SP 450 [illegible]

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

March 4, 1981

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKET NO. 50-266
SAFETY SIGNIFICANCE OF STEAM GENERATOR CREVICE DEFECTS
POINT BEACH NUCLEAR PLANT, UNIT 1

Appendix A of the Staff's Safety Evaluation Report Related to Point Beach Unit 1 Steam Generator Tube Degradation Due to Deep Crevice Corrosion dated November 30, 1979 provided a calculation of secondary-primary leakage under various primary-secondary pressure differentials and with certain assumptions concerning size and location of tube defects. This matter was also discussed briefly at the January 2, 1980 meeting of the Commission, at which time the in-leakage flow rate was conservatively stated to be 7 gpm per tube, and that on this basis, a large number of tubes had to be broken simultaneously in a guillotine manner to cause sufficient in-leakage so as to be a matter of concern regarding the steam binding effect that may impede the ECCS performance.

In order to verify this analysis and previous evaluations, we retained Combustion Engineering, Incorporated to conduct certain laboratory tests to determine more precisely the leakage characteristics of simulated tube defects occurring in the tube sheet crevice. We are pleased to enclose for your information a copy of the Combustion Engineering Test Report TR-ESE-411 entitled, "Steam Generator Crevice Flow Test for Wisconsin Electric Power Company", which presents the test data, procedures, and results of this program.

The highest steady-state flow achieved in these tests was approximately 4 gpm, or nearly half the values assumed in the Staff's SER. These tests included defects ranging from an 0.010" diameter hole to an 0.010" by 3.0" axial slit and located at various depths within the simulated tube sheet crevice. The tests were conducted at full temperature and at secondary-to-primary differential pressure of approximately 1,050 psi to simulate the highest differential pressure following the LOCA. The highest flow rate achieved, as noted above, occurred with the defect located just below the top of the tube sheet. As would be expected, the secondary-primary flows decreased as the distance between the defect location and the top of the tube sheet increased.

We believe the results of these experiments add pertinent information to the understanding of the safety significance of steam generator tube defects located within crevice regions. The test results confirm our conclusions and those of the NRC staff that crevice leakage will not, in a LOCA, cause steam binding within steam generator tubes that would render the ECCS incapable of cooling the reactor core. The test results indicate that the Staff safety evaluations are conservative by approximately a factor of two.

Please advise us if you wish to receive additional copies of this report or wish to discuss these tests in any further detail.

Very truly yours,

C. W. Fay, Director
Nuclear Power Department

Enclosure

Copies to NRC Resident Inspector
C. F. Riederer - PSCW
Peter Anderson - WED