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

December 29, 1998

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - AMENDMENT NOS. 62 AND 48 TO FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89 (TAC NOS. M97809 AND M97810)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment Nos. 62 and 48 to Facility Operating License Nos. NPF-87 and NPF-89 for the Comanche Peak Steam Electric Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 2, 1996 (TXX-96434), as supplemented by letters dated October 2, 1998 (TXX-98215), and November 13, 1998 (TXX-98241 and TXX-98244).

The amendment would increase the allowed outage time (AOT) for a centrifugal charging pump from 72 hours to 7 days (TSs 3.1.2.4 and 3.5.2) and add a Configuration Risk Management Program (TS 6.8.3 f).

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

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

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 62 to NPF-87
2. Amendment No. 48 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

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TPolich (2)	TGwynn, RIV	WBeckner	PDIV-1 r/f	CHawes
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Signed 12/30/98

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

December 29, 1998

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Docket Nos. 50-445 and 50-446

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cc:

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 2159
Glen Rose, TX 76403-2159

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, TX 75224

Mr. Roger D. Walker
TU Electric
Regulatory Affairs Manager
P. O. Box 1002
Glen Rose, TX 76043

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036-5869

Honorable Dale McPherson
County Judge
P. O. Box 851
Glen Rose, TX 76043

Office of the Governor
ATTN: John Howard, Director
Environmental and Natural
Resources Policy
P. O. Box 12428
Austin, TX 78711

Arthur C. Tate, Director
Division of Compliance & Inspection
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Jim Calloway
Public Utility Commission of Texas
Electric Industry Analysis
P. O. Box 13326
Austin, TX 78711-3326

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. 62
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 August 2, 1996 (TXX-96434), as supplemented by letters dated October 2, 1998 (TXX-98215), and November 13, 1998 (TXX-98241 and TXX-98244), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 62 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy J. Polich

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 29, 1998

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. 48
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 August 2, 1996 (TXX-96434), as supplemented by letters dated October 2, 1998 (TXX-98215), and November 13, 1998 (TXX-98241 and TXX-98244), 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. 48, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy J. Polich

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

Attachment: Changes to the Technical Specifications

Date of Issuance: December 29, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 62 AND 48

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

DOCKET NOS. 50-445 AND 50-446

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

REMOVE	INSERT
3/4 1-10	3/4 1-10
3/4 5-3	3/4 5-3
B3/4 1-3	B3/4 1-3
B3/4 5-2	B3/4 5-2
6-7	6-7
6-7a	6-7a

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least once per 92 days the above required positive displacement charging pump shall be demonstrated OPERABLE by verifying that the flow path required by Specification 3.1.2.1a. is capable of delivering at least 30 gpm to the RCS; or

4.1.2.3.2 The above required centrifugal charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2370 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.3 A maximum of two charging pumps shall be OPERABLE, one charging pump shall be demonstrated inoperable* at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

*An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 7 days or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least the value specified in the COLR at 200°F within the next 6 hours; and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 The required centrifugal charging pump(s) shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5.

4.1.2.4.2 The required positive displacement charging pump shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.2.c.

4.1.2.4.3 Whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 350°F, a maximum of two charging pumps shall be OPERABLE, except when Specification 3.4.8.3 is not applicable.

When required, one charging pump shall be demonstrated inoperable# at least once per 31 days by verifying that the motor circuit breakers are secured in the open position.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODES 3 and 4 for the charging pump declared inoperable pursuant to Specification 3.1.2.4 provided the charging pump is restored to OPERABLE status within 4 hours after entering MODE 3 or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

**In MODE 4 the positive displacement pump may be used in lieu of one of the required centrifugal charging pumps.

#An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve(s) with power removed from the valve operator(s) or by a manual isolation valve(s) secured in the closed position.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE safety injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically opening the containment sump suction valves during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable because of the inoperability of a centrifugal charging pump, restore the inoperable pump to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one ECCS subsystem inoperable for reasons other than an inoperable centrifugal charging pump, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pumps and the safety injection pumps declared inoperable pursuant to Specification 3.5.3 provided the centrifugal charging pumps and the safety injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
8802 A & B	SI Pump to Hot Legs	Closed
8808 A, B, C, D	Accum. Discharge	Open*
8809 A & B	RHR to Cold Legs	Open
8835	SI Pump to Cold Legs	Open
8840	RHR to Hot Legs	Closed
8806	SI Pump Suction from RWST	Open
8813	SI Pump Mini-Flow Valve	Open

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 442 psig the interlocks prevent the valves from being opened.

*Surveillance Requirements covered in Specification 4.5.1.1.

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of two charging pumps to be OPERABLE and the requirement to verify one charging pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

With one centrifugal charging pump (CCP) inoperable, the inoperable CCP must be returned to OPERABLE status within 7 days. The 7 day Allowed Outage Time is based on a risk-informed assessment to manage the risk associated with the equipment in accordance with the Configuration Risk Management Program and is a reasonable time for repair of the CCPs.

The limitation for minimum solution temperature of the borated water sources are sufficient to prevent boric acid crystallization with the highest allowable boron concentration.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 1,100 gallons of 7000 ppm borated water from the boric acid storage tanks or 7,113 gallons of 2400 ppm borated water from the RWST.

As listed below, the required indicated levels for the boric acid storage tanks and the RWST include allowances for required/analytical volume, unusable volume, measurement uncertainties (which include instrument error and tank tolerances, as applicable), margin, and other required volume.

Tank	MODES	Ind. Level	Unusable Volume (gal)	Required Volume (gal)	Measurement Uncertainty	Margin (gal)	Other (gal)
RWST	5,6	24%	98,900	7,113	4% of span	10,293	N/A
	1,2,3,4	95%	45,494	70,702	4% of span	N/A	357,535*
Boric	5,6	10%	3,221	1,100	6% of span	N/A	N/A
Acid	5,6	20%	3,221	1,100	6% of span	3,679	N/A
Storage	(gravity feed)						
Tank	1,2,3,4	50%	3,221	15,700	6% of span	N/A	N/A

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

*Additional volume required to meet Specification 3.5.4.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

For Specification 3.1.3.1 ACTIONS b and c it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus fall under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately four hours for this verification.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The required accumulator contained volume and nitrogen cover pressure specified in LCO 3.5.1 represent analytical limits. Measurement uncertainty has not been incorporated into the specified required values. Current control room instrumentation used for indication of accumulator volumes and pressures include a 5 percent measurement uncertainty. Using this instrumentation, indicated values of between 39% and 61% for accumulator level (based on the analytical limits of 6119 gallons and 6597 gallons, respectively plus a 1% tank tolerance) and between 623 psig and 644 psig (based on analytical limits of 603 psig and 693 psig, respectively) for accumulator nitrogen cover pressure, satisfy the acceptance criteria for SR 4.5.1. Other methods employed to verify these values in satisfying SR 4.5.1 shall account for measurement uncertainties.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required by BTP ICSB 18. This is accomplished via key-lock control board cut-off switches.

If the boron concentration of one accumulator is not within limits, it must be returned to within the required limits within 72 hours. In this condition, the ability to maintain subcriticality may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Thus, 72 hours is allowed to return the boron concentration to within limits.

If one accumulator is inoperable for any reason other than boron concentration, the accumulator must be returned to OPERABLE status within one hour. In this inoperable condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA, which may result in unacceptable peak cladding temperatures. Due to the severity of the consequences should a LOCA occur in these conditions, the one hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to

ACCUMULATORS (Continued)

return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period. With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of two charging pumps to be OPERABLE and the requirement to verify one charging pump and all safety injection pumps to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

With one centrifugal charging pump (CCP) inoperable, the inoperable CCP must be returned to OPERABLE status within 7 days. The 7 day Allowed Outage Time is based on a risk-informed assessment to manage the risk associated with the equipment in accordance with the Configuration Risk Management Program and is a reasonable time for repair of the CCPs.

The requirement to remove power from certain valve operators is in accordance with Branch Technical Position ICSB-18 for valves that fail to meet single failure considerations. Power is removed via key-lock switches on the control board.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests and ≤ 0.75 L_a for Type A tests;

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

   b) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves, which is specified in Specifications 4.6.1.7.2 and 4.6.1.7.3.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

6.9.1.1 Not used.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other individuals (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent greater than 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions;

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of 10 CFR 20.2206.

PROCEDURES AND PROGRAMS (Continued)

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR 190.

f. Configuration Risk Management Program (CRMP)

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed Allowed Outage Time has been granted. The program shall include the following elements:

1) Provisions for the control and implementation of a Level 1, at-power, internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

2) Provisions for performing an assessment prior to entering the LCO Action for preplanned activities.

3) Provisions for performing an assessment after entering the LCO Action for unplanned entry into the LCO Action.

4) Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action.

5) Provisions for considering other applicable risk significant contributors such as Level 2 issues, and external events, qualitatively or quantitatively.

g. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995".

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.3 psig.

The maximum allowable containment leakage rate, L_a at P_a, shall be 0.10% of containment air weight per day.

Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 62 AND 48 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 August 2, 1996 (TXX-96434), as supplemented by letters dated October 2, 1998 (TXX-98215), and November 13, 1998 (TXX-98241 and TXX-98244), Texas Utilities Electric Company (TU Electric/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 (CPSES), Units 1 and 2. The proposed changes would increase the allowed outage time (AOT) for one centrifugal charging pump (CCP) declared inoperable while in MODES 1, 2, 3, or 4 from 72 hours to 7 days. The affected TSs are 3/4.1.2.4, "Charging Pumps - Operating," and 3/4.5.2, "ECCS Subsystems - $T_{avg} > 350°F$." The licensee is also adding a Configuration Risk Management Program to Administrative Controls section of TS 6.8.3 f. The licensee is requesting this change to accommodate potential replacement of the rotating element of an inoperable CCP.

The October 2 and November 13, 1998, letters provided additional information and changed the no significant hazards consideration material to include a Configuration Risk Management Program in TS 6.8.3 f and was beyond the scope of the original scope of the August 2, 1996, application. The markup to the Improved Technical Specifications (ITS) (Attachment 4 to TXX-98241) is not addressed in the following safety evaluation because it was provided for information purposes. It will be addressed in connection with the CPSES ITS conversion dated May 15, 1997 (TXX-97105).

2.0 EVALUATION

The Comanche Peak charging system consists of two redundant centrifugal charging pumps and one reciprocating charging pump that operate as part of the Chemical and Volume Control System during normal operation and as part of the Emergency Core Cooling System (ECCS) following a LOCA.

During normal operation the charging system maintains the proper water inventory in the RCS, provides seal water flow to the RCP seals, and maintains water purity and boron concentration in the reactor coolant. In the event of a loss of coolant accident the CCPs act as components of the Emergency Core Cooling System and deliver high pressure water to the reactor coolant system cold legs. The current Final Safety Analysis Report indicates that the Chapter 15 transients are analyzed assuming one pump in operation. These functions, both normal and emergency operation, can be met with the use of one CCP.

Currently the TS allows for one CCP to be inoperable in MODES 1, 2, 3, 4 for 72 hours. The licensee is requesting to increase this AOT to seven days. This proposed AOT of 7 days is an extension of the current ACTION statement and has no impact on the current safety analysis. Therefore, the safety analysis remains valid, and it is concluded that there is no difference in the deterministic safety significance of a 72-hour AOT for the Comanche Peak CCP and a 7-day AOT. The difference in the current TS versus the proposed extension lies in the added risk due to the extension of the AOT which is reviewed in the following section of this evaluation.

2.1 <u>Evaluation of the Probabilistic Risk Assessment (PRA) Used to Support the Proposed Comanche Peak Steam Electric Station (CPSES) Amendment</u>

The licensee requested an allowed outage time (AOT) extension from 72 hours to 7 days for an inoperable centrifugal charging pump (CCP). The extended AOT would provide a sufficient time for performing most repairs on an inoperable CCP and could help avoid an unnecessary plant shutdown. Use of PRA insights constitutes the primary justification for the proposed change. The licensee's risk analysis indicates that the risk impact of the extended AOT for a CCP is small.

To gain risk insights, the staff used a three-tiered approach to evaluate the risk associated with the proposed amendment. The first tier evaluated the PRA model and the impact of the change on plant operational risk. The second tier addressed the need to preclude potentially high risk configurations if additional equipment will be taken out of service simultaneously or other risk significant operational factors such as concurrent system or equipment testing are involved. The third tier evaluated the licensee's configuration risk management program, to ensure that equipment removed from service prior to or during the proposed AOT will be appropriately assessed from a risk perspective. Each tier and associated findings are discussed below.

Tier 1: PRA Evaluation of AOT Extensions

The Tier 1 staff review of the licensee's PRA involved two aspects: (1) evaluation of the PRA model and its application to the proposed AOT extension, and (2) evaluation of PRA results and insights stemming from the application.

(1) Evaluation of PRA Model and Application to the AOT Extension.

The staff's review focused on the capability of the licensee's PRA model to analyze the risk stemming from the modified AOTs for CCPs. This activity, however, did not involve an in-depth review of the CPSES PRA to the extent necessary to validate the licensee's overall quantitative estimates. This was based on the staff's initial screening process which considered the information contained in the licensee's submittal, the Comanche Peak Individual Plant

Examination (IPE), the IPE for External Events (IPEEE) and the Staff Evaluation Report (SER)[1] on risk-informed inservice testing (RI-IST). Recent component reliability and availability experience and plant-specific features such as emergency core cooling system configurations were also considered. The staff subsequently determined that an in-depth review of the CPSES PRA would not be needed for this application.

The licensee's PRA used to support the proposed change is the same as the original IPE and IPEEE. The IPE has been reviewed by the Nuclear Regulatory Commission (NRC) Office of Research (RES) and RES staff has concluded that the CPSES IPE met the intent of Generic Letter 88-20. CPSES indicated that the IPE is adequate for this application due to the following considerations: (1) The IPE shows a relatively flat profile, meaning that the CDF is uniformly distributed by sequence type and initiating events. A significant change in plant design or operation would be necessary to significantly change this profile; (2) Plant changes and modifications have generally made the affected systems more reliable, and in some cases more redundant or diverse. For example, the new high temperature reactor coolant pump seals have been installed. CPSES indicates that the IPE was essentially revalidated to represent the as-built and as-operated plant prior to submitting the IPEEE in 1995. The licensee determined that there were no significant impacts on the models that would make the results non-conservative; (3) The updated plant-specific data, which has not been incorporated into the IPE model, shows that, in most cases, failure rates have decreased. Both internal and external independent reviews were conducted for the IPE and IPEEE process. A final independent review was performed after the IPE study was completed. The licensee intends to update the PRA periodically about every two refueling outages.

The recently completed NRC review of the licensee's RI-IST submittal included a review of the CPSES PRA, which include analyses stemming from internal and external initiating events for both at-power and shutdown modes, used in support of the submittal. Though the focus of the PRA review was to examine whether the PRA was of sufficient quality, scope and level of detail for use in support of that particular application, much of the same focus was determined to remain applicable for this charging pump application. The RI-IST SER concluded that the CPSES PRA was of sufficient quality, scope and level of detail to support the conclusions made in the IST submittal. Specifically, the IST PRA review found no significant deficiencies or shortcomings in areas such as initiating event types and frequencies, event trees and success criteria, data, common cause failures, and human reliability analysis.

The staff evaluated several areas in detail that are closely associated with the proposed charging pump AOT extension. For example, the success criteria and the reliability and unavailability data for CCPs were examined. The staff finds that the average unavailability of the Chemical and Volume Control (CVCS) trains for both units during last three years has been consistently lower than those used in the IPE. Likewise, the CCP reliability has been high and exceeds the IPE data. For this period, the run failure rate was estimated to be 2E-6/hr (3.4E-6/hr in the IPE) and the start failure rate 0.0/demand (3.3E-3/demand in the IPE). The data used in the CPSES IPE was based on the PLG 500 Database.

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[1] Letter, "Approval of Risk-Informed Inservice Testing Program for CPSES, Units 1 and 2," August 14, 1998

A plant-specific data update was performed for the four-year period between 1991 to 1994. The update showed that standby pump failure data used for IPE had been conservative. The CPSES PRA models the CVCS functions and their corresponding success criteria in the both accident sequence analysis and the system fault tree models. Review of both the system level success criteria and accident sequence level success criteria found that the CCPs, in general, were adequately modeled. The following summarizes the accident sequence level success criteria:

Feed and Bleed;	One of two CCPs or one of two Safety Injection Pumps (SIPs) and one of two Power Operated Relief Valves - Transient and small LOCA initiating events
Safety Injection;	One of two CCPs or one of two SIPs - LOCA and Steam Generated Tube Rupture (SGTR) initiating events
Recirculation;	One of two CCPs or one of two SIPs and one of two residual heat removal pumps - Transient, LOCA and SGTR initiating events
Seal LOCA;	Thermal barrier cooling by component cooling water or seal injection from CVCS - Non-LOCA Transient initiating event
Long-term Shutdown;	Boration of the RCS using CVCS - Anticipated Transient Without Scram initiating event

The supporting bases for these success criteria were based either on the design bases documents or on supporting calculations performed as part of the IPE study. The staff found that the success criteria and their bases are generally consistent with other IPEs. In addition, it is noted that the Safety Injection system essentially works as a redundant system for the accident sequences that are associated with feed and bleed, safety injection and high pressure recirculation.

The licensee performed sensitivity studies to examine the robustness of the importance measures for the CVCS system for the various failure modes. The studies showed that the CDF at CPSES is relatively insensitive to changes in CCP unavailability or reliability. The CDF changed less than 0.1% when the test and maintenance unavailability was doubled and less than 0.5% for a doubled demand failure rate.

The staff finds that the CPSES PRA used in support of the proposed change in CCP AOT extension is of sufficient quality, scope and level of detail for the application.

(2) Evaluation of PRA Results and Insights Associated with the Proposed Change

The IPE and IPEEE estimated the CDF contributions to be about 5.7E-5/yr and 2.5E-5/yr, respectively from internal and external initiating events. Fire and tornado initiators were the main contributors to the IPEEE CDF.

Risk measures on which the staff evaluation is based include CDF change, incremental conditional core damage probability (ICCDP) for a single outage, large early release frequency (LERF) change, and incremental conditional large early release probability (ICLERP) for a single outage. To calculate the risk impact, CPSES increased the expected maintenance unavailability by the ratio of 2.59, based on the PLG Database correlation, which relates corrective maintenance duration to TS AOT. The test unavailability was increased by a factor of 2.33. Summarized below are the results of the calculations for those risk measures;

Risk Measures	Results
ΔCDF	3E-8/yr
ICCDP for a single outage	1E-6
ΔLERF	9E-9/yr
ICLERP for a single outage	2E-8

These results are based on both internal and external initiating events (fire and tornado), and both test and maintenance unavailabilities are included. Review of the risk measures considered shows that the risk impact of the proposed change is small. In particular, the changes in average CDF and LERF are estimated to be much below the staff criteria in RG 1.174[2]. The single AOT risk for CPSES is about 1E-6 and is slightly in excess of the guideline value of 5E-7 in RG 1.177[3]; however, the result from external initiators, which constitute 50% of the total, has been included. This relatively large ICCDP, calculated with some conservatism, indicates that the instantaneous risk given a CCP out of service at CPSES is also relatively significant. The licensee should thus try to minimize the time in this configuration. The staff examined the licensee's response to the Tier 2 and Tier 3 guidance in this regard, and found that the licensee's Tier 2 and Tier 3 process could provide a reasonable means to accomplish the intent of the guidance. The licensee also provided a detailed discussion on the potential safety benefit stemming from being able to perform maintenance at power instead of having to shut the plant down to perform the maintenance. In several studies quoted in the licensee's RAI response, the averted risk from transition to shutdown was relatively significant as compared to the increased risk from performing the maintenance at power. The ICLERP result was still less than 5E-8. Therefore, the staff concludes that the risk impact of the proposed change is small and below the staff criteria and guidelines in RG 1.174 and 1.177.

Tier 2: Avoidance of Risk Significant Plant Configurations

CPSES identified a list of components and systems whose simultaneous unavailability might place the plant in a high risk configuration, based partially on their Risk Achievement Worth value. These components and systems include the following:

Electric Power - Opposite train motive and control power
Refueling Water Storage Tank - Tank and its associated discharge valves
Service Water System - Opposite train
Component Cooling Water System - Opposite train
Emergency Diesel Generator - Opposite train
ECCS Injection/Recirculation flow path valves

[2]RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998

[3]RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," September 1998

Electrical power - Same train motive and control power
Service Water - Same train
CVCS - Opposite train

This list was identified when CPSES developed their risk matrix. The risk matrix and work control procedure control simultaneous maintenance activities on these systems and components. To avoid or reduce the potential for risk-significant configurations from either emergent or planned work, CPSES has put in place a set of administrative guidelines which are consistent with the CRMP in Tier 3. The licensee has proposed to revise the Administrative Controls in the CPSES TS to include a CRMP in TS 6.8.3 f that is consistent with RG 1.177. The licensee stated that these guidelines control configurational risk by assessing the risk impact of equipment outage during all modes of operation to assure that the plant is being operated within acceptable risk. The weekly online maintenance schedules are train based and prohibit the scheduling of opposite train activities without additional review, approvals, and/or compensatory measures. In addition, restrictions are placed on the number and combination of systems/trains allowed to be simultaneously unavailable for scheduled work. Therefore, the staff finds that the licensee adequately addresses the intent of the Tier 2 guidance.

Tier 3: Risk-Informed Plant Configuration Management

Various procedures at CPSES embody programs that provide reasonable assurance that the risk impact of equipment outages is appropriately evaluated prior to and while performing any maintenance activity. The upper level procedure, Procedure No. STA-604, "Work Scheduling," prescribes the methods and assigns responsibilities for scheduling of test and maintenance activities during at-power operations and shutdown. The risk impact of on-line maintenance is required to be evaluated as prescribed by Instruction No. WCI-203, "Weekly Surveillance / Work Scheduling." For risk assessment of online maintenance activities, risk categories were developed using a methodology consistent with the overall public safety goal and the available industry guidelines, such as the draft EPRI PSA Applications Guide. The risk categories were defined based on internal events CDF as the following:

Risk Category 1: High Risk
Risk Category 2: High-Medium Risk
Risk Category 3: Medium Risk
Risk Category 4: Low Risk

Scheduling of online maintenance activities that fall in Categories 1 and 2 are generally prohibited. A risk matrix is provided for the work schedulers with the means of determining the category into which a work activity would fall. CPSES plans to acquire and use a safety monitor, a computerized online risk monitoring software, in the future.

An instantaneous risk measure for a specific plant configuration is used to control the instantaneous risk level and the duration. This was accomplished by keeping the configuration-specific core damage probability, i.e., the instantaneous CDF times the duration of a specific configuration, less than 1E-6.

CPSES assesses risk when scheduling planned outage activities. Procedure No. STA-627, " Control of Planned Outages," requires that an independent risk assessment of the outage schedule be performed. The Outage Risk Assessment and Management (ORAM) software tool is used to provide a means to evaluate the risk associated with planned outage activities.

The licensee has proposed to revise the Administrative Controls in the CPSES TS to include a CRMP that is consistent with RG 1.177. The proposed changes also revise appropriate BASES sections to reference the CRMP in TS 6.8.3 f.

The staff finds that the licensee has implemented a risk-informed CRMP to assess the risk associated with the removal of equipment from service prior to, or during, the proposed AOT. Therefore, the staff concludes that CPSES has met the Tier 3 guidance in RG. 1.177.

2.2 Implementation and Monitoring

The staff expects the licensee to implement these TS changes in accordance with the three-tiered approach described above. The licensee has also indicated that the maintenance scheduling practice and the tools used to implement a means of evaluating the impact of maintenance activities on plant configurations are consistent with the Maintenance Rule (10 CFR 50.65). The AOT extension will allow efficient scheduling and performance of on-line maintenance within the boundaries established by implementing the Maintenance Rule. The licensee will monitor CCP/CVCS performance in relation to the Maintenance Rule performance criteria. Therefore, application of these implementation and monitoring strategies will help to ensure that an extension of TS CCP AOT does not degrade operational safety over time and that the risk expected when a CCP is taken out of service is minimized.

3.0 EVALUATION CONCLUSION

The staff has reviewed the licensee submittal proposing to increase the AOT for one CCP. The staff concludes that because (1) there is no change to the Chapter 15 Safety Analysis, and (2) this is an extension of a condition for which the plant has already been analyzed, the deterministic aspect of this change is acceptable.

The CPSES PRA used in support of the proposed change in charging pump AOT extension is believed to be of sufficient quality, scope and level of detail for the proposed application. The staff did not identify any significant weaknesses or deficiencies associated with the licensee's risk analysis used to support the proposed change that could impact the overall quantitative conclusion. The results of the risk analysis indicate that the risk impact of the proposed change would be small. The licensee has put in place tools and guidance that provide a reasonable assurance to avoid potential risk significant configurations. The licensee has also implemented a risk-informed CRMP to assess the risk associated with the removal of equipment from service prior to or during the AOT, and to take appropriate measures in response to the outcome of the risk assessment. The staff concludes that the results and insights of the PRA analysis support the proposed charging pump AOT extension from 72 hours to 7 days. Therefore, the staff concludes that the proposed TS changes to TS 3.1.2.4, TS 3.5.2, and TS 6.8.3 f are acceptable, and the Bases have been changed to reflect these TS changes.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 65617). The amendment also changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (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 amendments.

6.0 CONCLUSION

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

Principal Contributors: I. Jung, SPSB
S. Brewer, SRXB

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