VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 19, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No.: 986B
FR/MLB: mvc

Docket No.: 50-338
50-339

License No.: NPF-4

SUPPLEMENTAL INFORMATION TO
AMENDMENT TO OPERATING LICENSE NPF-4
NORTH ANNA POWER STATION UNITS NO. 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGE NO. 27

My letter to you dated November 29, 1979, Serial No. 986, transmitted our proposed Technical Specifications Change No. 27 and a supporting LOCA-ECCS analysis which met the limits of 10 CFR 50.46. This analysis was determined to be in compliance with Appendix K to 10 CFR 50 even though the cladding heatup rate dependent burst curve used was revised and had not been explicitly reviewed by the NRC. This approach was taken because the revised modeling was more technically correct. To facilitate the NRC staff review and understanding of this revision, we are providing, as supplemental information to our transmittal of November 29, 1979, a LOCA-ECCS calculation for the C_D = 0.4 DECLG break (limiting break size) using the non-heatup rate dependent burst curve modeling documented in the February 1978 version of the Westinghouse ECCS Evaluation Model. This calculation resulted in a peak clad temperature of 2088°F.

Both the analysis provided in our November 29, 1979 submittal and the above analysis meet the criteria denoted in 10 CFR 50.46. However, as indicated in the Westinghouse (T. M. Anderson) to NRC (D. G. Eisenhut) letter of December 7, 1979 (Serial No. NS-TMA-2174), additional impacts on the limiting peak clad temperature may result if the flow blockage modeling documented in draft report NUREG 0630 (reference NRC (D. B. Vassalo) letter of November 28, 1979) is used. At this time, we do not believe that these potential impacts will be significant when the NRC review is completed. Further, these potential impacts can be offset by the modeling improvements documented by Westinghouse in their above letter of December 7, 1979. These benefits are expected to more than offset any potential impacts on North Anna 1 & 2 even if implementation of the current version of draft report NUREG-0630 is required. As a result, the reduction in the limiting F_Q values as proposed in our requested Technical Specifications Change No. 27 remains appropriate for supporting continued operation.

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Should you have questions, please contact our Mr. M. L. Bowling (804-771-3183) at your earliest convenience.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

cc: Mr. James P. O'Reilly, Director
Office of Inpsection and Enforcement, Region II

Mr. O. D. Parr, Chief Light Water Reactors Branch
No. 3

Mr. J. E. Rosenthall, Reactor Safety Branch
Division of Operating Reactors

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