April 1, 1996

Mr. C. Lance Terry
Group Vice President, Nuclear
TU Electric
Energy Plaza
1601 Bryan Street, 12th Floor
Dallas, TX 75201-3411

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - AMENDMENT NOS. 49 AND 35 TO FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89 (TAC NOS. M94167 AND M94204)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment Nos. 49 and 35 to Facility Operating License Nos. NPF-87 and NPF-89 for the Comanche Peak Steam Electric Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 21, 1995 (TXX-95288), as supplemented by letters dated December 15, 1995 (TXX-95306), and February 2, 1996 (TXX-96040).

The amendments revise the core safety limit curves and revised N-16 Overtemperature reactor trip setpoints as a result of the reload analyses for CPSES Unit 2, Cycle 3. In addition, the minimum required Reactor Coolant System (RCS) flow is increased and an administrative enhancement is included in the footnotes of the RCS flow-low reactor trip function setpoint for both Units 1 and 2.

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,

Original signed by:
Timothy J. Polich, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 49 to NPF-87
2. Amendment No. 35 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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

April 1, 1996

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Group Vice President, Nuclear
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1601 Bryan Street, 12th Floor
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The amendments revise the core safety limit curves and revised N-16 Overtemperature reactor trip setpoints as a result of the reload analyses for CPSES Unit 2, Cycle 3. In addition, the minimum required Reactor Coolant System (RCS) flow is increased and an administrative enhancement is included in the footnotes of the RCS flow-low reactor trip function setpoint for both Units 1 and 2.

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.

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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. 49
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 November 21, 1995 (TXX-95288), as supplemented by letters dated December 15, 1995 (TXX-95306), and February 2, 1996 (TXX-96040), 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. <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy J. Polich, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 1, 1996

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. 35
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 November 21, 1995 (TXX-95288), as supplemented by letters dated December 15, 1995, (TXX-95306), and February 2, 1996 (TXX-96040), 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. , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy J. Polich

Timothy J. Polich, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 49 AND 35

FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-3	2-3
2-6	2-6
2-9	2-9
2-11	2-11
B 2-7	B 2-7
3/4 2-12	3/4 2-12

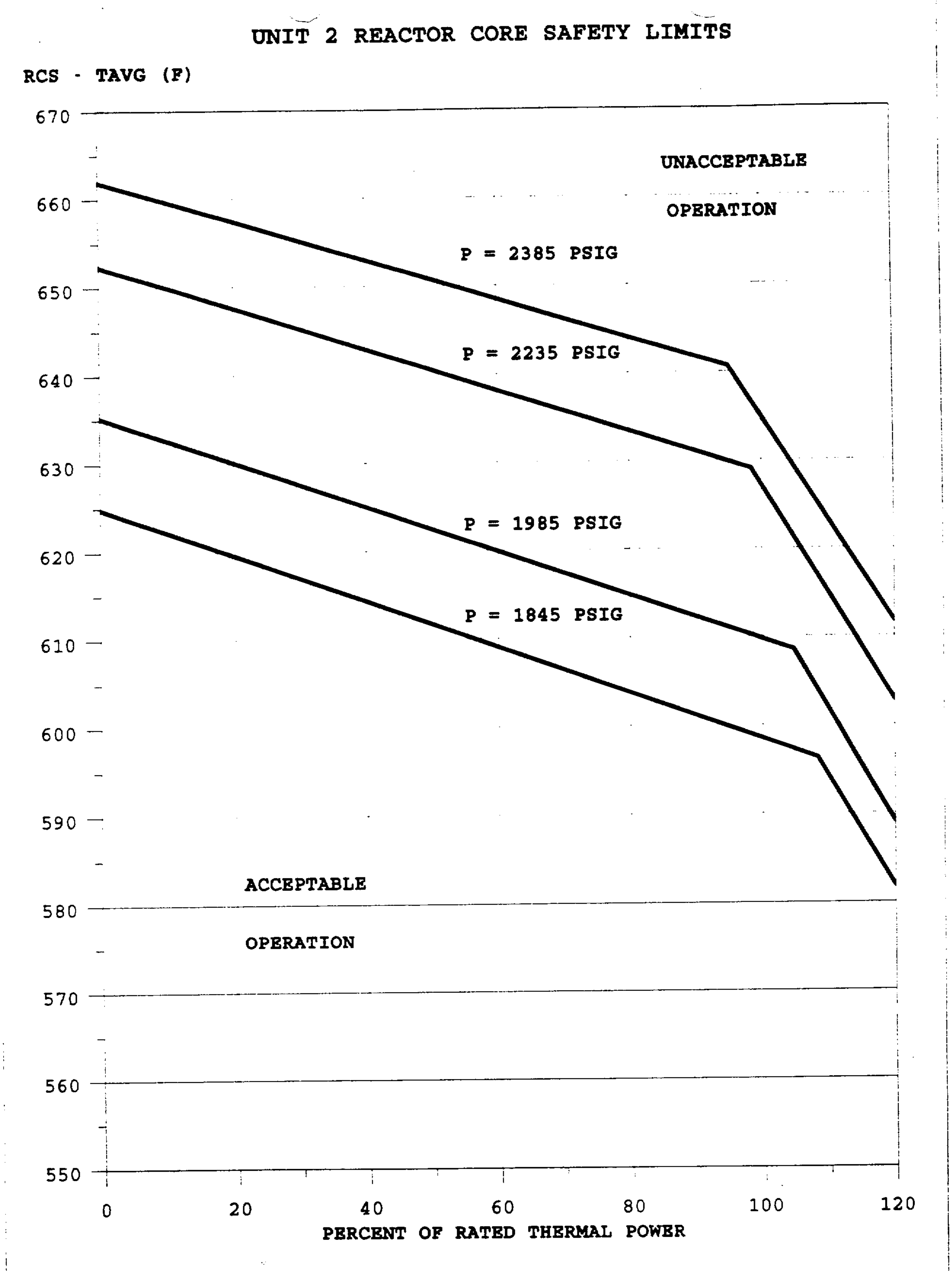

FIGURE 2.1-1b
UNIT 2 REACTOR CORE SAFETY LIMITS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤111.7% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.7 of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Not Used		
5. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31.5 of RTP*
6. Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature N-16	See Note 1	See Note 2

* RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
8. Overpower N-16	≤112% of RTP*	≤114.5% of RTP*
9. Pressurizer Pressure-Low		
a. Unit 1	≥1880 psig	≥1863.6 psig
b. Unit 2	≥1880 psig	≥1865.2 psig
10. Pressurizer Pressure-High		
a. Unit 1	≤2385 psig	≤2400.8 psig
b. Unit 2	≤2385 psig	≤2401.4 psig
11. Pressurizer Water Level-High	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low		
a. Unit 1	≥90% of instrument span	≥88.6% of instrument span
b. Unit 2	≥90% of instrument span	≥88.8% of instrument span

* RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: Overtemperature N-16

$$N = K_1 - K_2 \left[\frac{1 + \tau_1 s}{1 + \tau_2 s} T_c - T_c^{\circ}\right] + K_3 (P - P^1) - f_1 (\Delta q)$$

Where:

N	=	Measured N-16 Power by ion chambers,	
T_c	=	Cold leg temperature, °F,	
T_c°	=	560.5°F for Unit 1, 560.8°F for Unit 2 - Reference T_c at RATED THERMAL POWER,	|
K_1	=	1.150,	
K_2	=	0.0134/°F for Unit 1 0.0138/°F for Unit 2	|
$\frac{1 + \tau_1 s}{1 + \tau_2 s}$	=	The function generated by the lead-lag controller for T_c dynamic compensation,	
τ_1, τ_2	=	Time constants utilized in the lead-lag controller for T_c, $\tau_1 \geq 10$ s, and $\tau_2 \leq 3$ s,	
K_3	=	0.000719/psig for Unit 1 0.000720/psig for Unit 2	|

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P	=	Pressurizer pressure, psig,
P^1	$\geq$	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, s^{-1},

and $f_1(\Delta q)$ is a function of the indicated difference between top and bottom halves of detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

For Unit 1

(i) for $q_t - q_b$ between -65% and +4%, $f_1(\Delta q) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER,

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -65%, the N-16 Trip Setpoint shall be automatically reduced by 1.81% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +4%, the N-16 Trip Setpoint shall be automatically reduced by 2.26% of its value at RATED THERMAL POWER.

NOTE 1: (Continued)

For Unit 2

(i) for $q_t - q_b$ between -65% and +2.5%, $f_1(\Delta q) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER,

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -65%, the N-16 Trip Setpoint shall be automatically reduced by 1.86% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +2.5%, the N-16 Trip Setpoint shall be automatically reduced by 1.65% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.51% of span for Unit 1 or 1.88% of span for Unit 2.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-2.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), provides a backup block for Source Range Neutron Flux doubling, and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables the Reactor trip on low flow in one reactor coolant loop. On decreasing power, the P-8 automatically blocks the reactor trip on low flow in one reactor coolant loop.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and de-energizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Turbine first stage chamber pressure provides input to P-7.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE,

b. Calculating the ratio at least once per 12 hours when the alarm is inoperable, and

c. Calculating the ratio at least once per 12 hours when above 75% RATED THERMAL POWER with one Power Range channel inoperable.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the stated limits:

a. Indicated Reactor Coolant System $T_{avg} \leq 592°F$

b. Indicated Pressurizer Pressure ≥ 2219 psig*

c. Indicated Reactor Coolant System (RCS) Flow ≥ 403,400 gpm** for Unit 1
≥ 408,000 gpm** for Unit 2

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS total flow rate shall be verified to be within its limits at least once per 31 days by plant computer indication or measurement of the RCS elbow tap differential pressure transmitters' output voltage.

4.2.5.3 The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The channels shall be normalized based on the RCS flow rate determination of Surveillance Requirement 4.2.5.4.

4.2.5.4 The RCS total flow rate shall be determined by precision heat balance measurement after each fuel loading and prior to operation above 85% of RATED THERMAL POWER. The feedwater pressure and temperature, the main steam pressure, and feedwater flow differential pressure instruments shall be calibrated within 90 days of performing the calorimetric flow measurement.

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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 in excess of 10% of RATED THERMAL POWER.

**Includes a 1.8% flow measurement uncertainty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 49 AND 35 TO

FACILITY OPERATING LICENSE NOS. NPF-87 AND 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 November 21, 1995 (TXX-95288), as supplemented by letters dated December 15, 1995 (TXX-95306), and February 2, 1996 (TXX-96040), Texas Utilities Electric Company (TU Electric/the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License Nos. NPF-87 and NPF-89) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed changes would revise the core safety limit curves and revised N-16 Overtemperature reactor trip setpoints as a result of the reload analyses for CPSES Unit 2, Cycle 3. In addition, the minimum required Reactor Coolant System (RCS) flow is increased and an administrative enhancement is included in the footnotes of the RCS flow - low reactor trip function setpoint for both Units 1 and 2. The December 15, 1995, and February 2, 1996, supplemental letters were clarifying in nature and did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

TU Electric has changed the fuel supplier of CPSES for Units 1 and 2 from the Westinghouse Electric Company (WEC) to Siemens Power Corporation (SPC). SPC fuel will be supplied for Unit 2 in Cycle 3.

TU Electric has developed in-house analysis methodologies for the CPSES Units 1 and 2 which were approved by NRC prior to startup of Unit 1. TU Electric has expanded the referenced methodologies in TS Section 6.9.1.6b to include these methodologies developed in-house for the performance of the core reload licensing analyses. These methodologies are applicable to both CPSES Units 1 and 2, subject to the constraints of the applicable Safety Evaluations (SEs). These approved reload analysis methodologies are used to support CPSES Unit 2, Cycle 3 operation. For CPSES Unit 2 Cycle 3, the following analytical methods are used to determine the core safety limits and perform the departure from nucleate boiling (DNB)-related portion of the safety analyses:

WCAP-10079-P-A, "NOTRUMP, A NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," August 1985, (W Proprietary), (Ref. 1).

WCAP-10054-P-A, "WESTINGHOUSE SMALL BREAK ECCS EVALUATION MODEL USING THE NOTRUMP CODE," August 1985, (W Proprietary), (Ref. 2).

WCAP-11145-P-A, "WESTINGHOUSE SMALL BREAK LOCA ECCS EVALUATION MODEL GENERIC STUDY WITH THE NOTRUMP CODE," October 1986, (W Proprietary), (Ref. 3).

RXE-90-006-P, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," February 1991, (Ref. 4).

RXE-88-102-P, "TUE-1 Departure from Nucleate Boiling Correlation," January 1989, (Ref. 5).

RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1," December 1990, (Ref. 6).

RXE-89-002, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," June 1989, (Ref. 7).

RXE-91-001, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," February 1991, (Ref. 8).

RXE-91-002, "Reactivity Anomaly Events Methodology," May 1991, (Ref. 9).

RXE-90-007, "Large Break Loss of Coolant Accident Analysis Methodology," December 1990, (Ref. 10).

TXX-88306, "Steam Generator Tube Rupture Analysis," March 15, 1988, (Ref. 11).

RXE-91-005, "Methodology for Reactor Core Response to Steamline Break Events," May, 1991, (Ref. 12).

RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5," February 1994, (Ref. 13).

Using these methodologies and the changes in the RCS thermal design flow rate, calculations and analyses have been performed to identify the new core safety limit curves for Unit 2 (TS Figure 2.1-1b).

In addition to the analyses of the core safety limits and the DNB related parameters for the Unit 2, Cycle 3 core configuration (including revised Overtemperature N-16 setpoint equation coefficients), TU Electric intends to increase the RCS thermal design flow rate.

To enhance the DNB-related analyses of the mixed core configuration with the new analyses, TU Electric proposes to increase the thermal design flow value. Currently, the actual RCS flow is approximately 6.6 percent higher than the flow rate assumed in the CPSES Unit 2, Cycle 2 accident analysis. For Unit 2, Cycle 3, TU Electric proposes crediting 3.6 percent of the flow in the accident analyses, resulting in the definition of a higher RCS minimum required flow rate. Correspondingly, the TS minimum indicated total RCS flow requirement will also be increased from 395,200 gpm to 408,000 gpm.

3.0 EVALUATION

TU Electric proposed to use their in-house, NRC approved reload analysis methodologies for CPSES Units 1 and 2 to determine the core safety limits and to meet the applicable limits of the safety analyses. TU Electric will use a different DNB correlation, TUE-1, for performing the DNB-related analyses. The TUE-1 correlation has been approved by the NRC for use with Westinghouse and Siemens fuel, as well as in the mixed core configuration of Westinghouse standard fuel assemblies and Siemens fuel assemblies which will be co-resident in the core of CPSES Unit 2 during Cycle 3.

Because a different DNB correlation, TUE-1 (Ref. 8), is to be used for the CPSES Unit 2, Cycle 3 core configuration, new core safety limits have been calculated. The new core safety limits have been determined to insure that protective actions will be initiated to prevent the core from exceeding the DNB ratio limit and to prevent the core exit fluid conditions from reaching saturated conditions.

As a result of the new core safety limits, the Overtemperature N-16 trip setpoints were recalculated. In performing theses analyses, the RCS flow rate was increased.

Evaluations of the changes are described below:

3.1 Use of TU Electric topical reports that were approved by the NRC

The referenced methodologies in TS Section 6.9.1.6b, as amended on December 15, 1995, were expanded to include methodologies developed in-house, as listed above in Section 2.0, by TU Electric for the performance of core reload analyses. These methodologies can be applied to both CPSES Units 1 and 2, subject to the constraints of the applicable SEs. For CPSES Unit 2, Cycle 3, these methodologies will be used to determine the core safety limits and perform the DNB-related portion of the safety analyses. These methodologies will ensure that all applicable limits of the safety analyses are met for the reload core configuration. The NRC staff finds the use of these methodologies acceptable as they were previously reviewed and approved by the NRC.

3.2 Increase in the Unit 2 Reactor Coolant System Flow Rate

Using NRC approved methodologies for the DNB-related analyses of the mixed core configuration with the new analyses, TU Electric proposed to increase the RCS flow value assumed in the safety analyses by 3.6 percent for Unit 2, Cycle 3, to provide additional margin which may be used to demonstrate compliance with all applicable limits of the safety analysis. The proposed change in RCS flow rate also necessitates a change to the minimum indicated total RCS flow rate from 395,200 gpm to 408,000 gpm in TS 3.2.5c because of the relationship between the flow rate assumed in the safety analyses and the minimum required indicated flow. TU Electric stated that the current actual RCS flow rate (421,610 gpm) is unchanged and is approximately 6.6 percent higher than the RCS flow rate assumed in the previous CPSES Unit 2, Cycle 2 accident analyses. It is noted that this value of 6.6 percent is cycle dependant and is likely to decrease in succeeding cycles due to steam generator tube plugging and other degradations. The remaining 3.0 percent (6.6 percent minus 3.6 percent) RCS flow rate is sufficient to account for all uncertainties associated with the RCS flow rate (1.8 percent flow measurement uncertainty and 0.5 percent for the effects of the lower plenum flow anomaly) and the increased RCS flow resistance due to a full core of SPC fuel assemblies. TU Electric stated that the TS limits are consistent with the initial safety analysis assumption (plus uncertainties) and have been analytically demonstrated to be adequate to maintain a minimum DNBR at or above the safety analysis departure from nucleate boiling ratio (DNBR) limit throughout each analyzed transient. TU Electric used NRC approved methodologies for determining reactor core safety limits. A model of the CPSES Unit 2 mixed core configuration was developed to accurately account for the effects of the different co-resident fuel assembly designs.

The NRC staff finds the changes to the RCS flow rate in the Safety Analysis and the minimum indicated total RCS flow rate to be acceptable as approved methodologies were used and because there are acceptable margins available and the uncertainties are accounted for.

3.3 Revision to the Unit 2 Core Safety Limits

Beginning with Cycle 3, SPC will supply the nuclear fuel assemblies for Unit 2. During Cycle 3, the Siemens fuel assemblies will be co-resident with existing Westinghouse fuel assemblies.

TU Electric has used in-house reload analysis methodologies to determine the core safety limits and to meet applicable limits of the safety analyses for CPSES, Cycle 3. The in-house methodologies used by TU Electric to determine the core safety limits are wholly consistent with and represent no change to the TS 2.1 BASES for Safety Limits. TU Electric is using the NRC approved TUE-1 DNB correlation which has been approved by the NRC for the core configuration of Westinghouse standard fuel assemblies and Siemens fuel assemblies, including a mixture of these fuels which will be co-resident in the core of CPSES Unit 2 during Cycle 3. The calculation of the mixed core penalty used the approved code given in Reference 10. The effects of the mixed core on the large break LOCA analysis were evaluated in accordance with the approved methodology of Reference 10.

The core safety limits for CPSES Unit 2, Cycle 3 (TS Figure 2.1-1b) has been determined using the NRC approved TU Electric methodologies for determining core safety limits, an increase in the assumed RCS flow rate, and a safety analysis DNBR based on the NRC approved TUE-1 DNB correlation. The TUE-1 correlation DNBR limit plus margin constitutes the safety analysis DNBR limit.

The NRC staff has found the revisions to the Unit 2, Cycle 3 Core Safety Limits discussed above to be acceptable as they have been analyzed using NRC approved methodologies.

3.4 Revision to Unit 2 Overtemperature N-16 Reactor Trip Setpoints, Parameters and Coefficients

The Reactor Trip System setpoint limits specified in TS 2.2, Table 2.2-1 are the nominal values at which the reactor trips are set for each functional trip. The trip setpoints have been selected to ensure that the core and RCS are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. The Overtemperature N-16 trip function initiates a reactor trip which helps protect the core and RCS from exceeding their safety limits.

TU Electric stated that since the core safety limits have been changed for CPSES Unit 2, Cycle 3, the Overtemperature N-16 reactor trip setpoint was recalculated in accordance with the methods developed by TU Electric (listed in TS 6.9.1.6b). These are consistent with the BASES (BASES 2.2.1) for the Overtemperature N-16 reactor trip.

The Overtemperature N-16 reactor trip setpoint calculation includes the calculation of K_1, K_2, K_3 and $f_1(\Delta q)$ coefficients for the equation shown in TS 2.2, Table 2.2-1, Note 1. The $f_1(\Delta q)$ terms (the range for $q_t - q_b$ and the Overtemperature reductions when exceeding that range) are a function of axial flux difference and account for variation in the core axial power distributions.

The combination of the parameters affected by these coefficients in the Overtemperature N-16 reactor trip setpoint equation is designed to provide core safety limit protection by preventing DNB and core exit saturation for all combinations of pressure, power, coolant temperature, and axial power distributions. The K_1, K_2, K_3 coefficients are determined assuming a fixed reference (normal operations) axial power distribution; then, the compensation terms $f_1(\Delta q)$ are determined to account for variations in the axial power distribution during accident conditions. The combination of these parameters in the Overtemperature N-16 reactor trip setpoint equation is designed to provide reactor core safety limit protection by preventing DNB and core exit saturation for all combinations of pressure, power, coolant temperature, and axial power distribution.

The value of $T_c°$ (reference cold leg temperature at the minimum required RCS flow rate) for the Overtemperature N-16 trip setpoint equation in TS 2.2, Table 2.2-1, Note 1 was also changed. Due to the increase in the minimum required RCS flow rate as noted above in Section 3.2 above), the ΔT across the reactor vessel must decrease in order to maintain the same core power and reactor vessel average temperature. Performing an energy balance at rated thermal power with the higher value of the minimum required flow rate, a new value of $T_c°$ is determined.

After the safety analysis values for the Overtemperature N-16 reactor trip setpoint were determined, the instrumentation trip setpoints were determined.

These trip setpoints are defined by the Trip setpoint and Allowable Value in TS Table 2.2.1. The methodology to derive the Overtemperature N-16 reactor trip setpoint in Table 2.2-1 was based on the statistical combination of all of the uncertainties in the channels to arrive at a total uncertainty. Additional margin was applied in a conservative direction to arrive at the nominal trip setpoint value provided in TS Table 2.2-1. Because the safety analysis value for the Overtemperature N-16 reactor trip setpoint was changed, the nominal and allowable values also change. However, they are still calculated in a manner which is consistent with the current values.

The Overtemperature N-16 reactor trip setpoint helps prevent the core and RCS from exceeding their safety limits during normal operation and design basis anticipated operational occurrences. TU Electric stated in their supplemental submittal that all events were reviewed. Those events for which the Overtemperature N-16 trip function provides a primary protective or mitigative function were identified. The most relevant design basis analysis in Chapter 15 of the CPSES Final Safety Analysis Report (FSAR) which is affected by the change in the safety analysis value for CPSES Unit 2 Overtemperature N-16 reactor trip setpoint is the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (FSAR Section 15.4.2). With the exception of the Uncontrolled Rod Withdrawal from Power event (RWAP), none of the events are "limiting" with respect to the DNBR event acceptance criteria. TU Electric stated in their supplemental letter that the "resultant" DNBR for each transient is confirmed to be greater than the appropriate DNBR limit value (1.16 for deterministic methods, 1.429 for statistical methods for Unit 2 Cycle 3). TU Electric stated that the RWAP event has been re-analyzed with the revised safety analysis value for the Overtemperature N-16 reactor trip setpoint to demonstrate compliance with event specific criteria. A table was provided in the licensee's supplemental letter of the relevant event acceptance criterion for each non-LOCA FSAR Chapter 15 event considered during the core reload evaluation. The LOCA analysis were performed in accordance with References 1, 2, 3, and 10.

TU Electric stated that the CPSES Unit 2, Cycle 3 Overtemperature N-16 reactor trip setpoints are also sufficiently high such that the operational effects of the upper plenum flow anomaly on turbine runbacks or reactor trips will be minimized; thereby reducing the potential for challenges to the plant safety systems.

The NRC staff has found the changes the Overtemperature N-16 setpoints to be acceptable as they were recalculated using the TU Electric standard methods listed in TS 6.9.1.6b.

3.5 Deletion of the Footnotes Associated with the RCS Flow-Low Reactor Trip Function Setpoints, Unit 1 and Unit 2

Consistent with the Westinghouse Improved Standard Technical Specifications (ISTS) (NUREG-1431, Revision 1), the licensee proposed to delete the footnotes associated with RCS flow - low reactor trip function setpoints and to express the Allowable Value in percent of instrument span. The licensee stated that this will eliminate unnecessary information from the TS, thereby reducing the potential for cycle-specific changes. This change is proposed for Unit 2 in lieu of cycle-specific revision to footnote "**," loop design flow and "***," loop minimum measured flow and is also proposed for Unit 1 to maintain consistency between the units.

The licensee stated that in theory, with the current language of TS 2.2, Table 2.2-1, Functional Units 12.a and 12.b, the reactor trip setpoint on low RCS flow could be set such that the trip setpoint corresponded to 90 percent of the minimum RCS flow rate assumed in the accident analyses. Because the minimum RCS flow rate assumed in the accident analyses is less than the actual flow rate, the setpoint could potentially be set at some value less than 90 percent of instrument span. In practice, the trip setpoint is set at 90 percent of the instrument span, where the actual RCS loop flow corresponds to 100 percent (or perhaps slightly less) of the instrument span. The actual RCS flow is determined to be greater than the value assumed in the accident analysis through compliance with TS 3.2.5. The licensee stated that even though the deletion of the footnotes has no effect on the current practice, in theory, it could result in RCS - flow setpoints which are more restrictive than allowed with the current specifications. This restriction is conservative relative to the accident analysis assumptions, and has no impact with respect to actual plant operation. Due to the current method used to set the RCS flow - low setpoint, this change is essentially administrative in nature and is consistent with NUREG-1431. Therefore the NRC staff finds this change to be acceptable.

4.0 <u>STATE CONSULTATION</u>

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 <u>ENVIRONMENTAL CONSIDERATION</u>

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 (61 FR 185). 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.

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