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LIC-92-175R

U. S. Nuclear Regulatory Commission
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References:
1. Docket No. 50-285
2. Letter from LeBoeuf, Lamb, Leiby & MacRae (Attorneys for OPPD) to NRC (H. R. Denton) dated July 17, 1986
3. Letter from OPPD (R. L. Andrews) to NRC (Document Control Desk) dated December 21, 1987 (LIC-87-0692)
4. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988
5. Letter from OPPD (K. J. Morris) to NRC (Document Control Desk) dated May 18, 1989 (LIC-89-284)
6. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated December 16, 1991 (LIC-91-327R)
7. Letter from OPPD (W. G. Gates) to NRC (Document Control Desk) dated March 17, 1992 (LIC-92-093R)

Gentlemen:

SUBJECT: Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock (PTS) Events, 10 CFR 50.61(b), for Fort Calhoun Station (FCS)

In accordance with the Omaha Public Power District (OPPD) commitment of Reference 7, this letter is submitted to notify the NRC that a detailed fluence analysis for FCS has been completed. A schedule for implementation of further improvements to the flux reduction program has also been developed and is discussed later in this letter. These actions have been taken in accordance with 10 CFR 50.61(b)(4) and OPPD commitments made in Reference 6.

The FCS operating license currently expires in June 2008. Based upon conservative fluence extrapolations, Reference 6 concluded that the PTS screening criterion would not be reached until the year 2009. This follow-up submittal is required per 10 CFR 50.61 due to OPPD's request (Reference 2) for a five year extension of the license until the year 2013. As noted in Reference 5, OPPD has been closely monitoring the regulatory issues associated with mitigating PTS events.

Initial PTS flux reduction efforts included the implementation of a low radial leakage fuel management strategy in 1983 (Cycle 8) and improvements have continued during subsequent cycles. For example, during Cycle 10, FCS utilized a non-optimized extreme low radial leakage fuel management strategy. FCS has implemented an optimized extreme low radial leakage fuel management strategy for the current fuel cycle (Cycle 14).

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Core loading for Cycle 14 includes the use of hafnium flux suppression rods in twelve peripheral fuel assemblies located near the limiting reactor vessel welds, four natural uranium fuel assemblies located near the limiting reactor vessel welds and integral fuel burnable absorbers. This type of core loading scheme and continued improvements in core periphery flux reduction are the basis for OPPD's fuel management program.

To comply with the latest 10 CFR 50.61 requirements the Reference 6 assessment used best estimate neutron fluence values for Cycles 11 through 13. The accumulated surface fluence was estimated at 9.03 x 10^{18} n/cm^2 for the 60° and 300° 3-410 (lower course) longitudinal weld locations. The 180° 3-410 longitudinal weld was determined to be less limiting than the 60° and 300° 3-410 longitudinal welds. Future cycles were assumed to use core loading patterns equivalent to Cycle 14.

Cycle 14 fluence data did not exist prior to the Reference 6 submittal date. Therefore, Reference 6 conservatively assumed that Cycle 10 fluence data would be representative of fuel cycles following Cycle 13. Based upon a previous Cycle 10 DOT 4.3 analysis performed by ABB/Combustion Engineering (ABB/CE), the incremental fluence per effective full power year (EFPY) was estimated to be 0.41 x 10^{18} n/cm^2. Using best estimate fluence values through Cycle 13 combined with extrapolations based on the Cycle 10 DOT 4.3 analysis, it was determined in Reference 6 that the PTS screening criterion of 270°F would be reached in the year 2009.

To obtain a current fluence evaluation of the FCS reactor vessel and update future projections, OPPD recently contracted with ABB/CE to perform this work. Detailed flux calculations were performed using the DOT 4.3 computer code to determine the fast neutron fluence for Cycles 11-13 and to project the impact of implementing the Cycle 14 optimized extreme low radial leakage fuel management strategy. OPPD has received the final results from the new ABB/CE DOT 4.3 analysis as described below.

The new analysis determined that the accumulated fast neutron fluence at the reactor vessel surface for the 60° and 300° 3-410 longitudinal weld locations is 9.57 x 10^{18} n/cm^2 through the end of Cycle 13. The 180° 3-410 longitudinal weld was again determined to be less limiting than the 60° and 300° 3-410 longitudinal welds. Based upon cycle specific DOT 4.3 calculations for Cycles 11 through 13 and design DOT 4.3 calculations for Cycle 14, the projected accumulated fluence at the end of Cycle 14 is 1.00 x 10^{19} n/cm^2. The projected Cycle 14 incremental fluence per EFPY for the 60° and 300° 3-410 longitudinal weld locations is 0.34 x 10^{18} n/cm^2.

As a result of this new analysis, OPPD has extended its estimate for reaching the PTS screening criterion of 270°F. Using the best estimate fluence values from the new ABB/CE DOT 4.3 analysis and an estimated load capacity factor of 0.77, the PTS screening criterion of 270°F is now estimated to occur late in the year 2010 or early 2011.

OPPD has projected values for RT_{PTS} for FCS reactor vessel beltline materials based upon Reference 3 data and on the accumulated fluence data through Cycle 13 from the new ABB/CE DOT 4.3 analysis. Table 1 (attached) summarizes the chemistry factors and adjusted reference temperatures of nil ductility transition (ART_{NDT}) obtained through the application of 10 CFR 50.61(b)(2)(iv). The chemistry factors remain unchanged from those previously submitted in Table 2 of Reference 3.

Since the new ABB/CE DOT 4.3 analysis predicts that the PTS screening criterion of 270°F will be exceeded prior to the year 2013 (final year of the requested operating license extension), OPPD plans to continue evaluating additional flux reduction options. Currently identified flux reduction options include the use of stainless steel replacement pins in peripheral fuel assemblies and the use of full core fuel management rather than quarter core reflective fuel management to minimize thermal margin losses. Prior to Cycle 15 startup (currently scheduled for November 1993), OPPD intends to implement additional flux reduction fuel management strategies. Therefore, OPPD requests that the five year license extension application (TAC 82834) remain active. Additional information to justify the proposed license extension will be provided within ninety (90) days following Cycle 15 startup.

If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Division Manager
Nuclear Operations

WGG/sel

Attachment

c: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator, Region IV
R. P. Mullikin, NRC Senior Resident Inspector
D. L. Wigginton, NRC Senior Project Manager
S. D. Bloom, NRC Project Engineer

Table 1
ART_{NDT} for Fort Calhoun Beltline Materials

Using Reg. Guide 1.99, Rev. 02, and 10 CFR 50.61
60/300 Degree Angle, DOT 4.3 Results Incorporated

Weld Seam	Cu (w/o)	Ni (w/o)	Chemistry Factor	ART_{NDT}(°F) Cycle 14	ART_{NDT}(°F) 2008	ART_{NDT}(°F) 2013
2-410 (longitudinal)	0.17	0.17	89.45	99.42	107.66	109.85
3-410 (longitudinal)	0.22	1.02	234.50	244.4	266.02	271.78
8-410 (circumferential)	0.21	0.73	185.45	195.40	212.47	217.02
9-410 (circumferential)	0.21	0.74	187.10	197.05	214.28	218.86
D-4802 (intermediate shell-plate)	0.12	0.56	82.20	118.18	125.75	127.76
D-4812 (lower shell-plate	0.12	0.60	83.00	118.98	126.62	128.65

Regulatory Guide 1.99, Rev. 02 Equation:

ART_{NDT} = Adjusted Reference Temperature of Nil Ductility Transition = $I + M + (CF)\, f^{(0.28-0.10\log f)}$

Where;

CF = Chemistry Factor determined from Tables in Regulatory Guide 1.99 and 10 CFR 50.61

f = Calculated value of neutron fluence at the reactor vessel/clad interface divided by 10^{19}.

For Weld Material:

I = Generic mean value of initial reference temperature = −56°F for welds made with Linde 1092 and 124 fluxes.

M = Margin to cover uncertainties in initial RT_{NDT} = 66°F since generic value of I was used.

For Plate Material:

I = Initial reference temperature of irradiated material as defined in the ASME Code = −12°F for reactor vessel beltline plate material.

M = Margin to cover uncertainties in initial RT_{NDT} = 48° F since a measured value of I was used.

The proposed PTS criteria applied to the vessel ID for longitudinal weld seams and plate material is RT_{PTS} = 270°F and for circumferential weld seams is RT_{PTS} = 300°F