Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247

August 21, 1996
LIC-96-0119

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. WCAP-13027-P, "Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS," dated July 1991
3. Letter from OPPD (T. L. Patterson) to NRC (Document Control Desk) dated March 15, 1996 (LIC-96-0034)

SUBJECT: Report of Significant Change/Error in the Loss of Coolant Accident (LOCA)/Emergency Core Cooling System (ECCS) Models and Evaluations

In accordance with 10CFR50.46(a)(3)(ii), the Omaha Public Power District (OPPD) is submitting a report of a significant change/error in the LOCA/ECCS models and evaluations. 10CFR50.46(a)(3)(i) states that "a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model,..." This report identifies a significant change or error in the LOCA/ECCS models used by Westinghouse Electric Corporation (W) to model Fort Calhoun Station (FCS) Unit No. 1. Reference 2 describes the methodology utilized by W to model Combustion Engineering plants, such as FCS.

It was recently discovered that the current FCS LOCA/ECCS Analyses do not explicitly model the steam generator orifice plates. Subsequent to this discovery, W performed an evaluation with the orifice plates included using extremely conservative sensitivities and uncertainties to establish a maximum peak clad temperature (PCT) increase. W assessed the PCT increase to be no more than 112°F for both the small and the large break LOCAs. With this increase in PCT, the revised maximum PCTs for both small and large break LOCAs will not exceed the regulatory limit of 2200°F as defined in 10CFR50.46(b)(1). The conservatively calculated interim PCT values are 1542°F and 2153°F for the small and the large break LOCAs, respectively, based upon the 112°F increment and the values provided in Reference 3.

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Per 10CFR50.46(a)(3)(ii), if the change or error is significant, then a report must be provided within 30 days including a proposed schedule for providing a reanalysis or taking other action to show compliance with 10CFR50.46 requirements. W is developing a more detailed safety evaluation addressing the effect of the steam generator orifice plates on the FCS LOCA/ECCS Analyses. The completion of the W safety evaluation and subsequent reporting of the results to the NRC will be completed by the next annual reporting period. Therefore, the results of this safety evaluation will be included in the next annual report currently scheduled for March 1997.

If you should have any questions, please contact me.

Sincerely,

T. L. Patt

T. L. Patterson
Division Manager
Nuclear Operations

TLP/dll

c: Winston & Strawn
L. J. Callan, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector