March 27, 2001

Mr. S. K. Gambhir
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station, FC-2-4 Adm.
Post Office Box 399
Hwy. 75 - North of Fort Calhoun
Fort Calhoun, NE 68023-0399

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: EXTENDING PRESSURE-TEMPERATURE (P-T) LIMITS (TAC NO. MB0322)

Dear Mr. Gambhir:

The Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 18, 2000.

The amendment revises Section 2.1.2, Figures 2-1A and 2-1B, and associated Bases of the Fort Calhoun Station Technical Specifications to extend existing pressure-temperature (P-T) curves from 20 effective full power years (EFPY) to 24.25 EFPY. Additionally, the amendment deletes Figure 2-3, "Predicted Radiation Induced NDTT Shift" and updates the fluence analysis for projecting RT_{NDT} at 24.25 EFPY.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/
L. Raynard Wharton, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 197 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 27, 2001

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Docket No. 50-285

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 197
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee) dated October 18, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 197 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 27, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
vii	vii
2-4	2-4
2-5	2-5
2-6	2-6
Figure 2-1A	Figure 2-1A
Figure 2-1B	Figure 2-1B
Figure 2-3	Figure 2-3
2-7a	2-7a

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

Amendment No. ~~116,126,131,141,161,170,~~
~~172,188,192~~, 197

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

(a) Figures 2-1A and 2-1B are valid for a fast neutron (E ≥ 1MeV) fluence of $1.50X10^{19}$ n/cm^2 which corresponds to 24.25 EFPY.

(b) The limit line on the figures shall be updated for a new integrated power period as follows: the total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated fast neutron exposure (E ≥1 MeV).

(c) The limit lines in Figures 2-1A and 2-1B shall be moved parallel to the temperature axis (horizontal) in the direction of increasing temperature a distance equivalent to the transition temperature shift during the period since the curves were last constructed. The boltup temperature limit line shall remain at 82°F as it is set by the NDTT of the reactor vessel flange and not subject to fast neutron flux. The lowest service temperature shall remain at 182°F because components related to this temperature are also not subject to fast neutron flux.

(d) The Technical Specification 2.3(3) shall be revised each time the curves of Figures 2-1A and 2-1B are revised.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operation.

During unit startup and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon allowable heatup/cooldown rates and cyclic operation.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III(2) of the ASME Code including Appendix G, Protection Against Nonductile Failure, and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. This ASME Code assumes that a crack 10-11/16 inches long and 1-25/32 inches deep exists on the inner surface of the vessel. Furthermore, operating limits on pressure and temperature assure that the crack does not grow during heatups and cooldowns.

The reactor vessel beltline material consists of six plates. The nilductility transition temperature (T_{NDT}) of each plate was established by drop weight tests. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lbs. absorbed energy and 35 mils lateral expansion for the longitudinal direction. NRC technical position MTEB-5-2 was used to establish a reference temperature for transverse direction (RT_{NDT}) of -12°F.

The initial RT_{NDT} value for the Fort Calhoun submerged arc vessel weldments was determined to be -56°F with a standard deviation of 17°F. By applying the shift prediction methodology of Regulatory Guide 1.99, Revision 2, a weld material adjusted reference temperature (RT_{NDT}) was established at 10°F based on the mean value plus two standard deviations. The standard deviation was determined by using the root-mean-squares method to combine the margin of 28°F for uncertainty in the shift equation with the margin of 17°F for uncertainty in the initial RT_{NDT} value.

Similar testing was not performed on all remaining material in the reactor coolant system. However, sufficient impact testing was performed to meet appropriate design code requirements(3) and a conservative RT_{NDT} of 50°F has been established.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the T_{NDT} with operation. The techniques used to predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Reference 5 with the result that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73×10^{19} n/cm^2, including tolerance at the inside surface of the critical reactor vessel beltline weld material, over the 40 years design life of the vessel.

Since the neutron spectra and the flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured transition shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calibrated azimuthal neutron flux variation. The maximum integrated fast neutron ($E \geq 1$ MeV) exposure of the reactor vessel at the critical reactor vessel beltline location including tolerance is computed to be 1.73×10^{19} n/cm^2 at the vessel inside surface for 40 years operation at

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

1500 MWt and 85% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252°F, including margin, using the shift prediction equation of Regulatory Guide 1.99, Revision 2. The actual shift in T_{NDT} will be re-established periodically during the plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Section 4.5.3 and Figure 4.5-1 of the USAR. To compensate for any increase in the T_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. Analysis of the three removed irradiated reactor vessel surveillance specimens[8,9 and 10], combined with weld chemical composition data and implementation of extreme low radial leakage core loading designs beginning in Cycle 14, indicate that the fluence at the end of 24.25 Effective Full Power Years (EFPY) at 1500 MWt will be 1.50×10^{19} n/cm^2 on the inside surface of the reactor vessel. This results in a total shift of the RT_{NDT} of 238.5°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, and a shift of 187.5°F at the 3/4t location. Operation through fuel Cycle 22 will result in less than 24.25 EFPY.

The limit lines in Figures 2-1A and 2-1B are based on the following:

A. Heatup and Cooldown Curves - From Section III of the ASME Code, Appendix G-2215.

$$K_{IR} = 2\,K_{IM} + K_{IT}$$

K_{IR} = Allowance stress intensity factor at temperature related to RT_{NDT} (ASME III Figure G-2110.1).

K_{IM} = Stress intensity factor for membrane stress (pressure). The 2 represents a safety factor of 2 on pressure.

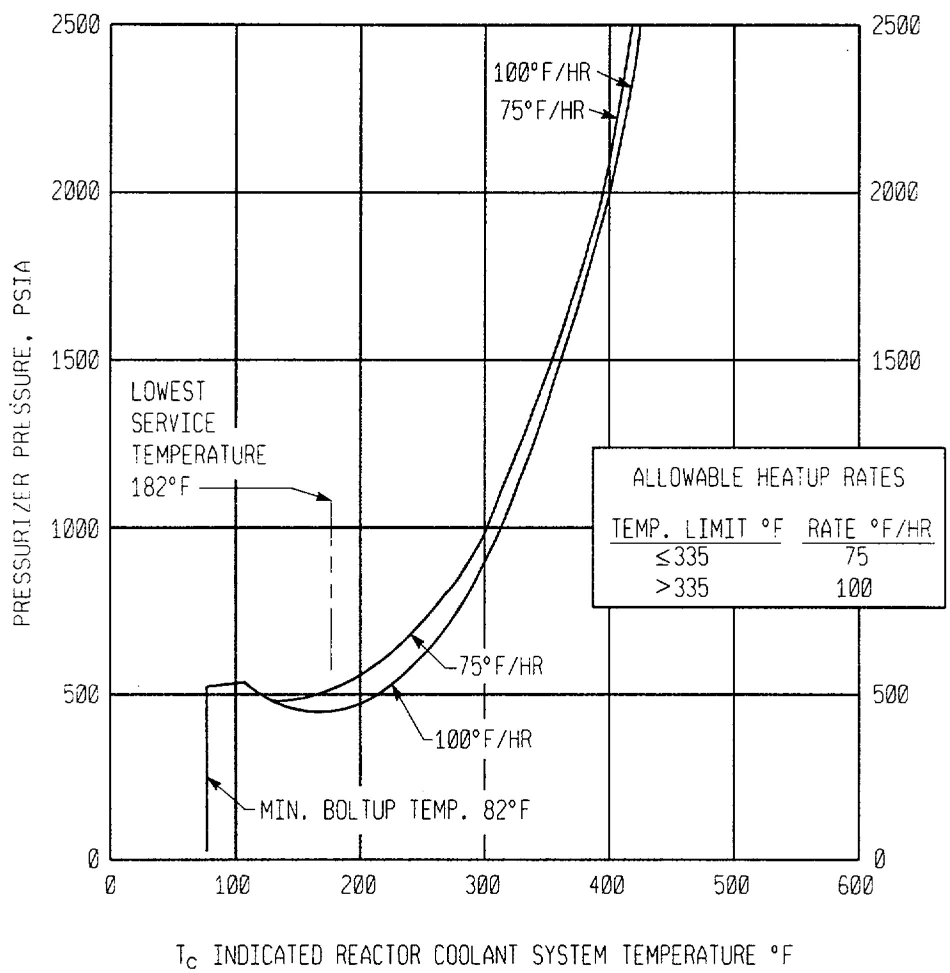
K_{IT} = Stress intensity factor radial thermal gradient.

The above equation is applied to the reactor vessel beltline. For plant heatup the reference stress intensity is calculated for both the 1/4t and 3/4t locations. Composite curves are then generated for each heatup rate by combining the most restrictive pressure-temperature limits over the complete temperature interval.

For plant cooldown thermal and pressure stress are additive.

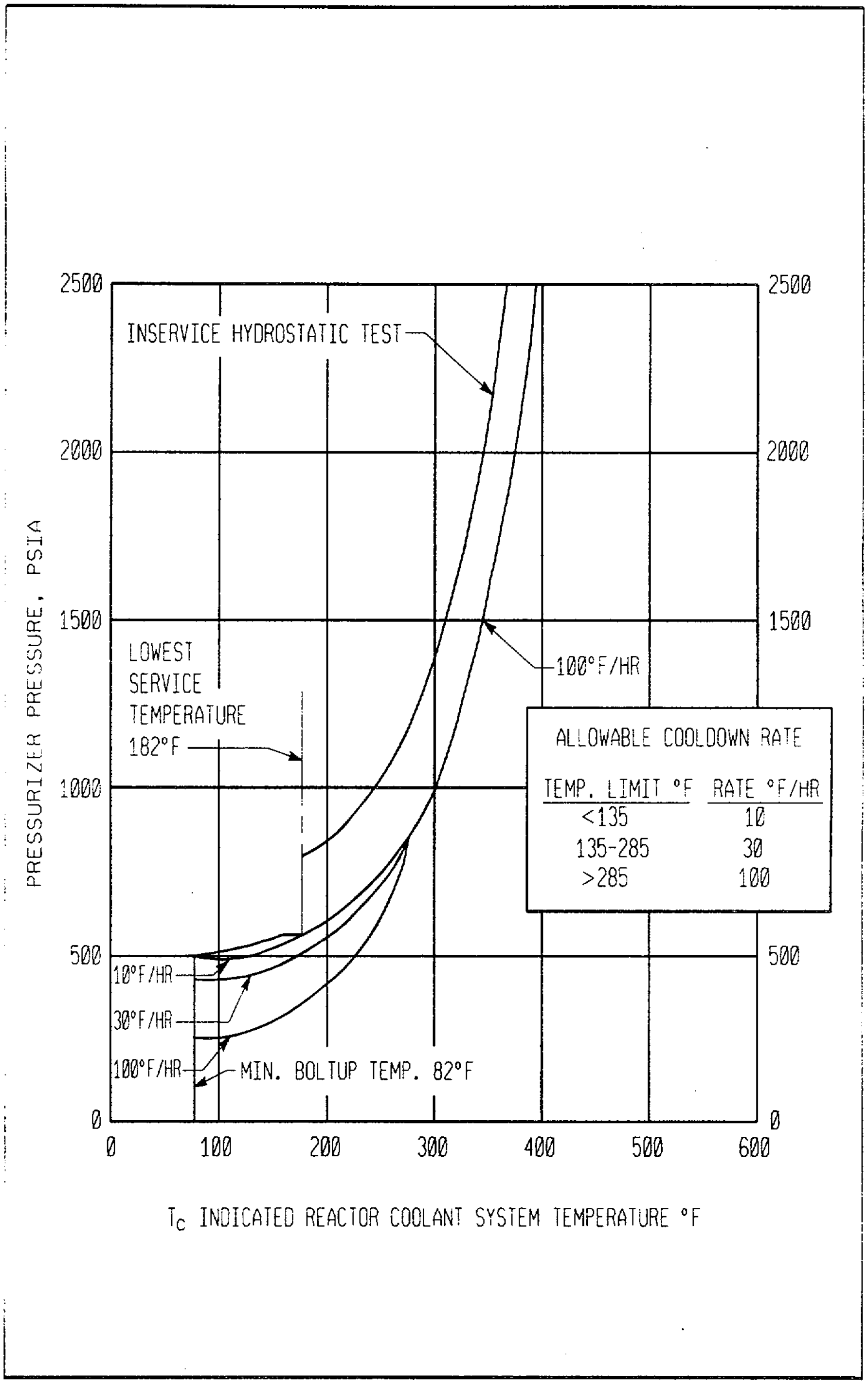
Figure 2-1A

FORT CALHOUN STATION UNIT 1 P/T LIMITS, 24.25 EFPY



RCS Pressure-Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station-Unit No. 1

Figure 2-1B
Fort Calhoun Station Unit 1 P/T Limits, 24.25 EFPY
Cooldown and Inservice Test



RCS Pressure-Temperature Limits for Cooldown	Omaha Public Power District Fort Calhoun Station-Unit No. 1

Amendment No. ~~74,77,100,114,153,161~~,197

FIGURE 2-3

This Figure has been deleted.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.1 Reactor Coolant System (Continued)

2.1.2 Heatup and Cooldown Rate (Continued)

This temperature is based on previous NDTT methods. This temperature corresponds to the measured 10°F NDTT of the reactor vessel flange, which is not subject to radiation damage, plus 60°F data scatter in NDTT measurements, plus 12°F instrument error.

E. The temperature at which the heatup and cooldown rates change in Figures 2-1A and 2-1B reflects the point at which the most limiting heatup and cooldown rates with respect to the inlet temperature (T_c) change.

References:

(1) USAR, Section 4.2.2

(2) ASME Boiler and Pressure Vessel Code, Section III

(3) USAR, Section 4.2.4

(4) USAR, Section 3.4.6

(5) WCAP-15443, Revision 0, Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel, July 2000.

(6) Technical Specification 2.3(3)

(7) Article IWB-5000, ASME Boiler and Pressure Vessel Code, Section XI

(8) TR-O-MCM-001, Revision 1, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, August 1980.

(9) TR-O-MCM-002, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-265, March 1984.

(10) BAW-2226, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-275, November 1994.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 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 application dated October 18, 2000, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS). The requested changes would revise Technical Specification (TS) 2.1.2 to extend the period of applicability of the pressure-temperature (PT) curves to 24.25 effective full power years (EFPYs) of operation. OPPD's request is based on the fact that the last few cycles were designed and operated with very low leakage loadings, thus, the vessel exposure savings can support another 4.25 EFPYs of operation to reach the fluence level (at the vessel critical elements) for which it is currently licensed to operate.

The fluence information required for this evaluation was submitted by application dated August 3, 2000, which included WCAP-15443, "Fast Neutron Fluence Evaluation for the Fort Calhoun Unit 1 Reactor Pressure Vessel," Westinghouse Electric Company LLC, July 2000. This report includes: (1) neutron transport and dosimetry evaluation methodologies, (2) methods qualification, (3) results of neutron transport calculations, (4) reevaluation of FCS surveillance capsules W225 and W265 (which were measured and analyzed by Combustion Engineering) and capsule W275 (which was measured and analyzed by B&W), and (5) projected neutron exposure for the FCS pressure vessel.

The licensee also requested approval of WCAP-15443. This is not feasible under a plant specific application. In a discussion with the staff conducted on November 29, 2000, OPPD clarified that the statement was not intended to request generic approval of WCAP-15443.

2.0 EVALUATION

2.1 WCAP-15443, Fast Neutron Fluence for the Ft. Calhoun Reactor Pressure Vessel

The purpose of the report is to describe and justify the methodology used for the calculation of the FCS fluence. At first the neutron transport method is described. The method uses the DORT/TORT systems of two and three-dimensional discrete ordinates transport codes (Reference 1). The cross sections used are from the BUGLE-93 version based on the ENDF/B-VI data base (Reference 2). The approximations use the P_3 scattering expansion and the S_8 angular quadrature approximation. In addition, the methodology proposes to use the

FERRET code to least-square adjust the dosimeter readings (Reference 3). The P_3 and S_8 approximations comply with the recommendations of Draft Regulatory Guide 1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (DG-1053). However, the DG-1053 requires the use of the latest available cross sections. At this time BUGLE-96 is available and should have been used. However, the licensee stated in discussions with the staff that: (1) the calculations in WCAP-15443 were performed before the BUGLE-96 became available and (2) BUGLE-96 would not make any difference in these calculations. Regarding the use of the FERRET code, the report states that only the calculated values were used. The FERRET values were limited in the comparison of calculation to measurement. This is in accordance with a standing OPPD commitment not to use adjusted dosimeter values.

At first the methodology is benchmarked to the pool critical assembly (PCA) simulator measurements. An analytic sensitivity study was performed to investigate the contribution of the main uncertainty components. The results of the available FCS surveillance capsules W225, W265 and W275 plant specific measurements were used to validate the calculational results.

The staff finds the methodology proposed in WCAP-15443, is consistent with the recommendations of DG-1053 for plant specific applications and thus, is acceptable.

2.2 Fort Calhoun Pressure Vessel Projected Fluence to the Axial Welds 3-410

As stated above, the proposed fluence values were calculated using the benchmarked code. The critical element consists of two 3-410 axial welds located at an azimuthal angle of 60°. Table 6.2-1 of WCAP-15443 summarizes the projected vessel fluence as a function of EFPYs and azimuthal angle. At 60° for 24.25 EFPYs, the projected value of the fluence is about 1.50×10^{19} n/cm^2 which is the value for which the P-T curves were licensed on March 23, 1994. This value was developed on the basis of the fuel management practices around 1990. However, in 1992 OPPD implemented extreme low leakage fuel loadings, incorporating full length hafnium flux suppression rods in the peripheral fuel assemblies nearest to the 3-410 axial welds. The extreme low leakage management also made up for the increase resulting from: (1) the calculational deficit created from the change of cross sections from ENDF/B-IV to ENDF/B-VI, and (2) the increase of the projected load factor from the 0.80 assumed in the 1994 calculations and 0.85 in the present projection. Other measures taken in the implementation of the low leakage included natural or depleted uranium rods in the peripheral assemblies and integral fuel burnable absorbers to minimize flux peaking.

Based on the methodology and the extreme low leakage measures implemented by the licensee, the staff finds that the proposed extension of the PT curves to 24.25 EFPYs to be acceptable.

2.3 TS Changes

- In TS 2.1.2, Figures 2-1A and 2-1B, the EFPYs are changed from 20 to 24.25.

- Figure 2-3 in TS 2.1.2 is deleted because the new fuel management scheme renders it obsolete.

- Also, in TS 2.1.2, WCAP-15443 is listed as the new fluence in reference 5.
- The reevaluated surveillance capsule reports are listed in the references.
- In the Bases of TS 2.1.2, the load factor has been changed to 0.85.
- Appropriate editorial changes were made in the TS as well as in the Bases.

The above technical specification changes are acceptable because they reflect the changes which were found acceptable in the review.

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 (65 FR 81926). 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. CCC-543 RSIC Computer Code Collection: "TORT-DORT Two and Three-Dimensional Discrete Ordinates Transport, Version 2.8.14," Radiation Safety Information Computational Center, January 1994.

2. DLC-175, "BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross Section Libraries Derived from ENDF/B-VI Nuclear Data," Radiation Safety Information Computational Center, April 1994.

3. HEDL-TME-79-40, "FERRET Data Analysis Code," by E. A. Schmittroth, Hanford Engineering Development Laboratory, Richland, Washington, September 1979.

Principal Contributor: L. Lois

Date: March 27, 2001