CATEGORY 1

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AUTH.NAME AUTHOR AFFILIATION
BEMIS,P.R. Washington Public Power Supply System
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SUBJECT: Provides addl perspective on apparent violation noted in insp rept 50-397/98-01.Clearer understanding of issue has been provided by addl testing & analysis by Siemens Power Corp.

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July 9, 1998
GO2-98-117

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2 OPERATING LICENSE NPF-21**
INSPECTION REPORT 50-397/98-01
EVALUATION OF APPARENT VIOLATION

The postponement of the predecisional enforcement conference has created an opportunity for the Supply System to provide additional perspective on the apparent violation identified in the subject inspection report (50-397/98-01). Since the issuance of the report, a clearer understanding of this issue has been provided by additional testing and analysis by Siemens Power Corporation (SPC).

Apparent Violation

Inspection Report 50-397/98-01 summarizes the apparent violation on page 12:

> In summary, licensee oversight of the previous and current fuel vendors did not assure that the core power distribution and fuel thermal limits submitted by letters dated August 2, 1990, February 28, 1991, and May 20, 1991, were accurate and conservative. 10 CFR Part 50, Appendix B, Criterion III requires that established measures assure that applicable regulations and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The inspectors determined that established measures did not assure the translation of regulatory requirements and the design basis for the reactor core into the facility license Technical Specifications. Consequently, the licensee failed to ensure that sufficient margin was available in its SLMCPR SAFDL. This was an apparent violation of Criterion III of Appendix B to 10 CFR Part 50 (50-397/9801-01).

JE01

It is the Supply System's interpretation that the apparent violation is based on the premise that the Supply System's vendor surveillance program should have identified an alleged inadequacy in a topical report data base used to calculate the fuel safety limits (i.e., NRC Approved SPC topical report ANF-1125(P)(A), "ANFB Critical Power Correlation," April 1990). The inspection report

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further notes the safety significance of this finding by concluding that, as a result of the alleged failure, a revised safety limit could have been exceeded by the occurrence of a limiting transient.

NRC Inspection Findings

The concern with the critical heat flux data base was identified by the NRC in 1997 during an inspection at SPC (Inspection Report 99900081/97-01). The inspection team questioned the adequacy and applicability of the ANFB critical power correlation to the ATRIUM-9 and ATRIUM-10 fuel types. The team believed that the data base for the ATRIUM-9 fuel did not contain an adequate number of test points nor was an adequate range of conditions tested to justify the uncertainty values for the additive constants used to determine the SLMCPR.

In 1998, SPC conducted confirmatory testing that demonstrated that the departures from the ANFB correlation anticipated by the NRC were primarily caused by the ULTRAFLOW spacer. Since the 9x9-9X fuel used at WNP-2 does not use the ULTRAFLOW spacer, it can be concluded that the NRC's finding related to the newer designs does not apply to the WNP-2 fuel. SPC has determined that a new correlation is necessary for those fuel assemblies using the ULTRAFLOW spacers.

The SPC test results had not been completed at the time of the February NRC inspection at WNP-2. SPC notified the Supply System of its findings regarding the ULTRAFLOW spacer and conclusions regarding the 9x9-9X fuel in a letter dated April 7, 1998.

9x9-9X Fuel Design

The SPC 9x9-9X fuel design was first used at WNP-2 during Cycle 7. Unlike the fuel designs with the ULTRAFLOW spacer, the 9x9-9X fuel type is an older ATRIUM-9 design containing an internal water canister and the standard non-vaned bi-metallic spacers (WNP-2 is the only reactor that utilizes this configuration). The 9x9-9X fuel does not use the innovative ULTRAFLOW vaned spacer found on the ATRIUM-9B and ATRIUM-10 fuel types. Although the ULTRAFLOW spacer was later offered as an option, the Supply System elected to continue using the standard bi-metallic design because of the lack of industry experience with the ULTRAFLOW spacer.

The geometric and thermal-hydraulic design features of the 9x9-9X fuel (array geometry, pin diameter, pin spacing, rod to wall spacing, full length rods, axial heat profiles, spacer designs, range of fluid conditions, etc.) are well represented by the test assemblies used to determine the ANFB critical power correlation. The data base and correlation address the effects upon boiling transition due to operating pressure level, mass velocity, enthalpy, axial power peaking, local power peaking, rod diameter, assembly hydraulic diameter, and heated length. Of the 2,842 total data points in the data base, more than half (approximately 1,500) are for bi-metallic spacer designs similar to the type used in the 9x9-9X fuel. The data base also contains more than 300 points for the 9x9 internal water canister design. Data were collected for cosine, upskew and uniform axial power profiles over thermal hydraulic conditions that cover the entire range of use of the ANFB critical heat flux correlation. Because the Supply System does not use fuel with

ULTRAFLOW spacers, it considers the ANFB correlation, as approved by the NRC, to be adequate to predict the dryout performance for the 9x9-9X fuel with standard bi-metallic spacers.

The ANFB critical power correlation includes factors called additive constants for each fuel design to account for differences in geometry. The approved ANF-1125(P)(A) methodology describes the method by which additive constants are established (Section 2.4 of Supplement 1). The use of additive constants was addressed by the NRC in SER Limitation 3:

> (3) The use of the ANFB correlation shall be limited to assessments with the additive constants given in Reference 2. If a new fuel design results in additive constants outside the range of Reference 2, then these must be justified. (Note: Reference 2 is ANF-1125(P)(A) Supplement 1)

The approved method states in Section 4.2 (ANF-1125(P)(A) Supplement 1) that "...only a few test points at typical operating conditions are required to determine the additive constants for one peaking pattern and a geometry/spacer combination." For the 9x9-9X fuel type, a full array assembly was tested at the Siemens (KWU) Karlstein facility and the amount of testing that was performed to determine the additive constants was consistent with the approved methodology (over 80 additional test points were acquired). The additive constants were determined to be within the range of those found in ANF-1125(P)(A) Supplement 1. The additive constant uncertainty for the 9x9-9X fuel was found to be consistent with the uncertainty reported for the KWU/ANF 9x9-IX fuel for which the full range of hydraulic conditions was tested (ANF-1125(P)(A) Table 6.2).

Since fuel types with bi-metallic spacers, 9x9 pin arrays, and 9x9 configurations with the internal water canister were contained in the ANFB correlation data base, the introduction of the 9x9-9X configuration only required the development of additive constants as described above.

Summary

The additional testing data recently acquired by SPC provides a better understanding of the issues identified by the NRC in the inspection report. It is now known that the performance of fuel assemblies using the ULTRAFLOW spacer cannot be satisfactorily predicted by the ANFB correlation and data base. A separate correlation is under development for these fuel types (i.e., ATRIUM-9B and ATRIUM-10).

Since the ULTRAFLOW spacer was never used at WNP-2, and because it can be demonstrated that the 9x9-9X fuel design is comprised of components that are contained in the original ANFB data base approved by the NRC, it can be concluded that the ANF-1125(P)(A) correlation adequately predicts the dryout performance of the WNP-2 fuel type.

Safety Significance

The safety significance of this event can be assessed by determining whether a revised safety limit would have been exceeded by the most limiting transient. At WNP-2, the most limiting transient is a generator load rejection with the bypass valves inoperable (LRNB). The NRC has concurred

that the use of actual plant initial conditions is appropriate for a retrospective evaluation of previous operating cycles.

A LRNB transient analysis was conducted by the Supply System (GO2-98-048, March 9, 1998) using actual plant conditions from the end of Cycle 8. This point was selected from a review of past cycles because WNP-2 was operating closer to the MCPR operating limit (OLMCPR) than at other times during Cycles 7 - 12. Also, this point represented more challenging conditions when compared to other points found in Cycles 7 - 12.

A quantitative evaluation of this transient was conducted by the Supply System using its in-house NRC approved methodologies. The RETRAN model was used to calculate system response to the transient (including peak power) and the VIPRE model was used to calculate the hot channel response (ΔCPR). Full arc operation of the turbine control valves was conservatively assumed (WNP-2 utilizes partial-arc operation) and results were calculated for both normal and Technical Specification control rod scram speeds.

The analysis demonstrated that the revised safety limit would not have been exceeded by the most limiting transient (load rejection without bypass) occurring during actual plant conditions. These results confirm that there is no safety significance associated with this event. Moreover, the analysis further demonstrates that there is sufficient conservatism built into the design process to assure that the safety limit is protected.

Should you have any questions or desire additional information regarding this matter, please contact Mr. D.W. Coleman at (509) 377-4342.

Respectfully,

P.R. Bemis (Mail Drop PE23)
Vice President, Nuclear Operations

cc: EW Merschoff - NRC RIV
KE Perkins, Jr. - NRC RIV, WCFO
JE Whittemore - NRC RIV
C Poslusny, Jr. - NRC NRR
TL Huang - NRC NRR
NRC Sr. Resident Inspector - 927N
DL Williams - BPA/399
PD Robinson - Winston & Strawn
TJ Rausch - BWROG