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VPNPD-88-526
NRC-88-103

October 28, 1988

U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
ADDITIONAL INFORMATION FOR
TECHNICAL SPECIFICATION CHANGE REQUEST 127
INCREASE ALLOWABLE CORE POWER PEAKING FACTORS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

The enclosed report provides additional information requested by your staff during our October 6, 1988 meeting regarding Technical Specification Change Request 127. The proposed changes provide for the design and operation of the Point Beach Nuclear Plant cores with enhanced Optimized Fuel Assembly Fuel and at higher core peaking factors than are allowed by current plant Technical Specifications. This report provides more detail on the reference core design used in the analyses, as well as on the results of the safety analyses.

The remainder of this letter briefly addresses questions raised at our October 6 meeting regarding the methodologies used in the analyses:

1. The steam generator tube rupture event reanalysis uses the same methodology presently used in Chapter 14 of the Point Beach Final Safety Analysis Report (FSAR) with some revised input assumptions, as outlined in the enclosed report. In addition, the tube uncovery issue for this event is being addressed generically through the Westinghouse Owners' Group (WOG) program presented to the NRC on July 27, 1988. Preliminary results of the WOG program are expected by January 1989.

2. Use of peripheral power suppression assemblies (PPSAs) as they affect the accident analyses has been bounded in the reanalyses, as described in the enclosed report. Other effects regarding the PPSAs and the specifics of their use will be discussed in a supplemental submittal to be made when further analysis is completed. We expect this supplement to be submitted in January 1989.

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3. New or revised methodologies employed in these reanalyses include the revised thermal design procedure (RTDP) and the WOG dropped rod methodology. The event analyses affected by the RTDP methodology include: Uncontrolled RCCA Withdrawal at Power, RCCA Drop, Excessive Load Increase Incident, Loss of External Electrical Load, and Loss of Reactor Coolant Flow. The WOG dropped rod methodology affects only the RCCA drop analysis. The WCAPs for these methodologies have been submitted for NRC review. A Safety Evaluation Report (SER) for the RTDP methodology is expected shortly. However, review of the 'OG dropped rod methodology has apparently been delayed. Since it is used in our justification for this Technical Specification change, we again emphasize the importance of NRC approval of the WOG dropped rod methodology.

4. The small-break LOCA analysis used the NOTRUMP code, which has been generically approved by the NRC for Westinghouse plants.

5. All the other analysis methodologies used, with the exception of the SGTR input changes already mentioned, are the same as those currently used in the FSAR analyses. It should be noted that the locked rotor reanalysis employed the same methodology and met the same acceptance criteria as our current FSAR analysis, which was accepted as a part of our OFA fuel submittal in 1984.

Proposed changes to the safety analyses, as described in Chapter 14 of the FSAR, and the results of the large-break LOCA best-estimate analysis will be submitted separately. These submittals will provide additional information supporting the request and may help you in your evaluation.

Please contact us should you have any questions regarding the information provided.

Very truly yours,

CW Fay

C. W. Fay
Vice President
Nuclear Power

Enclosure

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector
R. S. Cullen, PSCW