August 1999

Mr. C. Lance Terry
Senior Vice President
& Principal Nuclear Officer
TU Electric
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: RELOCATION OF CYCLE-SPECIFIC PARAMETERS (TAC NOS. MA5566 AND MA5567)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. NPF-87 and Amendment No. 67 to Facility Operating License No. NPF-89 for CPSES, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 24, 1999, as supplemented by letter dated July 9, 1999.

The amendments remove several cycle-specific parameter limits from the TSs. These parameter limits are added to the Core Operating Limits Report (COLR). Appropriate references to the COLR are inserted in the affected TSs. In addition, the core safety limit curves are replaced with safety limits more directly applicable to the fuel and fuel cladding fission product barriers. The affected TSs are (1) TS 2.0, "Safety Limits (SLs)," (2) TS 3.3.1, "Reactor Trip System Instrumentation Setpoints," (3) TS 3.4.1, "RCS [reactor coolant system] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and (4) TS 5.6.5, "Core Operating Limits Report."

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

Sincerely,

/s/

David H. Jaffe, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 67 to NPF-87
2. Amendment No. 67 to NPF-89
3. Safety Evaluation

070034

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

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. NPF-87

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

   A. The application for amendment by Texas Utilities Electric Company (TU Electric, the licensee) dated May 24, 1999, as supplemented by letter dated July 9, 1999, 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, as amended, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 67 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 30, 1999

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

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. NPF-89

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

   A. The application for amendment by Texas Utilities Electric Company (TU Electric, the licensee) dated May 24, 1999, as supplemented by letter dated July 9, 1999, 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, as amended, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 67 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This 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

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 30, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 67

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 67 TO

FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

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

Remove	Insert
2.0-1	2.0-1
2.0-2	2.0-2
2.0-3	2.0-3
3.3-21	3.3-21
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
5.0-32	5.0-32
5.0-33	5.0-33
5.0-34	5.0-34

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained ≥ the 95/95 DNB criterion for the DNB correlation(s) specified in Section 5.6.5.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Figure 2.1.1-1 (page 1 of 2)
Reactor Core Safety Limits (Unit 1)

[THIS FIGURE AND PAGE HAVE BEEN DELETED.]

Figure 2.1.1-1 (page 2 of 2)
Reactor Core Safety Limits (Unit 2)

[THIS FIGURE AND PAGE HAVE BEEN DELETED.]

Note 1: Overtemperature N-16

The Overtemperature N-16 Function Allowable Value shall not exceed the following setpoint by more than 1.72% of span for Unit 1, or 2.82% of span for Unit 2.

$$Q_{setpoint} = K_1 - K_2 \left[\frac{(1+\tau_1 s)}{(1+\tau_2 s)} T_C - T_C^o \right] + K_3(P - P^1) - f_1(\Delta q)$$

Where:

$Q_{setpoint}$	=	Overtemperature N-16 trip setpoint,
K_1	=	*,
K_2	=	*/°F
K_3	=	*/psig
T_C	=	Cold leg temperature
T_c^o	=	Reference T_C at RATED THERMAL POWER, *°F
P	=	Measured pressurizer pressure, psig
P^1	≥	* psig (Nominal RCS operating pressure)
s	=	the Laplace transform operator, sec^{-1}.
τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_c, $\tau_1 \geq$ * sec, and $\tau_2 \leq$ * sec

$f_1(\Delta q)$ =

$*\cdot\{(q_t - q_b) + *\%\}$	when $(q_t - q_b) \leq *\%$ RTP
0%	when $*\%$ RTP $< (q_t - q_b) < *\%$ RTP
$*\cdot\{(q_t - q_b) - *\%\}$	when $(q_t - q_b) \geq *\%$ RTP

Note 2: Not Used.

* as specified in the COLR

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ the limit specified in the COLR;

b. RCS average temperature ≤ the limit specified in the COLR; and

c. RCS total flow rate ≥ 389,700 gpm and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1

---NOTE---
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -------------NOTE------------- Only applicable prior to exceeding 85% RTP after a refueling outage. ----------------------------------- Measured RCS Flow not within limits.	B.1 Maintain THERMAL POWER less than 85% RTP.	Immediately
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ the limit specified in the COLR.	12 hours

(continued)

ACTIONS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.1.3	Verify RCS total flow rate is ≥ 389,700 and ≥ the limit specified in the COLR.	12 hours
SR 3.4.1.4	-------------------------------NOTE------------------------------- Not required to be performed until after exceeding 85% RTP after each refueling outage. -- Verify by precision heat balance that RCS total flow rate is ≥ 389,700 and ≥ the limit specified in the COLR.	18 months

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) Moderator temperature coefficient limits for Specification 3.1.3,

2) Shutdown Rod Insertion Limit for Specification 3.1.5,

3) Control Rod Insertion Limits for Specification 3.1.6,

4) AXIAL FLUX DIFFERENCE Limits and target band for Specification 3.2.3,

5) Heat Flux Hot Channel Factor, K(Z), W(Z), F_Q^{RTP}, and the $F_Q^C(Z)$ allowances for Specification 3.2.1,

6) Nuclear Enthalpy Rise Hot Channel Factor Limit and the Power Factor Multiplier for Specification 3.2.2.

7) SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6 and 3.1.8.

8) Refueling Boron Concentration limits in Specification 3.9.1.

9) Overtemperature N-16 Trip Setpoint in Specification 3.3.1.

10) Reactor Coolant System pressure, temperature, and flow in Specification 3.4.1.

11) Reactor Core Safety Limit figures (Safety Limit 2.1.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1) WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2) WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974 (W Proprietary).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (continued)

3) T. M. Anderson To K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980--Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

4) NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

5) WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).

6) WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary).

7) WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE", August 1985, (W Proprietary).

8) WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE", October 1986, (W Proprietary).

9) RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology, " June 1994.

10) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

11) RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1", December 1990.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (continued)

12) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

13) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

14) RXE-91-002-A, "Reactivity Anomaly Events Methodology", October 1993.

15) RXE-90-007-A, "Large Break Loss of Coolant Accident Analysis Methodology", April 1993.

16) TXX-88306, "Steam Generator Tube Rupture Analysis", March 15, 1988.

17) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

18) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3,4, and 5," February 1994.

19) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 67 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 67 TO

FACILITY OPERATING LICENSE NO. NPF-89

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated May 24, 1999, as supplemented by letter dated July 9, 1999, Texas Utilities Electric Company (the licensee) requested changes to the Technical Specifications (TSs) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed amendments would remove several cycle-specific parameter limits from the TSs. These parameter limits would be added to the Core Operating Limits Report (COLR). Appropriate references to the COLR would be inserted in the affected TSs. In addition, the core safety limit curves would be replaced with safety limits more directly applicable to the fuel and fuel cladding fission product barriers. The affected TSs are (1) TS 2.0, "Safety Limits (SLs)," (2) TS 3.3.1, "Reactor Trip System Instrumentation Setpoints," (3) TS 3.4.1, "RCS [reactor coolant system] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and (4) TS 5.6.5, "Core Operating Limits Report."

2.0 BACKGROUND

At the present time, the CPSES TSs contain parameters that are specifically applicable to the operation of CPSES for a particular fuel cycle. Guidance on the relocation of cycle-specific TS parameters to the COLR was developed by the NRC. This guidance was provided to all power reactor licensees and applicants by Generic Letter (GL) 88-16, dated October 4, 1988. Westinghouse Electric Company (Westinghouse) subsequently developed a topical report that describes how cycle-specific parameters, in TSs for facilities with nuclear steam supply systems designed by Westinghouse, can be transferred to the COLR. This topical report, WCAP-14483-A, "Generic Methodology for Expanding Core Operating Limits Report," was accepted for referencing by the NRC staff on January 19, 1999, and referenced by the licensee in its May 24, 1999, application.

3.0 EVALUATION

The following CPSES TSs have been evaluated concerning the transfer of cycle-specific parameters to the CPSES COLR:

TS 2.0, "Safety Limits (SLs)"

Figure 2.1.1-1 for Units 1 and 2 (Reactor Core Safety Limits) would be transferred to the COLR. The figure would be referenced in the Bases. The figure would be replaced by TS 2.1.1.1 which would state that in MODES 1 and 2 the departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to the 95/95 DNB criterion for the DNB correlation(s) specified in TS 5.6.5, and by TS 2.1.1.2, which would state that in MODES 1 and 2, the peak fuel centerline temperature shall be maintained less than 4700 °F. Therefore, the figures would be replaced with more specific requirements regarding the safety limits (i.e., the fuel DNB design basis and the fuel centerline melt design basis), conforming with WCAP-14483-A.

TS 3.3.1, Table 3.3.1-1, "Reactor Trip System Instrumentation Trip Setpoints"

The licensee has proposed that the numerical values pertaining to the overtemperature N-16 reactor trip setpoint be relocated to the COLR. The NRC has previously approved COLR additions for relocating the Overtemperature Delta T and Overpressure Delta T setpoint parameter values to the COLR for the Catawba, McGuire, South Texas, and Seabrook nuclear stations. This allows these setpoints to be based on cycle-specific core design parameters, which are verified on a cycle-specific basis, thereby avoiding the necessity of overly conservative TS limits. The applicable NRC-approved setpoint methodology, RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," dated June 1994, would be added to the list of approved analytical methods in TS 5.6.5.b.

TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

The licensee has proposed to relocate the pressurizer pressure, the RCS average temperature, and the RCS total flow rate to the COLR. These are cycle-specific values based on the total RCS flow (i.e., the sum of all four loop flows) and includes an allowance of 1.8 percent for the uncertainty associated with the measurement of the RCS flow.

The minimum RCS total flow rate of 389,700 gallons per minute would be retained in TS 3.4.1. This is the original licensed value approved by the NRC for CPSES, Unit 1, TS. Since this minimum value is retained in the TS, any reduction in RCS flow rate due to additional tube plugging or other physical plant change would have to be reviewed by the NRC staff.

TS 5.6.5, "Core Operating Limits Report (COLR)"

The COLR would be modified to reflect the relocation of the cycle-specific parameters, previously addressed, to the COLR and to add the appropriate approved references for the COLR parameters. In addition, the present references in TS 5.6.5.b would be updated to incorporate the approved versions, where appropriate.

The staff has reviewed the proposed TS revisions for CPSES, Units 1 and 2, and finds them in conformance with NRC GL 88-16 and with WCAP-14483-A and acceptable. Specifically, the revisions would relocate the reactor core limits figure for thermal power, pressurizer pressure, and the highest operating loop average coolant temperature for TS 2.1.1 to the COLR and replace the figure with the fuel integrity DNBR and peak fuel centerline melt temperature limits, which are the true safety limits. The numerical values pertaining to the overtemperature –16 trip setpoint values for TS 2.2 and the DNB-related parameters for RCS pressure, temperature, and flow would also be relocated to the COLR.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 35213 dated June 30, 1999, and 64 FR 40908 dated July 28, 1999). Accordingly, the amendments meet 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 amendments.

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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Kopp

Date: August 30, 1999

August 27, 1999

Mr. C. Lance Terry
Senior Vice President
& Principal Nuclear Officer
TXU Electric
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - CORRECTION OF AMENDMENT NO. 66 TO THE FACILITY OPERATING LICENSES

Dear Mr. Terry:

On August 3, 1999, the Commission issued Amendment No. 66 to Facility Operating License Nos. NPF-87 and NPF-89 for CPSES, Units 1 and 2, respectively. In the course of issuing the license amendments, a footnote on Technical Specification (TS) page 3.8-26 was incorrectly deleted. Enclosed with this letter is a corrected TS page 3.8-26.

We regret any inconvenience this matter has caused you.

Sincerely,

ORIGINAL SIGNED BY

David H. Jaffe, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosure: Corrected TS page 3.8-26

cc w/encls: See next page

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Comanche Peak Steam Electric Station

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.6	Verify each battery charger supplies ≥ 300 amps at ≥ 130 V for ≥ 8 hours.	18 months
SR 3.8.4.7	-------------------------------NOTES------------------------------ 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7*. 2. Verify requirement during MODES 3, 4, 5, 6 or with core off-loaded. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	 18 months

(continued)

* On a one time basis, for battery BT1ED2, a performance discharge test may be performed in lieu of the battery service test required by Specification 3.8.4.7, twice within a 60 month interval. This one time exception expires prior to entry into MODE 4 following the next Unit 1 outage of sufficient duration to perform a service test.