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DESCRIPTION:

Ltr re our 9-17-75 ltr .... furn addl info concerning the Core Cycle 4 Analysis ... trans the following:

ENCLOSURES:

Responses to Questions concerning the Core Cycle 4 Analysis ......

*(1 cy enc'l rec'd)

ACKNOWLEDGED

DO NOT ...

PLANT NAME: Nine Mile Point #1

10-9-75 JGB

FOR ACTION/INFORMATION

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Regulatory File Cy.

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD, WEST
SYRACUSE, N. Y. 13202

RECEIVED 10-7-75 U.S. NUCLEAR REGULATORY COMMISSION

October 3, 1975

DOCKETED USNRC OCT 7 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

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 September 17, 1975 requested additional information concerning the Core Cycle 4 Analysis for Nine Mile Point Unit 1. The enclosed information addresses itself to the attachment of your letter.

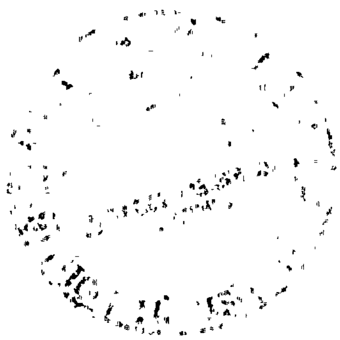
Sincerely,

NIAGARA MOHAWK POWER CORPORATION

Gerald K. Rhode
Vice President - Engineering

/sz
Enclosure

10560



1. QUESTION

Provide the value for the delayed neutron fraction applicable to the beginning and the end of Cycle 4.

RESPONSE

The value of the delayed neutron fraction applicable to the beginning of Cycle 4 is 0.006261; the value applicable to the end of Cycle 4 is 0.005361.

2. QUESTION

Provide the design conservatism factors for scram reactivity, Doppler coefficient, and void coefficient (include both values used in Table 6-1).

RESPONSE

The attached Table 2.1 provides the design conservatism factors for scram reactivity, Doppler coefficient and void coefficient.

Table 2.1

Design Conservatism Factors

Parameter	Percent of Nominal
Scram Reactivity	80%
Doppler Coefficient	90%
Void Coefficient	
Negative reactivity transients	90%
Positive reactivity transients	125%

3. QUESTION

Provide a statement verifying that the Neutron Effective Void model was used in the calculation of the void coefficient used for Cycle 4.

RESPONSE

The core transient data used in the transient analysis for Cycle 4 is based on the Neutron Effective Void (NEV) model for calculating void coefficients. For the abnormal operating transient analyses, the void coefficient calculated by the NEV model is converted to units of ¢/%.

4. QUESTION

Provide the control rod location used in the development of Figures 6-12 and 6-13 of NEDO-20772, "GE-BWR Reload-5 Licensing Submittal for NMP-1 Nuclear Power Station, Unit 1", for the Rod Withdrawal Error transient analysis.

RESPONSE

The control rod at position 22, 35 was used to develop information presented in Figures 6-12 and 6-13 of NEDO-20772.

5. QUESTION

Provide the time of Cycle 4 to which the curves in Figures 6-2 through 6-5, NEDO-20772, correspond.

RESPONSE

The curves provided in Figures 6-2 through 6-5 of NEDO-20772 correspond to the beginning of Cycle 4 conditions.

6. QUESTION

The analysis of the fuel loading error accident (page 6-7 of NEDO-20772) shows that a localized MCPR of 1.05 would occur in a misplaced bundle. Provide an explanation of the consequences of operating with a MCPR less than the Safety Limit of 1.06. Include the radiological consequences, if any.

RESPONSE

The results of the fuel loading error accident as presented in NEDO-20772 were calculated using overly conservative initial conditions. Particularly, the initial MCPR's were below the cycle 4 operating limits. The results of a re-analysis of this accident based upon operating limits show that the resulting MCPR of a misplaced bundle will be 1.097.

The MCPR value which determines boiling transition for the fuel loading error accident is 1.0. Therefore, there are no radiological consequences.

7. QUESTION

The GETAB transient analysis initial condition parameters, Table 4-4 of NEDO-20772, indicate that the initial condition MCPR values used in the transient analyses are lower than the operating limits derived from the analyses. Provide a discussion of how this affects the conservatism of the analyses. Include an explanation of the relationship between initial MCPR and ΔMCPR.

RESPONSE

The effect of initial condition MCPR values used in transient analyses was discussed in Response 9 of a January 20, 1975 letter from C. H. Frauenholz (General Electric) to A. J. Ignatonis (NCR).

The transient analysis as presented in NEDO-20772 has been reanalyzed to reflect proper initial conditions. Revisions to Tables 4-3 and 4-4 of NEDO-20772 are attached. No changes to MCPR operating limits are required.

Table 4-3 Revised

LIMITING PRESSURE AND POWER INCREASE TRANSIENTS

Event	Maximum ΔCPR 7x7	8x8
Turbine Type w/o Bypass Trip Scram, 94 percent Power 100 percent Flow, ECCS Scram Curve	.13	.16
Rod Withdrawal Error	0.30	0.32

Table 4-4 Revised

GETAB TRANSIENT ANALYSIS
INITIAL CONDITION PARAMETERS

	7x7	8x8
Peaking factors (local, radial and axial)	1.30, 1.47, 1.40	1.22, 1.612, 1.40
R-Factor	1.100	1.102
Bundle Power, MWt	4.707	5.158
Non-fuel Power Fraction	0.035	0.035
Core Flow, Mlb/hr	67.5	67.5
Bundle Flow, 10^3 lb/hr	114.1	101.4
Reactor Pressure, psia	1048.2	1048.2
Inlet Enthalpy, Btu/lb	526.1	526.1
Initial MCPR	1.36	1.38

8. QUESTION

Provide an explanation of the differences in the GETAB transient analysis initial condition parameters as presented in Table 4 of your June 30, 1975 submittal and Table 4-4 of NEDO-20772.

RESPONSE

The information presented in our June 30, 1975 submittal relates to cycle 3 operation whereas the NEDO-20772 document addresses cycle 4 operation. The differences in the GETAB transient analysis initial condition parameters are due to differences in core transient response. Because of a more negative void coefficient and a "worsened" scram reactivity curve, the transient response for cycle 4 is more severe. Therefore, initial MCPR's were necessarily higher. Additionally, the increased severity of the transient response is reflected in the reduction of allowable end of cycle power level for cycle 4 to 94 percent rated.

9. QUESTION

For the Rod Withdrawal Error transient analysis, provide a curve of APRM channel reading (% of initial level) versus control rod position (ft. withdrawn) for the case where no LPRM's are bypassed.

RESPONSE

The APRM response versus rod position curves presented in NEDO-20772 are for the worst permitted bypass conditions. The numerical data for other bypass conditions (including the case where no LPRM's are bypassed) have been reviewed to assure that the worst case curves are those previously submitted.

The standard rod withdrawal error analysis for the unique APRM system at Nine Mile Point Unit 1 does not create the computer generated graphs normally submitted with other plant analyses. As a result, the APRM channel reading curve with no LPRM bypasses is not readily available.

10. QUESTION

Provide an explanation for the difference in the void fraction values in Tables 5-1 and 6-1, NEDO-20772.

RESPONSE

The void fraction in Table 6-1 of NEDO-20772 is calculated from a void map which is a function of subcooling, exit quality, pressure and core flow. This value is used in the dynamic code from which transient calculations are performed (for further description see NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", issued in February, 1973).

The void fraction value presented in Table 5.1 of NEDO-20772 was calculated from a nuclear code and is typical of average expected conditions for cycle 4. This value is not used in safety analyses.

The intent historically was that the dynamic code should provide more conservative transient results.

[illegible]