Omaha Public Power District
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October 28, 1988
LIC-88-950

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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References:
1. Docket No. 50-285
2. Letter from OPPD (R. L. Andrews) to NRC (Document Control Desk) dated March 14, 1988 (LIC-88-145)
3. Amendment No. 114 to Fort Calhoun Station Technical Specifications, dated June 23, 1988
4. NRC Generic Letter 88-11 - "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations", dated July 12, 1988
5. Letter from OPPD (R. L. Andrews) to NRC (Document Control Desk), dated December 21, 1987 (LIC-87-692)

Gentlemen:

SUBJECT: Response to NRC Generic Letter 88-11: NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations

Omaha Public Power District (OPPD) was requested by Generic Letter 88-11 to submit a technical analysis to predict the effect of neutron radiation on reactor vessel materials by utilizing the methods contained in Revision 2 of Regulatory Guide 1.99. OPPD submitted in Reference 2, the Fort Calhoun Technical Specification pressure-temperature (P-T) limits which were reanalyzed using the "Draft" version of Regulatory Guide 1.99, Revision 2. The results were reviewed and approved by the NRC with the issuance of Amendment No. 114 (Reference 3) and are applicable through 14 Effective Full Power Years (EFPY).

The RT_{NDT} shift reported in Reference 2 was 285°F, including the 2σ uncertainty, at the 1/4t location. The corresponding reactor vessel inner surface fluence was 1.21×10^{19} n/cm^2. Since the fluence attenuation equation in the published Revision 2 changed from the "Draft" Revision 2, the RT_{NDT} shift was reanalyzed. The resulting shift at the 1/4t location was also calculated to be 285°F, including the 2σ uncertainty, and, therefore, the current Technical Specification P-T curves remain bounding. The Technical

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Specifications will, however, need to be revised to incorporate the new fluence attenuation equation in the RT_{NDT} shift trend curve, Technical Specification Figure 2-3. Also, the current Technical Specifications state that the reactor vessel inner surface fluence which corresponds to 14 EFPY is 1.4×10^{19} n/cm^2. The revised fluence is 1.21×10^{19} n/cm^2, which is the fluence that was used to calculate the 285°F RT_{NDT} shift included in the current Technical Specifications. OPPD will submit these changes by December 30, 1988.

Due to the increased shift of RT_{NDT} caused by the use of Regulatory Guide 1.99 Revision 2, as opposed to methods previously used, OPPD has experienced a decrease of the operating window which is common to many plants under the new regulation. Fort Calhoun Station utilizes variable setpoint PORVs in its Low Temperature Overpressure Protection (LTOP) system, and therefore the use of Revision 2 will not severely impact the operability of the plant. However, OPPD has awarded a contract for an LTOP analysis to be done by Combustion Engineering to review system hardware and operating procedures to allow for setpoint changes which could reduce the likelihood of LTOP challenges for the Fort Calhoun Station.

Pressurized Thermal Shock (PTS) calculations were repeated to determine RT_{PTS} using Revision 2 and submitted in Reference 5. Based on these calculations, Fort Calhoun will reach the PTS screening criteria before the expiration date of the operating license if Revision 2 is included in the PTS Rule. Therefore, the following actions are being taken by OPPD:

1) A DOT 4.3 calculation is being performed by Combustion Engineering for the Cycle 10 flux distribution so that credit may be taken for flux reduction to the beltline regions of the reactor vessel from the Cycle 10 extreme low leakage fuel management.

2) Performance of a Scoping Risk Assessment and/or a Regulatory Guide 1.154 analysis is being pursued for the Fort Calhoun Station. This program is tentatively scheduled to begin in 1989 and finish in 1991.

3) An extreme low radial leakage fuel management study is being planned which will further reduce the fluence to the critical beltline region materials.

4) OPPD is actively participating in the EPRI Reactor Vessel Embrittlement Management Program which is aimed at resolving reactor vessel embrittlement issues.

Based on the discussion above, OPPD believes it has taken the appropriate actions to fulfill the requirements of Generic Letter 88-11. If you have any questions concerning this matter, please contact us.

Sincerely,

K. J. Morris
Division Manager
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KJM/rh

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