTIONS

Nine Mile Point Unit 2
Docket.No. 50-220
DPR-63

1. Question

The Doppler coefficient used in previous NMP-1 transient analyses contained a conservative multiplier of 0.90 (See October 3, 1975 letter to G. Lear, NRC, from G.K. Rhode, NMPC, Question #2). Your present (Reload 6) topical report references NEDO-20965 ("Generation of Void and Doppler Reactivity Feedback for Application to BWR Design") for a discussion of conservatism factors. Table 2.3-2 in NEDO-20965 shows "Doppler used in PTA of a typical BWR at EOC". That table shows a 0.90 factor; we note that Table 4.1-2 in that report shows a factor of 0.95, but this factor has not been accepted.

In view of the above, our position is that the 0.95 conservatism factor used on the nominal Doppler coefficient in your analyses for Reload 6 is not acceptable.

Show quantitatively how much the lower (0.9 x nominal) coefficient would affect all significant safety related values which are calculated such as ΔMCPR, LHGR, maximum cal/g, etc. The transients and accidents considered must include rod drop, rod withdrawal error, turbine trip without bypass, and any other events significantly affected.

Response

The interpretation of the use of Tables 2.3-2 and 4.1-2 in NEDO-20964 is incorrect. Table 2.3-2 gives examples of the Doppler weighting factor including both safety and design margins. Table 4.1-2 shows the minimum safety conservatisms which are considered appropriate for transient licensing basis. These uncertainties are considered to account for "biases and uncertainties in the derivation of the nuclear data and its application to the transient model" and does not include any design margin.

In any case, the turbine trip without bypass transient is not the limiting transient with respect to ΔMCPR. Engineering evaluation using the 0.95 conservative factor instead of the 0.90 value shows a change of no more than 1 psi in steamline pressure and less than 0.01 ΔMCPR.

Additionally, the LHGR and maximum cal/gm are not affected since the nominal Doppler coefficient without conservative multipliers is used for the rod withdrawal error (RWE) and rod drop accident (RDA) analyses.

2. <u>Question</u>

You performed turbine-trip-w/o-bypass "maximum pressure analysis" at EOC, EOC minus 1000 Mwd/t, and EOC minus 2000 Mwd/t. When you reach EOC minus 2000 Mwd/t, are you proposing (in Section 6.3.3.2.2, "Power Level Profile") an immediate orderly decrease to the power level (94%) shown to be acceptable at EOC minus 1000 Mwd/t, or are you proposing a slow power decrease reaching 94% at approximately EOC minus 1000 Mwd/t, etc? Please clarify Section 6.3.3.2.2 and justify the conservatism of your proposed operation with respect to the analyses performed.

<u>Response</u>

Neither an immediate reduction in power at 2000 Mwd/st before end of cycle nor a slower reduction in power, reaching 94% at approximately 1000 Mwd/st before end of cycle, will be utilized. The term "coastdown" is used to define a method for power reductions. When the appropriate exposure point is reached (in this case 2000 Mwd/st before EOC) the control rods are "locked" (no longer pulled) until the reactor coasts down to the next lower calculated power level (in this case 94%). The new power level is then maintained by necessary rod motion until the next exposure point for further power reduction is reached (in this case 1000 Mwd/st before end of cycle). The process is then repeated to attain the recommended 92% power level for the last power reduction. This method is conservative since required rod inventories will be maintained. Also, the coastdown method as proposed is conservative compared to a reasonable extrapolation of analyzed exposure values.

3. Question

Provide the ΔMCPR resulting from loss of the maximum amount of feedwater preheat that can result from a single failure or operator error.

Response

The resulting ΔCPR from the loss of the maximum amount of feedwater preheat is less than 0.2. Since the rod withdrawal error is the limiting transient with ΔCPR's for 7 x 7 and 8 x 8 fuels (0.30 and 0.32 respectively), the result for loss of feedwater heater will have no effect on the current operating limits.

4. Question

Were all points on the scram-delta-K curves shown in Figure 6-8 multiplied by 0.8 for use in the transient analyses, or was the 0.8 factor used only to limit the maximum fully inserted value as shown on Table 6-1? All transient and accident analyses must be performed with an appropriate conservatism factor applied to all points on the scram-delta-K curve, not just to the maximum fully inserted value.

Response

All points on the scram-delta-K curve shown in Figure 6-8 were multiplied by the conservative factor of 0.8.

5. Question

Please specify the exact fuel loading error (fuel type, position moved from and to, etc.) that results in a 16.5 Kw/ft peak LHGR and a MCPR of 1.11 as described in Section 6.3.2.5.2. Discuss other potential loading errors that were analysed, and justify that the one reported represents the worst case.

Response

The potential fuel loading error that results in the reported 16.4 Kw/ft. peak LHGR and a MCPR of 1.11, is described in Section 6.3.2.5.2. This involved an exchange in position of a reload-4 8D250 bundle at an average exposure of approximately 12300 Mwd/st at location (25,28), with a fresh reload-6 8D274L bundle at core location (07,38).

While a number of other potential fuel loading errors were analyzed, some of the more severe are as follows:

SUMMARY OF ADDITIONAL FUEL LOADING ERROR & ANALYSES
FOR NINE MILE POINT 1, CYCLE 5

Bundles and Positions Exchanged Reactor Coordinates	Respective Bundle Average Exposure, Mwd/st		Percent Bundle Power Increase*
R-4, 8D250 at (25,28) and R-6 8D274L at (07,38)	12300	0	22.1
R-5, 8D250 at (49,28) and R-6 8D274L at (07,38)	7100	0	16.2
R-5, 8D250 at (17,28) and R-6 8D274L at (07,38)	7300	0	15.1
R-3, 7D250 at (33,28) and R-6 8D274L at (07,38)	14600	0	13.7
R-5, 8D250 at (31,28) and R-6 8D274L at (07,38)	5700	0	12.5
R-5, 8D250 at (41,28) and R-6 8D274L at (07,38)	5400	0	11.5
R-3, 7D250 at (23,28) and R-6 8D274L at (07,38)	15000	0	10.7

As can be seen from the above, the reported fuel loading error produced the largest bundle power increase, and therefore represents the worst case.

*Bundle Power Increase is $\frac{\text{Actual Power in Mislocated Bundle}}{\text{Power in Monitored Bundle in Mirror Symmetric Location}}$

6. Question

Please provide corrected versions of Figures 6-12 and 6-13. The sequence numbers of the plotted curves are not identified, and the Rod Block Line (identified as being at 105%) is plotted at 102.5%.

Response

The "Rod Block 105%" line was incorrectly drawn on Figures 6-12 and 6-13. The corrected figures are attached.

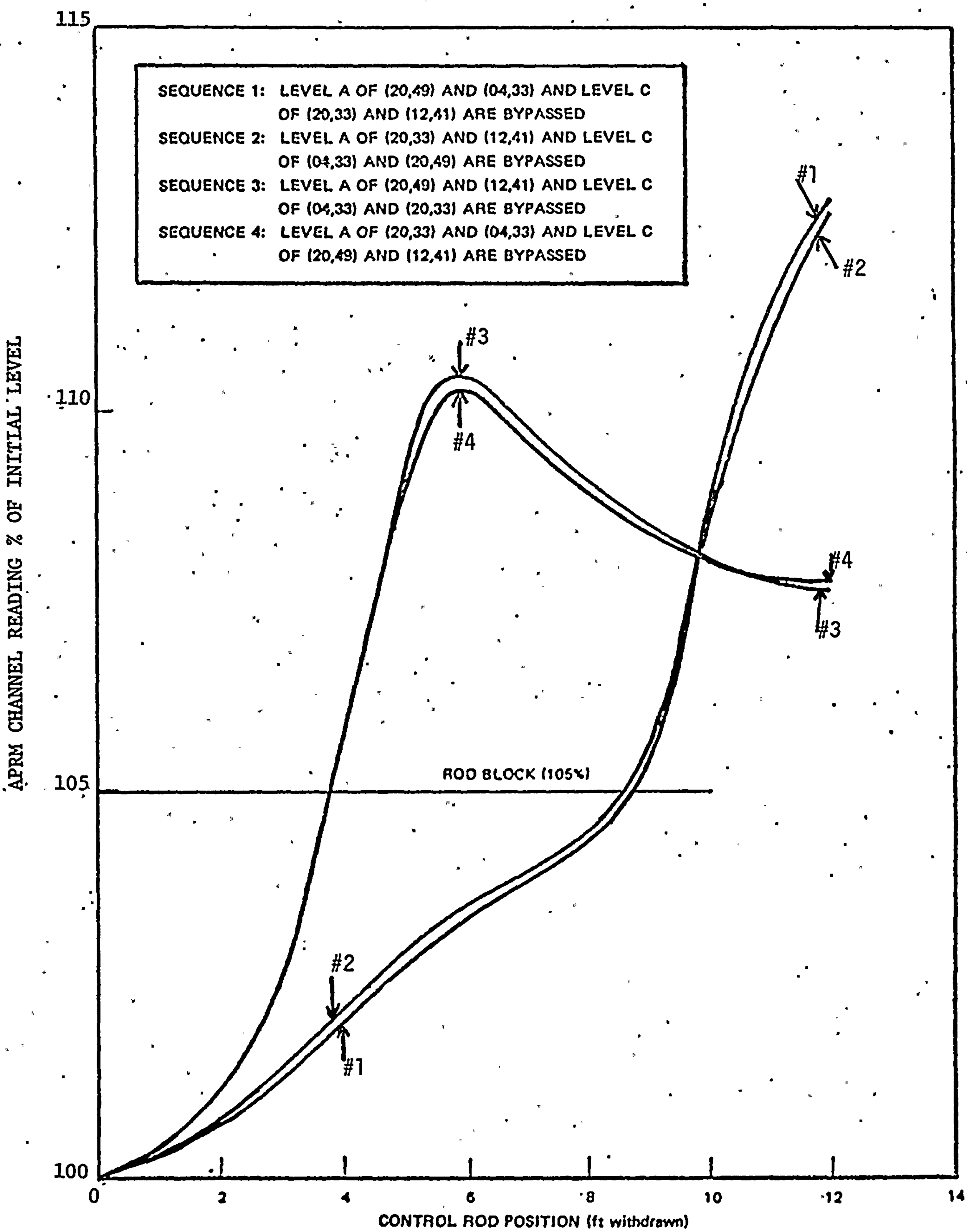

Figure 6-12. Nine Mile Point-1 Rod Block Response to Control Rod Motion for Rod Withdrawal Error - Limiting Case, Channel A+C for Reload-6

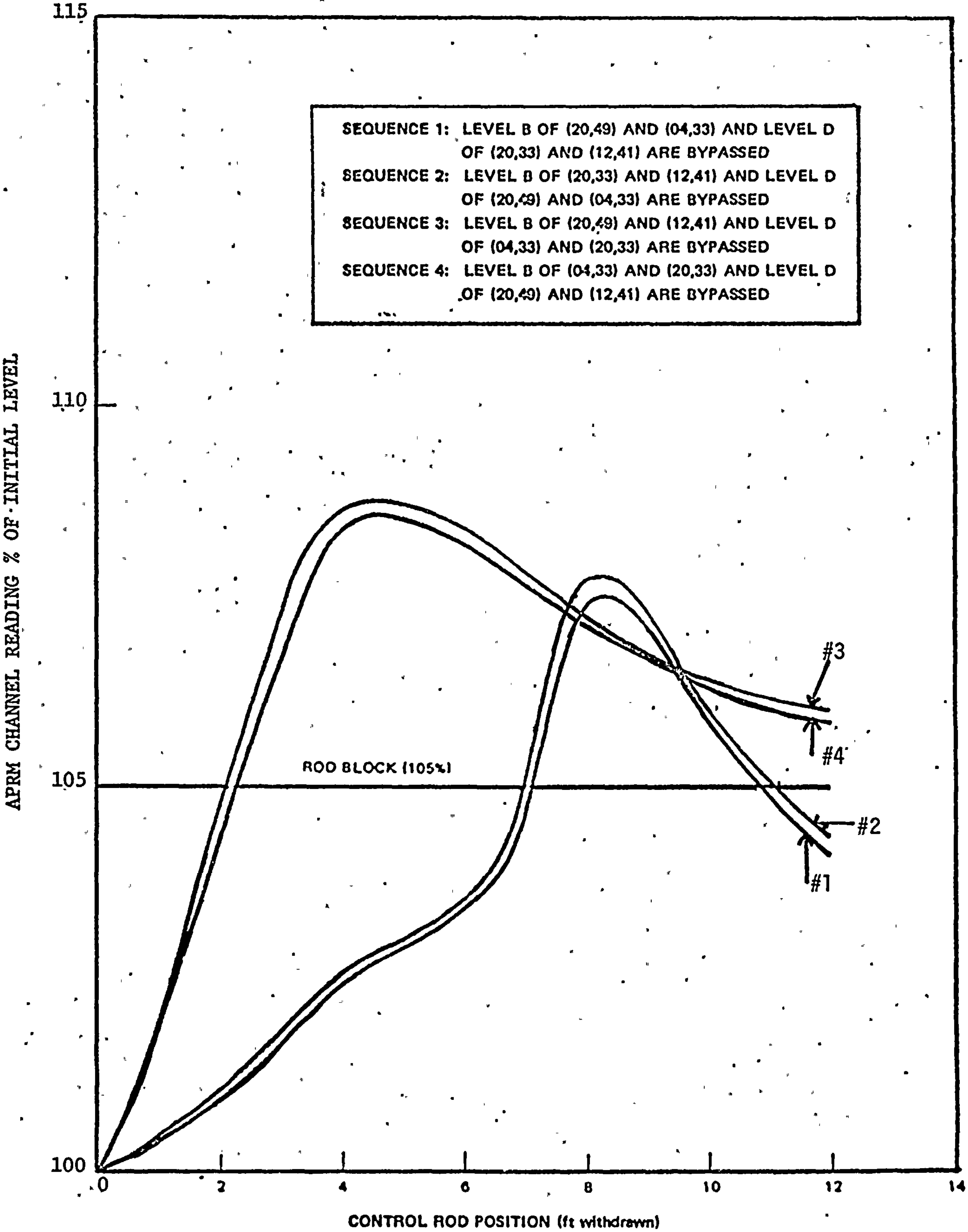

Figure 6-13. Nine Mile Point-1 Rod Block Response to Control Rod Motion for Rod Withdrawal Error - Limiting Case, Channel B+D for Reload-6

7. Question

Operational Assurance and Tests

A description of the actual reload fuel and irradiated fuel placement (location and orientation), if different from that planned according to Section 2.1.2, should be presented. It should identify loading sequence, verification techniques for placement, and any pertinent shutdown margin or verification tests performed during actual reloading. Startup physics tests selected from Reg. Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," should be performed. A test abstract summarizing the test objective, test method, and acceptance criteria should be presented. To validate the analytical models utilized for predicting plant responses to anticipated transients and postulated accidents, these tests should establish that measured responses are in accordance with predicted responses. The predicted responses should be developed using real or expected values of items such as beginning-of-cycle core reactivity coefficients, flow rates, pressures, temperatures, and the actual status of the plant and not those values or plant conditions assumed for conservative evaluations of postulated accidents. Acceptance criteria that are proposed should assure that the response of the plant to accidents and transients is in accordance with the design. Procedures to be followed if the acceptance criteria are exceeded should be discussed.

Recommended parameters to be tested should include, but not necessarily be limited to, the following:

a. Control Rod Drive Tests and Scram Time (Cold and Hot).

b. Verification of Shutdown Margin (Highest Worth Control Rod Withdrawn).

c. Comparison of Cold Critical Eigenvalue Calculation for a Fixed Control Rod Pattern (deviation of a percent or more in reactivity should be immediately reported and explained).

d. Power Distribution Comparison at a Given Control Rod Pattern and Power Level (>50% rated power with equilibrium xenon - anomalies should be reported and explained).

e. TIP Reproducibility Test (>75% rated power).

f. Core Power Symmetry Test (>75% rated power).

g. Instrumentation Calibration (LPRMs and APRMs).

7. Response

A description of the actual fuel placement, if different from that provided in NEDO 21466, will be provided prior to startup. Loading sequence and verification techniques will also be provided prior to startup.

Upon review of Reg. Guide 1.68, we conclude that the test program described is not applicable to reload 6. It is our understanding that Reg. Guide 1.68 applies to pre-operational and initial startup exclusively. Therefore, we believe certain tests are not justified for the reload 6 core.

In accordance with our Technical Specification, Niagara Mohawk will perform certain startup physic tests which include control rod drive scram tests (hot), shutdown margin tests, and instrumentation calibrations.

Additionally, Niagara Mohawk will compare a cold critical prediction and a power distribution calculation above 50% power with actual measurements.

Niagara Mohawk does not plan to perform control rod drive scram tests at cold conditions, TIP reproducibility tests or core power symmetry tests. It is believed that the results to be derived from these tests would not be of significant benefit.

8. Question

We request that you submit a summary report of the physics startup tests to NRC within 90 days following completion of the startup test program. This report should include both measured and predicted values. If the difference between the measured and predicted value exceeded the acceptance criterion, the report should discuss the actions that were taken and justify the adequacy of these actions. Appendix A presents an outline which may be used for the physics startup tests summary report.

Response

As stated in the Technical Specification, we are required to submit to NRC a summary report of plant startup and power escalation testing following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that have significantly altered the nuclear, thermal, or hydraulic performance of the plant. Since none of the above conditions apply to reload 6, it is concluded that there is no Technical Specification requirement to submit a summary startup test report.

However, in response to your direct request for this information, a summary report for the limited startup tests, as detailed in response 7, will be provided 90 days following completion of the test program.

The summary report will include test abstract, comparisons of measured and predicted responses and justification for deviation from acceptance criteria, if applicable.

9. Question

All tests that result in recalibration, new baseline settings, or other set point identifications should be reported according to Reg. Guide 1.16 "Reporting of Operating Information - Appendix A Technical Specifications."

Response

All tests that result in recalibration, new baseline settings, or other set point identifications will be reported according to Regulatory Guide 1.16 "Report of Operating Information - Appendix A Technical Specifications."

10. Question

Tests and inspections performed to verify and characterize design aspects and parameters of the reload fuel system components should be described. Planned operational surveillance and subsequent post irradiation testing of fuel rods, burnable poison rods, and control rods should also be described. Planned comparisons between characterization tests and inspections, surveillance, and post irradiation tests for the reload fuel and the irradiated fuel remaining in the core should be presented.

Response

Niagara Mohawk has not performed tests or inspections to verify and characterize design aspects and parameters of reload 6 fuel. Also, we do not have any plans to perform operational surveillance or post irradiation testing of any fuel rods, burnable poison rods or control rods presently at Nine Mile Point Unit 1.