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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-40

## OMAHA PUBLIC POWER DISTRICT

## FORT CALHOUN STATION, UNIT NO. 1

## DOCKET NO. 50-285

## 1.0 INTRODUCTION

By letter dated November 27, 1991, as supplemented February 12, March 6, and March 10, 1992, Omaha Public Power District (OPPD) submitted an application for an amendment to Facility Operating License No. DPR-40 that would modify the Fort Calhoun Station, Unit 1, Technical Specifications (TS) to support Cycle 14 operation. Specifically, the negative limit for the moderator temperature coefficient (MTC) would be changed to $-3.0 \times 10^{-4}$ delta rho/$^{o}$F from $-2.7 \times 10^{-4}$ delta rho/$^{o}$F. Other TS changes affected by the Cycle 14 reload are accounted for by the cycle/reload-specific parameter limits specified in the Core Operating Limits Report (COLR) which have been established in accordance with NRC approved methodologies. The supplemental submittals provided additional information and clarifications, which are within the scope of the initial notice and did not affect the initial no significant hazards determination.

## 2.0 EVALUATION

The fuel system, nuclear, thermal-hydraulic and safety analyses evaluations are presented herein. An evaluation of the proposed TS change is also presented.

### 2.1 Reload Methodology

In their amendment application, OPPD has identified reload methodology topical reports (Ref. 1, 2, 3) to be used for Cycle 14. This methodology has been reviewed and approved by the NRC in previous licensing actions.

### 2.2 Fuel System Design

The Cycle 14 core will contain reload fuel assemblies (Batch R) supplied by Westinghouse. Most of the Batch R assemblies will use an integral fuel burnable absorber (IFBA) instead of the traditional fuel displacing poison rods. These IFBA rods consist of fuel pellets treated with an electrostatically applied zirconium-diboride coating which surrounds the fuel pellet circumference. The use of IFBA rods has been approved by the NRC for numerous operating pressurized water reactors (PWRs) and is acceptable for use in Fort Calhoun.

The mechanical design for the Batch R reload fuel (Ref. 4) is slightly different than the previous cycle Batch P reload fuel supplied by Combustion Engineering (CE). The NRC has evaluated the Batch R mechanical design and has concluded that the Batch R fuel is mechanically, neutronically, thermally, and hydraulically compatible with the CE fuel remaining in the Cycle 14 core (Ref. 5). The staff has previously agreed that an analysis to show that compliance with the fuel assembly structural criteria in Appendix A of Standard Review Plan (SRP) Section 4.2 (Ref. 6) for the design seismic event is outside the scope of design basis for the Fort Calhoun Station and that an unreviewed safety question does not exist for a mixed core with respect to the design seismic event (Ref 7).

Based on the above, the NRC finds the fuel system design for Cycle 14 acceptable.

### 2.3 Nuclear Design

#### 2.3.1 Core Characteristics

The Cycle 14 fuel management uses an extreme low-radial leakage design with twice-burned fuel assemblies predominately loaded on the periphery of the core with hafnium flux suppression rods inserted into the guide tubes of selected peripheral fuel assemblies adjacent to the reactor vessel limiting welds. The hafnium flux suppression rods are similar to the part length poison rods previously used in Cycle 10 and are composed of hafnium metal over the full length of the active fuel. Since these rods are stationary, they are not subject to the safety concern of swelling and subsequent inability to fully insert associated with movable hafnium control rods. This fuel pattern is utilized to minimize the fluence to the pressure vessel welds and achieve the maximum in neutron economy. However, this fuel arrangement usually results in higher radial peaking factors than the standard out-in-in pattern. Therefore, the enrichment of the fuel pins adjacent to the fuel assembly water holes has been lowered in an attempt to reduce the maximum peaking factors for Cycle 14.

The Cycle 14 loading pattern incorporates 52 fresh Batch R fuel assemblies of which 48 contain the IFBA pellet design described in Section 2.2. The remaining four Batch R assemblies contain fuel rods that are loaded with naturally enriched uranium and also placed in locations near the limiting welds. All of these 48 assemblies employ intra-assembly uranium enrichment splits. Batches R2 through R5 contain a high pin U-235 enrichment of 4.00 weight percent (w/o) and a low pin U-235 enrichment of 3.50 w/o. Batches R6 and R7 contain a high pin U-235 enrichment of 3.75 w/o and a low pin enrichment of 3.25 w/o. Forty twice burned N assemblies are being returned to the core, along with 40 once burned P assemblies. One once burned M assembly, which was discharged into the spent fuel pool at the end of Cycle 12, will be returned to the core and used as the center assembly. This assembly arrangement will produce a Cycle 14 loading pattern with a cycle energy of 14,000 MWD/MTU with an additional 1,000 MWD/MTU of energy in a coastdown mode if required. The Cycle 14 core characteristics have been examined for a

Cycle 13 termination burnup between 14,250 MWD/MTU and 15,250 MWD/MTU. The Cycle 14 loading pattern is valid for any Cycle 13 endpoint between these values. Actual Cycle 13 termination burnup of 15,248.82 MWD/MTU was achieved on February 1, 1992, and limiting values based on this endpoint have been established for the safety analysis.

2.3.2 Moderator Temperature Coefficient

The TS require the MTC to be less positive than $+0.2 \times 10^{-4}$ delta rho/$^{o}$F for power levels at or above 80% of rated power and less positive than $+0.50 \times 10^{-4}$ delta rho/$^{o}$F for power levels below 80% of rated power. For Cycle 14, the end-of-cycle (EOC) hot full power MTC was predicted to be $-2.80 \times 10^{-4}$ delta rho/$^{o}$F including uncertainties. This value exceeds the Cycle 13 TS negative limit of $-2.70 \times 10^{-4}$ delta rho/$^{o}$F and, therefore, for Cycle 14 the negative limit of the MTC is revised from $-2.70 \times 10^{-4}$ delta rho/$^{o}$F to $-3.0 \times 10^{-4}$ delta rho/$^{o}$F. This new limit remains conservatively bounded by the steam line break cooldown curve used in the reference analysis (Cycle 8) as well as other cooldown transients. Since approved methods have been used and appropriate values incorporated in the safety analyses, the range of MTC TS limits for Cycle 14 is acceptable.

2.3.3 Control Requirement

The TS required shutdown margin remains at 4.0% delta k/k for Cycle 14. Based on this value and the available scram reactivity, including allowance for a maximum stuck control element assembly (CEA), and a calculational bias and uncertainty, OPPD has shown that sufficient excess reactivity exists between the available and the required scram reactivities for all Cycle 14 operating conditions. The staff has questioned the reduction in the calculational bias and uncertainty of the predicted scram worth used in previous cycles. This reduction is based on changes to the physics methods and codes originally described in Reference 8 as well as a new reevaluation of data by CE (Ref. 9).

The methodology changes which resulted in this update of the biases and uncertainties included the addition of the nodal expansion method (NEM) to the ROCS code, the use of anisotropic scattering and higher order interface currents in the DIT code, and the use of assembly discontinuity factors (ADFs) in ROCS. The first two changes have received previous approval by the NRC (Refs. 8 and 10). The use of ADFs improve the internal agreement between two existing modules of the approved code system (ROCS and DIT) and is widely used in the nuclear industry. It is, therefore, acceptable. The re-evaluation of biases and uncertainties used the same statistical methodology described in Reference 8 as well as an expanded data base which included recent reload cycles with low leakage and high burnup fuel management. As required by the NRC, a 95/95 tolerance limit is applied to the calculated results to assure that 95 percent of the calculated rod worths will be less than the "true" value with 95 percent confidence. In view of the small number of net (N-1) CEA worth measurements taken, the licensee chose to apply the larger bias and uncertainty associated with individual CEA bank worths to the (N-1) CEA worth. This is a conservative approach and is acceptable. The calculated net scram worth using these methodology changes and the associated bias and uncertainty have been shown by OPPD to be essentially the same as that calculated with the previous method and its associated bias and uncertainty (Ref. 14). The staff finds the revised CEA worth calculational bias and uncertainty acceptable.

#### 2.3.4 Power Distributions

The all-rods-out (ARO) planar radial power distributions at beginning-of-cycle (BOC), middle-of-cycle (MOC), and EOC have been calculated based on the high burnup end of the Cycle 13 shutdown window. This tends to increase the power peaking in the high power region of the core for Cycle 14. The calculated power densities also show that for both the departure-from-nucleate-boiling (DNB) and linear heat rate (LHR) safety and setpoint analyses in either the rodded or unrodded configurations, the power peaking values used are higher than those actually expected to occur at any time during Cycle 14.

Because of the methodology improvements and expanded data base discussed in Section 2.3.3, the power peaking factor calculational uncertainties have also been modified. As required in the NRC Safety Evaluation Report for ROCS (Ref. 8), the new biases and uncertainties are equivalent to those previously used in that the same conservative relationship is maintained between calculated and measured data. That is, a 95/95 tolerance limit is applied to the calculated results to assure that 95 percent of the calculated peaking factor will be greater than the "true" value with 95 percent confidence. The methods used to generate the new biases and uncertainties are the same as those therefore, acceptable.

The range of allowable axial peaking is defined by the limiting conditions for operation (LCOs) covering the axial shape index (ASI). The maximum three-dimensional or total peaking factor anticipated in Cycle 14 during normal base load, ARO operation at full power is 2.107, including uncertainty allowances. This is well below the values used in the CEA ejection accident analysis (2.52) and the large and small break loss of coolant accident (LOCA) analyses (2.545) and is, therefore, acceptable. Therefore, the radial peaking limits and the axial shapes are acceptable.

Based on the above, the NRC finds the nuclear design for Cycle 14 acceptable.

### 2.4 Thermal-Hydraulic Design

#### 2.4.1 Departure from Nucleate Boiling Ratio (DNBR) Analysis

The thermal-hydraulic analyses for Cycle 14 were performed using computer codes previously approved by the NRC for use by OPPD. As in previous cycles, the CETOP-D computer code was used in the setpoint analysis. However, for Cycle 14, the TORC code was used to calculate the minimum DNBR rather than the CETOP-D code. Both codes have been approved for use with the OPPD methods and, therefore, the use of the more accurate TORC code for DNBR analyses is acceptable.

The calculational factors, including the engineering heat flux factor, the engineering factor on hot channel heat input, rod pitch and clad diameter factor, were statistically combined with other uncertainty factors to arrive at the 95/95 confidence/probability DNBR design limit of 1.18. This statistical combination of factors has been approved by the NRC. This limit ensures that, with at least a 95% probability and at least a 95% confidence level, the limiting fuel pin will avoid DNB if the predicted minimum DNBR is not below the 1.18 limit.

2.4.2 Fuel Rod Bowing

The fuel rod bow penalty accounts for the adverse impact on minimum DNBR of random variations in spacing between fuel rods. Although Westinghouse has identified that the predicted amount of deflection does not require a DNB penalty to be applied to the Westinghouse fuel under Westinghouse analysis requirements (Ref. 4), the CE fuel bow DNBR penalty was applied to both the Westinghouse and CE fuel for Cycle 14. This is conservative and the methodology for determining this penalty was based on NRC approved methods.

Based on the above, the staff finds the thermal-hydraulic design for Cycle 14 acceptable.

2.5 Safety Analyses

OPPD has reviewed the parameters which influence the results of the transient and accident analyses for Cycle 14 to determine which, if any, would require reanalysis. With regard to non-LOCA safety analysis at 1500 MWt for Cycle 14, the design basis events (DBEs) were considered and a comparison of core parameters to bounding values was made. No reanalysis was performed for those DBEs in which the key transient input parameters were within the bounds of the reference cycle values (Ref. 11 and 12). For these DBEs, the results and conclusions presented in the reference cycle analysis remain valid for Cycle 14. All events were evaluated for an assumption of 6% steam generator tube plugging. This is acceptable since, currently, Fort Calhoun has 1.08% steam generator tubes plugged.

The NRC has reviewed the DBEs which were reanalyzed and the reasons for the reanalysis, the acceptance criteria used in judging the results, and the results obtained and finds them to be valid for Cycle 14 operation.

OPPD utilizes the CEA ejection accident analysis of the current fuel vendor, Westinghouse. This analysis methodology is documented in Ref. 1 and was performed by Westinghouse for Cycle 14. The results (Ref. 13) indicate that the radial average enthalpy of the hottest fuel pellet remains well below the 280 cal/gm limiting criterion for prompt fuel failure for both zero power and full power events. The peak reactor pressure during the event does not exceed 2750 psia (110% of design pressure) and therefore remains below the value that would cause stresses to exceed the emergency condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code. Clad failure is usually assumed for fuel rods that have a DNBR less than 1.18, the CE-1 critical heat flux correlation limit. However, instead of calculating an explicit number of rods which experience DNB, a conservatively bounding value of 10% was used to calculate offsite dose consequences. This is an acceptable approach and resulted in calculated thyroid and whole body doses well within the limits specified in 10 CFR Part 100, thus meeting the NRC radiological requirements.

2.6 Startup Tests

The startup testing program for Cycle 14 is identical to that used in Cycle 13 and is acceptable.

2.7 Technical Specification Changes

OPPD has proposed a change to the TS for Cycle 14. This change would increase the negative MTC limit in TS 2.10.2(3) from $-2.7 \times 10^{-4}$ delta rho/$^{o}$F to $-3.0 \times 10^{-4}$ delta rho/$^{o}$F. The NRC has reviewed this change and finds that it is properly incorporated in the supporting physics and safety analyses for Cycle 14 using the approved methods.

The NRC has reviewed the information presented in the Cycle 14 reload application and in the responses to requests for additional information. The NRC finds the proposed reload and the associated modified TS acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 exposure. 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 (57 FR 713). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

## 5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 REFERENCES

1. "Omaha Public Power District Reload Core Analysis Methodology Overview," OPPD-NA-8301-P, Revision 04, April 1991.

2. "Omaha Public Power District Reload Core Analysis Methodology Neutronics Design Methods and Verification," OPPD-NA-8302-P-A, Revision 02, April 1988.

3. "Omaha Public Power District Reload Core Analysis Methodology Transient and Accident Methods and Verification," OPPD-NA-8303-P Revision 03, March 1991.

4. "Westinghouse Reload Fuel Mechanical Design Evaluation for the Fort Calhoun Station Unit 1," WCAP-12977 (Proprietary), June 1991.

5. "Westinghouse Reload Fuel Design for Fort Calhoun," NRC Safety Evaluation, Memorandum from R. C. Jones to J. T. Larkins, dated January 27, 1992.

6. "USNRC Standard Review Plan," NUREG-0800, November 1988.

7. Letter from James R. Miller (NRC) to W. C. Jones (OPPD), dated August 29, 1983.

8. "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983.

9. "Physics Biases and Uncertainties," CE-CES-129, Rev. 1-P, August 1991.

10. "CE Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," CENPD-275-P, Revision 1-P-A, May 1988.

11. Fort Calhoun Station Updated Safety Analysis Report, dated July 1991

12. "Amendment to Operating License DPR-40, Cycle 11 License Application," Docket No. 50-285, dated May 4, 1987.

13. "Fort Calhoun Unit 1 Control Element Assembly Ejection Analysis Report," transmitted by letter from W. G. Gates (OPPD) to NRC, dated November 27, 1991.

14. "Additional Information Concerning Fort Calhoun Station Cycle 14 Reload," Letter from W. G. Gates (OPPD) to NRC, dated February 12, 1992.

15. "Additional Information Concerning Fort Calhoun Station Cycle 14 Reload Application," Letter from W. G. Gates (OPPD) to NRC, dated March 6, 1992.

Principal Contributor: L. Kopp, NRR/SRXB

Date: April 2, 1992