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

September 5, 2000
LIC-00-0069

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Application for Amendment of Facility Operating License No. DPR-40

Omaha Public Power District (OPPD) is submitting this "Application for Amendment of Facility Operating License" to permit the use of nuclear fuel fabricated by Siemens Power Corporation at Fort Calhoun Station (FCS).

OPPD proposes to amend the applicable sections in the Fort Calhoun Station Unit No. 1 Technical Specifications to allow use of NRC approved methodologies from Siemens Power Corporation for determining reactor core operating limits.

Attachment C contains a markup reflecting the proposed changes to the Definition Section and Sections 1.1, 1.3, 2.10, 3.10, and 5.9 of the Technical Specifications. Attachment D provides the "Discussion, Justification and No Significant Hazards Consideration."

OPPD will begin utilizing nuclear fuel from Siemens Power Corporation in the 2001 refueling outage, scheduled to begin in March 2001. Therefore, OPPD requests that the Nuclear Regulatory Commission review and approve the proposed amendment on or before February 1, 2001. OPPD respectfully 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.

A001

Employment with Equal Opportunity

Sincerely,

W. G. Gates
Vice President

WGG/BRH/brh

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 Definition Section, Bases of Technical Specifications 1.1 and 1.3, and Technical Specifications set forth in Section 2.10, 3.10, and 5.9 of the Facility Operating License be amended to allow the use of nuclear fuel fabricated by Siemens Power Corporation.

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 appended 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 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

By W Gary Gates
Vice President

Subscribed and sworn to before me this 5th day of September 2000

Kristei E. Thompson
Notary Public

GENERAL NOTARY-State of Nebraska
KRISTIE E. THOMPSON
My Comm. Exp. May 1, 2001

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 the Facility Operating License dated September 5, 2000 regarding the change to allow nuclear fuel fabricated by Siemens Power Corporations to be used in the Fort Calhoun Station reactor; 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

5th day of September 2000

Kristie E. Thompson
Notary Public

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

LIC-00-0069
Attachment C
Requested Changes to Technical Specifications Set Forth in Appendix A of the Facility Operating License
No. DPR-40

DEFINITIONS

Azimuthal Power Tilt - T_q

Azimuthal Power Tilt shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that axial half (upper or lower) of the core divided by the average power of all quadrants in that axial half (upper or lower) of the core.

~~Unrodded Planer Radial Peaking Factor - F_{xy}~~

~~The Unrodded Planar Radial Peaking Factor is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding azimuthal tilt, T_q. The maximum F_{xy} limit is provided in the Core Operating Limits Report.~~

Unrodded Integrated Radial Peaking Factor - F_R

The Unrodded Integrated Radial Peaking Factor is the ratio of the peak pin power to the average pin power in an unrodded core, excluding azimuthal tilt, T_q. The maximum F_R limit is provided in the Core Operating Limits Report.

Process Control Program (PCP)

The document(s) that contains the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid waste.

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. In other words,

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core

Applicability

This specification applies to the limiting combinations of reactor power and reactor coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures as shown in Figure 1-1 for 4-pump operation. The safety limit is exceeded if the point defined by the combination of reactor coolant cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At ~~this point~~ DNB there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through a ~~the CE-1~~ correlation.[1] ~~(1) The CE-1 DNB correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions.~~ The local DNB ratio (DNBR), defined as the ratio of the heat flux that

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.1 Safety Limits - Reactor Core (continued)

would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients ~~is limited to 1.18. A DNBR of 1.18~~ corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 1-1 represent the loci of points ~~or~~ for reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the ~~DNBR is 1.18.~~ minimum DNBR is not less than the minimum DNBR limit. The area of safe operation is below these lines.

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2.10 and axial shapes within the axial power distribution trip limits in the COLR ~~and a total unrodded planar radial peak (F_{xy}^T) as specified in the COLR.~~ The Thermal Margin/Low Pressure trip requirements shall be within the limits provided in the COLR. The Thermal Margin/Low Pressure trip is based on an unrodded integrated total radial peak (F_R^T) that is provided in the COLR.

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model test.[2] The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the minimum DNBR limit ~~1.18.~~[1]

References

(1) USAR, Section 3.6.6

(2) USAR, Section 1.4.6

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.3 Limiting Safety System Settings, Reactor Protective System (continued)

(3) High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the reactor and steam system safety valves to prevent reactor coolant system overpressure (Specification 2.1.6). In the event of loss of load without reactor trip, the temperature and pressure of the reactor coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. The power-operated relief valves are set to operate concurrently with the high pressurizer pressure reactor trip. This setting is below the nominal safety valve setting (2500 psia) to avoid unnecessary operation of the safety valves. This setting is consistent with the trip point assumed in the accident analysis.[1]

(4) Thermal Margin/Low Pressure Trip - The thermal margin/low pressure trip is provided to prevent operation when the DNBR is less than ~~1.18~~ the minimum DNBR limit, including allowance for measurement error. The thermal and hydraulic limits shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure, contained in the COLR, define the limiting values of reactor coolant pressure, reactor inlet temperature, axial shape index, and reactor power level which ensure that the thermal criteria[8] are not exceeded. The low set point of ~~a~~ 1750 psia trips the reactor in the unlikely event of a loss-of-coolant accident. The thermal margin/low pressure trip set points shall be set according to the equation given in the COLR for the Thermal Margin/Low Pressure Limit.

1.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

1.3 Limiting Safety System Settings, Reactor Protective System (continued)

(7) Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shut down simultaneously with the initiation of the safety injection system. The setting of this trip is identical to that of the containment high pressure signal which indicates safety injection system operation.

(8) Axial Power Distribution - The axial power trip is provided to ensure that excessive axial peaking will not cause fuel damage. The Axial Shape Index is determined from the axially split excore detectors. The set point functions, shown in the COLR ensure that neither a DNBR of less than ~~1.18~~ the minimum DNBR limit nor a maximum linear heat rate of more than 22 kW/ft (deposited in the fuel) will exist as a consequence of axial power maldistributions. Allowances have been made for instrumentation inaccuracies and uncertainties associated with the excore symmetric offset - incore axial peaking relationship. A variance of 5% between Δ T-Power and NI-Power is permitted due to the significant margins to local power density limits before calibration of NI-Power is performed at 30% power.

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than ~~1.18~~ the minimum DNBR limit nor a peak linear heat rate of more than 22 kW/ft occurs as a result of the loss of load to one steam generator.

(10) Physics Testing at Low Power - During physics testing at power levels less than 10^{-1}% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below 10^{-1}% of rated power. Written test procedures which are approved by the Plant Review Committee will be in effect during these tests. At reactor power levels less than 10^{-1}% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6.2.2, 14.6.4
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6
(9) USAR, Section 14.10

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

With $F_R^T \geq$ the limit provided in the COLR within 6 hours:

(a) Reduce power to bring power and F_R^T within the limits of the F_R^T, ~~F_{XY}^T~~ and Core Power Limitations Figure provided in the COLR, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the NTCEA's, or

(b) Be in at least hot standby.

(3) ~~Total Planar Radial Peaking Factor~~ Deleted

~~The calculated value of F_{xy}^T defined as $F_{xy}^T = F_{xy}(1+T_q)$ shall be within the limit provided in the COLR. F_{xy} shall be determined from a power distribution map with no non-trippable CEA's inserted and with all full length CEA's at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects. The azimuthal tilt, T_q, is the measured value of T_q at the time F_{xy} is determined.~~

~~With $F_{xy}^T \geq$ the limit provided in the COLR within 6 hours:~~

~~(a) Reduce power to bring power and F_{xy}^T to within the limits of the F_R^T, F_{xy}^T and Core Power Limitations Figure provided in the COLR, withdraw the full length CEA's to or beyond the Long Term Steady State Insertion Limits of Specification 2.10.2(7), and fully withdraw the NTCEA's or~~

~~(b) Be in at least hot standby.~~

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

(4) Azimuthal Power Tilt (T_q)

When operating above 70% of rated power,

(a) The azimuthal power tilt (T_q) shall not exceed 0.10 whenever Mini CECOR/BASSS is operable, the CEA's are at or above the Long Term Insertion Limit and Mini CECOR/BASSS is being utilized to monitor F_{xy}^{T} ~~and~~ F_R^T.

(b) The azimuthal power tilt (T_q) shall not exceed 0.03 whenever the provisions of 2.10.4(4)(a) do NOT allow Mini CECOR/BASSS to be utilized to monitor F_{xy}^{T} ~~and~~ F_R^T. With the indicated azimuthal power tilt determined to be >0.03 but <0.10, correct the power tilt within two hours or determine within the next 6 hours and at least once per subsequent 8 hours, that the total integrated radial peaking factor, F_R^T, is within the limit of Specification 2.10.4(2) ~~and that the total planar radial peaking factor, F_{xy}^{T}, is within the limit of 2.10.4(3),~~ or reduce power to less than 70% of rated power within 8 hours of confirming T_q >0.03.

(c) With the indicated power tilt determined to be ≥ 0.10, power operation may proceed up to 2 hours provided F_R^T ~~and F_{xy}^{T}~~ does not exceed the power limits of the F_R^T, ~~F_{xy}^{T}~~ and the Core Power Limitations Figure provided in the COLR, or be in at least hot standby within 6 hours. Subsequent operation for the purpose of measurement to identify the cause of the tilt is allowable provided the power level is restricted to 20% of the maximum allowable thermal power level for the existing reactor coolant pump combination.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following limits on DNB-related parameters shall be maintained:

(i)	Cold Leg Temperature (Core Inlet Temperature)	as specified in the COLR
(ii)	Pressurizer Pressure	≥2075 psia*
(iii)	Reactor Coolant Flow rate	≥206,000 gpm indicated
(iv)	Axial Shape Index	as specified in the COLR

t (b) With any of the above parameters exceeding the limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System, or the Incore Detector Monitoring System, provide~~s~~ adequate monitoring of the core power distribution and ~~are~~ is capable of verifying that the linear heat rate does not exceed its limit. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index (ASI) with the operable quadrant symmetric excore neutron flux detectors~~, and verifying that the~~ The axial shape index is maintained within the allowable limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR~~, as~~ This ASI is adjusted by Specification 2.10.4(1)(c) for the allowed linear heat rate of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR~~, RC Pump configuration, and F_{xy}^T of the F_R^T, F_{xy}^T~~ and the F_R^T Core Power Limitations Figure provided in the COLR. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.1(6) and long term insertion limits of Specification 2.10.1(7) are satisfied, and (2) the flux peaking augmentation factors are as shown in Figure 2-8~~, and (3) the total planar radial peaking factor does not exceed the limits of Specification 2.10.4(3).~~

* Limit not applicable during either a thermal power ramp in excess of 5% of rated thermal power per minute or a thermal power step of greater than 10% of rated thermal power.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

The Incore Detector Monitoring system provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be continuously maintained within the allowable limits of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR. The setpoints for these alarms include allowances, set in the conservative directions.

Calibration of the ex-core detector input to the APD calculator is required to eliminate ASI uncertainties due to instrument drift and axially nonuniform detector exposure. If the recalibration is not performed in the period specified, the prescribed steps will assure safe operation of the reactor.

Total ~~Planar and~~ Integrated Radial Peaking Factors (F_{xy}^T ~~and~~ F_R^T) and Azimuthal Power Tilt (T_q)

The limitations of F_{xy}^T ~~and~~ T_q ~~are~~ is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCO's and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations of F_R^T and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_R^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the action statements since these additional restrictions provide adequate assurance that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCO's and LSSS setpoints remain valid. An azimuthal power tilt >0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation F_{xy}^T ~~$= F_{xy}(1 + T_q)$ and~~ $F_R^T = F_R(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_R^T and T_q are within their limits provide assurance that the actual values of F_R^T ~~F_{xy}^T~~ and T_q do not exceed the assumed values. Verifying F_{xy}^T ~~and~~ F_R^T after each fuel loading prior to exceeding 70% of rated power provides additional assurance that the core was properly loaded.

DNBR Margin During Power Operation Above 15% of Rated Power

The selection of limiting safety system settings and reactor operating limits is such that:

1. No specified acceptable fuel design limits will be exceeded as a result of the design basis anticipated operational occurrences, and

2. The consequences of the design basis postulated accidents will be no more severe than the predicted acceptable consequences of the accident analysis in Section 14.

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the F_R^T, F_{xy}^T and Core Power Limitations Figure provided in the COLR along with the parameter limits on quadrant tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The Reactor Coolant System flow rate of 206,000 gallons per minute is the indicated value. It does not include instrumentation uncertainties.

The calorimetric methodology shall be used to measure the Reactor Coolant System flow rate.

3.0 **SURVEILLANCE REQUIREMENTS**

3.10 Reactor Core Parameters (Continued)

(2) Moderator Temperature Coefficient

The MTC shall be determined at the following frequencies and power conditions during each fuel cycle:

1. Prior to initial operation above 5% of rated power, after each fuel loading.
2. At any power level within 500 MWD/T of initial operation after each refueling.
3. At any power level within ± 14 EFPD of reaching a rated power equilibrium boron concentration of 300 ppm.

(3) Regulating CEA Insertion Limits

a. The position of each regulating CEA group shall be determined to be above the Transient Insertion Limits at least once per shift.

b. The accumulated times during which the regulating CEA groups are inserted beyond the Steady State Insertion Limits but above the Transient Insertion Limits shall be determined once per day.

(4) Linear Heat Rate Monitoring Systems

a. The incore detector monitoring system may be used for monitoring the core power distribution provided that at least once per 31 days of accumulated power operation the incore detector alarms generated by the plant computer are verified to be valid and satisfy the requirements of the core distribution map.

b. The excore detector monitoring system may be used for monitoring the core power distribution by:

1. Verifying at least once per 31 days of accumulated power operation that the axial shape index, Y_I, monitoring limit setpoints are maintained within the allowable limits of the Limiting Condition for Operations for Excore LHR Monitoring Figure provided in the COLR, as adjusted by Specification 2.10.4(1).

(5) Total Integrated ~~and Total Planar~~ Radial Peaking Factors (F_R^T ~~and F_{xy}^T~~)

F_R^T ~~and F_{xy}^T~~ shall be determined to be within the limits of Specification 2.10.4 at the following intervals:

a. After each refueling and prior to operation above 70 percent of rated power.

b. At least once per 31 EFPD's of accumulated power operation.

5.0 ADMINISTRATIVE CONTROLS

5.9.5 Core Operating Limits Report

a. Core Operating Limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as follows:

1. OPPD-NA-8301-P-A, "Reload Core Analysis Methodology Overview," ~~(Latest NRC approved revision as stated in the COLR).~~ approved version as specified in the COLR.

2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification," ~~(Latest NRC approved revision as stated in the COLR).~~ approved version as specified in the COLR.

3. OPPD-NA-8303-P-A, "Transient and Accident Methods and Verification," ~~(Latest NRC approved revision as stated in the COLR).~~ approved version as specified in the COLR.

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Report," April 1995 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

5. WCAP-13027-P, "Westinghouse ECCS Evaluation Model for Analysis of CE-NSSS," July 1991 (Westinghouse Proprietary) as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation dated March 26, 1992, and as applied in OPPD submittal to the NRC (LIC-96-0130) dated September 3, 1996, and as approved in the Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 178 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station Unit No. 1, Docket No. 50-285, dated October 25, 1996.

6. XN-75-32(P)(A) Supplements 1, 2, 3, & 4, "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

7. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

8. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

9. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

10. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

11. XN-NF-78-44(P)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

12. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

13. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for CE Reactors, Siemens Power Corporation," approved version as specified in the COLR.

14. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," approved version as specified in the COLR.

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

16. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

17. XN-NF-82-49(P)(A), Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," approved version as specified in the COLR.

18. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

19. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, approved version as specified in the COLR.

20. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," Siemens Power Corporation, approved version as specified in the COLR.

c. The core operating limits shall be determined so that all applicable limits of the safety analysis are met. The Core Operating Limits Report, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload

cycle, to the NRC Document Control Desk with copies to the Region IV Administrator and Senior Resident Inspector.

LIC-00-0069
Attachment D
Discussion, Justification and No Significant Hazards Consideration

DISCUSSION, JUSTIFICATION AND NO SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF AMENDMENT REQUEST

The Omaha Public Power District (OPPD) proposes to revise the Definition Section, Bases of Technical Specifications (TS) 1.1 and 1.3, and Technical Specifications set forth in Sections 2.10, 3.10, and 5.9 for Fort Calhoun Station (FCS) Unit No. 1. This revision will allow use of nuclear fuel fabricated by Siemens Power Corporation at Fort Calhoun Station.

Technical Specification 2.10.4(3) and the definition of unrodded planar radial peaking factor (F_{xy}) are being deleted, and TS 3.10 is being revised to reflect the deletion of this peaking factor. The Bases to TS 1.1 and 1.3 are being revised to delete discussion of the CE-1 correlation that is currently used to calculate minimum Departure from Nucleate Boiling Ratio (DNBR) and the value calculated by this method. Technical Specification 5.9.5 is being revised to incorporate NRC-approved methodologies necessary to determine core operating limits with nuclear fuel from Siemens Power Corporation.

JUSTIFICATION

Definition Section, Technical Specifications 2.10.4 and 3.10

It is proposed that the definition of the unrodded planar radial peaking factor (F_{xy}) and Technical Specification 2.10.4(3) be deleted and that Technical Specification 3.10(5) be revised to delete the surveillance requirement for the total planar radial peaking factor (F_{xy}^T).

The total planar radial peaking factor (F_{xy}^T) is being eliminated from the Technical Specifications. The current need for this parameter in the Technical Specifications is to protect assumptions about the maximum amount of planar peaking in the core. The limitation on F_{xy}^T is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density – High, Limiting Conditions for Operation, and Limiting Safety Systems Settings set-points remain valid during operation at the various allowable CEA group insertion limits. In a two-dimensional set-point analysis, as currently conducted, F_{xy}^T is combined with the maximum F_z (axial power profile) to produce the limiting F_q (maximum power point in the core) or equivalent Linear Heat Rate and remain conservative relative to assumptions on linear heat rate used as input to the LOCA and other transient analyses. In a three-dimensional analysis, as proposed with the use of Siemens methodology, those peaks are calculated directly during a series of pre-determined maneuvers (axial shape oscillation, power maneuver, or other transient). Direct calculation of these peaks negates the need to make inferences about the amount of planar radial peaking that occurs in any particular plane.

JUSTIFICATION (Cont.)

Bases of Technical Specifications 1.1 and 1.3

It is proposed to revise the Bases Sections to Technical Specifications 1.1 and 1.3 to delete the discussion on calculating the minimum Departure from Nucleate Boiling Ratio (DNBR) using the CE-1 correlation and its associated value.

The proposed methodology for determining the minimum DNBR for fuel supplied by Siemens Power Corporation is based on the NRC-approved HTP (high thermal performance) correlation. The HTP correlation is proposed for inclusion into TS 5.9.5, "Core Operating Limits Reports." The limit curves contained in TS Figure 1-1 describe the region of safe reactor conditions based on core power, reactor coolant pressure, and reactor coolant temperature conditions. The limit curves were developed for nuclear fuel supplied by Combustion Engineering or Westinghouse using the NRC-approved CE-1 correlation. As stated in the Basis of TS 1.1, using the CE-1 correlation with a minimum DNBR value of 1.18 provides a 95% probability at a 95% confidence level that DNB will not occur for any operating condition. For the Siemens fuel, using the HTP correlation with a minimum DNBR of 1.14 will provide a 95% probability at a 95% confidence level that DNB will not occur during any operating condition. The CE-1 correlation is more restrictive than the HTP correlation that will be used to predict the minimum DNBR limits for the Siemens fuel. For a given set of reactor coolant conditions, the CE-1 correlation provides a lower critical heat flux than the HTP correlation. Therefore, the current TS Figure 1-1 is conservative for Siemens fuel and no revision is proposed to TS Figure 1-1.

Technical Specification 5.9.5

It is proposed that Technical Specification 5.9.5, "Core Operating Limits Report," be revised to include NRC-approved methodologies necessary for evaluating core limits utilizing nuclear fuel from Siemens Power Corporation. The wording for identifying the version of the topical report, "approved version as specified in the COLR," is consistent with recommended wording provided to Siemens Power Corporation in a letter from the NRC dated December 15, 1999. The wording for the remaining topical reports contained in TS 5.9.5 (e.g., OPPD-NA-8301-P-A, etc.) are being revised to be consistent.

NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes have been evaluated against the standards in 10 CFR 50.92, "Issuance of Amendment," and have been determined to not involve a significant hazards consideration. In support of this determination, a discussion of each of the significant safety hazards consideration factors with respect to the proposed license amendment is provided.

1. *The proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed amendment is to incorporate Siemens Power Corporation topical reports for conducting reload analyses that have been previously reviewed and approved by the NRC. The applicable FCS Technical Specifications (TS) supported by these topical reports are being revised. These changes are necessary to support using nuclear fuel supplied by Siemens Power Corporation.

   It is proposed to revise the Bases of TS 1.1 and 1.3 to reflect changes in methodologies for calculating the minimum Departure from Nucleate Boiling Ration (DNBR). The proposed methodology for determining the minimum DNBR for fuel supplied by Siemens Power Corporation is the NRC-approved EMF-92-153(P)(A) and Supplement 1, *HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel.* As stated in the Basis of TS 1.1, Fort Calhoun Station currently uses the NRC-approved CE-1 correlation with a minimum DNBR value of 1.18, which provides a 95% probability at a 95% confidence level that DNB will not occur for any operating condition. For Siemens fuel, using the HTP correlation with a minimum DNBR of 1.14, as proposed, will continue to provide a 95% probability at a 95% confidence level that DNB will not occur during any operating condition. The CE-1 correlation is more restrictive than the HTP correlation that will be used to predict the minimum DNBR limits for the Siemens fuel. For a given set of reactor coolant conditions, the CE-1 correlation provides a lower critical heat flux than the HTP correlation. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

   It is proposed that the total planar radial peaking factor, F_{xy}^T, be eliminated from the Technical Specifications. The current need for this parameter is to protect assumptions about the maximum amount of planar peaking in the core. The limitation on the total planar radial peaking factor, F_{xy}^T, is provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density – High, Limiting Conditions for Operation, and Limiting Safety Systems Settings set-points remain valid during operation. In a two-dimensional set-point analysis, as currently conducted, F_{xy}^T is combined with the maximum axial power profile (F_z) to produce the maximum allowable peaking factor (F_q) or equivalent Linear Heat Rate. This ensures conservative operation relative to assumptions on linear heat rate used as input to the loss of coolant accident and other transient analyses. In a three-dimensional analysis, as proposed with the use of Siemens methodology, these peaks are calculated directly during a series of pre-determined maneuvers (axial shape oscillation, power maneuver, or other transient).

NO SIGNIFICANT HAZARDS CONSIDERATION (Cont.)

Direct calculation of these peaks negates the need to make inferences about the amount of planar radial peaking that occurs in any particular plane within the core. Therefore, this change will not significantly increase the probability or consequences of an accident previously evaluated.

It is proposed to add NRC-approved methodologies from Siemens Power Corporation to TS that are necessary to evaluate core parameters. The proposed additions of NRC-approved topical reports to the TS do not modify the manner in which the topical reports may be implemented. The core operating limits will continue to be determined using NRC-approved analytical methods. The plant will continue to operate within the limits specified by the Core Operating Limits Report and will take corrective actions as required by the current Technical Specifications should these limits be exceeded. Therefore, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

2. *The proposed amendment would not create the possibility of a new or different kind of accident from any other accident previously evaluated.*

   No new or different modes of operation are proposed as a result of these changes. The proposed revisions do not change any equipment required to mitigate the consequences of an accident. The proposed additions of NRC-approved topical reports to the TS do not modify the manner in which the topical reports may be implemented. The plant will continue to operate within the limits specified by the Core Operating Limits Report and will take corrective actions as required should these limits be exceeded. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed amendment would not involve a significant reduction in a margin of safety.*

   As required by TS 5.9.5, the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The proposed changes incorporate methodologies applicable for use with fuel supplied by Siemens Power Corporation that have been approved by the NRC as documented by Safety Evaluation Reports. Technical Specification 5.9.5 also requires that the core operating limits shall be determined so that all applicable limits of the safety analysis are met. These requirements will continue to be met. Therefore, OPPD concludes that the proposed changes do not involve a significant reduction in a margin of safety.

OPPD has determined that operation with the proposed amendment would not result in any significant change in the types, or significant increases in the amounts, of any effluents that may be released offsite, nor would it result in any significant increase in individual or cumulative occupational radiation exposure.

These proposed changes to the Technical Specifications and our determination of no significant hazards have been reviewed by our Plant Review Committee and Safety Audit and Review

NO SIGNIFICANT HAZARDS CONSIDERATION (Cont.)

Committee. These committees have concluded that implementation of these changes will not result in undue risk to the health and safety of the public.

Based on the above considerations, OPPD concludes that the proposed amendment to FCS Technical Specifications does not involve a significant hazards consideration as defined by 10 CFR 50.92 and that the proposed amendment will not result in a condition which significantly alters the impact of the station on the environment. Thus, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and, pursuant to the 10 CFR 51.22(b), no environment assessment need be prepared.