



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 1140 FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO.: 50-285

1.0 INTRODUCTION

In a letter from R.L. Andrews to USNRC dated February 19, 1988, the Omaha Public Power District (the licensee) requested to amend Facility Operating License No. DPR-40 for Fort Calhoun Station, Unit No. 1 (Fort Calhoun). The amendment proposes to reduce the effective period of the reactor coolant system pressure-temperature limits from 15 effective full power years (EFPY) to 14 EFPY. The revised pressure-temperature limits are contained in Section 2.1.2 of the Fort Calhoun Technical Specifications. The change in pressure-temperature limits is based on applying the method of predicting neutron irradiation that is proposed in Regulatory Guide 1.99, Revision 2 and on a revised estimate of the integrated fast neutron ($E \geq 1.0$ MeV) flux at the reactor vessel's inside surface. The revised integrated fast neutron flux for the Fort Calhoun reactor vessel was documented in a letter from R.L. Andrews to USNRC dated December 21, 1987.

2.0 DISCUSSION

Neutron Irradiation Flux

The staff approved the methodology and the prior neutron flux estimates for operation of Fort Calhoun to 2008 (CP+40). The staff's review of the prior neutron flux estimate is contained in a letter from W.A. Paulson (NRC) to R.L. Andrews (the licensee) dated March 5, 1987. The prior neutron flux estimate was based on the accumulated neutron irradiation for cycles 1 through 7, in which cores were loaded using standard out-in-in fuel management. Fuel cycles 1 through 7 accumulated a neutron fluence of $.88 \times 10^{19}$ n/cm² in 5.92 effective full power years (EFPY) of operation.

The licensee initiated low leakage loading patterns with cycle 8 which significantly reduced the neutron leakage. The same methods were applied for the estimation of the cycle 8 flux as for cycles 1-7; therefore, this estimate is acceptable. The cycle 8 results were used to estimate the fluence through 30 EFPY. The neutron fluence for cycles 8 through 30 EFPY was estimated using a .77 load factor. These projections are conservative, because the cycles 9, 10 and subsequent cycle leakages will be lower than cycle-8 leakage.

The estimated irradiation to the year 2008 is 26 EFPYs assuming a .77 load capacity factor and 30 EFPYs to the year 2013. The staff customarily assumes an .80 load factor. However, Fort Calhoun's historical record indicates a lower load factor. Therefore, the projection of .77 is acceptable, since the licensee verbally committed to notify the staff if the load capacity factor for the remaining duration of its license is projected to exceed .77.

Pressure-Temperature Limits: Pressure-Temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR Part 50, which became effective on July 16, 1983. Pressure-Temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR Part 50 are dependent upon the initial reference temperature (RT_{NDT}) for the limiting materials in the beltline, and closure flange regions of the reactor vessel and the increase in reference temperature resulting from neutron irradiation damage to the limiting beltline material. The Fort Calhoun reactor vessel was procured to earlier ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each vessel material. Appendix G, 10 CFR Part 50 indicates that vessels fabricated to earlier ASME Code requirements must provide supplementary data and analyses to demonstrate that the vessel material's fracture toughness data are equivalent to that specified in later editions of the ASME Code.

The Fort Calhoun reactor vessel was fabricated by Combustion Engineering (CE) The beltline was fabricated by welding plates together and the closure flange regions were fabricated by welding plates and forgings together.

The limiting materials in the beltline and closure flange regions were determined to have initial RT_{NDT} values of $-56^{\circ}F$ and $+50^{\circ}F$, respectively. The limiting material in the beltline region is the weld metal in longitudinal seam 3-410. The initial RT_{NDT} value for this weld metal corresponds to the value used by the staff in its Pressurized Thermal Shock evaluation, which is reported in the letter to the licensee dated March 5, 1987. The fracture toughness test data from the Fort Calhoun reactor vessel plate and forgings are reported in USAR Table 4.5-1. Based on the data reported in this table, the initial RT_{NDT} value of $+50^{\circ}F$ for the closure flange region is conservative.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated using the formulae in Regulatory Guide (RG) 1.99, Revision 2. Revision 2 of this guide has been approved and is awaiting publication as a final guide. This guide recommends two methods of estimating the increase in RT_{NDT} resulting from neutron irradiation. The formulae in the guide are to be used for determining the increase in RT_{NDT} when surveillance data is unavailable. To use these formulae, the amounts of copper and nickel in the limiting beltline weld metal must be known. When surveillance data becomes available, the guide recommends a "least square method" for extrapolation of the increase in RT_{NDT} . To extrapolate surveillance data, the amounts of copper and nickel in the limiting beltline weld metal and surveillance weld metal must be known. Fort Calhoun has surveillance data, which were reported to the staff in letters from W.C. Jones dated April 25, 1984 and January 23, 1981. The amounts of copper and nickel in the limiting beltline weld metal have been precisely determined, but the amounts of copper and nickel in the surveillance

weld has not been determined as accurately. Hence, we cannot completely evaluate the surveillance data. Since RG 1.99, Revision 2 permits the use of the formulae in the guide until surveillance data becomes available, the use of the formulae in the guide to calculate pressure-temperature limits is acceptable at this time.

To confirm that the pressure-temperature limits proposed by the licensee will meet the safety margins of Appendix G, 10 CFR Part 50 for the proposed operating period, the staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 and the formulae in RG 1.99, Revision 2 to calculate neutron irradiation damage. The staff's calculation includes the neutron flux estimates reported in the licensee's submittal dated March 5, 1987. Our calculations confirm that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR Part 50 for length of time proposed by the licensee, 14 EFPY.

FINDINGS

1. The initial RT_{NDT} for the limiting beltline and closure flange materials of -56°F and $+50^{\circ}\text{F}$, respectively, are acceptable for use in calculating the reactor vessel's pressure-temperature limits.
2. The revised neutron flux estimates and chemical compositions for beltline materials that are documented in the licensee's letter dated December 21, 1987 are acceptable for determining irradiation damage to reactor vessel materials and may be used for calculating pressure-temperature limits. Since the neutron flux estimates are based on a capacity factor of .77, the licensee must inform the staff when the projected or actual load capacity factor exceeds .77 and indicate whether new pressure-temperature limits are necessary.
3. Since the chemical composition of the surveillance weld metal has not been precisely determined, the formulae in RG 1.99, Revision 2 may be used to calculate pressure-temperature limits.
4. Based on the above conclusions and the staff's confirmation calculations, the Fort Calhoun Pressure-Temperature Limits are acceptable for 14 EFPY and Facility Operating License No. DPR-40 may be amended as requested in the licensee's letter dated February 19, 1988.

Recommendation

The licensee should precisely determine the amount of copper and nickel in its surveillance weld metal. Based on the chemical composition of its surveillance weld metal and beltline weld metal, and the test data from the Fort Calhoun surveillance program, the licensee should evaluate the Fort Calhoun Pressure-Temperature Limits in accordance with Section C.2.1 of RG 1.99, Revision 2.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 28, 1988

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