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

May 12, 2009

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS TO CORRECT TYPOGRAPHICAL ERRORS AND MAKE ADMINISTRATIVE CLARIFICATIONS (TAC NO. MD9186)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 259 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 2, 2008, as supplemented by emails dated February 18 and May 5, 2009.

The amendment corrects several typographical errors and makes administrative clarifications to the TSs. The NRC staff denies the heading changes to TS Limiting Condition for Operation (LCO) 2.13 Table 2-11 and TS LCO 2.14 Table 2-1 which are not editorial or administrative in nature and, therefore, are not acceptable. All other changes associated with this LAR are approved as requested.

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,

for.

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 259 to DPR-40
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 259
Renewed 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 July 2, 2008, as supplemented by emails dated February 18 and May 5, 2009, 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, Renewed Facility Operating License No. DPR-40 is amended by changes as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 259, 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 180 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-40 and Technical Specifications

Date of Issuance: May 12, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 259

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of the Renewed Facility Operating License No. DPR-40 and the 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.

License Page

| REMOVE | INSERT |
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Technical Specifications

| REMOVE | INSERT |
|---|---|
| 2.1 – Page 1 | 2.1 – Page 1 |
| 2.1 – Page 4 | 2.1 – Page 4 |
| 2.8 – Page 8 | 2.8 – Page 8 |
| 2.10 – Page 15 | 2.10 – Page 15 |
| 2.13 – Page 5 | 2.13 – Page 5 |
| 2.14 – Page 5 | 2.14 – Page 5 |
| 5.0 – Page 5 | 5.0 – Page 5 |
| 5.0 – Page 16 | 5.0 – Page 16 |
| 5.0 – Page 17 | 5.0 – Page 17 |

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is, subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 259 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan," submitted by letter dated May 19, 2006.

# 2.0 **LIMITING CONDITIONS FOR OPERATION**

## 2.1 Reactor Coolant System

### 2.1.1 Operable Components

Applicability

Applies to the operable status of the reactor coolant system components.

Objective

To specify certain conditions of the reactor coolant system components.

Specifications

Limiting conditions for operation are as follows:

(1) Reactor Critical

All four (4) reactor coolant pumps shall be in operation.

Exceptions

The limitations of this specification may be suspended during the performance of physics tests provided the power level is $\leq 10^{-1}$% of rated power and the flow requirements of Table 2-11 No. 2 are met.

(2) Hot Shutdown or 350°F $\leq T_{cold} \leq$ 515°F

(a) The reactor coolant loops listed below shall be operable:

(i) Reactor coolant loop 1 and at least one associated reactor coolant pump.

(ii) Reactor coolant loop 2 and at least one associated reactor coolant pump.

(b) At least one of the above reactor coolant loops shall be in operation.

Exceptions

All reactor coolant pumps may be de-energized for up to one hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

(12) Reactor Coolant System Pressure Isolation Valves

(a) The integrity of all pressure isolation valves listed in Table 2-9 shall be demonstrated, except as specified in (b). Valve leakage shall not exceed the amounts indicated.

(b) In the event that the integrity of any pressure isolation valve specified in Table 2-9 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a nonfunctional valve are in and remain in the mode corresponding to the isolated condition. Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supply deenergized.

(c) If Specifications (a) and (b) above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

Basis

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation and maintain DNBR above the minimum DNBR limit during all normal operations and anticipated transients.

When Specification 2.1.1(2) is applicable, the reactor coolant pumps (RCPs) are used to provide forced circulation heat removal during heatup and cooldown. Under these conditions, decay heat removal requirements are low enough that a single reactor coolant system (RCS) loop with one RCP is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to provide redundant paths for decay heat removal. Only one RCP needs to be OPERABLE to declare the associated RCS loop OPERABLE. Reactor coolant natural circulation is not normally used but is sufficient for core cooling. However, natural circulation does not provide turbulent flow conditions. Therefore, boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be assured.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature ($T_c$) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature ($T_c$) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

1. Immediately initiate action to restore the temperature or pressure to within the limit.

2. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

3. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

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

(c) When the linear heat rate is continuously monitored by the excore detectors, withdraw the full length CEA's beyond the long term insertion limits of Specification 2.10.2(7) and maintain the Axial Shape Index, $Y_I$ within the limits of Limiting Condition for Operations for the Excore Monitoring of LHR Figure provided in the COLR. If the linear heat rate is exceeding its limits as determined by the Axial Shape Index, $Y_I$, being outside the limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR:

(i) Restore the reactor power and Axial Shape Index, $Y_I$, to within the limits of the Limiting Condition for Operations for Excore Monitoring of LHR Figure provided in the COLR within 2 hours, or

(ii) Be in at least hot standby within the next 6 hours.

(d) When calibration of the ex-core detectors has not been accomplished within the previous 30 equivalent full power days, then:

(i) reduce the axial power distribution monitoring trip setpoints as shown in the Axial Power Distribution LSSS for 4 Pump Operation Figure provided in the COLR by 0.03 ASI units; and

(ii) reduce the axial power distribution monitoring Limiting Condition for Operations (LCO for Excore Monitoring of LHR and LCO for DNB Monitoring Figures provided in the COLR) by 0.03 ASI units.

When calibration of the ex-core detectors has not been accomplished within the previous 200 equivalent full power days, the power shall be limited to less than that corresponding to 75% of the peak linear heat rate permitted by Specification 2.10.4(1).

(2) Total Integrated Radial Peaking Factor

The calculated value of $F_R^T$ defined by $F_R^T = F_R (1+T_q)$ shall be within the limit provided in the COLR. $F_R$ is 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. The azimuthal tilt, $T_q$, is the measured value of $T_q$ at the time $F_R$ is determined.

## 2.0 LIMITING CONDITIONS FOR OPERATION
2.13 Limiting Safety System Settings, Reactor Protective System (continued)

**TABLE 2-11**

**RPS LIMITING SAFETY SYSTEM SETTINGS**

| No. | Reactor Trip | Trip Setpoints |
|---|---|---|
| 1 | High Power Level (A)<br>4-Pump Operation | ≤ 109.0% of Rated Power |
| 2 | Low Reactor Coolant Flow (B)(F)<br>4-Pump Operation | ≥ 95% of 4 Pump Flow |
| 3 | Low Steam Generator Water Level | ≥ 31.2% of Scale |
| 4 | Low Steam Generator Pressure (C) | ≥ 500 psia |
| 5 | High Pressurizer Pressure | ≤ 2400 psia |
| 6 | Thermal Margin/Low Pressure (B)(F) | 1750 psia to 2400 psia (depending on the reactor coolant temperature as shown in the Thermal Margin/Low Pressure 4 Pump Operation Figure provided in the COLR) |
| 7 | High Containment Pressure (D) | ≤ 5 psig |
| 8 | Axial Power Distribution (E) | (as shown in the Axial Power Distribution LSSS for 4 Pump Operation Figure provided in the COLR) |
| 9 | Steam Generator Differential Pressure | ≤ 135 psid |

A Setpoint cannot be set greater than 10% above measured power whenever reactor power is greater than 10% of rated power.

B May be bypassed below $10^{-4}$% power.

C May be bypassed below 600 psia.

D Bypass allowed for containment leak test.

E Inhibited below 15% power.

F For physics testing at power levels less than $10^{-1}$% of rated power, the low reactor coolant flow and thermal margin/low pressure trips may be bypassed until their reset points are exceeded if automatic bypass removal of $10^{-1}$% of rated power is operable.

TABLE 2-1 (continued)

**ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS**

| Functional Unit | Channel | Setting Limit |
|---|---|---|
| 6. 4.16 KV Emergency Bus Low Voltage | a. Loss of Voltage | (2995.2 + 104, -20.8) volts } Trip<br>≤ 5.9 seconds(4) } |
| | b. Degraded Voltage | |
| | i) Bus 1A3 Side | ≥ 3988.8 volts } Trip<br>(4.8 ± .5) seconds } |
| | ii) Bus 1A4 Side | ≥ 3990.6 volts } Trip<br>(4.8 ± .5) seconds } |
| 7. Low Steam Generator Water Level | Auxiliary Feedwater Actuation | ≥ 28.2% of wide range tap span |
| 8. High Steam Generator Delta Pressure | Auxiliary Feedwater Actuation | ≤ 119.7 psid |

(1) May be bypassed below 1700 psia and is automatically reinstated prior to exceeding 1700 psia.
(2) May be bypassed below 600 psia and is automatically reinstated prior to exceeding 600 psia.
(3) Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds).

# 5.0 **ADMINISTRATIVE CONTROLS**

5.7 Not used.

## 5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.24.

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

# 5.0 **ADMINISTRATIVE CONTROLS**

## 5.18 Process Control Program (PCP) (Continued)

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.

b. Shall become effective after the review and acceptance by the Plant Review Committee and the approval of the plant manager.

c. Temporary changes to the PCP may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire PCP as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the PCP was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

## 5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature $T_{cold}$ > 210°F.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

b. The containment design accident pressure ($P_a$) is 60 psig.

# 5.0 ADMINISTRATIVE CONTROLS

## 5.19 Containment Leakage Rate Testing Program (Continued)

c. The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, shall be 0.1% of containment air weight per day.

d. Leakage Rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is ≤ 1.0 $L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 $L_a$ Maximum Pathway Leakage Rate (MXPLR) for Type B and C tests and ≤ 0.75 $L_a$ for Type A tests.

2. Personnel Air Lock testing acceptance criteria are:

   (i) Overall Personnel Air Lock leakage is ≤ 0.1 $L_a$ when tested at ≥ $P_a$.

   (ii) For each PAL door, seal leakage rate is ≤ 0.01 $L_a$ when pressurized to ≥ 5.0 psig.

e. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

For each Containment Purge Valve, leakage rate is < 18,000 SCCM when tested at ≥ $P_a$.

f. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds 0.60 $L_a$ Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

g. The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

h. The provisions of Specifications 3.0.4 and 3.0.5 are applicable to the Containment Leakage Rate Testing Program.

## 5.20 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 259 TO RENEWED 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 July 2, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081890156), as supplemented by emails dated February 18 and May 5, 2009 (ADAMS Accession Nos. ML091170346 and ML091250264, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Technical Specifications (TSs) Limiting Conditions for Operation (LCO) for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed changes would correct several typographical errors and make administrative clarifications to the TSs.

The supplemental emails dated February 18 and May 5, 2009, provided additional information that clarified the application, 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 as published in the *Federal Register* on November 4, 2008 (73 FR 65697).

## 2.0 REGULATORY EVALUATION

In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the Nuclear Regulatory Commission (NRC) established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS. These proposed changes are editorial or administrative in nature as they correct the current FCS TSs to reflect changes previously approved by the NRC.

## 3.0 TECHNICAL EVALUATION

OPPD submitted this license amendment request (LAR) seeking to correct the following typographical errors and make administrative clarifications to the FCS TSs.

TS LCO 2.1.1(1)

The objective of TS LCO 2.1.1 is to specify certain conditions of the reactor coolant system components. As such, TS LCO 2.1.1(1) states that all four reactor coolant pumps shall be in operation. The exceptions for TS LCO 2.1.1(1) allow the limitations of this specification to be suspended during low power physics testing provided the reactor coolant flow requirements of Table 1.1 are met. Amendment No. 252, dated February 4, 2008 (Reference 8), relocated Table 1-1 to TS LCO 2.13 and renamed it Table 2-11. Thus, the correct reference for TS LCO 2.1.1(1) for the reactor coolant flow requirements is Table 2-11. The NRC staff has reviewed this proposed change and concludes that it corrects a typographical error and, therefore, is acceptable.

TS LCO 2.1.1(12)(a)

Table 2.9 of TS LCO 2.1.1(12)(a) is incorrectly spelled. The reference to Table 2.9 should be Table 2-9. The NRC staff has reviewed the proposed change to TS LCO 2.1.1(12)(a) and concludes that it corrects a typographical error and, therefore, is acceptable.

TS LCO 2.1.2c.

TS LCO 2.1.2c. states that the heatup rate of the pressurizer shall not exceed "I"00 degrees Fahrenheit in any one-hour period. The letter "I" is replaced with a "1" to correct this typographical error. The NRC staff has reviewed the proposed change to TS LCO 2.1.2c and concludes that it corrects a typographical error and, therefore, is acceptable

TS LCO 2.10.4(1)(d)

In Amendment No. 170, dated September 1, 1995 (Reference 1), OPPD deleted Figure 1-2 titled "Axial Power Distribution LSSS for 4 Pump Operation" from the FCS TS and relocated it to the Core Operating Limits Report (COLR) in accordance with NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 3, 1988. However, OPPD inadvertently overlooked a reference to Figure 1-2 contained in TS LCO 2.10.4(1)(d) and thus the reference, which should have been deleted, was not. To correct this oversight, the reference to Figure 1-2 is deleted and replaced with a reference to the Axial Power Distribution LSSS for 4 Pump Operation Figure, which is Figure 7 in the COLR. This change is consistent with the changes approved by the NRC in Amendment No.170 and thus is editorial in nature to remove an obsolete reference. The NRC has reviewed the proposed changes and agrees that it is administrative in nature as it corrects a reference citation and, therefore, is acceptable.

The licensee also proposes to delete an extraneous period in the reference to Specification 2.10.4(1). The licensee proposes to change "2.10.4.(1)" to "2.10.4(1)" in the last paragraph of TS LCO 2.10.4(1)(d). The NRC staff agrees that this is a typographical error and, therefore, is acceptable.

TS LCO 2.13, Table 2-11, RPS Limiting Safety System Settings

Table 2-11 lists each reactor trip and its associated setting. The licensee proposes to change the heading from "Trip Setpoint" to "Allowable Value". The NRC staff does not consider the heading change from "Trip Setpoint" to "Allowable Value" to be an editorial change as these are two different settings used for each instrument function. Therefore, in order for the NRC staff to approve this change, OPPD would need to provide revised instrument setpoint methodology and associated calculations. The licensee has decided to not provide revised instrument setpoint methodology and calculations and has acknowledged this denial by submitting new clean TS pages in its email dated February 18, 2009 (ADAMS Accession No. ML091170346). The NRC staff has reviewed this proposed change and concludes it is not editorial in nature and, therefore, is not acceptable.

In its application, the licensee stated that the reactor protective system (RPS) generates a reactor trip signal at 31.2 percent narrow-range steam generator water level, which is an abnormally low level. A water level this low indicates a loss of steam generator secondary water inventory, which if not corrected could result in the loss of the ability to remove heat from the reactor coolant system. Since the purpose of the column is to list the allowable value of the various trip settings. Therefore, the licensee states that it is more accurate to precede the low steam generator water level allowable value (31.2 percent) with a greater than or equal to sign. Adding a greater than or equal sign is consistent with the designation of allowable values in Table 3.3.1-1 of the ISTS, as well as with the designation of other RPS settings in Table 2-11. This change does not allow less conservative settings (i.e., less than 31.2 percent) but would allow settings greater than 31.2 percent, which would provide additional margin for protecting steam generator water levels. The NRC staff has reviewed this proposed change and concludes it is editorial in nature as it improves accuracy and readability and, therefore, is acceptable.

In its application, the licensee states:

> The change proposed to the title of the Axial Power Distribution LSSS for 4 Pump Operation Figure is consistent with the title of the applicable figure in the COLR (Figure 7) and with the changes made to TS 2.10.4(1)(d) as described above.
>
> Insertion of a comma into Footnote F is a grammatical correction that improves readability of the footnote when bypassing the low reactor coolant flow and thermal margin/low pressure trips during low-power physics testing.

The NRC staff has reviewed the proposed changes and concludes it is administrative in nature and, therefore, is acceptable.

The NRC staff has concluded that, aside from the column name change, the above changes are editorial in nature, are consistent with the ISTS and with the designations of other settings in Table 2-11, or otherwise improve accuracy or readability and, therefore, are acceptable.

<u>TS LCO 2.14, Table 2-1, Engineered Safety Features System Initiation Instrument Setting Limits</u>

OPPD proposes to make two clarifications to Table 2-1 which lists the engineered safety features (ESF) system trip setting limit. The NRC staff does not consider the heading change from “Setting Limit” to “Allowable Value” to be an editorial change as these are two different settings used for each instrument function. Therefore, in order for the NRC staff to approve this change, OPPD would need to provide revised instrument setpoint methodology and associated calculations. The licensee has decided to not provide revised instrument setpoint methodology and calculations and has acknowledged this denial by submitting new clean TS pages in its email dated February 18, 2009 (ADAMS Accession No. ML091170346). The NRC staff has reviewed this proposed change and concludes it is not editorial in nature and, therefore, is not acceptable.

The licensee stated in its application that a review of regulatory correspondence associated with amendments to Table 2-1 determined that the loss of voltage setting limit error has been manifesting itself since Amendment No. 157, dated November 22, 1993 (Reference 2). Prior to Amendment No. 157, the -20.8 (volts) was shown immediately below the +104 volts near the superscript for the reference to Footnote (4). In Amendment No. 157, OPPD did not request changes to Item 6. OPPD surmises, that due to its location near the footnote reference, a typographical error related to the issuance of Amendment No. 157 caused the -20.8 to relocate to the line below in the superscript to Footnote (4). This footnote is associated with the time delay for loss of voltage.

The proposed change corrects this discrepancy by placing both loss of voltage limits on the same line and clarifies the time delay setting by relocating the superscript (4) away from the numeric value to follow the word "seconds," which is a more appropriate location for the footnote.

The NRC agrees that, aside from the column name change, these changes are limited to the correction of typographical errors and the relocation of text for legibility, are editorial in nature and, therefore, are acceptable.

<u>TS 5.8.1d.</u>

TS 5.8.1d. lists additional programs for which written procedures and administrative policies are required. In recent years, several new programs have been added to TS 5.0, "Administrative Controls," and the need to update TS 5.8.1d. to reflect the addition of these programs was overlooked. Specifically, the new programs are:

- TS 5.22, Diesel Fuel Oil Testing Program
- TS 5.23, Steam Generator (SG) Program
- TS 5.24, Control Room Habitability Program

Accordingly, TS 5.8.1d. is revised to apply through TS 5.24. The NRC staff has reviewed the TS and found that the new programs are listed and the proposed change is administrative in nature and, therefore, is acceptable. The licensee has stated it has instituted an administrative control to evaluate TS 5.8.1d. whenever new programs are added to TS 5.0. As this evaluation of TS 5.8.1d. occurs prior to submitting an LAR to the NRC, this error should not recur.

TS 5.19

OPPD proposes the following changes to TS 5.19:

The first sentence of TS 5.19, "Containment Leakage Rate Testing Program," is revised to correct a punctuation error where a period, which should be a comma, separates the citation of 10 CFR 50, Appendix J, Option B. The NRC staff has reviewed this change and agrees that it is editorial in nature and, therefore, is acceptable.

In order to set the provisions of TS 5.19 apart from each other, OPPD proposes to format TS 5.19 using an alphanumeric numbering scheme that is similar to the format of TS 5.5.16, "Containment Leakage Rate Testing Program of ISTS" (Reference 3). This change is limited to segregating the provisions of TS 5.19 for clarity. The NRC staff has reviewed this change and agrees that it is editorial in nature and, therefore, is acceptable.

OPPD proposes to delete Exception (4), which was added to TS 5.19 by Amendment No. 220, dated August 15, 2003 (Reference 9). In Reference 4, as supplemented by References 5 and 6, OPPD had requested a one-time 5-year extension to the 10-year test interval for the Type A integrated leak rate test. OPPD successfully completed the Type A test during the 2008 refueling outage, which ended in June 2008. With the deletion of Exception (4), OPPD reverts to a 10-year test interval based on performance history in accordance with Option B of 10 CFR 50, Appendix J, as approved by the NRC in Amendment No. 185, dated March 23, 1998 (Reference 7). The proposed change is editorial in nature, as the exception no longer serves a purpose as the Type A test has been completed. The NRC staff has reviewed this change and agrees that it is editorial in nature and, therefore, is acceptable.

OPPD proposes to correct a typographical error in the leakage rate of the containment purge valves. Amendment No. 185 resulted in the establishment of the Containment Leakage Rate Testing Program of TS 5.19. The licensee's markup of TS 5.19 specified that the leakage rate acceptance criteria for the containment purge valves is less than 18,000 standard cubic centimeters per minute (SCCM) when tested at greater than or equal to (≥) Pa (containment design accident pressure). However, the TS page issued with Amendment No. 185 inadvertently changed the comma indicating thousand to a decimal point. This error was not discovered at the time the amendment was issued. TS 5.19 is revised to correct this typographical error and as such is editorial in nature. The NRC staff has reviewed this change and agrees that it is a typographical error and, therefore, is acceptable.

## 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 published in the *Federal Register* on November 4, 2008 (73 FR 65697). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 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.

## 7.0 REFERENCES

1. S. D. Bloom, U.S. Nuclear Regulatory Commission to T. L. Patterson, Omaha Public Power District, Letter, "Fort Calhoun Station. Unit No. 1 - Amendment No. 170 to Facility Operating License No. DPR-40 (TAC No. 92057)," dated September 1, 1995 (NRC-95-0184).

2. S. D. Bloom, U.S. Nuclear Regulatory Commission to T. L. Patterson, Omaha Public Power District, Letter, "Fort Calhoun Station, Unit 1 - Amendment No. 157 to Facility Operating License No. DPR-40 (TAC No. M86928)," dated November 22, 1993 (NRC-93-0397).

3. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications Combustion Engineering Plants," NUREG-1432, Vol. 1, Rev. 3.0, June 2004 (ADAMS Accession No. ML041830597).

4. D. J. Bannister, Omaha Public Power District to U.S. Nuclear Regulatory Commission, Letter, "Fort Calhoun Station, Unit 1, License Amendment Request, Risk-Informed One-Time Increase in Integrated Leak Rate Test Surveillance Interval," dated October 8, 2002 (ADAMS Accession No. ML022900093).

5. R. T. Ridenoure, Omaha Public Power District to U.S. Nuclear Regulatory Commission, Letter, "Response to Request for Additional Information, Integrated Leak Rate Testing Surveillance Interval Amendment Request (TAC No. MB6473)," dated April 11, 2003 (ADAMS Accession No. ML031070506).

6. R. T. Ridenoure, Omaha Public Power District to U.S. Nuclear Regulatory Commission, Letter, "Supplemental Response to Request for Additional Information, Integrated Leak Rate Testing Surveillance Interval Amendment Request (TAC No. MB6473)," dated May 21, 2003 (ADAMS Accession No. ML031490377).

7. L. R. Wharton, U.S. Nuclear Regulatory Commission to S.K. Gambhir, Omaha Public Power District, Letter, "Fort Calhoun Station, Unit No. 1 - Amendment No. 185 to Facility Operating License No. DPR-40 (TAC No. M99546)," dated March 23, 1998 (NRC-98-0051).

8. Michael T. Markley, U.S. Nuclear Regulatory Commission to David J. Bannister, Omaha Public Power District, Letter, "Fort Calhoun Station, Unit No. 1 – Issuance of Amendment [No. 252] Re: Adoption of Technical Specification Task Force (TSTF) 448-A, Revision 1 (TAC No. MD6808)," dated February 4, 2008 (ADAMS Accession No. ML073440385).

9. Alan B. Wang, U.S. Nuclear Regulatory Commission to R.T. Ridenoure, Omaha Public Power District, Letter, "Fort Calhoun Station, Unit No. 1 - Issuance of Amendment [No. 220] (TAC No. MB6473)," dated August 15, 2003 (ADAMS Accession No. ML032300003).

Principal Contributor: L. Wilkins

Date: May 12, 2009

May 12, 2009

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS TO CORRECT TYPOGRAPHICAL ERRORS AND MAKE ADMINISTRATIVE CLARIFICATIONS (TAC NO. MD9186)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 259 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 2, 2008, as supplemented by emails dated February 18 and May 5, 2009.

The amendment corrects several typographical errors and makes administrative clarifications to the TSs. The NRC staff denies the heading changes to TS Limiting Condition for Operation (LCO) 2.13 Table 2-11 and TS LCO 2.14 Table 2-1 which are not editorial or administrative in nature and, therefore, are not acceptable. All other changes associated with this LAR are approved as requested.

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 by N. Kalyanam for/

Alan B. Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 259 to DPR-40
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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LWilkins, NRR/DORL/LPL4

**ADAMS Accession No. ML083260697** *See previous concurrence

| OFFICE | NRR/LPL4/GE | NRR/LPL4/PM | NRR/LPL4/LA | DIRS/ITSB/BC | DE/EICB/BC | OGC-NLO wcomments | NRR/LPL4/BC | NRR/LPL4/PM |
|---|---|---|---|---|---|---|---|---|
| NAME | LWilkins* | AWang* | JBurkhardt* | RElliott* | WKemper* | AJones* | MMarkley | AWang NKalyanam for |
| DATE | 1/5/09 | 1/7/09 | 12/1/08 | 3/24/09 | 2/18/09 | 4/7/09 | 5/12/09 | 5/12/09 |

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