March 14, 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 (TAC NO. MB0083)

Dear Mr. Gambhir:

The Commission has issued the enclosed Amendment No. 196 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 September 5, 2000, as supplemented by letters dated September 28, 2000, December 1, 2000, and December 11, 2000.

The amendment revises TS Sections 1.1, 1.3, 2.10, 3.10, and 5.9 and associated Bases to allow use of NRC-approved Siemens Power Corporation (SPC) methodologies for determining reactor core operating limits in conjunction with use of SPC fabricated nuclear fuel. Additionally, the revised SPC fuel assembly growth model for Cycle 20 core reload was reviewed and approved.

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. 196 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 14, 2001

Mr. S. K. Gambhir
Division Manager - Nuclear Operations
Omaha Public Power District
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L. Raynard Wharton

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. 196 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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
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 September 5, 2000, as supplemented by letters dated September 28, 2000, December 1, 2000, and December 11, 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. 196 , 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 14, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 196

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
7	7
1-1	1-1
1-2	2-1
1-8	1-8
1-9	1-9
2-57a	2-57a
2-57b	2-57b
2-57c	2-57c
2-57d	2-57d
2-57e	2-57e
3-63a	3-63a
5-17a	5-17a
---	5-17b
5-18	5-18

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 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 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 correlation. The local DNB ratio (DNBR), defined as the ratio of the heat flux

1.1 Safety Limits - Reactor Core (continued)

that 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 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 for reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the 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. 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]

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 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 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 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 ΔT-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 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

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 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) Deleted

(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_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_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) 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 does not exceed the power limits of the F_R^T and 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.

(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

(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, provides adequate monitoring of the core power distribution and 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. 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. 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 and the F_R^T and 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.

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* 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.

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 Integrated Radial Peaking Factor (F_R^T) and Azimuthal Power Tilt (T_q)

The limitation of T_q 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_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 $F_R^T = F_R(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_R^T and T_q are within their limits provide assurance that the actual values of F_R^T and T_q do not exceed the assumed values. Verifying 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 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.

(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 $\pm$ 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_L 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 Radial Peaking Factor (F_R^T)

F_R^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.

5.9.5 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," approved version as specified in the COLR.

2. OPPD-NA-8302-P-A, "Neutronics Design Methods and Verification," approved version as specified in the COLR.

3. OPPD-NA-8303-P-A, "Transient and Accident Methods and Verification," 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.

5.10 Record Retention

5.10.1 The following records shall be retained for at least five years:

a. Records, and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Licensee Event Reports (LER).

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. DELETED.

h. Records of annual physical inventory of all source material of record.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 196 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 September 5, 2000, as supplemented by letters dated September 28, 2000, December 1, 2000, and December 11, 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 allow the use of NRC-approved Siemens Power Corporation (SPC) methodologies for determining reactor core operating limits and permit the use of SPC fabricated nuclear fuel at FCS. Additionally, the September 28, 2000, submittal requested staff review and approval of a revised fuel assembly growth model developed by SPC for the Cycle 20 core reload and future core loadings in FCS.

OPPD has experienced a significant number of Westinghouse fuel failures at FCS over several operating cycles. OPPD has changed fuel vendors from Westinghouse to Siemens and has submitted the subject amendment in an effort to address the fuel failures at FCS.

The September 28, December 1 and 11, 2000, supplemental letters provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the *Federal Register* on December 27, 2000 (65 FR 81925).

2.0 SIEMENS FUEL EVALUATION

In support of the fuel transition program from Combustion Engineering/Westinghouse nuclear fuel to SPC nuclear fuel, OPPD, proposed changes to Technical Specification (TS) 2.10.4 and 3.10, as well as proposed revisions to the Bases of TS 1.1 and 1.3. These changes reflect the improvement in computer code development and methodologies that are incorporated into the NRC-approved SPC computer codes and methodologies regarding transition cores.

The Cycle 20 core loading will be composed of 53 SPC high thermal performance (HTP) fuel assemblies, 40 Batch X fuel assemblies manufactured by Westinghouse with Inconel grids, and 40 assemblies from Batch T with Zircaloy grids which were manufactured by Westinghouse and are currently in the FCS spent fuel pool. OPPD pointed out that any failed fuel rods in the Batch T assemblies will be repaired prior to insertion into the reactor. The SPC supplied fuel

will also consist of SPC-designed HTP spacer grid and FUELGUARD™ lower tie plates. The HTP spacer provides line contact with the fuel rods thus reducing the potential for fretting. The FUELGUARD™ lower tie plate provides protection against debris and improves resistance to flow induced fretting. The burnable absorber used in the SPC fuel will be gadolinia. The SPC fuel assembly components are part of the standard SPC design and have been used previously with no reported fuel failures at HTP spacer locations.

The reload analyses for Cycle 20 will be performed by both OPPD and SPC. The reload analyses will consist of mechanical, neutronics, thermal-hydraulic, transient, setpoint and LOCA analyses. OPPD has the primary responsibility for the reload analyses but has contracted with SPC to perform selected analyses for Cycle 20. The use of SPC analyses necessitated a change to the list of approved methodology in Section 5 of the Administrative Controls of the FCS Technical Specifications. The list of methods to be used in the reload analyses is specified in TS 5.9.5.b. The interrelationship between the analyses performed by OPPD and the analyses performed by SPC is depicted in Figure 1 of the licensee's December 1, 2000, letter.

The performance of the analyses by both OPPD and SPC was performed in a planned and controlled manner to assure that problems do not arise due to the interfaces. The analysis methods used to perform the analysis have been previously reviewed and approved by the NRC, with the exception of the assembly growth correlation which is discussed in Section 3.0 of this safety evaluation. In a meeting conducted with OPPD, SPC, and the staff on May 31, 2000, it was indicated that appropriate interfaces had been established between OPPD and SPC starting with formal documents, the design interface document, and calculation plans. Assumptions inherent to the required data exchanges were mutually understood and documented, and that meetings were held between OPPD and SPC to discuss the shared analyses and results.

Typically, the departure from nucleate boiling ratio (DNBR) is defined by the particular type of correlation used in the analysis. The CE-1 (Combustion Engineering or Westinghouse nuclear fuel) correlation DNBR limit will be different from that of the high thermal performance (SPC nuclear fuel) correlation DNBR limit. Consequently, no specific value for the DNBR limit is stated in the Bases of TS 1.1 and 3.1. This is because OPPD will use the appropriate NRC-approved DNB correlation and corresponding DNBR limit in their cycle specific analysis to ensure that the thermal margin DNBR limit is not violated for any of the anticipated combinations of transient conditions initiated within the limiting conditions of operations in combination with the reactor protection systems. Use of the appropriate NRC-approved DNB correlation is specified in TS 5.9.5.

2.1 Changes to Technical Specifications 2.10.4 and 3.10

OPPD proposed to delete the definition of the unrodded planer radial peaking factor (F_{xy}) and TS 2.10.4(3). They also proposed to revise TS 3.10(5) by deleting the surveillance requirement for the total planer radial peaking factor (F_{xy}^T).

In a two-dimensional setpoint analysis, as currently conducted at FCS, 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 (LHR). In a three-dimensional analysis, as that used by the SPC

methodology, the peaking factors are calculated directly at a three-dimensional given point in the core during a series of pre-determined maneuvers such as axial shape oscillation, power maneuver, or some other transient. Thus in a three-dimensional calculation, there is no need for a planer peaking factor. The staff agrees with this proposal.

2.2 Changes to Bases of Technical Specifications 1.1 and 1.3

Since OPPD is switching fuel vendors from ABB-CE/W to SPC, it is necessary to switch DNB correlations to correspond to the appropriate vendor. This necessitates a revision to the Bases of TS 1.1 and 1.3 to delete the discussion on calculating the minimum DNBR using the CE-1 correlation and its associated value.

Operating limit curves contained in TS Figure 1-1 describe the region of safe operations based on core power, reactor coolant pressure, and reactor coolant temperature conditions. FCS current limit curves were developed by ABB-CE/W using the NRC approved CE-1 correlation with a DNBR value of 1.18. The DNBR limit for the SPC fuel is 1.14, obtained by using the HTP correlation. Both these DNBR numbers were obtained based on a 95 percent probability at a 95 percent confidence level. However, since the ABB-CE correlation bounds (i.e., the CE-1 correlation is more restrictive than the HTP correlation), the SPC correlation, OPPD decided to continue using the current TS Figure 1-1. Consequently, OPPD concluded that no revisions need be made to TS Figure 1-1. The staff agrees with this conclusion.

OPPD also proposed revision to TS 5.9.5, "Core Operating Limits Report," to include NRC approved methodologies necessary for evaluating core limits utilizing nuclear fuel from SPC. The wording for identifying the version of the topical report, "approved version as specified in the COLR," is consistent with recommended wording provided to SPC in a letter from the NRC dated December 15, 1999. The staff agrees with this revision.

2.3 Conclusion

The staff reviewed the submitted information regarding the proposed change of fuel vendor and the affected FCS TS and finds that it is acceptable, because both OPPD and SPC used NRC-approved methodologies that have been applied successfully to similar fuel transitions. The revisions and changes to the above mentioned TS are also acceptable because they are in keeping with the change in methodologies.

3.0 REVISED FUEL ASSEMBLY GROWTH MODEL EVALUATION

The Cycle 20 core design is a mixed fuel design consisting of 53 new SPC HTP fuel assemblies, 40 Batch X fuel assemblies, and 40 Batch T fuel assemblies. Both Batches T and X fuel assemblies were manufactured by Westinghouse, and have been irradiated in previous cycles. The reload analyses for Cycle 20 will be performed by both OPPD and SPC.

Traditionally, the FCS fuel design is a CE type fuel design of a 14x14 array without hold-down springs in the upper end fitting. All the SPC and Westinghouse fuel assemblies for FCS reload design are consistent with this CE design. During reactor operations, fuel assemblies and fuel rods will grow axially under irradiation. In most circumstances, fuel rods are observed to grow faster than the rest of the fuel assembly. When fuel rods start reaching both upper and lower

end fittings, a phenomenon called shoulder gap closure occurs. The shoulder gap closure may cause the assemblies as well as the fuel rods to bow in a way that may impede fuel thermal-hydraulic performance. Thus, maintaining an adequate shoulder gap to prevent bowing is one of the major concerns in the fuel design.

Both the fuel assembly growth model and the fuel rod growth model are required to analyze the shoulder gap tolerance. For SPC fuel, there were four different approved fuel assembly growth models. However, the model for CE 14x14/16x16 type assemblies considers fuel with hold-down springs, while the FCS fuel design has no hold-down springs. A fuel assembly without hold-down springs tends to grow faster than a fuel assembly with hold-down springs. Therefore the SPC model for CE 14x14/16x16 type assemblies is not adequate for FCS. There is another SPC growth model for CE 15x15 type assemblies with no hold-down springs. SPC intends to apply the growth model for CE 15x15 type assemblies to demonstrate that this model is expected to conservatively overpredict and thus bound the FCS fuel design in end-of-life (EOL) assembly growth analysis.

An assembly growth model is a plot of assembly growth versus assembly average fast fluence. SPC revised the assembly growth model by extending the data base to the high fluence regime. The revised model shows two different growth rates: a smaller growth rate in the low fluence regime, and a larger growth rate in the high fluence regime. The larger growth rate is based on the observation that the growth rate of recrystallized zircaloy approaches the larger growth rate of cold worked stress-relieved (CWSR) zircaloy in the high fluence regime. The guide tubes responsible for the assembly growth are fabricated from recrystallized zircaloy. Due to the limited amount of the growth data for guide tubes, SPC used growth rate data from CWSR fuel cladding to augment the capability of the assembly growth model in the high fluence regime. The revised growth model for CE 15x15 type assemblies shows a plot of best-estimate and upper as well as lower uncertainty bound curves. Based on the staff's understanding of the zircaloy material characteristics, the staff agrees with the SPC proposal to incorporate CWSR behavior in the high fluence into the revised assembly growth model.

The revised assembly growth model was compared with Westinghouse type assembly growth measured data at EOL. The result showed that the revised growth model conservatively bounds the Westinghouse type growth data for best-estimate and 95 percent upper bound calculations. Furthermore, SPC compared the revised growth model to an independent source of data for fully recrystallized zircaloy in the unrestrained condition. The results showed that the revised growth model also bounds the independent data. Based on these two comparisons against available data, SPC concluded that the use of the revised growth model for CE 15x15 type assemblies bounds the FCS fuel design.

The staff has examined the overall analyses, and found that there is enough conservatism in the revised growth model to adequately predict the FCS assembly growth at EOL. Therefore, the staff concludes that the revised assembly growth model is acceptable for FCS reload licensing applications.

The staff has reviewed OPPD's submittal for the revised assembly growth model for CE 15x15 type assemblies. Based on the conservatism in the model predictions, which bound the available data, the staff concludes that the revised assembly growth model is acceptable for FCS Cycle 20 reload licensing applications.

4.0 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.

5.0 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 81925). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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.

6.0 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.

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Date: March 14, 2001