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SUBJECT: Responds to 931026 RAI re power uprate review.

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January 6, 1994
G02-94-006

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION, POWER UPRATE REVIEW (TAC M87075)**

Reference: Letter, dated October 26, 1993, JW Clifford (NRC) to JV Parrish (SS), "Request for Additional Information, Power Uprate (TAC No. M87075)

Attachment 1 provides the response to the request for additional information included with the reference.

Also attached are fifteen copies of pages 1-9/1-10 and 1-11/1-12 of NEDC-32141P to be inserted in the NRC copies of this document.

Sincerely,

J V. Parrish

J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

WCW/bk
Attachment

cc: KE Perkins - NRC RV
NS Reynolds - Winston & Strawn
JW Clifford - NRC
DL Williams - BPA/399
NRC Site Inspector - 927N

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ATTACHMENT

1. *Should RPS and RRC be defined in the glossary of terms?*

The Reactor Protection System (RPS) and Reactor Recirculation Control (RRC) were inadvertently omitted from Table 1-1 of NEDC-32141P. These terms have been correctly delineated in the updated Table 1-1 as attached.

2. *In Section 2.4 of NEDC-32141P, the stability discussion did not contain a commitment to follow the guidance contained in NRC Bulletin 88-07 and the Supplement 1. Do you plan to follow this guidance?*

To maintain the same level of instability protection during uprated operation, the plant will operate as described in Section 3.2 of NEDO-31984P.

On-going activities by the BWR Owner's Group (BWROG) and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that occasionally have been observed in certain BWR operating conditions. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, Technical Specifications and associated implementing procedures have been incorporated which restrict plant operation in the high power, low core flow regions which have been defined in accordance with this bulletin. Specific operator actions have been established to provide clear instructions for the possibility that the reactor inadvertently (or under controlled conditions) enters any of the defined regions.

The interim agreements and constraints will continue to be followed for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWR Owners Group and the NRC.

3. *In Section 3.4 of NEDC-32141P, the licensee did not commit to perform power uprate tests on the RRC system to demonstrate flow control over the entire recirculation flow range that WNP-2 plans to operate, as discussed in the GE generic report NEDO-31897 (power uprate generic guidance report). These tests provide assurance that no undue vibration or other phenomena occur at power uprate and/or Extended Load Line Limit (ELLL) operation. Is the licensee planning to perform these tests prior to operating in these regimes?*

The "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate", NEDC-31897P-1, noted that power uprate requires only limited startup tests and a testing plan will be included in the uprate licensing application. WNP-2 licensing submittal, NEDC-32141P, Section 10.3 details the plant-specific testing at the time of implementation of power uprate. Based on Section 10.3 commitments, a startup test procedure will be developed for WNP-2. Because ELLL represents a minor variation from the new uprated rod line, the normal surveillance as required by WNP-2 Technical Specifications will be performed to ensure proper systems operation.

As noted in the generic guideline,"the planned approach to achieve an increase in rated power requires no increase in the maximum core flow". WNP-2 is currently licensed to operate at 106% of the rated core flow, which is the same core flow at which the power uprate evaluation was performed (i.e., no change in the recirculation pump speed because WNP-2 is a Flow Control Valve plant). The main effect of the power uprate on the recirculation system is the slight increase in the two-phase flow resistance due to an increase in the core average void fraction. A plant-specific evaluation was performed for WNP-2 to confirm that the recirculation system will accommodate the expected increase in the flow resistance at uprated power conditions when operating at maximum core flow. Potential increases in internals vibration and piping components were also evaluated at uprated power (Sections 3.3 and 3.5 of NEDC-32141P). The results of these evaluations showed that the vessel internals and reactor recirculation piping systems have sufficient margin to perform at the analyzed uprated conditions. The Supply System will continue surveillance of various systems in accordance with WNP-2 Technical Specifications during power ascension and steady state operation, as is currently the practice, to ensure proper operation at uprated conditions.

The Supply System is planning tests to be incorporated into the Uprate Startup Program which will address ELLL operation. The tests planned for ELLL operation can be divided into four main areas. They are 1) Stability tests, 2) CPR performance verification, 3) TIP measurements and 4) Bi-stable flow. The purpose of these tests is to validate ELLL assumptions.

4. *In Section 9.3.2 of NEDC-32141P, the Station Blackout (SBO) event is discussed. Will this event be evaluated with power uprate and including ELLL operating conditions to assure that systems needed to respond after power has been restored are designed to operate for the peak suppression pool temperature expected for this event?*

The containment response to the SBO event was analyzed for WNP-2 at power uprate conditions for a 4 hour coping period. The bounding analysis was performed at the thermal power level of 3629 MWt which is 4.1% higher than the new uprated power level of 3486 MWt. The suppression pool temperature at the end of the 4 hour coping period will be controlled by the initial vessel energy and decay heat during the 4 hour period. The initial vessel energy and decay heat used for the analysis with power uprate bounds the corresponding value for the ELLL operating region. Analyses were performed for the uprated power at 3629 MWt. A suppression pool temperature of 209°F was calculated for the thermal power level of 3629 MWt. The calculated suppression pool temperature at the end of the 4 hour coping period for power uprate conditions is within the design limits of the systems needed to respond after the power has been restored.

5. *In Section 9.3.1 of NEDC-32141P, provide an explanation as to why the Loss Of Feedwater (LOFW) event is more limiting for WNP-2 than the Loss Of Offsite Power (LOOP) in the generic Anticipated Transients Without Scram (ATWS) analysis. Provide assurance that the generic evaluation referenced by WNP-2 is valid for SNP fuel. Provide the results of the analysis or reference where they are contained if they are elsewhere in your submittal.*

Table 3-6 of the generic power uprate evaluation (NEDC-31984P,Supplement 2) shows that the LOOP event was the mildest of the four events evaluated. The LOFW event was selected for the WNP-2 Power Uprate evaluation because it is the only event that reaches the low level ATWS setpoint (L2) before it reaches the high pressure setpoint. However, the LOFW and LOOP events are similar in that there is a total loss of feedwater at the start of the event. Both events are quite mild and neither represents a challenge to ATWS evaluation criteria (i.e., reactor coolant pressure boundary, containment pressure, suppression pool temperature and fuel cladding temperature).

The generic power uprate evaluation was performed for GE fuel. However, for the WNP-2 power uprate evaluation, GE performed a cycle specific core simulation with major fuel parameters and characteristics which were provided by the Supply System based on cycle 8 which included SNP fuel. The resulting GE calculated void and Doppler coefficients were within the range of the sensitivity studies which were performed for NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II," December 1979, which showed the impact of important fuel design related parameters, void coefficient and Doppler coefficient on the ATWS event results.

6. *The transient analysis results for uprate are presented in Table 9-2 of NEDC-32141P. Please define $MCPR_f$ for the ΔMCPR column for the Slow Recirculation Increase transient.*

The analysis of abnormal operational occurrences for BWR core reloads is generally performed assuming rated and/or maximum allowable core flow conditions. The purpose of the $MCPR_f$ curve is to adjust the MCPR operating limits when the plant is operating at less than rated core flow. This is necessary because some transient events are more severe (i.e., larger ΔMCPR) when initiated at reduced core flow. In the manual flow control mode, similar to WNP-2, the $MCPR_f$ curve assures that the safety limit MCPR will not be violated should the most limiting (e.g., flow increase) transient occur.

The transient analysis performed for WNP-2 with RRC valve flow control included analysis of the $MCPR_f$ curve for WNP-2 for cycle 8 for uprate conditions. A copy of this curve is attached. The cycle specific Slow Flow Runout Event analysis and the corresponding $MCPR_f$ will be completed for the specific reload cycle in which power uprate is to be initiated by the Supply System and its fuel vendor. The results of the analysis will be shown in the Core Operating Limits Report of the specific fuel cycle.

7. *In your SAFER/GESTR-LOCA analysis report (NEDC-32115P) you provide a discussion for Single Loop Operation (SLO) thermal limits. The evaluation assumes there will be no MAPLHGR or PLHGR reduction for SLO. The SLO nominal and Appendix K PCTs are higher than for two loop operation. Provide assurance that the upper bound PCT is less than 1600°F. Are there any reductions on these thermal limits for operating in the ELLL regions such that the PCTs for SLO will be less than for two loop operation? Will there be any increase in the SLMCPR for SLO because of increased uncertainties?*

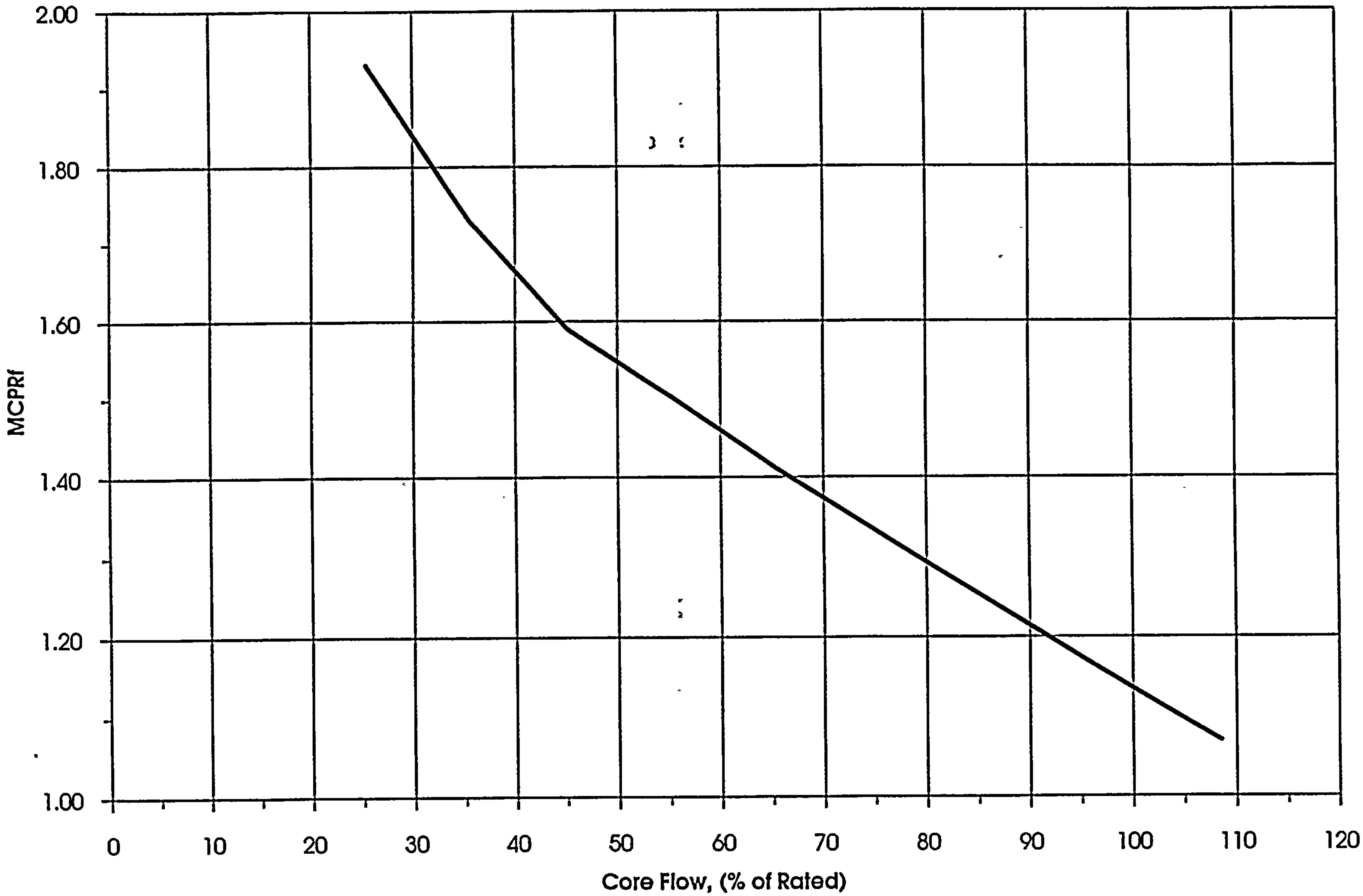
The calculated nominal and Appendix K PCTs for single loop operation at WNP-2 are higher than that of the two loop operation due to conservative dryout times which were used in the SLO analysis. However, the calculated upper bound PCT for SLO, assuming no MAPLHGR or PLHGR reduction, is 1450 °F, which is below the 1600 °F limit. This calculation included a plant specific evaluation of the plant variable uncertainties under SLO conditions. The evaluation for SLO conservatively bounds the ELLL conditions because the SLO analysis was performed at full power with no credit for recirculation flow coastdown.

The current WNP-2 Technical Specifications applies a 0.01 adder to the SLMCPR when in SLO due to increased uncertainties. This Technical Specification will continue to apply.

8. *Not transmitted to the Supply System*

9. *In your GE-NE-187-24-0992, Rev.2 report discussing SRV setpoint tolerance and the number of SRV's out-of-service (OOS) evaluation, there appears to be no reference to a generic GE Nuclear Energy Licensing Topical Report (NEDC-31753P). This report explains what plant specific analyses are needed to assure no safety limits are exceeded. There also was an NRC Safety Evaluation Report prepared upon review of this generic report. Why did you not reference these reports?*

   WNP-2 plant did not participate in the "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report,"NEDC-31753P; thus,WNP-2 does not have access to the findings and resolution of the report. Therefore, a plant specific independent study was performed for WNP-2 SRV OOS evaluation.


2.00
1.80
1.60
1.40
1.20
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MCPRf
0
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60
70
80
90
100
110
120
Core Flow, (% of Rated)