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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

STOLZ,J.F. Operating Reactors Branch 4

SUBJECT: Forwards revised responses to Questions 4 & 11,addressed in util 801113 ltr,per NRC 810316 request for addl clarification on reload design methodology technical rept.

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March 18, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: J. F. Stolz, Chief
Operating Reactors Branch No. 4

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Oconee Reload Design Methodology Technical Report

Dear Sir:

My letter of November 13, 1980 provided Duke Power Company's responses to the Staff's questions concerning the subject report. Attached are revised responses to Questions 4 and 11 which were addressed by that letter. These responses have been revised to provide additional clarification, as requested informally by the Staff on March 16, 1981.

Very truly yours,

William O. Parker, Jr.

FTP:pw
Attachment

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Q. 4. Paragraph 8.3.2 Start-up Accident

Give the variation of the total (and its components) reactivity for the start-up accident for the first 10 seconds after the accident initiation, (these would complement Fig. 14-1 and 14-2 of the Oconee FSAR Rev. 16).

A. 4. The approach taken in the review of FSAR transient analyses as an integral part of the reload design methodology is discussed in Section 8 of NFS-1001. For each FSAR analysis the main parameters of interest have been identified and documented in the FSAR. In order to assure that a reload core is in conformance with the assumptions in the analysis, it is necessary to determine that the parameters associated with the reload core are bounded by the parameters assumed in the FSAR. If this criterion is met, it can be concluded that the existing FSAR analysis remains valid for the reload core.

Question 4 requests additional information for the start-up accident concerning the variation of the components of the reactivity response. These parameters are an intermediate output of the analysis whose response is indicated by other documented parameters such as power level, but are not normally included in the analysis documentation. However, the components of the reactivity response are determined by the parameters which are reviewed and shown to be within the bounds of the FSAR analysis. The reactivity response determined by those parameters remains valid until the value of a parameter is no longer bounded for a reload core. The safety review methodology of Section 8 assures the identification of all pertinent reload core parameters affecting the reference safety analysis, confirmation of the validity of the reference safety analysis for the reload core, and the resolution of any non-conservative parameter.

In order to respond to the question, the variation of the total reactivity and its components were calculated from the results presented in FSAR Figures 14-1 and 14-2, utilizing the analysis assumptions specified in the FSAR. The variation of the total reactivity during a startup accident is the sum of three reactivity effects. The withdrawal of the control rod banks adds positive reactivity which causes the neutron power level to increase and raise the average core temperature. The increase in fuel temperature causes a negative reactivity feedback due to the negative Doppler coefficient. The increase in power level increases heat transfer from the fuel to the coolant, resulting in an increase in moderator temperature. This causes a positive reactivity feedback since a positive beginning of cycle moderator coefficient is assumed. The transient response is primarily determined by the rate of positive reactivity addition from the withdrawal of rods, and the Doppler feedback which slows or terminates the nuclear excursion. The moderator feedback has a smaller effect. Figures 4-1 and 4-2 show the variation of the reactivity consistent with FSAR Figures 14-1 and 14-2 respectively. It should be noted that these figures do not represent the first 10 seconds of the transients, considering that the initial conditions are 10E-9 rated power and 1% k/k subcritical. Figures 4-1 and 4-2 illustrate the time interval of greatest interest during the transient, Figure 4-1 is the same scale as Figure 14-1, and Figure 4-2 is the first one second of the response in Figure 14-2. For both transients the reactivity addition for the first 10 seconds following initiation of rod withdrawal would only cause a reduction in the subcriticality margin.

Q. 11. Supplement 2, Paragraph 3.2, Comparison of ARMP PDQ07 to Cold Criticals.

The two-dimensional simulation of the criticals has not been performed at Duke nor with PDQ07, yet it was concluded that the results would have been identical with the PDQ07 results. Justify the above conclusion.

A. 11. The cold criticals have been simulated with PDQ07. The results have been published in Part I Chapter 2, Rev. 1 of the ARMP System Documentation. This work was performed under EPRI Research Project 118-1.

These benchmark calculations use standard ARMP methodology, standard ARMP codes (EPRI-CELL, NUPUNCHER, PDQ07) and Duke Power also uses these codes and methodology. Duke Power Company has been actively involved in developing in-core fuel management capability since 1969. Currently in the Nuclear Fuel Services Section, there are a total of nine employees with a cumulative thirty-two (32) man-years of PDQ experience. The level of individual experience ranges from one to nine years, and includes experience with Combustion Engineering, Westinghouse, and Babcock & Wilcox core design calculations. Therefore, Duke Power considers that if it had performed these benchmark calculations, the results would have been identical.