Omaha Public Power District
1623 Harney Omaha, Nebraska 68102
402/536-4000

January 31, 1984
LIC-84-029

Mr. James R. Miller, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Operating Reactors Branch No. 3
Washington, D.C. 20555

Reference: Docket No. 50-285

Dear Mr. Miller:

Fort Calhoun Reload Core Analysis
Methods and Verification

Pursuant to discussions held with Mr. E. G. Tourigny of your staff on January 17, 1984, the attached information is provided in response to verbal questions asked in regard to the above subject.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/JJF:jmm

Attachment

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N.W.
Washington, D.C. 20036

Mr. E. G. Tourigny, Project Manager
Mr. L. A. Yandell, Senior Resident
Inspector

A001
1/1

Attachment

NRC Question

The District has stated that the fuel mechanical performance calculations are performed by the nuclear fuel vendor.

How does the District determine the fuel dependent parameters such as gap conductance that are used in the safety analysis?

Response

The District has obtained from both Combustion Engineering and Exxon Nuclear values for the various parameters which will bound the fuel during its lifetime in the Fort Calhoun reactor core. These values were either directly transmitted to the District or have been derived by the District in conjunction with the nuclear fuel vendor and confirmed by the nuclear fuel vendor. If another fuel type is used in the Fort Calhoun reactor, values for these parameters will be discussed with the nuclear fuel vendor and suitable values selected for safety analysis with the concurrence of the nuclear fuel vendor.

NRC Question

It appears that the few group neutron cross-sections used in the verification of neutronics models have been generated for the District by Combustion Engineering. Therefore, the ability of the District to generate few group cross-sections has not been demonstrated. Is this true of both CEPAK and DIT generated cross-sections? In view of this, are the few group cross-section table sets generated by Combustion Engineering for the District for all future cycles in the Fort Calhoun core? If not, who will generate cross-sections for future cycles?

Response

The CEPAK cross-sections were generated by the District for both Combustion Engineering and Exxon Nuclear fuel. The DIT cross-sections were generated by Combustion Engineering for the District utilizing their own parameters for the Combustion Engineering fuel and parameters for Exxon Nuclear fuel supplied by the District. The DIT cross-section table sets are applicable for the current fuel designs in the Fort Calhoun reactor for all future cycles. If new fuel designs are utilized in the Fort Calhoun reactor, Combustion Engineering will generate DIT cross-sections utilizing information supplied by the District for fuel that is not manufactured by Combustion Engineering.

NRC Question

Since the computer codes used by the District are maintained by Combustion Engineering on the Combustion Engineering computer system in Windsor and the District accesses these codes through a time-sharing system, what procedures are in effect to alert the District to possible code modifications, updates, or errors?

Response

The Combustion Engineering Computer Services Engineer and the Combustion Engineering Omaha Public Power District Project Manager are both on internal distribution at Combustion Engineering for all code related items. These two individuals have the responsibility to transmit this information to the District. The District has been alerted to changes in the codes which are being made at Combustion Engineering by this method.

NRC Question

The Exxon methodology is referenced for the analysis of a control rod ejection. How does the District access the XTRAN code which is used by Exxon for this analysis? What type of training was received by the District for analyzing the rod ejection accident and for utilizing Exxon methods?

Response

The Exxon methodology for analyzing the control rod ejection transient is discussed in XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors" (January, 1979). This topical report discusses the fact that it is not necessary to access the XTRAN code to calculate the post ejection energy in the pellets using this generic methodology. Rather, it is necessary to calculate the physics parameters which include the control rod worth, the Doppler coefficient, power peaking factors, and delayed neutron fractions. Using these values with the appropriate figures in the topical report, the appropriate additive and multiplicative factors are determined which are used to calculate the total deposited energy in the pellet. This is done in the same fashion as that discussed in Chapter 7 of the topical report where a sample problem is worked. For each reload cycle, the District determines that the resultant enthalpy calculated for a rod ejection accident is less than the value calculated by Exxon Nuclear for Cycle 6, which is the reference cycle for this transient. Because of the simplicity of this method, Exxon and the District jointly determined that XN-NF-78-44 contained sufficient information to analyze the rod ejection accident and that District personnel did not require additional training by Exxon.