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

April 4, 1984
LIC-84-098

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 Station
Cycle 9 Core Reload Submittal

Attached is Omaha Public Power District's response to your request for additional information dated March 22, 1984 concerning the Fort Calhoun Station's Cycle 9 core reload submittal. This information was requested by your staff to enable them to continue their review of the Cycle 9 reload facility license change application.

Sincerely,

W C Jones

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

Add: Reg File
Original

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ATTACHMENT

NRC Question 1: What was the EFPY when the unit shut down at the end of Cycle 8?

Response: 6.77 EFPY

NRC Question 2: Results of the evaluation of Surveillance Capsule W-265 were referenced. However, the report was not submitted. When does the licensee plan to submit the report per 10 CFR 50, Appendix H?

Response: The Surveillance Capsule W-265 final report is currently in the process of being printed and bound. The District expects to submit this report to the Commission in April.

NRC Question 3:

A) What material is in weld metal block 5 and block 3 heat affected zone?

B) Were they fabricated from materials used in making of the Fort Calhoun beltline?

C) Are there base line tests data for these materials?

Responses:

A) The material in weld block 5 and block 3 heat affected zone is material fabricated as part of the Fort Calhoun reactor vessel materials irradiation surveillance program which was archived by the District. Further information concerning this material is contained in Reference 1.

B) These materials were fabricated from materials used in making the beltline of the Fort Calhoun reactor vessel.

C) There are baseline test data for these materials. This information is contained in Reference 2. In addition, material from the manufacturer of these capsules was retained for further benchmarking.

NRC Question 4: What is the ΔT between (the reactor vessel) metal 1/4T and T_w and 3/4T and T_w at (a heatup rate of) 100°F/hour?

Response: The ΔT between 1/4T and T_w for the heatup curve was zero. For the case of heatup, the thermal stresses are opposite in sign from the pressure stresses. Therefore, it is conservative, when calculating K_{IR}, to assume an isothermal heatup (K_{IT} equal to zero). A correction from the 1/4T location to the reactor vessel fluid is not made for the heatup case because there is no thermal gradient through the reactor vessel wall for the isothermal condition. If thermal stresses were taken into consideration for heatup (non-isothermal), the reactor vessel fluid temperature would be higher than the reactor vessel 1/4T wall temperature because the vessel wall lags the fluid temperature

during a change in fluid system temperature. The curves are based on an isothermal heatup and are found to be conservative as compared to similar heatup curves that account for thermal stresses and a fluid temperature correction due to the temperature gradient in the reactor vessel wall.

The ΔT between 3/4T and T_W was not calculated because the 3/4T location was not considered in the generation of the heatup curves. The basis for the Fort Calhoun Station Technical Specifications is a crack sized in accordance with Section III, Appendix G, of the ASME Code which extends from the inner surface to the 1/4T location. Therefore, calculations were not performed at the 3/4T location.

NRC Question 5: What is the RT_{NDT} at the 3/4T Location?

Response:

Basis	Fluence* (n/cm^2)	RT_{NDT}** (°F)
BOC to EOC7	1.5×10^{18}	174
BOL to 8.5 EFPY assuming 30% fluence reduction following Cycle 7	2.0×10^{18}	185

* Peak fluence axial and azimuthal locations

** Adjust $RT_{NDT}(°F) = -56 + 283 (\Phi/10^{19})^{0.194} + 34$

The 1/4T fluence and RT_{NDT} are discussed in the basis section of the proposed Technical Specification.

NRC Question 6: The 2/14/84 application references an end of life vessel ID fluence of 3.6×10^{19} n/cm^2 on pages 2-4, 2-5, and 2-6. However, the justification contained in Attachment B uses 4.8×10^{19} n/cm^2. In addition, the licensee answers dated 1/27/84 to questions posed in another license ammendment request (R. V. Surveillance Capsules) used 4.8×10^{19} n/cm^2. Which fluence is correct?

Response: Both values are correct as they are described. The 4.8×10^{19} n/cm^2 value represents the end of life (EOL) fluence based on a linear extrapolation of the BOL to EOC7 fluence calculated for the W-265 surveillance capsule removed at the end of Cycle 7 (EOC 7). Data from this capsule indicated a reactor vessel wall fluence of 0.88×10^{19} n/cm^2. Beginning with Cycle 8, the District implemented a low leakage fuel management program which guarantees a vessel flux reduction of at least 30%. Therefore, based on an EOC7 fluence distribution resulting from standard fuel management, and a 30% flux reduction resulting from low leakage fuel management for Cycle 8 through EOL, the

EOL fluence is projected to be 3.6×10^{19} n/cm^2. Thus, the EOL fluence used in the Technical Specifications is correct. The 4.8×10^{19} n/cm^2 EOL fluence is also correct, as described, but does not reflect the EOL fluence based on the current fuel management strategy.

rh/J

REFERENCES

1. TR-0-MCM-001, Revision 1, Evaluation of Irradiated Capsule W-225, August, 1980.

2. TR-0-MCD-001, Evaluation of Baseline Specimens, March, 1977.