January 12, 1993

Docket No. 50-445

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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, UNIT 1 - AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-87 (TAC M80290)

The Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-87 for the Comanche Peak Steam Electric Station, Unit 1. The amendment consists of changes to the Technical Specifications in response to your application dated April 26, 1991 (TXX-91084), and as supplemented by letters dated May 4, 1992 (TXX-92227) and October 15, 1992 (TXX-92483).

The amendment revises Technical Specification Tables 3.3-1, 3.3-2, and 4.3-2 and associated Bases to increase the surveillance test intervals and allowed outage times for the analog channels of the reactor trip system and the engineered safety features actuation systems. The technical specification changes are based on WCAP-10271, Supplement 2, and WCAP-10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System." The NRC staff has previously reviewed and approved these documents in safety evaluations dated February 22, 1989, and April 30, 1990. The NRC has encouraged licensees to incorporate the changes approved in these safety evaluations.

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. 13 to NPF-87
2. Safety Evaluation

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cc w/enclosures:
See next page

OFFICE	PDIV-2/LA	PDIV-2	PDIV-2/PM	OGC NLO	PDIV-2/D	SICB/BC
NAME	EPeyton	DSkay;nb	TBergman		SBlack	~~SNewberry~~
DATE	12/22/92	12/22/92	12/28/92	/92	1/11/93	12/5/92

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

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.*

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
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) acting for itself and as agent for Texas Municipal Power Agency (licensees) dated April 26, 1991, and as supplemented by letters of May 4, 1992 and October 15, 1992, 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.

*The current owners of the Comanche Peak Steam Electric Station are: Texas Utilities Electric Company and Texas Municipal Power Agency. Transfer of ownership from Texas Municipal Power Agency to Texas Utilities Electric Company was previously authorized by Amendment No. 9 to Construction Permit CPPR-126 on August 25, 1988 to take place in 10 installments as set forth in the Agreement attached to the application for Amendment dated March 4, 1988. At the completion thereof, Texas Municipal Power Agency will no longer retain any ownership interest.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 13, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

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: January 12, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 13

FACILITY OPERATING LICENSE NO. NPF-87

DOCKET NO. 50-445

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
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-35	3/4 3-35
B 3/4 3-1	B 3/4 3-1
B 3/4 3-2	B 3/4 3-2

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3[a], 4[a], 5[a]	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1[c], 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2
5. Intermediate Range, Neutron Flux	2	1	2	1[c], 2	3
6. Source Range, Neutron Flux					
a. Reactor Trip and Indication					
1) Startup	2	1	2	2[b]	4
2) Shutdown	2	1	2	3, 4, 5	5.1
b. Boron Dilution Flux Doubling*	2	1	2	3[h],4, 5	5.1, 5.2
7. Overtemperature N-16	4	2	3	1, 2	12
8. Overpower N-16	4	2	3	1, 2	12
9. Pressurizer Pressure--Low	4	2	3	1[d]	6
10. Pressurizer Pressure--High	4	2	3	1, 2	6

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1[d]	6
12. Reactor Coolant Flow--Low					
a. Single Loop	3/loop	2/loop in any loop	2/loop	1[f]	6
b. Two Loops	3/loop	2/loop in any two loops	2/loop	1[g]	6
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen.	1, 2	6
14. Undervoltage--Reactor Coolant Pumps	4-1/bus	2	3	1[d]	6
15. Underfrequency--Reactor Coolant Pumps	4-1/bus	2	3	1[d]	6
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1[i]	6
b. Turbine Stop Valve Closure	4	4	4	1[i]	10
17. Safety Injection Input from ESFAS	2	1	2	1, 2	13

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2[b]	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1,2	7
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-9	4	2	3	1[i]	7
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
19. Reactor Trip Breakers	2	1	2	1, 2	8, 11
	2	1	2	3[a], 4[a], 5[a]	9
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	13
	2	1	2	3[a], 4[a], 5[a]	9

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

[a]Only if the reactor trip breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

[b]Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

[c]Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

[d]Above the P-7 (At Power) Setpoint.

[e]Deleted.

[f]Above the P-8 (3-loop flow permissive) Setpoint.

[g]Above the P-7 and below the P-8 Setpoints.

[h]The boron dilution flux doubling signals may be blocked during reactor startup.*

[i]Above the P-9 (Reactor trip on Turbine trip Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1., and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER LEVEL:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint,

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5.1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or within the next hour open the reactor trip breakers and suspend all operations involving positive reactivity changes. With no channels OPERABLE complete the above actions within 4 hours.

ACTION 5.2*- With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or within the next hour verify either valve 1CS-8455 or valves 1CS-8560, FCV-111B, 1CS-8439, 1CS-8441, and 1CS-8453 are closed and secured in position, and verify this position at least once per 14 days thereafter. With no channels OPERABLE, complete the above actions within 4 hours and verify the positions of the above valves at least once per 14 days thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1 or maintenance, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status, during which time ACTION 8 applies.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing per Specifications 4.3.1.1 or 4.2.5.4.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3[a], 4[a], 5[a]
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1[c], 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1[c], 2
6. Source Range, Neutron Flux	S	R(4, 13)	S/U(1), Q(9)	R(12)*	N.A.	2[b], 3, 4, 5
7. Overtemperature N-16	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	Q	N.A.	N.A.	1, 2
8. Overpower N-16	S	D(2, 4) R(4, 5)	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(8)	N.A.	N.A.	1[d]
10. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2

*Boron Dilution Flux Doubling requirements become effective for Unit 1 six months after criticality for Cycle 3.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (ECCS, Reactor Trip, Feedwater Isolation, Control Room Emergency Recirculation, Emergency Diesel Generator Operation, Containment Vent Isolation, Station Service Water, Phase A Isolation, Auxiliary Feedwater-Motor Driven Pump, Turbine Trip, Component Cooling Water, Essential Ventilation Systems, and Containment Spray Pump.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	26
c. Containment Pressure--High-1	3	2	2	1, 2, 3	17
d. Pressurizer Pressure--Low	4	2	3	1, 2, 3[a]	17
e. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3[a]	17

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair operated simultaneously	2 pair	1, 2, 3, 4	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	26
c. Containment Pressure--High-3	4	2	3	1, 2, 3	14
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	26
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	See Item 2.a above. Phase "B" isolation is manually initiated when containment spray function is manually initiated.			1, 2, 3, 4	16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	26
3) Containment Pressure--High-3	4	2	3	1, 2, 3	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
c. Containment Vent Isolation					
1) Manual Initiation	See Item 2.a and 3.a.1 above. Containment vent isolation is manually initiated when Phase "A" isolation function or containment spray function is manually initiated.			1, 2, 3, 4	15
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	15
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual Steam Line	1/steam line	1/steam line	1/operating steam line	1, 2^c, 3^c	21
2) System	2	1	2	1, 2^c, 3^c	20
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2^c, 3^c	19
c. Containment Pressure--High-2	3	2	2	1, 2^c, 3^c	17

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2[c], 3[a,c]	17
e. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line in any steam line	2/steam line	3[b,c]	17
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Steam Generator Water Level--High-High	3/stm. gen.[e]	2/stm. gen. in any stm. gen.	2/stm. gen.	1, 2	17
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	19
b. Stm. Gen. Water Level--Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	17

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3[d]	17
c. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	1/train	1/train	1/train	1, 2, 3	16
e. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	2/AFW pump	2/AFW pump	1/AFW pump	1, 2	23
7. Automatic Initiation of ECCS Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	26

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Automatic Initiation of ECCS Switchover to Containment Sump (Continued)					
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	17
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Loss of Power (6.9 kV & 480 V Safeguards System Undervoltage)					
a. 6.9 kV Preferred Offsite Source Undervoltage	2/bus	2/bus	1/bus	$1^f, 2^f, 3^f, 4^f$	23
b. 6.9 kV Alternate Offsite Source Undervoltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
c. 6.9 kV Bus Undervoltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
d. 6.9 kV Degraded Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
e. 480 V Degraded Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
f. 480 V Low Grid Undervoltage	2/bus	2/bus	1/bus	1, 2, 3, 4	23
9. Control Room Emergency Recirculation					
a. Manual Initiation	2	1	2	All	24

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	18
b. Reactor Trip, P-4	2	2	2	1, 2, 3	20
11. Solid State Safeguards Sequencer (SSSS)					
a. Safety Injection Sequence	1/train	1/train	1/train	1, 2, 3, 4	26
b. Blackout Sequence	1/train	1/train	1/train	1, 2, 3, 4	25

TABLE 3.3-2 (Continued)

TABLE NOTATIONS

[a]Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

[b]Trip function automatically blocked above P-11 and may be unblocked below P-11 by blocking the Safety Injection on low steam line pressure.

[c]Not applicable if each affected main steam isolation valve and its associated upstream drain pot isolation valve per steam line is closed.

[d]The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

[e]The channel which provides a steam generator water level control signal (if one of three specific trip channels is selected to provide input into steam generator water level control) must be placed in the tripped condition within 6 hours and maintained in the tripped condition with the exception that the channel may be taken out of the tripped condition for up to 4 hours to allow testing of redundant channels.

[f]Not applicable if Preferred Offsite Source Breaker is open.

ACTION STATEMENTS

ACTION 12 - Deleted.

ACTION 13 - Deleted.

ACTION 14 - With the number of OPERABLE channel one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 15 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment pressure relief valves are closed within 4 hours and maintained closed.

ACTION 16 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met.

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or initiate and maintain operation of the Control Room Emergency Recirculation System.

ACTION 25 - With the number of OPERABLE channels on one or more trains less than the Minimum Channels OPERABLE requirement, declare the diesel generator(s) associated with the affected train(s) inoperable and apply the appropriate ACTION for Specification 3.8.1.1.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Automatic Initiation of ECCS Switchover to Containment Sump (Continued)								
b. RWST Level-Low-Low Coincident With	S	SR	Q	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power (6.9 kV & 480 V Safeguards System Undervoltage)								
a. 6.9 kV Preferred Offsite Source Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 6.9 kV Alternate Offsite Source Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
c. 6.9 kV Bus Under-voltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
d. 6.9 kV Degraded Voltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
e. 480 V Degraded Voltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4
f. 480 V Low Grid Undervoltage	N.A.	R	N.A.	(3, 2)	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Control Room Emergency Recirculation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
11. Solid State Safeguards Sequencer (SSSS)								
a. Safety Injection Sequence	N.A.	R	N.A.	M(1,3,4)	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Blackout Sequence	N.A.	R	N.A.	M(1,3,4)	N.A.	N.A.	N.A.	1, 2, 3, 4

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the reactor protection and engineered safety features instrumentation, and (3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", WCAP-10271 Supplement 2, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System" and supplements to these reports as approved by the NRC and documented in the SER (letters to the Westinghouse Owners Group (WOG) dated February 21, 1985, February 22, 1989, and April 30, 1990).

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-3 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-3. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-3, in percent span, is the statistical summation of errors assumed in the analysis excluding those

REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-3, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time specified in the Technical Requirements Manual at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) ECCS pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) station service water pumps start and automatic valves position, (11) Control Room Emergency Recirculation starts, and (12) essential ventilation systems (safety chilled water, electrical area fans, primary plant ventilation ESF exhaust fans, battery room exhaust fans, and UPS ventilation) start.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-87

TEXAS UTILITIES ELECTRIC COMPANY, ET AL.

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

1.0 INTRODUCTION

By letter dated April 26, 1991, and supplemented by letters dated May 4, 1992, and October 15, 1992, Texas Utilities Electric Company (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-87) for the Comanche Peak Steam Electric Station, Unit 1. The proposed changes would be in accordance with the staff's safety evaluation (SE) and supplemental safety evaluation (SSE) described below to modify the engineered safety features actuation systems (ESFAS) and reactor trip system (RTS) instrumentation surveillance and testing.

2.0 BACKGROUND

In response to growing concerns over the impact of current testing and maintenance requirements on plant operation, particularly as related to instrumentation systems, the Westinghouse Owners Group (WOG) initiated a program to develop a justification to be used to revise generic and plant-specific instrumentation technical specifications. Operating plants have experienced inadvertent reactor trips and safeguards actuations during performance of instrumentation surveillances, causing unnecessary transients and challenges of safety systems. Significant time and effort on the part of operating staffs have been devoted to performing, reviewing, documenting, and tracking the various surveillance activities.

In response to this concern, the WOG submitted WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," to the NRC on February 3, 1983. This resulted in the NRC publishing a SE in February 1985, that allowed for an increase in the time allowed for an inoperable channel to be placed in the tripped condition from 1 to 6 hours, and increased time for surveillance and maintenance testing in bypass for analog channels of the RTS. The SE also required quarterly testing to be conducted on a staggered basis.

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On March 20, 1986, the WOG submitted WCAP-10271, Supplement 2, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Systems Actuation Systems." In addition, on May 12, 1987, the WOG submitted WCAP-10271, Supplement 2, Revision 1. After reviewing these WCAPs, the NRC issued an SE on February 22, 1989 (letter from C.E. Rossi to R.A. Newton) and an SSE on Revision 1 on April 30, 1990 (letter from C.E. Rossi to G.T. Goering). These SEs approved quarterly surveillance test intervals (STIs), an increase in the time allowed for an inoperable channel to be placed in the tripped condition from 1 to 6 hours, increased time for surveillance and maintenance testing, and testing in bypass for analog channels of the ESFAS. Staggered testing was not required for ESFAS analog channels and the requirement was removed from the RTS analog channels. The SE also concluded that 4-hour test and 12-hour maintenance allowed outage times (AOTs) are acceptable for ESFAS automatic actuation logic and actuation relays.

The reduction in testing associated with these changes is expected to result in fewer inadvertent reactor trips and less frequent spurious actuations of ESFAS components. As stated in the staff's SE and SSE described above, the increase of AOTs and STIs for the ESFAS analog channels and AOTs for the actuation logic and relays will result in a slight increase in the probability of core damage accidents. The staff concluded that an overall upper bound for the core damage frequency (CDF) increase due to the proposed STI/AOT changes is less than 6 percent for Westinghouse PWR plants. The staff also concluded that actual CDF increases for individual plants are expected to be substantially less than 6 percent.

The staff considered this CDF increase to be small compared to the range of uncertainty in the CDF analyses and therefore acceptable. Based on the WOG analyses and subsequent staff review, the staff concluded that the proposed STI and AOT changes for the ESFAS and RTS would have only a small, and therefore acceptable, impact on plant risk.

In WCAP-10271 Supplement 3, the Reactor Water Storage Tank (RWST) level channels were evaluated and found to be identical in configuration to the Steam Generator level channel and therefore the RWST level channels would be bounded by the unavailability analysis for the steam generator level channel. This amplified the analysis presented in the other supplements to encompass RWST level.

In both the February 22, 1989, SE and April 30, 1990, SSE, the NRC encouraged licensees to propose changes to their technical specifications that are consistent with the guidance provided by the staff. An acceptable format for proposing technical specification changes based on the WCAPs was provided to the WOG in the February 22, 1989 SE. In order for the staff to find the licensee's submittals acceptable, the licensees were told that they must:

1. Confirm the applicability of the generic analyses of WCAP-10271, Supplement 2 and WCAP-10271, Supplement 2, Revision 1, and

2. Confirm that any increase in instrument drift due to the extended STIs is properly accounted for in the setpoint calculation methodology. (Licensees were advised to use the letter from C.E. Rossi to R.F. Janecek, dated April 27, 1988, for guidance on this issue.)

CPSES incorporated most of the suggested STI and AOT relaxations in the original Technical Specifications issued in February, 1990; however, in the months prior to license issuance and during Technical Specification development, all review and acceptance of the WCAPs and supplements was not completed. As a result, most, but not all provisions of WCAP-10271 and supplements were incorporated. This change proposes to incorporate the remainder of the changes.

3.0 EVALUATION

As described above, in order to be considered for the generic technical specification changes, licensees must first make two specific findings, the first being that the licensee must confirm the applicability of the generic analyses of WCAP-10271, Supplement 2 and WCAP-10271 Supplement 2, Revision 1 to its facility. In the CPSES letter of April 26, 1991, the licensee stated that the methodology of WCAP-10271 and its supplements were applied to specific RTS and ESFAS function implemented via the Westinghouse Solid State Protection System (SSPS). A review was performed to assure that the functions used in the generic analysis and the employment of the SSPS to perform ESFAS function, as described in the generic analysis, are applicable to the CPSES design. Based on their review, the licensee concluded that all changes proposed in this amendment are addressed by the generic analysis except for RWST level. RWST level was separately evaluated on a plant specific basis in WCAP-10271 Supplement 3. This analysis concluded that RWST level is identical in configuration to the steam generator level channel; therefore, the system unavailabilities resulting from relaxed STIs and AOTS were essentially the same and will have no impact on plant safety.

The second finding that needed to be addressed was the concern over instrument drift over the extended STIs. In their submittals, the licensee stated that it had implemented quarterly analog testing upon receipt of the operating license in February 1990, except for RWST level. The setpoint methodology contains adequate allowance to bound anticipated drift over a three month period. Additionally, setpoint drift data has been trended since prior to licensing to confirm this allowance. No excessive drift has been noted over this period.

Specific changes proposed to be made to the CPSES technical specifications include the following:

1. Table 3.3-1, Functional Units 17 and 20

The licensee proposed to add new Action Statement 13 to Functional Unit 17 (Safety Injection Input from ESFAS) and Functional Unit 20 (Automatic Trip and Interlock Logic) to Table 3.3-1. This change would increase the maintenance AOT of an inoperable channel from 6 to 12 hours and increase the allowed bypass time for surveillance testing from 2 to 4 hours. The staff found this change acceptable in the SE of April 30, 1990.

2. Table 3.3-1, Note "e"

The licensee proposed to delete note "e" of Table 3.3-1. This change reflects the staff's finding that the AOT for instruments common to RPS and ESFAS do not have differing RPS and ESFAS requirements. Therefore, the change is acceptable.

3. Table 3.3-1, Action Statement 12

The licensee proposed changing Action Statement 12 of Table 3.3-1 to increase the AOT for surveillance testing from 2 to 4 hours. This change, which was found acceptable in the staff's SE, is applicable to Functional Units 7 (Overtemperature N-16) and 8 (Overpower N-16).

4. Table 3.3-2, Note "e"

The licensee proposed to revise note "e" of Table 3.3-2 to make the time allowed to place the steam generator high level channel into trip and the surveillance AOT consistent with the provisions for inoperable channels in Action Statement 17. This is an editorial change which results from changing the action statement to 17 for the functional units to which note "e" applies, and is acceptable.

5. Table 3.3-2, Functional Units 4.e and 5.b

The licensee proposes to increase the time to place an inoperable channel in the tripped position from 1 to 6 hours and to add a provision that allows one additional channel to be bypassed for up to 4 hours for surveillance testing. These changes give 3 channel systems the same provisions as 4 channel systems. These changes, which were found acceptable in the staff's SE, apply to Functional Units 4.e (Steam Line Pressure - Negative Rate High) and 5.b (Steam Generator Water Level -- High-High).

6. Table 3.3-2, Functional Units 1.b, 2.b, 3.a.2, 3.b.2, 7.a, and 11.a

The licensee proposes to delete Actions 13 and 26 to Table 3.3-2 that are no longer used and renumber ESF Action 12 to the new Action 26. This is an administrative change that avoids assigning the number 12 to two Action Statements.

7. Table 3.3-2, Action Statements 17 and 23

The licensee proposed to revise Actions 17 and 23 to Table 3.3-2 to delete specific reference to STARTUP and/or POWER OPERATION and apply the broader term of "operation". This change avoids confusion with the Table 1.2 definitions of startup and power operation but does not change the applicable modes of these functions. Therefore, this change is acceptable.

8. Table 3.3-2, Action Statements 19, 22, and 26

The licensee proposed to revise Actions 19, 22, and 26 of Table 3.3-2 to break the 12 hour shutdown time requirement into the more standard 6 hours to restore or be in at least HOT STANDBY within the next 6 hours. The proposed time limits, like the time limits in 3.0.3, will provide ample time for an orderly shutdown without excess stress on the plant. Therefore, the staff finds these changes acceptable.

The licensee also proposes to change Action Statements 19, 22, and 26 to Table 3.3-2 to increase the AOT for surveillance testing from 2 to 4 hours. This change, which was found acceptable in the staff's SE , is applicable to Functional Units 1.b, 2.b, 3.a.2, 4.b, 5.a, 6.a, 7.a, 7.b, and 11.a.

9. Table 4.3-2, Functional Unit 7.b

The licensee has proposed to increase the STI from monthly to quarterly for Functional Unit 7.b (RWST Level - Low-Low Coincident with Safety Injection). WCAP-10271 Supplement 3 concluded that the RWST level channels are identical in configuration to the steam generator level channels which were approved for quarterly STI in the staff's SER. Since the proposed change would result in essentially the same system unavailabilities and risks associated with relaxed SITs and AOTs for the steam generator level channel, and will have no impact on plant safety, the staff finds this change acceptable.

10. BASES 3/4.3.1 and 3/4.3.2

The licensee has proposed modifying Technical Specification Bases 3/4.3.1 and 3/4.3.2, REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION, to incorporate reference WCAP-10271, Supplement 2 and the staff's SER. The staff finds these changes appropriate and acceptable.

Since the licensee's submittal has been based on the recommendations of the staff's SE and SSE, and the licensee has adequately addressed the required findings regarding the applicability of the analyses to the CPSES site, and the concerns about setpoint drift, the staff finds the above changes to the CPSES technical specifications acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 32577 and 57 FR 55594). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Donna Skay, PDIV-2/NRR

Date: January 12, 1993