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

October 18, 2000
LIC-00-0086

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

References:
1. Docket 50-285
2. Operating License DPR-40 Amendment No. 161
3. WCAP-15443, Revision 0, *Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel*, dated July 2000
4. Letter from OPPD (W.G. Gates) to NRC (Document Control Desk), dated August 3, 2000 (LIC-00-0064)
5. Letter from OPPD (S.K. Gambhir) to NRC (Document Control Desk), dated September 28, 1998 (LIC-98-0124)

SUBJECT: Application for Amendment of Operating License

Omaha Public Power District (OPPD) is submitting this "Application for Amendment of Operating License" to amend the Fort Calhoun Station Unit No. 1 Technical Specifications. The proposed changes include revision to the labels for Figures 2-1A (RCS Pressure –Temperature Limits for Heatup) and 2-1B (RCS Pressure –Temperature Limits for Cooldown). Specifically, the validity of the existing curves is being extended from 20 effective full power years (EFPY) to 24.25 EFPY, based on crediting reductions in projected reactor vessel fast neutron fluence as a result of implementation of extreme low radial leakage fuel management in 1992 (Cycle 14). The NRC staff also noted the existing conservatism in Section 2.3 of the Safety Evaluation Report of Amendment 161 dated March 23, 1994 (Reference 2). The assumed limiting 3-410 axial weld material properties inputs remain unchanged from Reference 2.

In addition, OPPD proposes deletion of Technical Specification Figure 2-3 (Predicted Radiation Induced NDTT Shift), because the figure does not reflect use of the current NRC-approved embrittlement correlation (as contained in Regulatory Guide 1.99, Revision 2), and it is not directly applicable to plant operation. Accordingly, OPPD also proposes revision of Technical Specification 2.1.2 (6) (a) and (b) to reflect deletion of Figure 2-3 and to incorporate definition of the validity range of Figures 2-1A and 2-1B with respect to fast neutron ($E \geq 1$MeV) fluence up to 1.50×10^{19} n/cm^2 and operation to 24.25 EFPY.

A001

Employment with Equal Opportunity

Finally, OPPD proposes the following associated revisions to the Basis section of Technical Specification 2.1.2: (1) change the wording of "the proposed Regulatory Guide 1.99, Revision 2" to "Regulatory Guide 1.99, Revision 2"; (2) incorporate the predicted values at 40 years of operation for fluence and 1/4T RT_{NDT}; (3) revise the assumed load factor from 80% to 85% consistent with Reference 3; (4) change the validity of the P-T curves from Cycle 19 to Cycle 22 consistent with the fluence projection changes; (5) revise the RT_{NDT} values at 24.25 EFPY at the 1/4T and 3/4T locations consistent with the NRC Safety Evaluation of Reference 2; (6) describe the use of extreme low radial leakage fuel management beginning in Cycle 14; and (7) add an additional reference for the Surveillance Capsule W-275 Report.

Included in a previously submitted Application for Amendment of Operating License (Reference 4) was the most current Fort Calhoun Station fluence analysis (Reference 3). This analysis (WCAP-15443, Revision 0, *Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel*) used the ENDF/B-VI nuclear cross-section library, versus the ENDF/B-IV cross-section library previously used which conservatively predicts a greater fast neutron fluence to the limiting locations of the Fort Calhoun Station reactor vessel. The present 20 EFPY P-T curves are based on a fluence of 1.50×10^{19} n/cm^2. Based on the Reference 3 analysis, the fluence of 1.50×10^{19} n/cm^2 presently corresponds to 24.29 EFPY. The margin between 24.29 and 24.25 is being utilized as an additional conservatism. Reference 3 forms the basis of revising 20 EFPY to 24.25 EFPY; therefore, NRC approval of Reference 3 is needed to support the proposed Technical Specification and Basis changes included in this application.

Attachment C contains a mark-up reflecting the requested Technical Specification changes and Basis changes. Attachment D provides the Discussion, Justification and No Significant Hazards Considerations.

Fort Calhoun Station Unit No. 1 will exceed 20 EFPY on approximately June 1, 2001. Therefore, OPPD requests NRC approval of this proposed amendment before the start of the 2001 refueling outage, currently scheduled for March 17, 2001. OPPD requests 30 days to implement the proposed specifications following NRC approval. If you have additional questions, or require further information, please contact me or members of my staff.

Sincerely,

W. G. Gates
Vice President

WGG/dls

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)	)	

APPLICATION FOR AMENDMENT
OF
FACILITY OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U. S. Nuclear Regulatory Commission ("the Commission"), Omaha Public Power District, holder of Facility Operating License No. DPR-40, herewith requests that the Facility Operating License be amended to extend the validity of the RCS pressure-temperature limits for heatup and cooldown from 20 EFPY to 24.25 EFPY. This amendment will affect Technical Specification 2.1.2, Figure 2-1A, Figure 2-1B, Figure 2-3 and the corresponding Basis.

The proposed changes to the Technical Specifications are provided in Attachment C of this Application. A Discussion, Justification, and No Significant Hazards Consideration Analysis, which demonstrates the proposed changes do not involve significant hazards considerations, is provided in Attachment D. The proposed changes to Appendix A, Technical Specifications of the Facility Operating License, would not authorize any change in the types or any increase in the amounts of effluents or any change in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Appendix A of the Facility Operating License be amended in the form attached hereto as Attachment C.

A copy of this Application, including its attachments, has been submitted to the Director - Nebraska State Division of Environmental Health, as required by 10 CFR 50.91.

OMAHA PUBLIC POWER DISTRICT

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this

18th day of October 2000

Kristie E. Thompson

GENERAL NOTARY-State of Nebraska
KRISTIE E. THOMPSON
My Comm. Exp. May 1, 2004

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

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President in charge of all nuclear activities of the Omaha Public Power District; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached information concerning the Application for Amendment of Facility Operating License dated October 18, 2000, regarding the change to extend the validity of the RCS pressure-temperature limits for heatup and cooldown from 20 EFPY to 24.25 EFPY; 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.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to me, a Notary Public in and for the State of Nebraska on this

18th day of October 2000

Kristie E. Thompson
Notary Public

GENERAL NOTARY-State of Nebraska
KRISTIE E. THOMPSON
My Comm. Exp. May 1, 2004

LIC-00-0086
Attachment C
Requested Changes to Technical Specifications Set Forth in Appendix A of the Facility Operating License No. DPR-40, with Associated Basis Changes

TECHNICAL SPECIFICATIONS - FIGURES

TABLE OF CONTENTS

FIGURE	DESCRIPTION	PAGE WHICH FIGURE FOLLOWS
1-1	TMLP Safety Limits 4 Pump Operations	1-3
2-1A	RCS Pressure-Temperature Limits for Heatup	2-6
2-IB	RCS Pressure-Temperature Limits for Cooldown	2-6
~~2-3~~	~~Predicted Radiation Induced NDTT Shift~~	~~2-6~~
2-3	Deleted	2-6
2-12	Boric Acid Solubility in Water	2-19h
2-10	Spent Fuel Pool Region 2 Storage Criteria	2-39e
2-8	Flux Peaking Augmentation Factors	2-53

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 1.50×10^{19} n/cm^2 which corresponds to 24.25 EFPY. ~~The curve in Figure 2-3 shall be used to predict the increase in transition temperature based on integrated fast neutron flux. If measurements on the irradiation specimens indicate a deviation from this curve, a new curve shall be constructed.~~

(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). ~~The predicted transition temperature shift to the end of the new period shall then be obtained from Figure 2-3.~~

(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 ~~mean~~ 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 ~~the proposed~~ 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] ~~(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 ~~Section 3.4.6 of the USAR, except~~ that the integrated fast neutron flux ($E \geq 1$ MeV) is 1.73~~2.55~~x10^{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.[5]

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~~2.55~~x10^{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~~80~~% load factor. The predicted shift at this location at the 1/4t depth from the inner surface is projected to be 252~~332~~°F, including margin, ~~and was calculated~~ 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~~second~~ removed irradiated reactor vessel surveillance specimens[8, 9 and 10], combined with weld chemical composition data and implementation of extreme low radial leakage ~~reduced fluence~~ core loading designs beginning~~initiated~~ in Cycle 14~~8~~, indicate~~d~~ that the fluence at the end of 24.25~~20.0~~ 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~~298~~°F, including margin, for the area of greatest sensitivity (weld metal) at the 1/4t location using Regulatory Guide 1.99, Revision 2, ~~as determined from Figure 2-3~~ and a shift of 187.5~~241~~°F at the 3/4t location. Operation through fuel Cycle 22~~19~~ will result in less than 24.25~~20.0~~ 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.

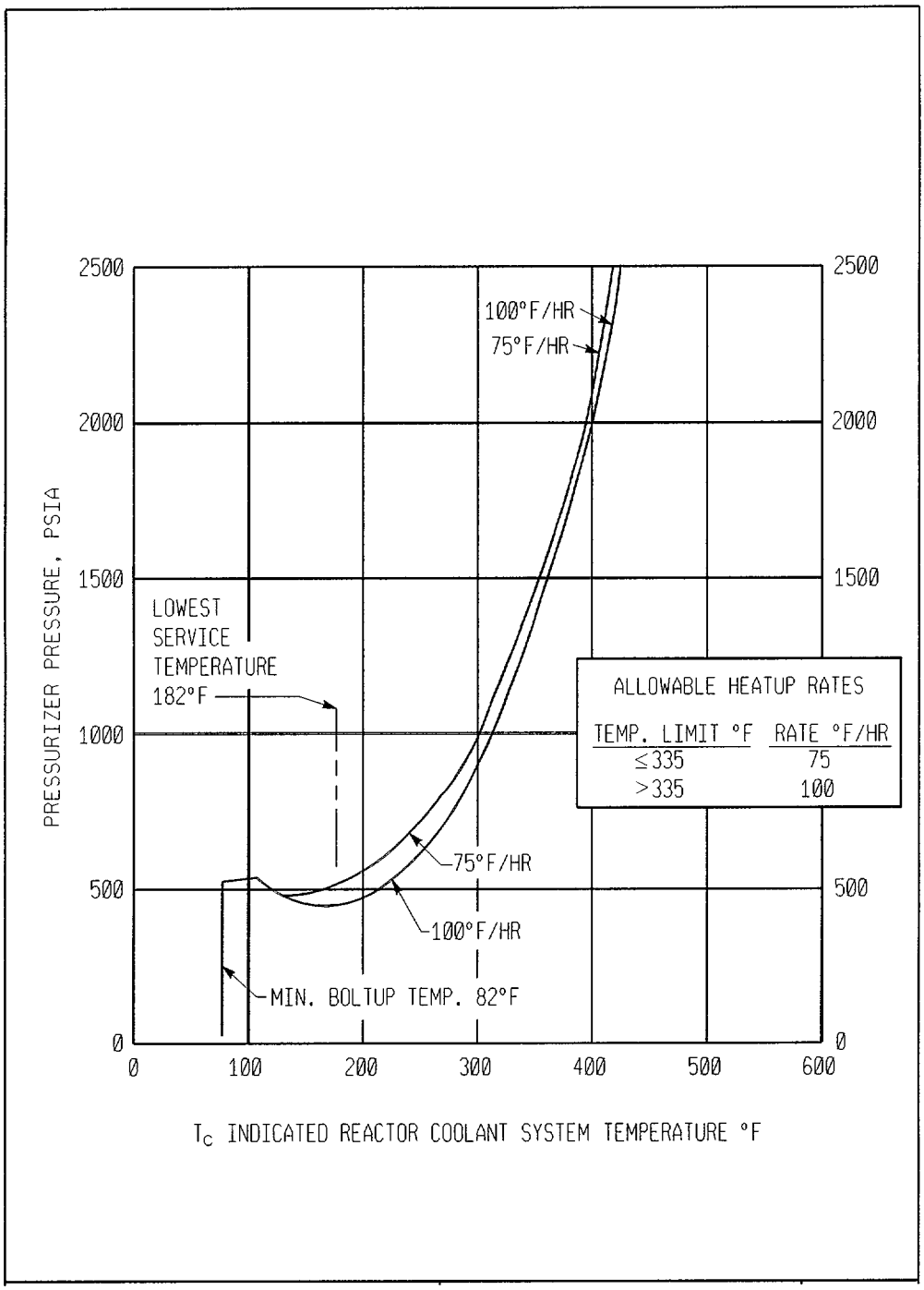
K_{IT} = Stress intensity factor radial thermal gradient.

The above equation is applied to the reactor vessel beltline. For plant heatup the reference stress intensity is ~~in~~ 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.

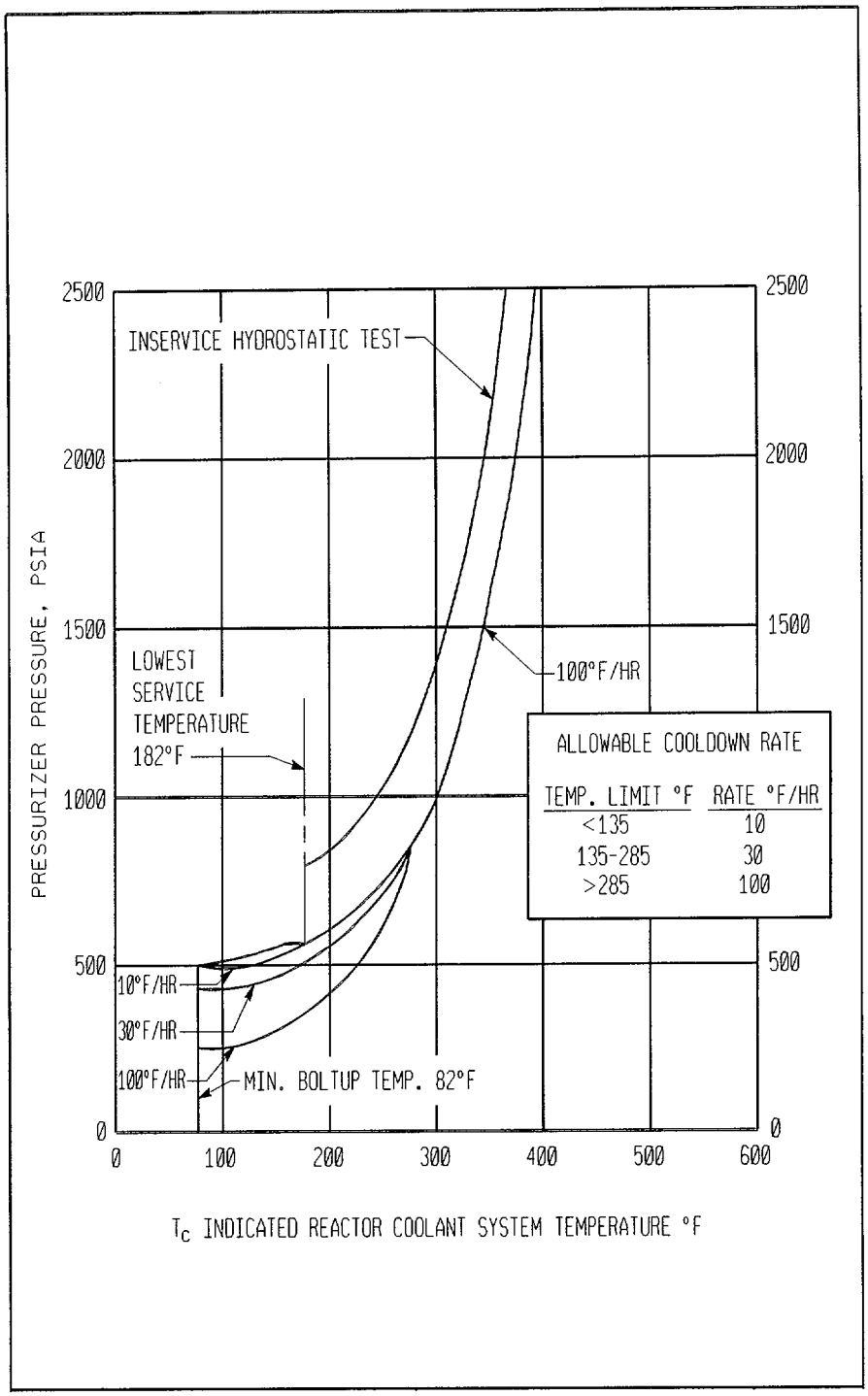
Figure 2-1A

Fort Calhoun Station Unit 1 P/T Limits, 24 25 ~~20~~ EFPY



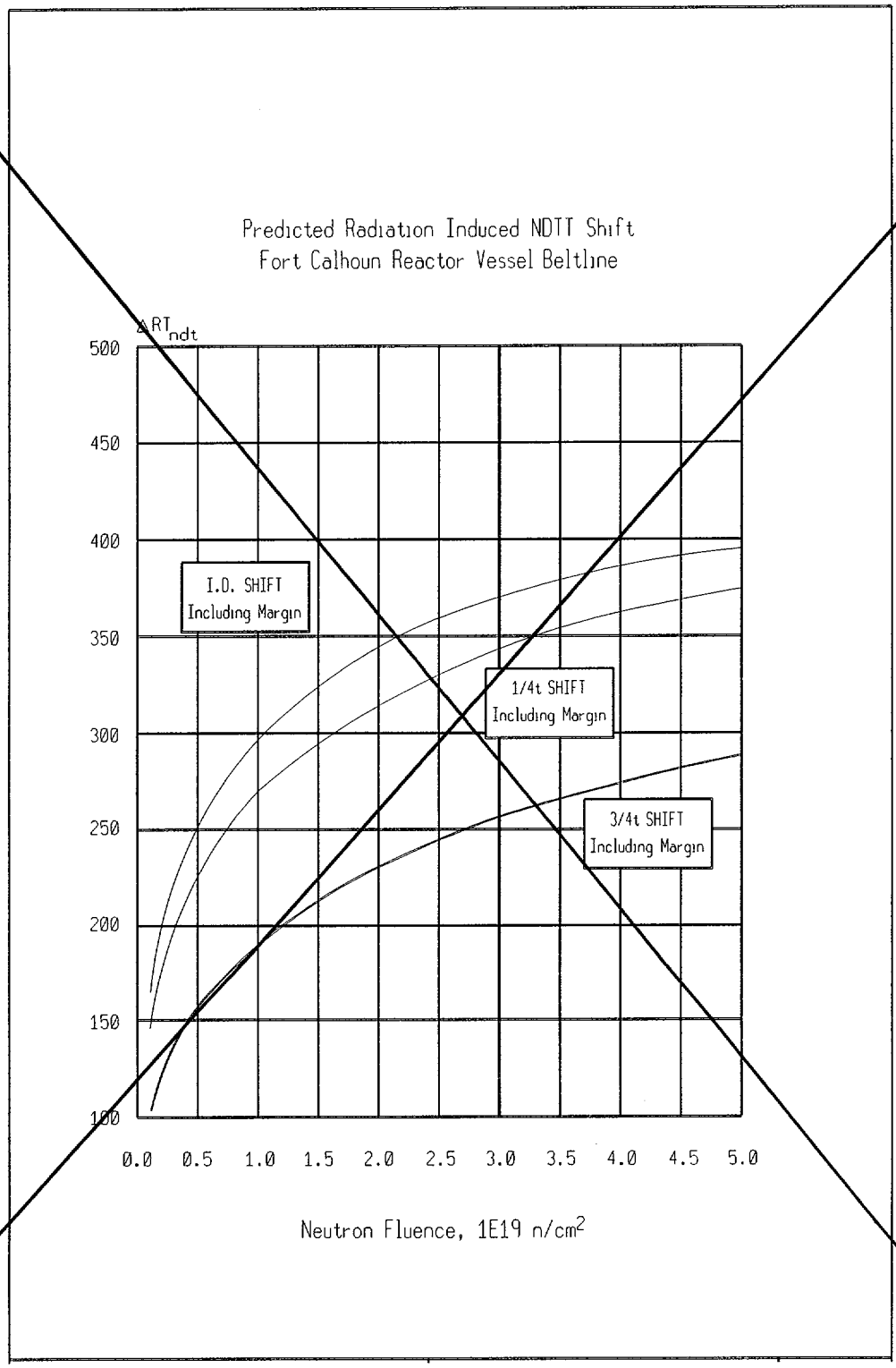
RCS Pressure-Temperature Limits for Heatup	Omaha Public Power District Fort Calhoun Station-Unit No. 1

Figure 2-1B
Fort Calhoun Station Unit 1 P/T Limits, 24.25 ~~20~~ EFPY
Cooldown and Inservice Test



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

Figure 2-3 - Deleted



Predicted Radiation Induced NDTT Shift	Omaha Public Power District Fort Calhoun Station-Unit No. 1

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.

~~(5) Omaha Public Power District, Fort Calhoun Station Unit No. 1, Evaluation of Irradiated Capsule W-225, Revision 1, August 1980.~~

(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~~8) 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.

LIC-00-0086
Attachment D
Discussion, Justification, and No Significant Hazards Analysis

DISCUSSION AND JUSTIFICATION

The Omaha Public Power District (OPPD) proposes to amend the Technical Specifications for Fort Calhoun Station (FCS) Unit 1 Operating License No. DPR-40 by (1) extending the validity of the existing Technical Specification Figure 2-1A (RCS Pressure-Temperature Limits for Heatup) and Figure 2-1B (RCS Pressure-Temperature Limits for Cooldown) from 20.0 EFPY to 24.25 EFPY, (2) deleting Figure 2-3 (Predicted Radiation Induced NDTT Shift), and (3) providing replacement guidance in Technical Specifications 2.1.2 (6) (a) and (b) for use of the most current fluence analysis and Regulatory Guide 1.99, Revision 2 for projecting RT_{NDT} at 24.25 EFPY.

OPPD also proposes the following associated revisions to the Basis section of Technical Specification 2.1.2: (1) change "proposed Regulatory Guide 1.99, Revision 2" to "Regulatory Guide 1.99, Revision 2"; (2) update the predicted values at 40 years of operation for fluence and 1/4T RT_{NDT} consistent with the most current fluence analysis (WCAP-15443, Revision 0, *Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel*, dated July 2000); (3) revise the assumed load factor from 80% to 85% consistent with the present plant operating history and as input to WCAP-15443, Revision 0; (4) change the validity of the P-T curves from Cycle 19 to Cycle 22 consistent with the fluence projection changes; (5) revise the RT_{NDT} values at 24.25 EFPY at the 1/4T and 3/4T locations consistent with the NRC Safety Evaluation Report for Amendment No. 161; (6) describe the use of extreme low radial leakage fuel management beginning in Cycle 14; and (7) add additional references for WCAP-15443 and the Surveillance Capsule W-275 Report.

Each of the above inter-related items is discussed below.

The existing P-T curves for heatup and cooldown were developed in 1990 and approved by the NRC in the Safety Evaluation Report for Amendment No. 161, dated March 23, 1994. The P-T curve analysis used a 20.0 EFPY projected fast neutron fluence of 1.50×10^{19} n/cm^2 based on continued use of low radial leakage fuel management in place at that time and a fluence analysis that used the ENDF/B-IV Nuclear Cross-Section Library. In 1992, OPPD implemented extreme low radial leakage fuel management incorporating the use of full length hafnium flux suppression rods inserted in fuel assemblies located nearest the limiting reactor vessel welds, natural or depleted uranium fuel assemblies in the peripheral locations near the limiting welds, and integral fuel burnable absorbers to minimize core flux peaking. The result was a significant reduction in the fast neutron flux to the limiting welds for the new fuel management strategy.

Based on WCAP-15443, the most current fluence analysis which uses the updated ENDF/B-VI Nuclear Cross-Section Library (providing better/conservative representation of iron cross sections), and continued use of extreme low radial leakage fuel management projected to the end of plant life, the existing P-T curves fluence value of 1.50×10^{19} n/cm^2 now corresponds to greater than 24.25 EFPY. Thus, assurance is provided of the acceptability of the proposed changes to Figures 2-1A and 2-1B for extending the validity of the present curves from 20.0 EFPY to 24.25 EFPY.

DISCUSSION AND JUSTIFICATION (Continued)

Technical Specification 2.1.2(6)(d) states that Technical Specification 2.3(3) shall be revised each time Figures 2-1A and 2-1B are revised. Since the P-T curves are not being changed in this proposed amendment, no changes to the LTOP setpoints, the subject of Tech Spec 2.3(3), are required.

Review of Section 2.1.2 of the Technical Specifications identified an inconsistency between the RT_{NDT} values predicted by Figure 2-3 and those predicted by Regulatory Guide 1.99, Revision 2. The NRC staff reviewers also noted this discrepancy in the Safety Evaluation for Technical Specification Amendment No. 161, dated March 23, 1994, but accepted the changes at that time due to the conservatisms inherent in Figure 2-3. Since this figure has no operational use, and the values are easily obtained using Regulatory Guide 1.99, Revision 2, deletion of this figure is deemed appropriate. With the deletion of this figure, revisions to Specifications 2.1.2(6) (a) and (b) are also proposed to define the fast neutron fluence and number of EFPY for which Figures 2-1A and 2-1B are valid. These proposed changes represent an enhancement to the Technical Specifications, eliminate a figure not directly applicable to the operation of the plant, and maintain consistency with the presently applicable regulatory guide.

Related changes to the Basis for Specification 2.1.2 are also necessary:

- Delete the words " the proposed" from "the proposed Regulatory Guide, Revision 2", because the revision was issued in May 1988.
- Incorporate the revised 40-year projected values of fluence and 1/4T RT_{NDT} resulting from incorporation of extreme low radial leakage fuel management. The projected fluence value is reduced from 2.55×10^{19} n/cm^2 to 1.73×10^{19} n/cm^2 and the corresponding 1/4T RT_{NDT} is reduced from 332°F to 252°F, consistent with the fluence projections of WCAP-15443, Revision 0.
- Revise the predicted load factor from 80% to 85%, which is consistent with the current operating history and the input assumption to WCAP-15443, Revision 0.
- Revise the expected fuel cycle for which the P-T curves are valid from Cycle 19 for 20.0 EFPY to Cycle 22 for 24.25 EFPY.
- Revise the 1/4T RT_{NDT} and 3/4T RT_{NDT} values at 24.25 EFPY from 298°F to 238.5°F and 241°F to 187.5°F, consistent with FCS Technical Specification Amendment No. 161.
- Add text to document the implementation of extreme low radial leakage fuel management in Cycle 14, replacing the description of low fluence core designs initiated in Cycle 8.
- Add the WCAP-15443, Revision 0 fluence analysis as Reference 5 (moving the existing Reference 5 to Reference 8) and add new Reference 10 to include the Surveillance Capsule W-275 report as a reference for the third capsule removed and evaluated.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes do not involve a significant hazards consideration because operation of Fort Calhoun Station (FCS) Unit 1 in accordance with these changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The NRC previously approved Technical Specification Amendment No. 161 in March 1994 for the use of RCS Pressure-Temperature (P-T) Limits good to 20.0 EFPY. The proposed changes in this submittal reflect the validity of these same curves from 20.0 EFPY to 24.25 EFPY based on the implementation of extreme low radial leakage fuel management in 1992 (Cycle 14). Significant reductions in the fast neutron flux to the limiting 3-410 axial weld in the Fort Calhoun Station reactor pressure vessel were obtained, thus significantly increasing the time to when the fast neutron fluence input to the derivation of the previously approved P-T curves will be reached. Since no inputs (including assumed material properties of the limiting weld) to the existing analysis are being changed, extension of the validity of the curves from 20.0 EFPY to 24.25 EFPY is justified. In addition, deletion of Figure 2-3 and references to it are proposed. This proposed change removes an outdated figure which is non-operational in nature. The application of the current Regulatory Guide 1.99, Revision 2 is more appropriate for these purposes. Administrative changes to the Basis section of TS 2.1.2 are proposed to reflect the extension to 24.25 EFPY.

No accidents previously analyzed are affected by these changes, and it can be concluded that there is no significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes do not physically alter the configuration of the plant and no new or different mode of operation is proposed. Extending the validity of the P-T curves more accurately projects reactor vessel embrittlement by accounting for improvements in FCS fuel management which have significantly reduced the fast neutron fluence to the limiting 3-410 axial weld, incorporates improved operating cycle efficiency, and applies the WCAP-15443, Revision 0 fluence analysis. The revised fluence analysis uses the ENDF/B-VI Nuclear Cross Section Library. Deletion of Figure 2-3 represents a change which does not affect plant operations. Figure 2-3 is administrative in nature, and proposed revisions to Specifications 2.1.2(6) (a) and (b) provide guidance consistent with the current Regulatory Guide for P-T curves updates. Update of the Technical Specification 2.1.2 Basis section represents an administrative change that does not affect plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

(3) Involve a significant reduction in a margin of safety.

The proposed changes to extend the validity of Technical Specification Figures 2-1A and 2-1B to 24.25 EFPY are consistent with the extreme low radial leakage fuel management implemented in 1992 (Cycle 14) and performance/application of the updated fluence analysis described above. With no changes to the inputs of the existing P-T limits analysis, there is no reduction in the margin of safety. Figure 2-3 is not used to provide limits on plant operation, and deletion of this figure, which uses a pre-Regulatory Guide 1.99, Revision 2 embrittlement correlation, is considered an improvement in the consistency of the requirements outlined in the Technical Specifications. This Figure is not used in plant operation and provides only a general indication of the RT_{NDT} shift. The TS 2.1.2 Basis section changes are administrative in nature and do not affect the margin of safety. The changes serve to maintain consistency with the NRC approval of Amendment No.161.

In summary, the proposed changes do not involve a significant reduction in the margin of safety.

Therefore, based on the above, OPPD's position is that this proposed amendment does not involve a significant hazards consideration as defined by 10 CFR 50.92, and the proposed change will not result in a condition which significantly alters the impact of FCS on the environment. Thus, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared.