Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

November 15, 1999
LIC-99-0107

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

References:
1. Docket No. 50-285
2. Operating License DPR-40 Amendment No. 158
3. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated June 23, 1993 (LIC-93-0119)
4. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated August 12, 1993 (LIC-93-0200)
5. Letter from OPPD (S. K. Gambhir) to NRC (Document Control Desk) dated January 30, 1998 (LIC-98-0009)

SUJECT: Supplemental Information for Application for Amendment of Operating License

Omaha Public Power District (OPPD) submitted the Reference 5 *Application for Amendment of Operating License* in order to delete Section 3.E, *License Term*, from the Fort Calhoun Station (FCS) Unit No. 1 Operating License No. DPR-40. This application included an updated fluence analysis (Westinghouse calculation SE-REA-95-003, *Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel*, dated November 1995). As noted in the Discussion and Justification portion of the application, calculation SE-REA-95-003 indicates that the adjusted reference temperature RT_{PTS} at the end of the current license term is 265.8°F, which is within the pressurized thermal shock screening criteria value of 270°F.

The Westinghouse fluence methodology for FCS included use of a plant-specific bias in calculating the projected RT_{PTS}. However, OPPD has received comments from the NRC Staff indicating that use of plant-specific bias may not be acceptable. Therefore, OPPD proposes to substitute for the plant-specific bias the application of Position 2.1 of Regulatory Guide 1.99. This revised methodology for calculating RT_{PTS} is documented in report CEN-636, Rev. 0, from ABB Combustion Engineering Nuclear Power, *Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials – Basis for Prediction of RT_{PTS} for the Fort Calhoun RPV*. Using the methodology based on Regulatory Guide 1.99 in this report, the revised projected RT_{PTS} at the end of the current license term is 253.8°F, well within the pressurized thermal shock screening criteria value of 270°F.

Accordingly, this letter provides a revised Discussion and Justification section, as well as the aforementioned ABB report, as supplemental information relative to Reference 5, Attachment B. (Approval of Amendment 184 resulted in re-lettering of the *License Term* condition from 3.E to 3.D and succeeding conditions to 3.E and F; the new Discussion and Justification section incorporates this change.) ABB Report CEN-636 is intended to be a new Attachment D to Reference 5.

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The Basis for No Significant Hazards Consideration and the deletion of the *License Term* condition previously provided in Reference 5 do not require a technical change. As previously noted, however, administrative re-lettering of affected License Conditions is required.

OPPD requests expeditious review and approval of this proposal by January 14, 2000, which will result in an NRC-approved method for calculating RT_{PTS} for FCS. This approval is necessary for decisions related to operating license renewal activities.

Please contact me if you have any questions.

Sincerely,

S. K. Gambhir
Division Manager
Nuclear Operations

TCM/tcm

Attachments

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
B. E. Casari, Director - Environmental Health Division, State of Nebraska
Winston & Strawn

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

S. K. Gambhir, being duly sworn, hereby deposes and says that he is the Division Manager-Nuclear Operations of the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached supplemental information concerning the Application for Amendment dated January 30, 1998 concerning deletion of Section 3.E, *LICENSE TERM* of Facility Operating License No. DPR-40; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information, and belief.

S. K. Gambhir
S. K. Gambhir
Division Manager
Nuclear Operations

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 15th day of November, 1999.

Julie B Kuhr
Notary Public

GENERAL NOTARY-State of Nebraska
JULIE B. KUHR
My Comm. Exp. April 4, 2000

LIC-98-0009
Attachment B

Revised Discussion and Justification

DISCUSSION AND JUSTIFICATION:

The Omaha Public Power District (OPPD) proposes to delete Section 3.D, *License Term* from Fort Calhoun Station (FCS) Unit 1 Operating License No. DPR-40.

The long-term load factor described in Section 3.D is used for calculation of the RT_{PTS} value to ensure that the screening criteria for reactor vessel integrity are not exceeded. The previous fluence analysis performed by Combustion Engineering (ABB/CE) used a 0.77 load factor in conjunction with the ENDF/B-IV cross section library. As shown in Attachment C, Westinghouse Electric Corporation (W) has completed an analysis (Westinghouse calculation SE-REA-95-003, *Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel*, dated November 1995) to update the ABB/CE calculation.

In the updated analysis, the long-term load factor was increased from 0.77 to 0.85 to reflect improvement in FCS Unit 1 operating efficiency. The updated analysis also used the ENDF/B-VI cross section library with updated uncertainties as required by Operating License No. DPR-40, Section 3.D. The neutron fluence calculations are carried out using forward and adjoint formulations in r,θ geometry of the two dimensional Discrete Ordinates Transport (DOT) code. The anisotropic scattering is treated with a P_3 expansion of the scattering cross section and the angular discretization is modeled with a S_8 order of quadrature. The actual core power distribution and neutron source distributions from 14 cycles of operation (13.6 Effective Full Power Years) were utilized, which included the spectral changes due to plutonium accumulation. The BUGLE-93 cross section library which is based on the data set of the Evaluated Nuclear Data File/B-VI (ENDF/B-VI) was used. The Westinghouse DOT code was benchmarked to the ENDF/B-VI cross sections using the Poolside Critical Assembly (PCA) simulator experiment at the Oak Ridge National Laboratory (ORNL), surveillance capsule and cavity dosimetry measurements.

The results of these fluence evaluations demonstrate that the best estimate fast neutron exposure of the pressure vessel can be determined with a 1σ uncertainty of ±13% for Φ (E>1.0MeV), ±19% for Φ (E>0.1MeV) and ±14% for dpa. These uncertainties are within the ±20% guidelines contained in Draft Regulatory Guide, DG-1053, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*.

The methodology used, as summarized above, is the same as the neutron fluence calculation section of WCAP-14040, Revision 1, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Curves* (TAC# M91749). Application of the exposure methodology to the FCS reactor vessel indicates that at the conclusion of Cycle 14 the critical weld material (i.e., weld 3-410) had accumulated a maximum unbiased fast neutron fluence (E>1.0MeV) of 1.057E19 n/cm^2 and had reached a corresponding RT_{PTS} value of 244.6°F based on the correlations provided in Regulatory Guide 1.99, Revision 2.

DISCUSSION AND JUSTIFICATION: (Continued)

Based on the use of low leakage fuel management as embodied in the design of FCS Operating Cycles 15 and 16, including actual power generation through Cycle 17 to date and projections through the last operating cycle in which Operating License DPR-40 expires, the critical weld material will have accrued a maximum fast neutron fluence of 1.728E19 n/cm^2. This number represents the unbiased fluence value used in Westinghouse calculation SE-REA-95-003, and was derived by dividing the EOL (30 EFPY) fluence number in Table 6.3-1 of SE-REA-95-003 (1.51E19) by 0.874, which was the bias used to create the table. This unbiased value does not credit FCS-specific surveillance data or a derived "fleet bias." An ABB Combustion Engineering Nuclear Power Report, *CEN-636, Rev. 0, Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials – Basis for Prediction of* RT_{PTS} *for the Fort Calhoun RPV,* outlines the evaluation of surveillance data and justifies reduction of the 10CFR 50.61 margin term to 44°F (see Attachment D). Using a chemistry factor of 231.06°F for the limiting 3-410 axial weld, the 1.728E19 n/cm^2 unbiased fluence value, the 44°F margin term, an $RT_{NDT}(0)$ value of –56°F, and a long term load factor of 0.85 results in an RT_{PTS} value of 253.8°F. This value is well within the PTS screening criteria of 270°F.

In accordance with 10 CFR 50.61, this assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon request for a change in the expiration date of the facility. Thus, Section 3.D can be deleted from Operating License No. DPR-40 based upon the analysis contained in Attachment C and the fact that Section 3.D is redundant to 10 CFR 50.61 requirements.

LIC-99-0107
Attachment

LIC-98-0009
New Attachment D

ABB Combustion Engineering Nuclear Power Report
CEN-636, Rev. 0
Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials – Basis for Prediction of RT_{PTS} for the Fort Calhoun RPV