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SUBJECT: Submits rept on impact of non-significant changes & errors discovered in ECCS evaluation methodology used in licensing o WNP-2 core. Rept submitted per requirements of 10CFR50.46.

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October 2, 1997
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Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
ANNUAL 10CFR50.46 REPORT
CHANGES AND ERRORS IN ECCS EVALUATION MODELS

References:

1) NEDC-32115P, Revision 2, "Washington Public Power Supply System Nuclear Project 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," GE Nuclear Energy, July 1993

2) CE NPSD-801-P, Revision 1, "WNP-2 LOCA Analysis Report," ABB Combustion Engineering Nuclear Operations, June 1996

The purpose of this letter is to report the impact of non-significant changes and errors discovered in the Emergency Core Cooling System (ECCS) evaluation methodology that was used in licensing of the WNP-2 core. This report is submitted pursuant to the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

Both General Electric (GE) and Asea Brown Boveri (ABB) methodologies are applied to the WNP-2 core. The GE methodology was used to license Siemens Power Corporation fuel (Reference 1). The ABB methodology was used to license ABB SVEA-96 fuel (Reference 2).

This report covers the period from July 1996 to the present. One error was identified by ABB in the application of the ABB ECCS evaluation. The impact is a cumulative 45 degree Fahrenheit increase in the reported Peak Cladding Temperature (PCT) of 2035 degrees Fahrenheit, and is associated with single loop operation. However, the final PCT remains less than the 10 CFR 50.46 acceptance limit of 2200 degrees Fahrenheit. The calculated impact for two-loop operation is less than one degree Fahrenheit.

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The GE methodology using the SAFER/GESTR-LOCA analysis did not contain any changes or errors during the past 12 months. The cumulative impact in the last five years is approximately 13 degrees Fahrenheit.

In addition, a concern was raised by a utility in 1995 pertaining to the reactor pressure vessel bottom head drain (BHD) impact on the LOCA analysis. The specific concern was that, because the bottom head drain line is directly connected to the reactor recirculation loops, a recirculation line break LOCA would have additional flow contribution from the BHD, and the vessel would depressurize to the drywell faster than assumed in current models. During the event, some water required to keep the core covered to the 2/3 core height would exit the core due to either gravity or core pressure by means of the interconnected recirculation and bottom head reactor water cleanup suction lines. The utility requested that GE perform an evaluation of this concern.

A GE evaluation concluded that, although no analysis had been performed to precisely evaluate the PCT impact of this event, it was believed that the impact was less than 10 degrees Fahrenheit. This determination was based on engineering judgment and extrapolation of previous LOCA analyses.

The impact of this concern was also evaluated by the Supply System. In the power uprate LOCA analysis model, we penalized Low Pressure Core Spray (LPCS) system flow by 625 gpm. In the model, the LPCS system initiates at about 60 seconds. Water level reaches the bottom of the fuel at approximately 40 seconds and does not recover to 2/3 core height until about 80 seconds. The relaxation of 625 gpm in LPCS flow more than accounts for the additional drain path from a recirculation line break. Accordingly, there is no impact on the PCT for WNP-2.

In each case, these items do not represent a significant change or error as defined by 10 CFR 50.46. Therefore, no further reanalysis or any Technical Specification amendments are required.

If you have any questions or require additional information pertaining to this letter, please contact P.J. Inserra at (509) 377-4147.

Respectfully,

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

cc:	EW Merschoff - NRC RIV	NRC Senior Resident Inspector - 927N
	KE Perkins, Jr. - NRC RIV, WCFO	DL Williams - BPA/399
	TG Colburn - NRR	PD Robinson - Winston & Strawn