November 16, 1993

Docket Nos. 50-445
and 50-446

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Mr. William J. Cahill, Jr.
Group Vice President, Nuclear
TU Electric
400 North Olive Street, L.B. 81
Dallas, Texas 75201

Dear Mr. Cahill:

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - AMENDMENT NOS. 21 AND 7 TO FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89 (TAC NOS. M86654 AND M86655)

The Commission has issued the enclosed Amendment Nos. 21 and 7 to Facility Operating License Nos. NPF-87 and NPF-89 for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated May 28, 1993, as supplemented by letter dated September 24, 1993.

The amendments incorporate changes related to Cycle 4 operations in Unit 1; specifically, core safety limit curves and N-16 overtemperature reactor trip setpoints are revised. In addition, the amendments permit the use of additional NRC approved methodologies, increase the minimum required reactor coolant system flow, remove a penalty on pressurizer pressure uncertainty, and include an operational enhancement for the treatment of the uncertainty allowance for the N-16 power indication.

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

Thomas A. Bergman, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 21 to NPF-87
2. Amendment No. 7 to NPF-89
3. Safety Evaluation

cc w/enclosures:
See next page

*See Previous Sheet for Concurrence

OFFICE	PDIV-2/LA	PDIV-2/PM	OGC*	PDIV-2/D	HICB*
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DF01

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Honorable Dale McPherson
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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. 21
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 28, 1993, as supplemented by letter dated September 24, 1993, 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. 21, 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William D Reckley for

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1993

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. 7
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 28, 1993, as supplemented by letter dated September 24, 1993, 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. Technical Specifications and Environmental Protection Plan

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 7, 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. The license amendment is effective as of its date of issuance to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William D Reckley for

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 16, 1993

ATTACHMENT TO LICENSE AMENDMENT NOS. 21 AND 7

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

DOCKET NOS. 50-445 AND 50-446

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-2	2-2
2-5	2-5
2-6	2-6
2-9	2-9
2-10	2-10
2-11	2-11
3/4 2-12	3/4 2-12
B 3/4 2-4	B 3/4 2-4
B 3/4 2-6	B 3/4 2-6
6-21	6-21
--	6-21a
6-22	6-22

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

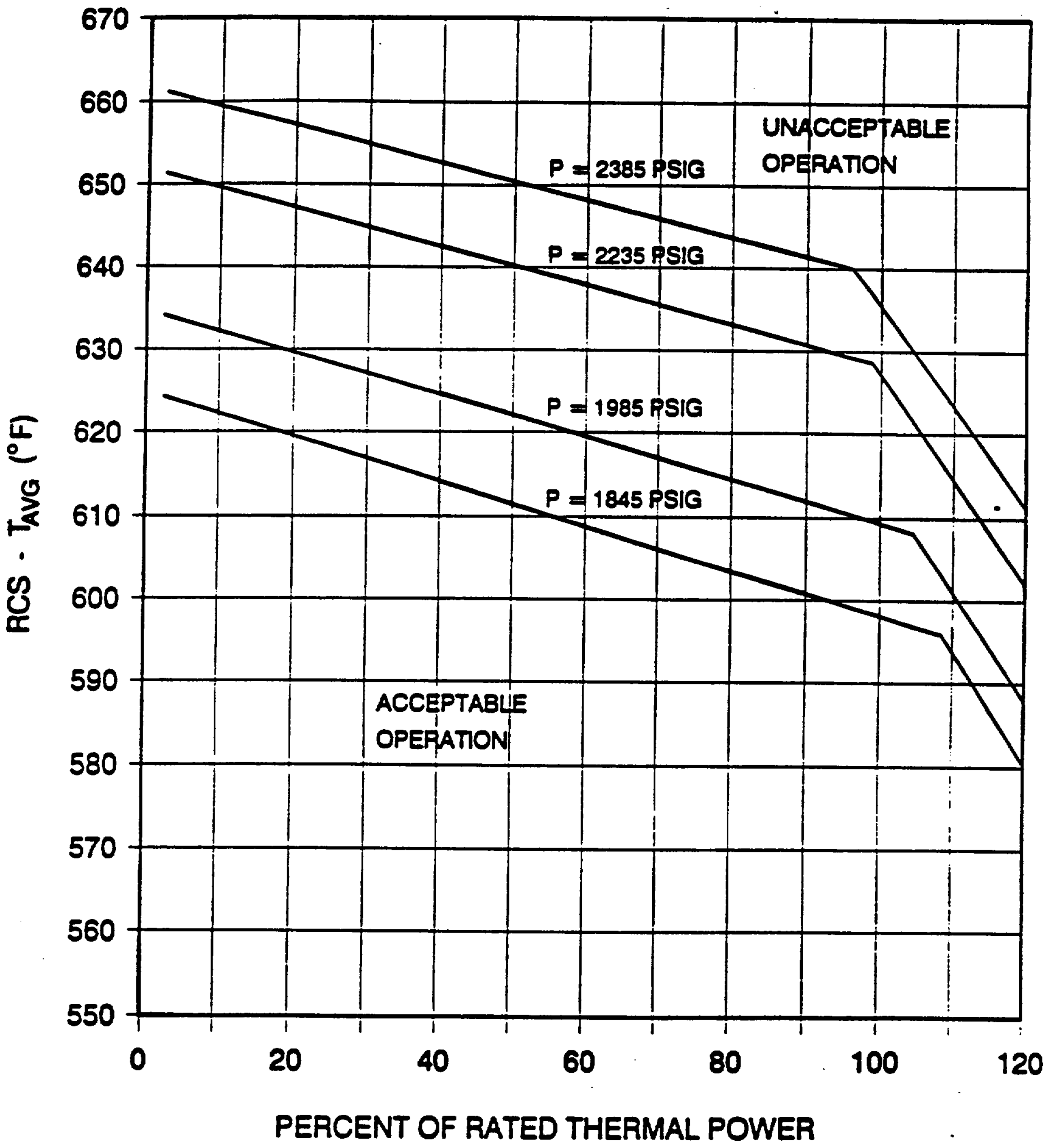

FIGURE 2.1-1a

UNIT 1 REACTOR CORE SAFETY LIMITS

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	1.25	≤109% of RTP*	≤111.7% of RTP*
b. Low Setpoint	8.3	4.56	1.25	≤25% of RTP*	≤27.7 of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3 of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.5 of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature N-16					
a. Unit 1	10.53	6.70	1.0+1.10+0.76(1)	See Note 1	See Note 2
b. Unit 2	10.0	6.75	1.0+1.38+0.96(2)	See Note 1	See Note 2

*RTP = RATED THERMAL POWER
(1) 1.0% span for N-16 power monitor, 1.10% for T_{cold} RTDs and 0.76% for pressurizer pressure sensors.
(2) 1.0% span for N-16 power monitor, 1.38% for T_{cold} RTDs and 0.96% for pressurizer pressure sensors.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Overpower N-16	4.0	2.05	1.0+0.05(3)	≤112% of RTP*	≤114.5% of RTP*
9. Pressurizer Pressure-Low					
a. Unit 1	4.4	0.71	2.0	≥1880 psig	≥1863.6 psig
b. Unit 2	4.4	1.12	2.0	≥1880 psig	≥1863.6 psig
10. Pressurizer Pressure-High					
a. Unit 1	7.5	5.01	1.0	≤2385 psig	≤2400.8 psig
b. Unit 2	7.5	1.12	2.0	≤2385 psig	≤2401.4 psig
11. Pressurizer Water Level-High					
a. Unit 1	8.0	2.18	2.0	≤92% of instrument span	≤93.9% of instrument span
b. Unit 2	8.0	2.35	2.0	≤92% of instrument span	≤93.9% of instrument span
12. Reactor Coolant Flow-Low					
a. Unit 1	2.5	1.18	0.6	≥90% of loop design flow**	≥88.6% of loop design flow**
b. Unit 2	2.5	1.25	0.87	≥90% of loop minimum measured flow***	≥88.8% of loop minimum measured flow***

(3) 1.0% span for N-16 power monitor and 0.05% for T_{cold} RTDs.
* RTP = RATED THERMAL POWER
** Loop design flow = 99,050 gpm
*** Loop minimum measured flow = 98,500 gpm

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: Overtemperature N-16

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

Where:

N	=	Measured N-16 Power by ion chambers,
T_c	=	Cold leg temperature, °F,
T_c^{o}	=	560.5°F for Unit 1, 560.3°F for Unit 2 - Reference T_c at RATED THERMAL POWER,
K_1	=	1.150,
K_2	=	0.0134/°F for Unit 1 0.016856/°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 0.000898/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.

TABLE 2.2-1 (Continued)
TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

For Unit 2

(i) for $q_t - q_b$ between -52% and +5.5%, $f_1(\Delta q) = 0$, 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 -52%, the N-16 Trip Setpoint shall be automatically reduced by 2.15% of its value at RATED THERMAL POWER, and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +5.5%, the N-16 Trip Setpoint shall be automatically reduced by 2.17% 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 2.88% of span for Unit 2.

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, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric thimble locations or

b. Using the Movable Incore Detection System to monitor the QUADRANT POWER TILT RATIO.

POWER DISTRIBUTION LIMITS

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
≥ 395,200 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 75% 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.

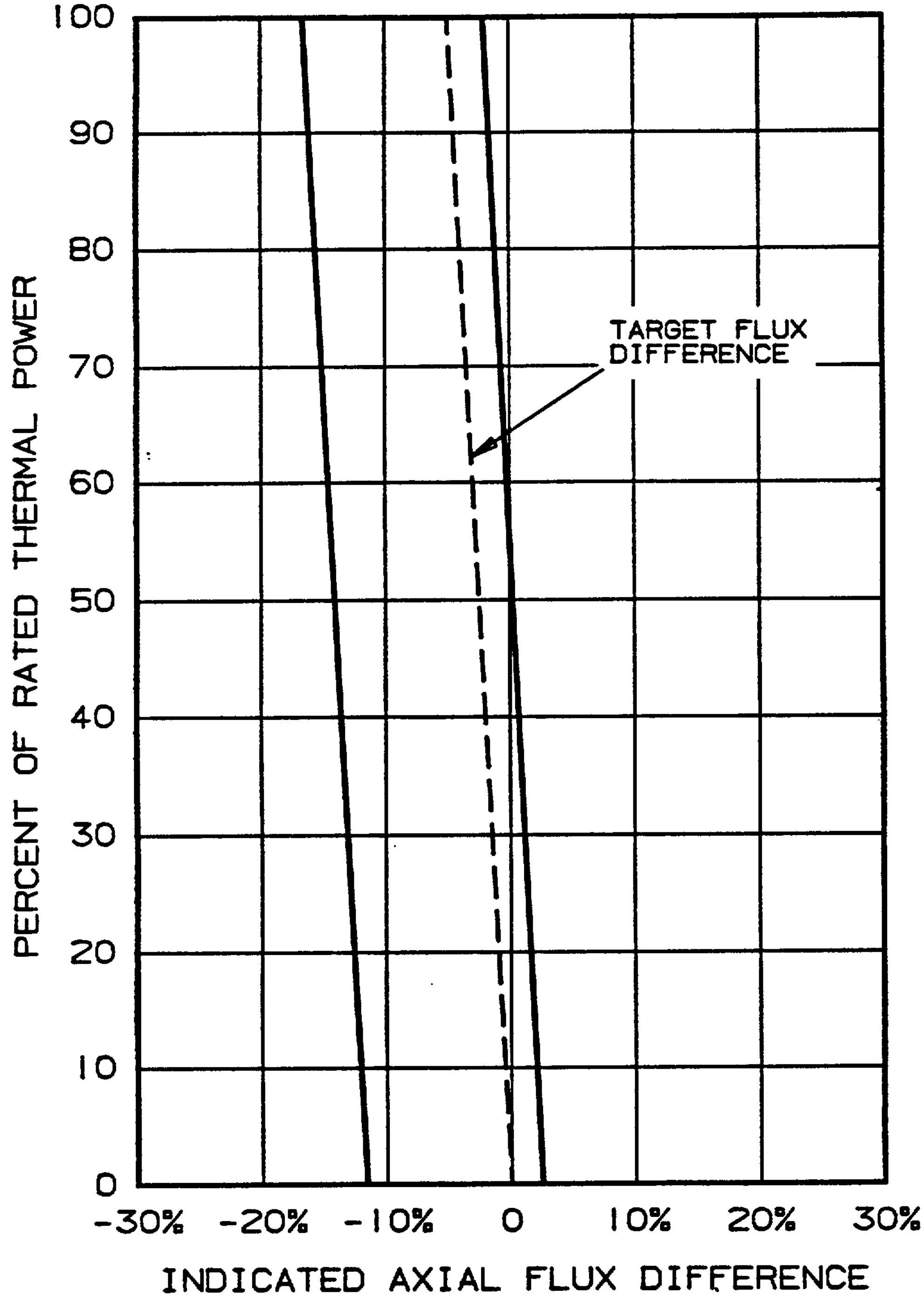

FIGURE B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the generic margin. The DNBR generic margin, totaling 18.1% for Unit 1 and 10.1% for typical cells and 9.5% for thimble cells for Unit 2 for DNBR completely offset any rod bow penalties. The margin for Unit 1 and Unit 2 is included by establishing a fixed difference between the safety analysis limit DNBR and the design limit DNBR equal to the percent margin of the safety analysis limit DNBR.

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The heat flux hot channel factor $F_Q(Z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to Constant Axial Offset Control (CAOC) operation, W(Z), to provide assurance that the limit on the heat flux hot channel factor, $F_Q(Z)$, is met. W(Z) accounts for the effects of normal operation transients within the AFD band and was determined from expected power control maneuvers over the range of burnup conditions in the core. The W(Z) function is provided in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.6.

When $F^N_{\Delta H}$ is measured, an adjustment for measurement uncertainty must be included for a full-core flux map taken with the Incore Detector Flux Mapping System.

$F_Q(Z)$ should be measured with the reactor core at, or near, equilibrium conditions. Therefore, the effects of transient maneuvers, such as power increases, should be permitted to decay to the extent possible while assuring that flux maps are taken in accordance with the specified surveillance schedules.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt. the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles.

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR at or above the safety analysis limit value throughout each analyzed transient. The Unit 1 indicated T_{avg} value of 592.7°F (conservatively rounded to 592°F) and the Unit 1 indicated pressurizer pressure value of 2219 psig correspond to analytical limits of 594.7°F and 2205 psig respectively, with allowance for measurement uncertainty. The Unit 2 indicated T_{avg} value of 592.8°F (conservatively rounded to 592°F) and the Unit 2 indicated pressurizer pressure value of 2219 psig correspond to analytical limits of 595.16°F and 2205 psig respectively, with allowance for measurement uncertainty. The indicated uncertainties assume that the reading from four channels will be averaged before comparing with the required limit.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation, and to detect any significant flow degradation of the Reactor Coolant System (RCS).

The additional surveillance requirements associated with the RCS total flow rate are sufficient to ensure that the measurement uncertainties are limited to 1.8% as assumed in the Improved Thermal Design Procedure Report for CPSES.

Performance of a precision secondary calorimetric is required to precisely determine the RCS temperature. The transit time flow meter, which uses the N-16 system signals, is then used to accurately measure the RCS flow. Subsequently, the RCS flow detectors (elbow tap differential pressure detectors) are normalized to this flow determination and used throughout the cycle.

CORE OPERATING LIMITS REPORT (Continued)

5). WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary). (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor (W(z) surveillance requirements for F_Q Methodology).)

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, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," February 1991. (Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor.)

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

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

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

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

14). RXE-91-002, "Reactivity Anomaly Events Methodology", May 1991. (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

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

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

Reference 17) is for Unit 1 only:

17). WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL, February 1978 Version," February 1978 (W Proprietary).

CORE OPERATING LIMITS REPORT (Continued)

Reference 18) is for Unit 2 only:

18). WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL- 1981 Version", February 1982 (W Proprietary).

6.9.1.6c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

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

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by the Technical Specifications, Technical Requirements Manual, and Fire Protection Report, except as explicitly covered in Specification 6.10.3;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

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

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 21 AND 7 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 May 28, 1993, Texas Utilities Electric Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. NPF-87 and NPF-89) for the Comanche Peak Steam Electric Station, Unit Nos. 1 and 2. The licensee supplemented the application by letter dated September 24, 1993. The amendments would incorporate changes to the Technical Specifications (TS) for Cycle 4 operations in Unit 1; specifically, revised core safety limit curves and revised N-16 overtemperature reactor trip setpoints. In addition, the amendments increase the minimum required reactor coolant system flow, remove a penalty on pressurizer pressure uncertainty, and include an operational enhancement for the treatment of the uncertainty allowance for the N-16 power indication. The September 24, 1993, supplemental letter provided clarifying information and did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

TU Electric has changed the fuel supplier of CPSES Unit 1 from the Westinghouse Electric Company (WEC) to Siemans Power Corporation (SPC). SPC fuel will be supplied for Unit 1 for Cycle 4 and for Unit 2 for Cycle 3.

TU Electric has developed in-house analysis methodologies for the CPSES Units 1 and 2, which are scheduled to be 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 can be applied to both CPSES Units 1 and 2, subject to the constraints of the applicable Safety Evaluations (SEs). For CPSES Unit 1 Cycle 4, these methodologies will be used to determine the core safety limits and perform the departure from nucleate boiling (DNB) related portion of the safety analyses. The reload analysis methodologies have been approved by the NRC as listed below and can be used to support CPSES Unit 1, Cycle 4 operation.

RXE-88-102-P (Ref. 1), SE dated June 11, 1992
RXE-88-102-P, Supplement 1 (Ref. 2), SE dated June 11, 1992
RXE-91-002 (Ref. 3), SE dated January 19, 1993
RXE-90-007 (Ref. 4), SE dated April 26, 1993
TXX-88306 (Ref. 5), SSER 23, Section 15.4.4 issued February 1990
RXE-90-006-P (Ref. 6), SE dated August 5, 1993
RXE-89-002 (Ref. 7), SE dated August 5, 1993
RXE-91-001 (Ref. 8), SE dated July 16, 1993

Using these methodologies and the changes in (1) and (2) below, calculations and analyses have been performed to identify the new core safety limit curves for Unit 1. The departure form nucleate boiling ratio (DNBR) generic margin will increase from 9.1 percent to 18.1 percent for Unit 1.

In addition to the determination of the core safety limits and the DNB related parameters for the Unit 1, Cycle 4 core configuration (including revised Overtemperature N-16 setpoint equation coefficients), TU Electric intends to:

(1) Increase the reactor coolant system (RCS) thermal design flow rate.

To enhance the DNB-related analysis 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 7.9 percent higher than the thermal design flow (TDF) assumed in the CPSES Unit 1, Cycle 3 accident analyses. For Unit 1, Cycle 4, TU Electric proposes crediting 3.5 percent of the flow in the accident analyses, resulting in the definition of a higher RCS TDF rate. Correspondingly, the TS minimum measured RCS flow requirement will also be increased from 389,700 gpm to 403,400 gpm. Unit 2 is not affected by this change.

(2) Remove the bias on the system pressure uncertainty on the Barton 763 pressure transmitters.

Previously, the CPSES Unit 1 safety analysis assessed a penalty on the pressurizer pressure uncertainty associated with the Barton 763 pressure transmitters. This was due to the non-repeatability of the transmitters at high temperatures. However, the transmitters have now been refurbished by the vendor. Therefore, the penalty is no longer necessary and will be removed from the setpoint determination. The minimum indicated pressurizer pressure value will be increased from 2207 psig to 2219 psig. Also, the analytical limit, with allowance for measurement uncertainty will be increased from 2193 psig to 2205 psig. Unit 2 is not affected by this change.

(3) Provide an allowance for the normalization of the N-16 power to the daily plant calorimetric measurement in the statistical setpoint study.

Because of the new Unit 1, Cycle 4 core safety limits, the Overtemperature N-16 reactor trip setpoints must be recalculated to ensure that the new core safety limits are met. With this recalculation it is proposed to add an operational enhancement. The TS require the readjustment of indicated N-16 power indicator if the N-16 power indication differs by more than plus or minus 2 percent of rated thermal power determined by the daily power calorimetric measurement. Currently, the sensor measurement and test equipment (SMTE) allowance for the N-16 power indication is subtracted directly form the allowable power difference. This reduces the allowed tolerance between the indicated N-16 power and the calorimetric power and results in an unnecessarily high N-16 readjustment frequency. To reduce this readjustment frequency, the SMTE allowance associated with the indicated N-16 power will be included in the channel statistical allowance of the statistical setpoint studies for N-16 power.

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 departure from nucleate boiling 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 1 during Cycle 4.

The licensee stated that the methods used for the calculation of the mixed core DNB penalty are the same as used for the DNB analyses described in the NRC approved TU Electric report RXE-89-002. The effect of the mixed core on the large break LOCA analysis was evaluated in accordance with the NRC approved TU Electric report RXE-90-007. The mechanical and thermal-hydraulic compatibility between the existing Westinghouse fuel assemblies and the co-resident SPC fuel assemblies was evaluated in the reload safety evaluation 10 CFR 50.59 evaluation. The licensee stated that it was confirmed that both the SPC and Westinghouse performed evaluations demonstrate that their respective fuel assembly designs meet all applicable design criteria including those pertaining to the interaction between the two fuel types.

Because a different DNB correlation, TUE-1, is to be used for the CPSES Unit 1, Cycle 4 core configuration, new core safety limits have been calculated.

The new core safety limits have been determined to ensure 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 these analyses, the RCS thermal design flow rate was increased and the bias on the system pressure uncertainty due to the thermal non-repeatability of the pressurizer pressure transmitters was removed. Also, an operational enhancement was added to statistically include the sensor measurement and test equipment (SMTE) allowance associated with the N-16 power indication into the statistical setpoint determination of the reactor trip system instrumentation trip setpoints which will reduce the required frequency of N-16 power adjustment.

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 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 1, Cycle 4, 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. We find the use of these methodologies acceptable as they were previously reviewed and approved by the NRC.

3.2 Increase in the Unit 1 Thermal Design Flow

TU Electric has proposed to increase the thermal design flow (TDF) rate by 3.5 percent (from 95,700 gpm per loop to 99,050 gpm per loop or from 382,800 gpm to 396,200 gpm for four loops). TU Electric stated that the current difference between the actual measured RCS flow rate and the TDF rate assumed in the CPSES Unit 1, Cycle 3 safety analyses is approximately 7.9 percent. This leaves a remaining difference of approximately 4.4 percent (7.9 percent-3.5 percent) for the RCS flow. This 4.4 percent difference is sufficient to account for all uncertainties associated with measuring 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. The 1.8 percent RCS flow measurement uncertainty is indicated in the footnote ** of TS 3.2.5 and remains valid. The proposed change in TDF also necessitates a change to the minimum indicated total RCS flow rate from 389,700 gpm to 403,4000 gpm in TS 3.2.5c because of the relationship between the TDF flow rate assumed in the safety analyses and the minimum required indicated flow. The licensee stated that the measured RCS flow rates for CPSES Units 1 and 2 for the last cycles are 413,127 gpm and 418,993 gpm respectively. We find the changes to the TDF rate and the minimum indicated total RCS flow rate to be acceptable as there are acceptable margins available and the uncertainties are accounted for.

3.3 Increase in the Unit 1 Minimum Pressurizer Pressure

CPSES Unit 1 was assessed a penalty on the pressurizer pressure uncertainty associated with the Barton 763 transmitters which provide indication of pressurizer pressure. The penalty (-12 psi, treated as a bias on pressurizer pressure uncertainty) was due to non-repeatability of the transmitters at high temperatures. The penalty was assessed in the safety analyses value for pressurizer pressure which was decreased by the amount of the penalty. TU

Electric had all of the Barton 763 pressurizer pressure transmitters refurbished by the vendor prior to initial fuel load. The removal of the penalty allows TU Electric to raise the analytical limit for pressurizer pressure for the safety analyses from 2193 psig to 2205 psig. Consequently, the minimum required indicated value, which never included the non-repeatability penalty, is also higher, increasing from 2207 psig to 2219 psig. Removal of the penalty results in the same safety analyses analytical limit for Unit 1 as for Unit 2. These changes are related to TS 3.2.5b and BASES 3/4.2.5. The limits on pressurizer pressure are consistent with the Final Safety Analysis Report (FSAR) initial condition assumptions and have been analytically demonstrated adequate for Unit 1, Cycle 4 to maintain a minimum DNBR at or above the safety analysis limit value throughout each analyzed transient. The staff finds the increase in the minimum pressurizer pressure to be acceptable.

3.4 Revision to the Unit 1 Core Safety Limits

Beginning with Cycle 4, CPSES Unit 1, Siemens Power Corporation will supply the nuclear fuel assemblies for Unit 1. 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 4. 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 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 1 during Cycle 4.

The core safety limits for CPSES Unit 1, Cycle 4 (TS 2.1, Figure 2.2-1a) have been determined using the NRC approved TU Electric methodologies for determining core safety limits, an increase in the assumed RCS TDF rate, an increase in the minimum assumed pressurizer pressure, and a safety analysis DNBR based on the NRC approved TUE-1 DNB correlation.

The TS BASES (3/4.2.2 and 3/4.2.3) description of DNBR generic margin was revised due to the change from the W-3 R-grid critical heat flux (CHF) correlation to the TUE-1 DNB correlation for the Unit 1, Cycle 4 DNB analyses.

The generic margin was established for these two correlations by different methods. The current method of allocating the DNBR generic margin for Unit 1 quantifies the change in the DNBR predicted by W-3 R-grid CHF correlation due to various modeling conservatism. The total change in the DNBR due to the selected modeling conservatism is then presented as a percent of the calculated DNBR. This approach was used by Westinghouse in arriving at the 9.1 percent DNBR generic margin for Unit 1.

The method of allocating the DNBR generic margin used by TU Electric for Unit 1 was changed. It is similar to the method used by Westinghouse in allocating the DNBR generic margin for Unit 2 for which the WRB-1 CHF correlation is used. This method sets a DNBR limit to be utilized in the safety analyses (i.e., the DNBR safety analysis limit) above the 95/95 DNBR correlation limit (i.e., the DNBR design limit) by an amount which will be used to offset known and potential DNBR penalties. The TU Electric method of allocating DNBR generic margin for CPSES Unit 1, Cycle 4, results in a generic margin of 18.1 percent above the TUE-1 95/95 DNBR correlation limit.

We have found the revisions to the Unit 1 core safety limits discussed above to be acceptable as they have been analyzed using NRC-approved methodology.

3.5 Revision to Unit 1 Overtemperature and Overpower 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 and Overpower N-16 trip setpoints are reactor trips which help protect the core and RCS from exceeding their safety limits.

The licensee stated that the method (WCAP-12123) used by TU Electric for performing the statistical setpoint calculations for CPSES Units 1 and 2 was licensed from Westinghouse. This method has been previously used for the calculation of the RTS and ESFAS setpoints for CPSES Units 1 and 2.

The Overtemperature N-16 setpoint is automatically varied with coolant temperature, pressurizer pressure, and axial power distribution. With a normal operation axial power distribution, the Overtemperature N-16 reactor trip limit is always below the core safety limit. If the axial flux difference is greater than design, the Overtemperature N-16 reactor trip setpoint is automatically reduced according to the notations (Note 1) in TS 2.2, Table 2.2-1, to provide protection consistent with the core safety limits.

Since the core safety limits have been changed for CPSES Unit 1, Cycle 4, the Overtemperature N-16 reactor trip setpoint was recalculated in accordance with the methods developed by TU Electric. 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 (Δq) coefficients for the equation shown in TS 2.2, Table 2.2-1, Note 1. 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 distribution.

The value of $T_c°$ (reference cold leg temperature at rated thermal power) for the Overtemperature N-16 trip setpoint equation in TS 2.2, Table 2.2-1, Note 1 was also changed. This change is the result of the calculation of a new value of $T_c°$ from an energy balance at rated thermal power using the higher TDF rate.

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 total allowance (TA), sensor errors (S), trip setpoint and allowable value, in TS Table 2.2-1. The methodology to derive the Overtemperature N-16 reactor trip setpoints 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.

An operational enhancement was added to the CPSES Unit 1, Cycle 4, Overtemperature N-16 reactor trip system instrumentation trip setpoint. TS 4.3.1.1 (Note 2 to Table 4.3-1) requires the indicated N-16 power be readjusted if the indicated N-16 power differs by more than plus or minus 2 percent of rated thermal power (RTP) as calculated from the daily power calorimetric measurement. This involves the subtraction of the sensor measurement and test equipment allowance for the indicated N-16 power (plus or minus 1.5 percent of RTP) from the plus or minus 2 percent of RTP difference. This reduces the allowed tolerance between the N-16 power indication and the calorimetric power to plus or minus 0.5 percent of RTP and results in an unnecessarily high N-16 readjustment frequency. The readjustment requires entry into the Westinghouse 7300 process cabinets, which increases the potential for personnel errors. To reduce this readjustment frequency, the SMTE allowance associated with the indicated N-16 power is proposed to be included into the channel statistical allowance calculation of the Overtemperature N-16 reactor trip setpoint (which uses the N-16 power signal) instead of being subtracted from the allowable power difference. This increases the channel total uncertainty and is accounted for in Table 2.2-1 by a change in the "S" term only. This change to include the indicated N-16 power SMTE in the statistical treatment of the nominal Overtemperature N-16 reactor trip setpoint is acceptable because the Overtemperature N-16 measurements continue to be made with an acceptable level of accuracy which will assure that the accident analyses are valid. This change will also make the Unit 1 requirements consistent with Unit 2.

The possibility of spurious turbine runbacks or reactor trips due to a slight observed upper plenum flow anomaly has been considered and determined not to be a concern given the magnitude of the actuation setpoints.

Since the N-16 signal is also part of the Overpower N-16 reactor trip setpoint, the Overpower N-16 reactor trip setpoint values for allowance (TA), a sensor error (S), and allowable value (AV) were recalculated to include the SMTE allowance discussed above.

No change to the safety analysis value of the Overpower N-16 reactor setpoint occurred and instrument uncertainties are properly accounted for in determining the trip instrumentation values of TA, Z, S, and AV.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

The technical specifications were changed as a result of the use of the new TU Electric in-house reload analysis methodologies for CPSES Unit 1, Cycle 4, revision of the RCS flow rate, removal of the penalty on pressurizer pressure uncertainty, and enhancement of the treatment of the uncertainty allowance for N-16 power indication. The following technical specifications were evaluated for changes.

(1) Figure 2.1-1a, page 2-2, Unit 1 reactor core safety limits

TS Figure 2.1-1a was revised because of the use of the new TU Electric methodologies for reload analyses, the increase in the TDF rate, and the increase of the minimum pressurizer pressure.

We find this figure to be acceptable as discussed in the evaluation in Section 3.0.

(2) Table 2.2-1, reactor trip system instrumentation setpoints

Page 2-5, Functional Unit 7., Overtemperature N-16, a. Unit 1

The total allowance (TA) was changed to 10.53.
The Z value was changed to 6.75.
The sensor error (S) was changed to 1.0 + 1.10 + 0.76[1].
The note (1) was changed to 1.0% span for N-16 power monitor, 1.10% for T_c RTDs and 0.76% for pressurizer pressure sensors.

Page 2-6, Functional Unit 8., Overpower N-16

Since both Unit 1 and Unit 2 will have the same values, the listing titles for the Unit 1 and Unit 2 were eliminated. The values for Unit 2 for TA, Z, S, Trip Setpoint, and Allowable Value were kept to represent both Units 1 and 2.
The footnote ** was changed to: **Loop design flow = 99,050 gpm

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

Page 2-9, TABLE NOTATIONS

The value for $T_c°$ was changed for Unit 1 to 560.5°F.
The value for K_1 for Unit 1 and Unit 2 was made the same as for Unit 2. Since both units now have the same value, the Unit 1 and Unit 2 designation was removed.
The value of K_2 was changed for Unit 1 to 0.0134.

The value of K_3 was changed for Unit 1 to 0.000719 psig.

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

Page 2-10, TABLE NOTATIONS

For Unit 1, the values associated with $q_t - q_b$ and the N-16 trip setpoint were changed in Note 1 as follows:

(i) for $q_t - q_b$ between -65% and +4%, ...,

(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.

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

(3) 3.2.5, Page 3/4 2-12, DNB PARAMETERS

The indicated pressurizer pressure values for Unit 1 and Unit 2 were changed to the same value of equal to or greater than 2219 psig.

The indicated reactor coolant system flow was changed for Unit 1 to equal or greater than 403,400 gpm, which includes a 1.8% flow measurement uncertainty.

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

(4) BASES, Page B 3/4 2-4, HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The total DNBR generic margin was changed to 18.1% for Unit 1 and the listing of where these margins came from was deleted. An editorial change was made to indicate that for Unit 1 as well as Unit 2 the margin is included by establishing a fixed difference between the safety analysis limit DNBR and the design limit DNBR equal to the percent margin of the safety analysis limit DNBR.

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

(5) BASES 3/4.2.5, Page 3/4 2-6, DNB PARAMETERS

The pressures in the following statement were changed for Unit 1 to state ..."the Unit 1 indicated pressurizer pressure value of 2219 psig correspond to analytical limits of 594°F and 2205 psig respectively, with allowance for measurement uncertainty."

These changes were found to be acceptable as discussed in the evaluation in Section 3.0.

(6) CORE OPERATING LIMITS REPORT, Page 6-21

An insert was made to list eight new references for topical reports pertaining to the TU Electric in-house analyses. Editorial changes were made to change the numbering of the listings and to eliminate some descriptions assigned to the previously existing reports.

The addition of the new references are acceptable as discussed in Section 3.1 of the evaluation. These editorial and other changes are acceptable as they are made to provide editorial additions and modifications.

5.0 REFERENCES

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

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

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

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

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

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

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

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

6.0 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.

7.0 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 (58 FR 43934). The amendment also change recordkeeping or reporting requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 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: H. Balukjian

Date: November 16, 1993