UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Title: AFFIRMATION/DISCUSSION AND VOTE

Location: ROCKVILLE, MARYLAND

Date: APRIL 6, 1993

Pages: 4 PAGES

SECRETARIAT RECORD COPY

NEAL R. GROSS AND CO., INC.

COURT REPORTERS AND TRANSCRIBERS
1323 Rhode Island Avenue, Northwest
Washington, D.C. 20005
(202) 234-4433

DISCLAIMER

This is an unofficial transcript of a meeting of the United States Nuclear Regulatory Commission held on April 6, 1993, in the Commission's office at One White Flint North, Rockville, Maryland. The meeting was open to public attendance and observation. This transcript has not been reviewed, corrected or edited, and it may contain inaccuracies.

The transcript is intended solely for general informational purposes. As provided by 10 CFR 9.103, it is not part of the formal or informal record of decision of the matters discussed. Expressions of opinion in this transcript do not necessarily reflect final determination or beliefs. No pleading or other paper may be filed with the Commission in any proceeding as the result of, or addressed to, any statement or argument contained herein, except as the Commission may authorize.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

\- - - -

AFFIRMATION/DISCUSSION AND VOTE

\- - - -

PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Tuesday, April 6, 1993

The Commission met in open session, pursuant to notice, at 12:04 p.m., Ivan Selin, Chairman, presiding.

COMMISSIONERS PRESENT:

IVAN SELIN, Chairman of the Commission
KENNETH C. ROGERS, Commissioner
JAMES R. CURTISS, Commissioner
FORREST J. REMICK, Commissioner

STAFF SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary

WILLIAM C. PARLER, General Counsel

P-R-O-C-E-E-D-I-N-G-S

12:04 p.m.

CHAIRMAN SELIN: Good afternoon, ladies and gentlemen.

This is an affirmation session and we have two items to come before us.

Mr. Chilk, would you please lead us through the session?

SECRETARY CHILK: The first item, Mr. Chairman, is SECY-93-083. It's a Motion for Stay of Full Power License for Comanche Peak Unit 2 and Petition to Intervene in Comanche Peak Proceeding.

The Commission is being asked to act on an order that responds to two motions to stay the issuance of the full power licence by B. Irene Orr and D.I. Orr and a petition to intervene by R. Micky Dow and Sandra Long Dow.

All Commissioners have approved an order which denies both a stay motion and a petition to intervene. Commissioner de Planque, although not present, has indicated her approval of the order.

Would you please affirm your votes?

CHAIRMAN SELIN: Aye.

COMMISSIONER ROGERS: Aye.

COMMISSIONER CURTISS: Aye.

COMMISSIONER REMICK: Aye.

SECRETARY CHILK: The second item is the subject of Comanche Peak Unit 2 Full Power Operating License.

The Commission is being asked to act on a recommendation that the Director of the Office of Nuclear Regulatory Regulation be authorized to issue a full power license for Comanche Peak Unit 2.

All Commissioners have voted to approve the authorization. Commissioner de Planque, although not present, has approved this authorization.

Would you please affirm your votes?

CHAIRMAN SELIN: Aye.

COMMISSIONER ROGERS: Aye.

COMMISSIONER CURTISS: Aye.

COMMISSIONER REMICK: Aye.

SECRETARY CHILK: I have nothing further.

CHAIRMAN SELIN: I would like to point out that my vote and perhaps those of others was conditioned by the later staff response of specific question on April 2nd in addition to the original submission from the staff. Thank you.

This session is completed.

(Whereupon, at 12:05 p.m., the above-entitled matter was concluded.)

CERTIFICATE OF TRANSCRIBER

This is to certify that the attached events of a meeting of the United States Nuclear Regulatory Commission entitled:

TITLE OF MEETING: AFFIRMATION/DISCUSSION AND VOTE

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: APRIL 6, 1993

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

Carol Lynch

Reporter's name: Peter Lynch

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

April 5, 1993

NOTE TO: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: COMANCHE PEAK UNIT 2 FULL POWER LICENSE BRIEFING BOOK

On March 8, 1993, the subject briefing book was provided to you in support of the March 16, 1993 Commission Meeting to consider authorizing a full power license for Comanche Peak Steam Electric Station Unit 2. The March 8, 1993 note included information regarding fire barrier and ampacity derating testing recently conducted by TU Electric. You were informed that updates to this information would be described in SSER 27, to be published upon full power license issuance. SSER 27 (Draft) is enclosed for your information. The final ORAT Inspection Report was issued on March 9, 1993. The changes from the draft, which was included in the briefing book, were editorial in nature.

James H Sniezek /for
James M. Taylor
Executive Director
for Operations

Enclosure:
SSER 27 (Draft)

cc: SECY (w/enclosure)
OGC
OCA
OPA
PDR

DRAFT

NUREG-0797
Supplement No. 27

Safety Evaluation Report

related to the operation of Comanche Peak Steam Electric Station, Unit 2

Docket No. 50-446

Texas Utilities Electric Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1993

UNITED STATES NUCLEAR REGULATORY COMMISSION

DRAFT

ABSTRACT

Supplement 27 to the Safety Evaluation Report related to the operation of the Comanche Peak Steam Electric Station (CPSES), Unit 2 (NUREG-0797), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC). The facility is located in Somervell County, Texas, approximately 40 miles southwest of Fort Worth, Texas. This supplement reports the status of certain issues that had not been resolved when the Safety Evaluation Report and Supplements 1, 2, 3, 4, 6, 12, 21, 22, 23, 24, 25, and 26 to that report were published. This supplement deals primarily with Unit 2 issues.

Supplement 5 was cancelled. Supplements 7, 8, 9, 10, and 11 were limited to the staff's evaluation of allegations investigated by the NRC Technical Review Team. Supplement 13 presented the staff's evaluation of the Comanche Peak Response Team (CPRT) Program Plan, which was formulated by the applicant to resolve various construction and design issues raised by sources external to TU Electric (applicant). Supplements 14 through 19 presented the staff's evaluation of the CPSES Corrective Action Program: large- and small-bore piping and pipe supports (Supplement 14); cable trays and cable tray hangers (Supplement 15); conduit supports (Supplement 16); mechanical, civil/structural, electrical, instrumentation and controls, and systems portions of the heating, ventilation, and air conditioning (HVAC) system workscopes (Supplement 17); HVAC structural design (Supplement 18); and equipment qualification (Supplement 19). Supplement 20 presented the staff's evaluation of the CPRT implementation of its Program Plan and the issue-specific action plans, as well as the CPRT's investigations to determine the adequacy of various types of programs and hardware at CPSES.

Items identified in Supplements 7, 8, 9, 10, 11, and 13 through 20 are not included in this supplement, except to the extent that they affect the licensee's Final Safety Analysis Report.

This twenty-seventh supplement, which is in support of the full-power license for Unit 2, provides updated information on the issues that had been considered previously, as well as the evaluation of issues that have arisen since the twenty-sixth supplement was issued. This evaluation addresses all of the issues necessary to support the issuance of a full-power license for Unit 2.

TABLE OF CONTENTS

APPENDICES

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Nuclear Regulatory Commission (NRC) Safety Evaluation Report (SER), NUREG-0797, on the application of the Texas Utilities Generating Company (TUGCO)* (the applicant) for a license to operate the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, was issued in July 1981. Since then the following supplements have been issued:

- Supplement 1 (SSER 1) was issued in October 1981. It described the resolution of a large portion of the outstanding and confirmatory issues identified in the SER.

- Supplement 2 (SSER 2) was issued in January 1982. It included the report of the Advisory Committee on Reactor Safeguards (ACRS) to the NRC Chairman by letter dated November 17, 1981, which was appended as Appendix F. Applicant and staff responses to comments by the ACRS were also included.

- Supplement 3 (SSER 3) was issued in March 1983. It addressed outstanding and confirmatory issues resolved since SSER 2 was issued. The staff's evaluation of the applicant's emergency plans was also described.

- Supplement 4 (SSER 4) was issued in November 1983. It included the staff's evaluation report on design modifications made to the Westinghouse model D4 and D5 steam generators installed at CPSES.

- Supplement 5 (SSER 5) has been canceled. It was to have been limited exclusively to the CYGNA Independent Assessment Program. The issues from the CYGNA Independent Assessment Program have been addressed in the applicant's corrective action program. The staff's evaluations of the CYGNA issues are provided in the respective SSERs (14-19) for each corrective action program design workscope. Therefore, the planned supplement was never issued.

- Supplement 6 (SSER 6) was issued in November 1984. It addressed outstanding and confirmatory issues resolved since SSER 4 was issued. Noteworthy in this supplement was a partial exemption to General Design Criterion (GDC) 4 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) deleting the requirement for installing jet impingement shields for the Unit 1 primary coolant loop piping at postulated break locations.

---

*On January 16, 1987, TUGCO informed the NRC that it had adopted a new corporate signature and would be known as TU Electric (Texas Utilities Electric Company).

- Supplement 7 (SSER 7) was issued in January 1985. It was limited exclusively to the staff's evaluation of allegations investigated by the NRC's Technical Review Team (TRT) pertaining to plant electrical/ instrumentation systems and testing programs.

- Supplement 8 (SSER 8) was issued in February 1985. It was limited exclusively to the staff's evaluation of allegations investigated by the TRT pertaining to the plant's civil/structural and other miscellaneous construction and plant-readiness testing items.

- Supplement 9 (SSER 9) was issued in March 1985. It was limited exclusively to the staff's evaluation of coating requirements inside containment and allegations of coating deficiencies investigated by the TRT.

- Supplement 10 (SSER 10) was issued in April 1985. It was limited exclusively to the staff's evaluation of allegations investigated by the TRT pertaining to the mechanical and piping areas.

- Supplement 11 (SSER 11) was issued in May 1985. It was limited exclusively to the staff's evaluation of allegations investigated by the TRT pertaining to quality assurance/quality control (QA/QC) practices in the design and construction of CPSES.

- Supplement 12 (SSER 12) was issued in October 1985. It updated the SER further by providing the results of the staff's review of information submitted by the applicant by letter and in Final Safety Analysis Report (FSAR) amendments addressing several of the issues and license conditions listed in Sections 1.7, 1.8, and 1.9 of the SER that were unresolved at the time SSER 6 was issued. SSER 12 also listed several new issues that had been identified since SSER 6 was published and that were unresolved.

- Supplement 13 (SSER 13) was issued in May 1986. It presented the staff's evaluation of the Comanche Peak Response Team (CPRT) Program Plan, which was formulated by the applicant to resolve various design and construction issues raised by the Atomic Safety and Licensing Board, allegers, the Citizens Association for Sound Energy (CASE), and NRC inspections, as well as those raised by CYGNA Energy Services during its independent design assessment.

- Supplement 14 (SSER 14) was issued in March 1988. It presented the staff's evaluation of the applicant's corrective action program related to large- and small-bore piping and pipe supports.

- Supplements 15 and 16 (SSERs 15 and 16) were issued in July 1988; Supplements 17 through 19 (SSERs 17-19) were issued in November 1988. They presented the staff's evaluation of the corrective action program as related to cable trays and cable tray hangers (SSER 15); conduit supports (SSER 16); the mechanical, civil/structural, electrical, and instrumentation and controls workscopes, and systems portions of the heating, ventilation, and air conditioning (HVAC) system workscope (SSER 17); HVAC structural design (SSER 18); and equipment qualification (SSER 19).

- Supplement 20 (SSER 20) was issued in November 1988. It presented the staff's evaluation of the CPRT implementation of the CPRT Program Plan and the issue-specific action plans, as well as the CPRT's investigations to determine the adequacy of various types of programs and hardware at CPSES.

- Supplement 21 (SSER 21) was issued in April 1989. It updated the SER further by providing the results of the staff's review of information that the applicant submitted by letter and in FSAR amendments. It addressed several of the issues and license conditions listed in Sections 1.7, 1.8, and 1.9 of the SER that were unresolved at the time SSER 12 was issued. Of note from an administrative standpoint, SSER 21 renumbered items appearing in Sections 1.7, 1.8, and 1.9, and deleted all items that were previously resolved but listed in SSER 12.

- Supplement 22 (SSER 22) was issued in January 1990. It updated the SER by presenting the results of the staff's review of information that the applicant submitted by letter and in FSAR amendments. The staff review addressed several of the issues and license conditions listed in Sections 1.7, 1.8, and 1.9 of the SER that were unresolved at the time SSER 21 was issued.

- Supplement 23 (SSER 23) was issued in February 1990 with the low-power operating license for CPSES Unit 1. It documented resolution of the remaining outstanding issues appearing in Section 1.7 of SSER 22.

- Supplement 24 (SSER 24) was issued with the full-power operating license for CPSES Unit 1. Confirmatory issues remaining at the time of license issuance, as well as proposed license conditions, were listed in Sections 1.8 and 1.9, respectively.

- Supplement 25 (SSER 25) was issued in September 1992. It updated the SER and subsequent SSERs, by presenting the results of the staff's review of information that the applicant submitted by letter and in FSAR amendments; specifically documenting reviews in support of the licensing of Unit 2. The staff review also addressed the translation of the Unit 1 and common area Corrective Action Program to Unit 2.

- Supplement 26 (SSER 26) was issued in February 1993. It updated the SER and subsequent SSERs by presenting the results of the staff's review of information that the applicant submitted by letter and in FSAR amendments. Significant issues contained in this SSER included TU Electric's fire barrier qualification testing program, preservice inspection and inservice testing programs and relief requests, an optimized fuel assembly review, and the plant's dual-unit station blackout review. This evaluation addressed all of the issues necessary to support the issuance of a low-power license for Unit 2.

SSER 27 updates the SER and subsequent SSERs by presenting the results of the staff's review of information that TU Electric has submitted by letter. It addresses all of the issues necessary to support the issuance of a full-power license for Unit 2. Each section or appendix of this supplement is numbered and titled so that it corresponds to the section or appendix of the SER that has

been affected by the staff's additional evaluations and, except where specifically noted, does not replace the corresponding SER section or appendix. Appendix A is a continuation of the chronology of correspondence between the NRC and TU Electric that updates the correspondence listed in the SER and in SSERs 1 through 26. Appendix B includes references other than NRC documents and correspondence cited in this supplement.* Appendix C contains information concerning the status of Three Mile Island (TMI) issues for CPSES Unit 2. Appendix D contains a list of principal contributors to this supplement. No changes were made to SER Appendices E, F, G, H, I, J, K, L, M, N, O, P, Q, R, S, T, U, V, W, X, Y, Z, AA, BB, CC, DD, EE, or FF by this supplement.

Copies of this supplement are available for public inspection at the NRC's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555; and at the University of Texas at Arlington Library, Government Publications/Maps, 701 South Cooper, P.O. Box 19447, Arlington, Texas 76019.

The NRC Project Manager for Comanche Peak Steam Electric Station, Unit 2, is Brian E. Holian. Mr. Holian may be contacted by calling (301) 504-1334 or by writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.7 Summary of Outstanding Issues

Section 1.7 of the SER, as supplemented, identified no open issues at the time SSER 26 was issued. Those issues that were resolved in previous supplements were not listed in SSER 26.

1.8 Confirmatory Issues

Section 1.8 of the SER, as supplemented, identified no confirmatory issues at the time SSER 26 was issued.

1.9 License Conditions

In Section 1.9 of SSER 26, the staff listed three proposed license conditions. Those license conditions that were resolved in previous supplements were not listed in SSER 26.

License conditions discussed in previous SSERs that were included in the Unit 1 license, and are similarly included in the Unit 2 license, follow:

(1) The applicant shall continue to control mineral exploration within the exclusion area; that is, at distances beyond 2250 feet from safety-related structures per GDC 4, 10 CFR Part 50, Appendix A.

---

*Availability of all material cited is described on the inside front cover of this document.

(2) The applicant must implement and maintain in effect all provisions of the approved fire protection program, as described in the Final Safety Analysis Report (as amended) and as approved in the SER and its supplements, subject to the following provision: "The applicant may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."

(3) The applicant shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans, previously approved by the Commission, and all amendments made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, are entitled: "Comanche Peak Steam Electric Station Physical Security Plan" with revisions submitted through July 21, 1992; "Comanche Peak Steam Electric Station Security Training and Qualification Plan" with revisions submitted through June 10, 1991; and "Comanche Peak Steam Electric Station Safeguards Contingency Plan" with revisions submitted through December 1988.

9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

In Supplement 21 to the SER (SSER 21), which was issued in April 1989, the staff reviewed the Comanche Peak fire protection program as documented in the FSAR through Amendment 71 and as described in Revision 1 to the Fire Protection Report.

In Supplement 26 to the SER (SSER 26), the staff documented a review of fire protection-related changes and modifications made to the FSAR through Amendment 87 and through Revision 6 to the Fire Protection Report. In SSER 26, the staff concluded that the fire protection program for Unit 2 adheres to the guidance in Appendix A to Branch Technical Position (BTP) APCSB 9.5-1, and with Sections G, J, and O of Appendix R to 10 CFR Part 50.

The following TU Electric commitments related to fire protection issues were documented in SSER 26:

- 36-inch wide fire barrier: Perform a confirmatory test or provide additional information addressing the staff's concerns (SSER 26, pp. 9-10 and 9-23).

  This issue is discussed in this SSER (below). The NRC staff witnessed the fire test and reviewed the preliminary test results submitted by TU Electric. The staff will review the final fire test report and will prepare a safety evaluation. This action is tracked by NRC TAC No. M85998.

- Ampacity derating testing: Perform plant-specific testing (SSER 26, pp. 9-20 and 9-32).

  This issue is discussed in this SSER (below). TU Electric provided preliminary results from their plant-specific ampacity derating tests. The staff observed a portion of the testing, and will review the final ampacity test report and will prepare a safety evaluation. This action is tracked by NRC TAC No. M85999.

- "Box enclosure" barriers: Establish qualification (SSER 26, pp. 9-22 and 9-23).

  This issue is closed in this SSER (below) based on NRC staff's onsite inspection and review of the fire barrier upgrades and a review of the engineering documentation associated with these upgrades.

- Compensatory measures: Provide fire watches in accordance with the CPSES fire protection plan until the barriers have cured for 30 days, and until the box enclosures are qualified (SSER 26, pp. 9-22 and 9-26).

  This issue is closed in this SSER (below) based on the preliminary results of the 36-inch-wide cable tray fire test, with a seven-day cure time, and the completion of "box enclosure" upgrades.

- Alternative shutdown system design enhancements: Implement design changes, as necessary, to ensure that the torque and limit switches in the affected motor-operated valve operators are electrically connected downstream of the contacts located in the motor control center (SSER 26, pp. 9-36).

  This issue is not addressed below. In SSER 26, the staff reported TU Electric's commitment to perform the necessary design changes before startup from the next refueling outage (first refueling outage for Unit 2 and the third refueling outage for Unit 1). This action is tracked by NRC TAC No. M86000.

In letters of February 26, 1993 (TU Electric letter TXX-93101 to NRC), March 10, 1993 (TU Electric letter TXX-93125 to NRC), and March 23, 1993 (TU Electric letter TXX-93136 to NRC), TU Electric submitted updated information regarding the fire barrier commitments discussed above.

In addition to the information that TU Electric submitted to the staff in the three letters, the staff visited the site on February 19 and 26, 1993. Through these letters and visits, the staff gathered the information to update the material contained in SSER 26. Four of the issues are discussed separately below.

The fifth issue is not addressed below as TU Electric's commitment to perform the necessary changes is recorded in SSER 26.

36-INCH CABLE TRAY TEST

Background

In a letter of October 29, 1992, the staff stated that TU Electric's proposed acceptance criteria, as supplemented, were acceptable. In summary, the approved fire test acceptance criteria were

(1) External conduit, cable tray rail, and cable jacket temperatures should not exceed a rise of 250 °F (139 °K) plus ambient temperature (using thermocouple averaging), and no single thermocouple reading should exceed 30 percent above the specified average temperature rise.

(2) The fire barrier should not burn through or develop any openings through which either the test specimen raceway or cables were visible.

(3) If the temperature rise criteria were not satisfied, the cables should be inspected for visible damage. The following attributes constitute cable damage: jacket swelling, splitting, discoloration, hardening, blistering,

cracking, or melting; conductor insulation exposed, degraded, or discolored; shield exposed; or bare copper conductor exposed.

(4) If the fire barrier burned through during the fire exposure, or if a visual cable inspection revealed any of the damage attributes listed above, then the barrier was considered to have deviated from the acceptance criteria. Use of the fire test results to qualify a deviating fire barrier would require that cable functionality be demonstrated. Cable functionality test methodology and criteria were specified in the staff's October 29, 1992, letter.

In a letter of October 29, 1992, the staff concluded that TU Electric's acceptance criteria, as supplemented by the conditions stated in the October 29, 1992, letter, ensured that adequate cable and barrier tests would be performed and that satisfactory results from these tests would constitute an acceptable basis for qualifying the CPSES Unit 2 fire barriers.

In a letter of February 1, 1993 (TU Electric letter TXX-93076 to NRC), TU Electric committed to either perform a confirmatory test of a 36-inch cable tray, participate in an industry testing program to resolve concerns over a 36-inch-wide barrier, or submit additional information that adequately addresses the staff's concerns. TU Electric committed to perform one of these actions by the end of the first refueling outage for Unit 2.

Update

In a letter of February 26, 1993 (TU Electric letter TXX-93101 to NRC), TU Electric committed to perform the confirmatory fire endurance test. A 36-inch cable tray "straight run" configuration was constructed using licensee- proposed upgrades for the Unit 1 plant Thermo-Lag fire barrier configurations (stress skin reinforcement on joints instead of stress skin and stitching as used on Unit 2). The test configuration was built with the application of Thermo-Lag topcoat material approximately 72 hours following completion of the raceway envelope. The fire test was conducted four days after the topcoat was applied. Circuit integrity was not monitored during the test.

The 36" x 4" ladderback cable tray (straight run with 90 degree sweeping bends) was tested on March 4, 1993; the staff observed the test. The cable tray was protected with 1/2" (nominal) Thermo-Lag panels with longitudinal, vertical, and bottom joints reinforced with stress skin and trowel-grade material. TU Electric summarized the test data in a letter of March 10, 1993 (TU Electric letter TXX-93125 to NRC). Temperatures were below the acceptance criteria (which allows a 250 °F rise above ambient). The proper conduct of the fog hose stream test was observed. The hose stream test did not damage the barrier; no fire barrier burn-through was noted. Post-fire cable visual inspections were satisfactory. There were no signs of cable damage.

Texas Utilities Fire Barrier Testing for Comanche Peak Unit 2
(Conducted March 4, 1993, at Omega Point Laboratories)

Thermocouple locations	Average temperatures in °F (Ambient - 68 °F)	Maximum temperatures in °F (Ambient - 68 °F)
Power cable	241	277
Control cable	210	224
Instrument cable	217	240
Front tray rail	244	285
Rear tray rail	247	292

Scheme 15-1 - 36" Cable Tray

These preliminary test results meet the acceptance criteria and are indicative of a satisfactory test, subject to staff review of the final fire test report. This test was conducted in an identical method as the previous upgraded testing (documented in SSER 26), with the exception of a shorter material cure time and the absence of circuit integrity measurements. The cure time difference will be discussed below. The circuit integrity measurements were not taken for this test since TU Electric considered them unnecessary. The staff does not consider circuit integrity measurements an adequate test of cable functionality, and has determined that post-fire megger testing (described in the acceptance criteria as appropriate cable functionality testing) should be conducted as soon as possible following the test. Therefore, TU Electric's minor change to their test methodology (not performing circuit integrity measurements) is acceptable.

This 36" wide cable tray test was performed by TU Electric to satisfy the SSER 26 commitment regarding a confirmatory test of the widest cable tray. The staff will review the final test report when it becomes available and document the results of its review in a safety evaluation report. Staff actions will be tracked by NRC TAC No. M85998.

AMPACITY DERATING

Background

Cables enclosed in electrical raceways protected with fire barrier materials are derated because of the insulating effect of the fire barrier material. Other factors that affect ampacity derating include cable fill, cable loading, cable type, raceway construction, and ambient temperature. The National Electrical Code, Insulated Cable Engineers Association (ICEA) publications, and other industry standards provide general ampacity derating factors for open air installations, but do not include derating factors for fire barrier systems. Historically, ampacity derating factors for raceways enclosed with fire barrier material have been determined for specific installation configurations by

testing. In SSER 26, the staff discussed its concerns with inconsistent ampacity derating test data, but recognized that the ampacity derating concern is an aging issue rather than an immediate operability issue. In SSER 26, the staff (1) documented TU Electric's commitment to complete plant-specific ampacity derating testing by the end of the first refueling outage and (2) concluded that the use of TU Electric's interim ampacity derating factors is acceptable.

Update

After SSER 26 was issued, TU Electric conducted a series of ampacity derating tests for Thermo-Lag fire barrier configurations at Omega Point Laboratories (OPL) in San Antonio, Texas from March 3 through March 13, 1993. The NRC staff observed test preparation and testing from March 2 to 7, 1993. The first test group, conducted from March 2, 1993 to March 3, 1993, consisted of a 3/4"- diameter conduit with a single 3/C #10 AWG 600-volt copper cable and a 2"- diameter conduit with a single 3/C #6 AWG 600-volt copper cable. The second test group, conducted from March 5 to March 8, 1993, consisted of a 24" x 4" cable tray filled to a 2.95-inch depth with 3/C #6 AWG 600-volt copper cables and a free air drop (small) made of a single 3/C #6 AWG 600-volt copper cable. The final test group, conducted from March 10 to 14, 1993, consisted of a 5"- diameter conduit with four 1/C 750MCM 600-volt copper cable and a free air drop (large) made of three 1/C 750MCM 600-volt copper cable. The ampacity derating factor test results are summarized below.

The TU Electric ampacity derating test methodology followed the guidance detailed in the proposed standard IEEE-P848, "Procedure for the Determination of the Ampacity Derating of Fire Protected Cables," Draft 11, dated April 6, 1992, except for the following changes described further in TU Electric's ampacity test plan, revision 3, dated March 3, 1993:

(1) Conduit/air drop test articles were selected to be consistent with CPSES installation including the enhanced Thermo-Lag configurations.

(2) Test articles were supported by wood blocks during the performance of the tests.

(3) Type T special accuracy thermocouples were used for the conduit/air drop test articles and for all ambient temperature measurements. Type K thermocouples were used for tray configurations, with directions to make adjustments, if necessary, for the difference in accuracy.

(4) Baseline tests may be run before or after the ampacity derating test.

(5) Three thermocouples were installed at each location for the conduit/air drop test articles.

(6) Both the baseline and ampacity derating test shall utilize measured current normalized as outlined in ICEA P-46-426 for final conductor and ambient temperatures (that were not 90 °C and 40 °C, respectively).

[Note: By letter of March 23, 1993 (TU Electric letter TXX-93136 to NRC), TU Electric referenced Revision 4 of their ampacity test plan. The staff's review of this latest revision will be included in the staff's review of the final test reports, as discussed below].

In addition, the subject test plan supplemented elements of the Draft IEEE-P848 document in the following manner:

- Use a clamp-on ammeter with an accuracy of ±1 percent to take the final current measurements.
- Base the data interpretation of the ampacity derating factor on the measured values irrespective of the published ICEA values in accordance with the TU Electric letter of February 26, 1993 (TU Electric letter TXX-93101 to NRC).

The ampacity derating test procedure used for all test articles was performed in two steps, as follows:

(1) An ampacity product (or derating) test was conducted with the Thermo-Lag material configured around the test article.

(2) Then the baseline test was conducted on the same instrumented article without the Thermo-Lag product.

Each ampacity test was performed by raising the conductor temperature from ambient (i.e., 40 °C) to its rated temperature limit (i.e., 90 °C), allowing the test article to reach thermal equilibrium, and then measuring the final current or ampacity value for the test article. The ampacity derating factor was calculated as follows:

$$\text{Ampacity derating factor} = 1 - \frac{I_f}{I_o}$$

where:

I_f = ampacity value for product test

I_o = ampacity value for baseline test

TU Electric performed a series of calculations to establish the existing design margin for cable ampacity derating. These calculations were performed for the cables fed from the various switchgear, as follows:

Calculation	Cables	Calculated excess ampacity margin
#EE-CA-0008-3097	From 6.9 kV	Cable tray - 40% Conduit - 40%
#EE-CA-0008-3038	From 480 V	Cable tray - 38% Conduit - 23%
#2-EE-053	All other	Cable tray - 40% Conduit - 35%

Calculation	Cables	Calculated excess ampacity margin
#16345-EE(B)-140	Air drops	Cable tray - 39% Conduit - 35%

TU Electric letters of March 10, 1993 (TU Electric letter TXX-93125 to NRC) and March 23, 1993 (TU Electric letter TXX-93136 to NRC), supplied preliminary information regarding both TU Electric's calculated excess ampacity margin and the test result data for the plant-specific ampacity derating tests. Based on its testing, TU Electric is revising its design basis document to reflect the following derate factors: 11% for cables in conduits; and, 32% for cables in trays and air drops. The following table summarizes the preliminary test data, and provides the ampacity derate margin based on the effects of the fire barrier (calculated excess ampacity margin minus the actual test data):

Raceway	Ampacity derate test value	Excess ampacity derate margin
3/4" conduit	9.1%	25.9%
2" conduit	6.5%	28.5%
5" conduit	10.7%	12.3%
24" cable tray	31.4%	6.6%
Small air drop	23.0%	12.0%
Large air drop	31.7%	3.3%

The NRC staff finds that the preliminary ampacity test results provided by TU Electric are acceptable since the test derate factor data are bounded by the calculated (design) ampacity margins. However, the NRC staff is still reviewing TU Electric's plant-specific ampacity derating program. The NRC staff will complete its review of the plant-specific test program and results after TU Electric's submits the final test reports (consistent with the schedule published in SSER 26). Staff actions can be tracked under NRC TAC No. M85999.

"BOX ENCLOSURE" UPGRADES

Background

In a letter of January 19, 1992 (TU Electric letter TXX-93038 to NRC), TU Electric submitted engineering report ER-ME-082, "Evaluation of Unit 2 Thermo-Lag Configurations," to the staff for review in order to (1) establish the design basis for the Thermo-Lag fire barriers installed at CPSES Unit 2 that were configured differently from the tested configurations and (2) provide reasonable assurance that these Thermo-Lag fire barrier configurations will provide sufficient fire resistance to ensure that at least one train of safe shutdown systems will remain free of fire damage.

TU Electric's fire testing program established the technical and installation attributes for most of the Thermo-Lag fire barrier configurations installed at CPSES Unit 2. TU Electric documented about 180 cases in which the application of Thermo-Lag fire barrier materials used to protect electrical raceways and structural steel varied from the tested configurations. The staff recognized that there are actual field conditions that cause the application of fire

barrier assemblies to differ from the tested configurations. These cases may require the creation of a unique fire barrier design to address structural steel, other raceways, or mechanical equipment interferences. The staff also recognized that it was not feasible to qualify all aspects of the in-plant fire barriers through configuration-specific fire endurance testing. In Generic Letter 86-10, the staff provided guidance for performing engineering evaluations of raceway fire barrier systems that differed from the tested configurations. TU Electric used this guidance to establish its fire barrier evaluation criteria for configurations that differed from the tested configurations.

The following summarizes TU Electric's criteria: the continuity of the fire barrier material applied was consistent with the tested configuration; the effective thickness of the fire barrier material applied to the unique configuration was consistent with the thickness of the fire barrier material that was tested; the nature and effectiveness of the fire barrier support assembly was consistent with the tested configurations; and the application and end use of the fire barrier material was consistent with the tested configuration. In its engineering report, TU Electric evaluated the following: unique fire barrier configurations, minor protected commodity deviations, protruding and interfering item coverage deviations, and structural steel deviations.

In reviewing TU Electric's engineering report, the staff sampled those unique configurations where the fire barrier installations on safe shutdown raceways were constructed differently from those raceway fire barrier configurations tested by TU Electric's fire test program. The staff reviewed the engineering report and selected approximately 27 configurations for onsite review. The sample represented typical and unique configurations that varied from the tested configurations.

In SSER 26, the staff documented specific reviews of six representative configurations from this sample; for three of these, the staff requested additional actions. Configurations 1 and 3 represented "box-type" configurations, which the staff determined were not adequately justified in the engineering report. Specifically, two layers of Thermo-Lag material had been used for the qualification testing of junction box barriers, and the staff determined that designs similar to Configurations 1 and 3 would be more appropriately bounded by that type of construction. The staff considered Configuration 2, consisting of two parallel horizontal cable trays (18 and 12-inches wide), acceptable subject to the confirmatory resolution of staff concerns regarding the 36-inch wide cable tray fire barrier.

Update

As discussed, the preliminary results of the 36-inch cable tray fire test appear to have been satisfactory. Subject to staff review of the final fire test report, this confirmatory test satisfies staff concerns regarding the appropriate testing of the widest span cable trays. On the basis of the preliminary test results, the staff has reasonable assurance that Configuration 2 is acceptable. Any questions arising from the staff's review of the final test report for the 36-inch cable tray will be tracked by NRC TAC No. M85998.

Regarding the "box-type" configurations (Configurations 1 and 3), in letters of February 26, 1993 (TU Electric letter TXX-93101 to NRC), March 10, 1993 (TU Electric letter TXX-93125 to NRC), and March 23, 1993, (TU Electric letter TXX-93136 to NRC), TU Electric discussed the upgrades and documented their completion. TU Electric verified that the staff's concern with "box-type" configurations was limited to 13 plant configurations. In a letter of February 26, 1993, (TU Electric letter TXX-93101 to NRC), TU Electric committed to either increase the material thickness or rework the configurations in accordance with designs bounded by the previous fire barrier testing.

The NRC staff performed "walkdowns" of the subject configurations on February 19 and 26, 1993. On February 19, 1993, various elevations in the auxiliary building were walked down to ensure that TU Electric had properly selected the configurations for upgrade. No additional examples of cable tray box-type enclosures which would necessitate an additional layer of Thermo-Lag were identified. On February 26, 1993, walkdowns of all 13 upgrades were conducted while work was in progress. TU Electric "Minor Modification Forms" 93-123 through 93-126 were reviewed; these documented the upgrades by building elevation - 810' auxiliary, 832' auxiliary, 810' safeguards, and 790' auxiliary, respectively. The minor modification forms were verified to include engineering-basis discussions addressing the acceptability of the upgrades in regard to ampacity derating and the added weight on the supports. TU Electric redesigned one of the 13 upgrades instead of adding an additional layer of Thermo-Lag material. The previous box design had covered an airdrop from two cable trays to through-wall sleeves. The redesign incorporated three layers of flexi-blanket Thermo-Lag material covering the air drops, and the installation of an elastomer fire stop material. The redesign is appropriately bounded by laboratory-tested airdrop configuration, Scheme 11-1.

In a letter of March 10, 1993 (TU Electric letter TXX-93125 to NRC), TU Electric certified the completion of these upgrades. The NRC staff concludes from its review of the engineering documents and walkdown of the specific configurations that the barriers are properly bounded by acceptable test schemes and are, therefore, acceptable.

COMPENSATORY MEASURES

Background

In a letter to the staff of October 5, 1991, the vendor stated that Thermo-Lag trowel-grade material takes about 30 days to reach its optimum properties. In a letter of January 19, 1993 (TU Electric letter TXX-93038 to NRC), TU Electric stated that it considered its Thermo-Lag fire barriers to be functional (capable of performing their design function) immediately after completion of the barrier installation and inspection. In a letter of January 25, 1993 (TU Electric letter TXX-93060 to NRC), TU Electric submitted additional information regarding cure time, stating that its vendor concurred with TU Electric's recommendation on cure time.

TU Electric cured its fire test specimens for at least 30 days preceding the conduct of the fire endurance tests. The staff was concerned that Thermo-Lag fire barriers are not functional until they are either cured for 30 days in accordance with the vendor's original recommendation or until the installed barriers reflect the tested conditions.

In a letter of January 28, 1993 (TU Electric letter TXX-93061 to NRC) TU Electric committed to provide fire watches as compensatory measures in accordance with the CPSES fire protection plan for the Thermo-Lag fire barriers installed in areas that contain fire-safe shutdown conduits or cable trays until the barriers have cured for 30 days, and where box enclosures are located, until this issue is adequately resolved with the staff.

The use of fire watches is consistent with the compensatory measures implemented by TU Electric for the CPSES Unit 1 Thermo-Lag fire barriers in response to NRC Bulletin 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage," June 24, 1992. The staff concluded in SSER 26, therefore, that TU Electric's commitment was acceptable and ensures that an adequate level of fire protection is provided at CPSES Unit 2 until the Thermo-Lag fire barriers are cured (1) to reflect the condition of the fire test specimens and (2) the box enclosure issue is resolved.

Update

On March 4, 1993, TU Electric tested a 36-inch cable tray with a seven-day cure time (topcoat was applied after the assembly had cured for three days; four days later the fire test was conducted). The utility performed this test for two reasons: to satisfy their commitment to perform a confirmatory cable tray test bounding their widest tray, and to perform a test with a shorter cure time in order to demonstrate TU Electric's position that the configurations can be considered operable less than 30 days after completion of the installation and inspection.

The 36-inch cable tray passed its confirmatory test, demonstrating the operability of Comanche Peak fire barrier designs with a seven-day cure time. Accordingly, TU Electric informed the staff that compensatory measures such as fire watches are no longer required for configurations that have exceeded a seven-day cure time.

The staff considers TU Electric's position on cure time acceptable based on preliminary results of the 36-inch-wide cable tray fire test. A majority of the fire tests were conducted with a 30-day cure time in order to ensure that the moisture content in the Thermo-Lag material had reached equilibrium, thereby providing conservative qualification fire test results. Additionally, a test was conducted with a shorter cure time, and the fire barrier did not exhibit any seam separation.

On the basis of its findings from the fire barrier testing program, TU Electric has demonstrated that its installed configurations are bounded by test results; therefore, the staff concludes that compensatory measures are not required for fire barrier installations that exceed a seven-day cure time.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

13.1.1 Management and Technical Resources

In SSER 26 the staff recorded TU Electric's commitment to submit organizational changes resulting from the change to two-unit operation in a future FSAR amendment. In a letter of February 26, 1993 (TU Electric letter TXX-93102 to NRC), TU Electric submitted an advanced change to the FSAR to revise the TU Electric corporate structure. The organization was revised to include the following divisions: Operations, Production, Bulk Power and Technical Support, and Finance and Corporate Support. The Production Division retained corporate responsibility for the design, construction, and operation of CPSES.

Within the Production Division, the nuclear group, redesignated the Nuclear Production Group (formerly the Nuclear Engineering and Operations Group), provides the design, engineering, construction, licensing, operation, and fuel management support for CPSES. The Nuclear Production Group was reorganized into four organizations to better focus resources on operation of the dual-unit CPSES facility. These four organizations are Nuclear Operations, Nuclear Engineering and Support, Nuclear Overview, and Regulatory Affairs.

All previously assigned responsibilities and duties have been reassigned to appropriate positions within the new management structure. Positions associated with construction activities have been deleted as part of the transition from Unit 2 construction completion to dual-unit operation. The description of the responsibilities for the Manager, Administrative Services has been removed because this position does not perform a safety function and, therefore, does not need to be in the FSAR. The staff finds this acceptable. The new organizational structure is shown in revised FSAR Figure 13.1-2, included as an attachment to the February 26, 1993, letter.

The changes to the corporate organization made by the licensee primarily reflect an organizational restructuring to focus resources on dual-unit operation of the CPSES facility. The new lines of management authority and communication have been clearly defined. Other changes made to the corporate organization reflect changes in name, not in function. Therefore, they do not change the staff's previous conclusion that the corporate level management structure is acceptable.

The staff concludes that the revised organization continues to meet the acceptance criteria of Section 13.1.1 of the Standard Review Plan (SRP) (NUREG-0800) for appropriate lines of authority, and is, therefore, acceptable.

13.1.2 Operating Organization

In SSER 26 the staff recorded TU Electric's commitment to submit organizational changes resulting from the change to two-unit operation in a future FSAR amendment. In a letter of February 26, 1993 (TU Electric letter TXX-93102 to NRC), TU Electric submitted an advanced change to the FSAR to revise the TU Electric operating organization structure. The Nuclear Operations organization, under the direction of the Vice President of Nuclear Operations, is responsible for plant operations and operating support. The Vice President of Nuclear Operations has assumed the duties of the Plant Manager and is responsible for the operation and maintenance of CPSES. Reporting to the Vice President of Nuclear Operations are the Manager, Operations; Manager, Maintenance; Manager, Plant Support; Manager, Work Control; and the Radiation Protection Manager. The Manager, Nuclear Training has been reassigned from the Nuclear Operations organization and now reports to the Director of Nuclear Overview.

All previously assigned responsibilities and duties have been reassigned to appropriate positions within the new operating organization. The Chemistry Manager (formerly the Chemistry and Environmental Manager) now reports to the Manager, Operations. Environmental responsibilities have been reassigned under the Manager of Design/Support Engineering, in the Nuclear Engineering and Support organization. The new organizational structure is shown in revised FSAR Figure 13.1-3, included as an attachment to the February 26, 1993, letter.

The changes to the operating organization made by the licensee primarily reflect an organizational restructuring to focus resources on dual-unit operation of the CPSES facility. The new lines of management authority and communication have been clearly defined. These changes do not affect the staff's previous conclusion that the operations organization is acceptable.

The staff concludes that the revised organization continues to meet the acceptance criteria of Section 13.1.2 of the SRP for appropriate lines of authority, and is, therefore, acceptable.

13.3 Emergency Planning

The Federal Emergency Management Agency (FEMA) evaluated the offsite radiological emergency response plans site-specific to Comanche Peak during an exercise conducted on November 19, 1991, and a remedial drill conducted on February 6, 1992. In a letter of June 24, 1992, FEMA stated that on the basis of these evaluations, the offsite radiological emergency response plans and preparedness site-specific to Comanche Peak are adequate to give reasonable assurance that appropriate measures can be taken offsite to protect the health and safety of the public in the event of a radiological emergency at the site. Before issuing low-power and full-power licenses, the NRC confirmed with FEMA that there were no offsite emergency preparedness issues that would potentially affect startup of CPSES Unit 2.

The NRC conducted a special inspection on May 18-21, 1992 of TU Electric's emergency preparedness program as it related to the licensing of Unit 2. No areas were identified that would preclude the licensing of Unit 2. Additionally, TU Electric successfully passed the last annual exercise evaluated

by the NRC (conducted on November 18, 1992). TU Electric has responded with a corrective action plan to three onsite exercise weaknesses identified during this inspection. These issues are being tracked by the NRC's Region IV staff and will be evaluated during a future inspection.

The staff concludes that the overall state of emergency preparedness at Comanche Peak is adequate to support dual-unit operations.

14 INITIAL TEST PROGRAM

Pre-operational Test Deferral

In SSER 26, the staff documented its review of TU Electric's preoperational test program changes for Unit 2 (TU Electric letters of December 23, 1992, TXX-92586 to NRC; January 8, 1993, TXX-93011 to NRC; and January 25, 1993, TXX-93051 to NRC). TU Electric proposed to defer certain preoperational tests until after fuel load. The staff verified that TU Electric's letters contained commitments for completing the tests at the appropriate plant power levels or plant milestones. The staff determined that the schedule for performing the deferred testing ensured that systems required to prevent, limit, or mitigate the consequences of postulated accidents would be tested before the systems would be required to be operable and ensured that the safety of the plant would not be dependent on the performance of untested systems, structures, and components. Therefore, the staff considered that TU Electric's justification for deferred testing and its subsequent schedule for conducting the tests was acceptable.

In a letter of March 22, 1993 (TU Electric letter TXX-93140 to NRC), TU submitted additional information regarding the status of several deferred preoperational tests which had been reviewed by the staff before the low-power license was issued. The additional information contained updated test methodology, results, schedules and deletions regarding plant computer, plant communication system, pressurizer spray valve, and steam dump valve testing. The pressurizer spray valve re-test was completed; however, a maintenance item is being tracked to correct a slightly higher valve leak-by rate. The steam dump valves will be retested "hot," but with the downstream block valves closed (similar to Unit 1 preoperational testing). Additionally, the schedule for testing the availability of the safety parameter display system and submitting the test report was clarified.

The staff reviewed the additional information and determined that the conclusions reached in SSER 26 are still valid; that is, the systems will be adequately tested before they will be required to be operable, and the safety of the plant will not be dependent on the performance of untested systems, structures, and components. Therefore, TU Electric's additional information, including test methodology and scheduler changes, regarding the deferred testing is acceptable.

16 TECHNICAL SPECIFICATIONS

The NRC issued the "Final Draft Combined Technical Specifications for Comanche Peak Unit 1 and Unit 2" to TU Electric on September 9, 1992. TU Electric certified on November 4, 1992 (TU Electric letter TXX-92536 to NRC), that the final draft accurately reflects the as-built plant and the Final Safety Analysis Report. TU Electric also noted certain minor corrections. The staff discussed the corrections with TU Electric and appropriate changes were made to the Final Draft Technical Specifications (TS). The staff issued the Final Draft TS to TU Electric in a letter of January 22, 1993. Editorial corrections were discussed and TU Electric recertified the TS in a letter of January 30, 1993 (TU Electric letter TXX-93001 to NRC). The "Combined Comanche Peak Unit 1 and 2 Technical Specifications" were included as Appendix A to the low-power license issued on February 2, 1993. The same TS were reissued with the full-power license.

17 QUALITY ASSURANCE

The staff reviewed TU Electric's operations phase quality assurance (QA) program organization in SSER 22. In a letter dated February 26, 1993 (TU Electric letter TXX-93102 to NRC), TU Electric submitted an advance FSAR change to update the organizational structure, as discussed in Section 13.1 of this SER supplement. These changes resulted in some revisions to the description of the QA program organization described in Section 17.2 of the FSAR. The staff's reevaluation of the licensee's revised QA organization is presented below.

17.2 Organization of the QA Program

The Group Vice President, Nuclear Production, is responsible for the overall management and operation of CPSES, including the establishment of company nuclear policies. The Group Vice President has the overall responsibility for establishing and executing the CPSES QA program for operations. The Group Vice President has assigned to the Vice President of Nuclear Engineering and Support the overall responsibility for engineering and support of CPSES, and for implementation of the QA program for the nuclear engineering and support function at CPSES. The Group Vice President has assigned to the Vice President of Nuclear Operations the overall responsibility for operating CPSES and for implementing the QA program for operations at CPSES.

The Vice President of Nuclear Operations is responsible to the Group Vice President, Nuclear Production for operating activities at CPSES. Duties and responsibilities of the Vice President of Nuclear Operations include technical and administrative direction of the Manager, Operations; the Manager, Maintenance; the Radiation Protection Manager; the Manager, Work Control; the Manager, Plant Support; and the technical and administrative direction for implementing QA controls at nuclear plants operated by the licensee.

The Vice President of Nuclear Engineering and Support is responsible to the Group Vice President, Nuclear Production for providing engineering related technical services in support of CPSES operations. Duties and responsibilities of the Vice President of Nuclear Engineering and Support include technical support to the nuclear operations organization, and assistance in the procurement of equipment, materials, and services for the operation, maintenance, and modification of CPSES.

The Director of Nuclear Overview reports directly to the Group Vice President, Nuclear Production and is responsible to the Group Vice President for ensuring effective implementation of the QA program. This reporting relationship ensures that the Director of Nuclear Overview has sufficient authority, organizational freedom, and independence from undue influence from, or responsibility for, costs and schedules to effectively ensure implementation of and compliance with the CPSES operations QA requirements and controls.

The Director of Nuclear Overview communicates directly with the Nuclear Production Group supervisory and management personnel and with appropriate management levels in consultant and contractor QA organizations to identify quality problems; initiate, recommend, or provide solutions; and to verify implementation of solutions to quality problems. The Director has the authority to "stop work" during the operations phase. Specific duties of the Director of Nuclear Overview include the direction of Nuclear Overview Department personnel; technical and administrative direction of the Manager, Operations Quality Control; Manager, QA; Manager, Independent Safety Engineering Group; Manager, Plant Analysis; and Manager, Nuclear Training; verification that procedures for the control of quality-related activities comply with QA requirements; verification of the implementation of the QA program within the Nuclear Production Group; verification that consultants, contractors, and suppliers providing quality-related items or services have established and implemented an adequate QA program; and membership or representation on the Operations Review Committee.

The Nuclear Overview Department, under the Director of Nuclear Overview, functions to ensure effective implementation of the QA program. The department performs internal and external audits, surveillances, and inspections. The audits, surveillances, and inspections are performed by qualified individuals other than those who performed or directly supervised the work.

On the basis of its review and evaluation, the staff concludes that the applicant's QA organization has (a) sufficient independence from cost and schedule, (b) authority to effectively carry out the operations QA program, and (c) access to management at a level necessary to perform the QA functions. The staff concludes that the applicant's description of the QA organization is in compliance with applicable NRC regulations and is acceptable for the operation of CPSES.

22 TMI-2 REQUIREMENTS

After the accident at Three Mile Island (TMI) Unit 2, the NRC staff developed NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," to provide a comprehensive and integrated plan to improve safety at power reactors. The Commission approved specific items from NUREG-0660 for implementation at reactors. NUREG-0737, "Clarification of TMI Action Plan Requirements," was issued in November 1980; this document included items approved by the Commission and additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions.

In Chapter 22 of the SER and its supplements, the staff discussed TMI issues relative to CPSES. In the table of TMI action plan issues, included in Appendix C of this SSER, the staff summarizes each TMI item, including the SER (or supplement) that documents issue resolution and the date of inspection verification and associated inspection report number (if applicable). The staff adhered to the TMI action plan numbering scheme in SSER 24 (except where items were consolidated for inspection activity performed in accordance with Revision 2 of Temporary Instruction (TI) 2515/065, "TMI Action Plan Requirement Follow-up").

APPENDIX A

CONTINUATION OF CHRONOLOGICAL LIST OF CORRESPONDENCE

This appendix continues the chronological listing of routine licensing correspondence, regarding Unit 2 and Unit 1/Unit 2 common issues, between the U.S. Nuclear Regulatory Commission (NRC) staff and the applicant (Texas Utilities Electric Company) since Supplement 26 was issued.

January 4, 1993	Summary of November 4, 1992, meeting with applicant regarding pressurizer surge line leak-before-break analysis.
January 6, 1993	Summary of December 17, 1992, meeting with applicant regarding fire protection issues.
January 11, 1993	Letter to applicant transmitting environmental assessment for exemption from 10 CFR 70.24.
January 11, 1993	Letter to applicant transmitting environmental assessment for exemption from 10 CFR Part 50, Appendix J, Section III.D.2(b)(ii).
January 18, 1993	Letter from applicant transmitting information regarding augmented inservice testing for CVCS valves.
January 19, 1993	Letter from applicant transmitting response to Generic Letter 92-08.
January 19, 1993	Letter to applicant transmitting safety evaluation regarding topical report RXE-1-002, "Reactivity Anomaly Events Methodology".
January 20, 1993	Letter from applicant transmitting final response for Unit 2 to NRC Bulletin 88-08.
January 21, 1993	Letter from applicant transmitting information regarding ASME IST and Inservice Test Program relief request.
January 22, 1993	Letter to applicant transmitting final draft version of combined technical specifications.
January 25, 1993	Letter from applicant forwarding information regarding Thermo-Lag testing data and engineering evaluations.
January 25, 1993	Letter from applicant forwarding information regarding scheduled completion of primary plant ventilation system and plant computer testing.

January 28, 1993	Letter from applicant forwarding supplemental response to Bulletin 88-08.
January 28, 1993	Letter from applicant forwarding clarifying information regarding test scheme 1, conduit support modifications, and use of test scheme 9 results.
January 29, 1993	Letter from applicant transmitting interim change request to preservice program plan.
January 29, 1993	Letter from applicant transmitting information regarding HVAC design validation.
January 29, 1993	Letter to applicant transmitting significant findings of the Operational Readiness Assessment Team Inspection.
February 1, 1993	Letter from applicant transmitting response to concerns regarding turnover process, fire seals for piping penetrations and containment spray system nozzle completion.
February 2, 1993	Letter to applicant transmitting Facility Operating License No. NPF-88 for Comanche Peak Unit 2.
February 2, 1993	Memo to File regarding request for stay of issuance of the low power operating license.
February 3, 1993	Board Notification 93-01 regarding new information regarding Comanche Peak Unit 2.
February 3, 1993	Letter from licensee forwarding Revision 11 to Technical Requirements Manual.
February 4, 1993	Letter from licensee forwarding Revision 1 to IST plan for pumps and valves first interval.
February 5, 1993	Letter to licensee transmitting correction to Appendix B of Facility Operating License No. NPF-88.
February 9, 1993	Letter to licensee transmitting correction to Indemnity Agreement No. B-96.
February 18, 1993	Letter from licensee forwarding overview of self-assessment plans for power operation above 5 percent and above 50 percent.
February 19, 1993	Letter from licensee forwarding results of engineering review of plant record to address issues in NRC Bulletin 90-01.
February 22, 1993	Letter to licensee forwarding NUREG-1275, Volume 8, "Operating Experience Feedback Report - Human Performance in Operating Events."

February 24, 1993	Letter from licensee forwarding summary of personnel monitoring ending December 31, 1992.
February 25, 1993	Letter from licensee forwarding documentation of discussions with NRC regarding planned method of treating DNB penalties.
February 25, 1993	Letter from licensee forwarding documentation of sensitivity study performed to evaluate effect of variations in core noding on calculated peak cladding temperature.
February 26, 1993	Letter from licensee forwarding revisions to Sections 13.1.17.1 and 17.2 to FSAR reflecting organizational changes.
February 26, 1993	Letter from licensee forwarding clarification on ampacity derating test and Thermo-Lag fire endurance test.
March 2, 1993	Summary of January 21, 1993, meeting concerning Comanche Peak Steam Electric Station fire protection issues.
March 9, 1993	Letter to licensee forwarding operation readiness assessment team inspection report.
March 10, 1993	Letter from licensee forwarding an updated status of open issues in Section 9.5 of SSER 26 regarding preliminary fire endurance and ampacity test results.
March 10, 1993	Letter to licensee forwarding clarification of staff safety evaluation on Topical Report RXE-91-002, "Reactivity Anomaly Events Methodology."
March 11, 1993	Letter from licensee forwarding RXE-93-003, CPSES Unit 2 Cycle 1 Core Operating Limits Report.
March 11, 1993	Letter to licensee forwarding "Toxicological Evaluation of the Combustion Products from a Thermal Barrier Material Decomposed under Flaming and Nonflaming Conditions."
March 17, 1993	Letter from licensee forwarding supplemental information to include Unit 2 in license amendment requests 92-05, 92-06, 92-07, and 92-08.
March 22, 1993	Letter from licensee describing approach for analysis of large break LOCA with mixed cores delineated in Topical Report RXE-90-007.
March 23, 1993	Letter from licensee submitting results of ampacity testing of upgraded Thermo-Lag installations and provides information on box configurations.

March 28, 1993	Letter from licensee transmitting certification for readiness for full power operating license.
March 31, 1993	Letter to licensee transmitting documents filed by the staff with the Commission relating to Comanche Peak Unit 2 full-power licensing.

APPENDIX B

BIBLIOGRAPHY

Miscellaneous

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C., 1991.

National Electrical Code, Insulated Cable Engineers Association (ICEA) publications.

IEEE-P848, "Procedure for the Determination of the Ampacity Derating of Fire Protected Cables," Draft 11, April 6, 1992.

NRC Bulletins

Bulletin 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free From Fire Damage," June 24, 1992.

NRC Generic Letters

Generic Letter 86-10, "Implementation of Fire Protection Requirements," April 28, 1986.

NRC Letters

See Appendix A.

NRC NUREG-Series Reports

NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980.

NUREG-0737, "Clarification of TMI Action Plan Requirements," October 1980.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

APPENDIX C

TMI ACTION PLAN ISSUES

TMI ACTION PLAN ISSUES

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
I.A.1.1.1	Shift Technical Advisor; On Duty	SSER 1 & 23 [11/14/90; 50-446/90-40]
I.A.1.1.2	Shift Technical Advisor	SSERs 1 & 23 & TS 6.2.4
I.A.1.1.3	Shift Technical Advisor; Training	SSERs 1 & 23 [11/14/90; 50-446/90-40]
I.A.1.I.4	Shift Technical Advisor; Long-Term Program	SSERs 1 & 23
I.A.1.2	Shift Supervisor; Administrative Duties	SSER 1 [11/14/90; 50-446/90-40]
I.A.1.3.1	Shift Manning; Overtime	SSER 1 & TS 6.2.2.f [11/14/90; 50-446/90-40]
I.A.1.3.2	Shift Manning; Minimum Shift Crew	SSER 1 & TS T6.2-1 [11/14/90; 50-446/90-40]
I.A.2.1.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications; SRO Experience	SER & letter dated 3/8/85
I.A.2.1.2	Immediate Upgrading of Operator and Senior Operator Training and Qualifications; Training	SER & letter dated 3/8/85
I.A.2.1.3	Immediate Upgrading of Operator and Senior Operator Training and Qualifications; Facility Certification and Fitness of Applicants	SER & letter dated 3/8/85
I.A.2.1.4	Immediate Upgrading of Operator and Senior Operator Training and Qualifications; Modify Training	SER & letter dated 3/8/85 [11/14/90; 50-446/90-40]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
I.A.2.1.5	Immediate Upgrading of Operator and Senior Operator Training and Qualifications; Facility Certification	SER & letter dated 3/8/85
I.A.2.3	Administration of Training Programs for Licensed Operators	SER & SSER 23
I.A.3.1.1	Revised Scope and Criteria for Licensing Examination - Increase Scope	SER
I.A.3.1.2	Revised Scope and Criteria for Licensing Examination - Increase Passing Grade	SER
I.A.3.1.3.A	Revised Scope and Criteria for Licensing Examination - With Simulators	SER
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Team Operating License Applicants	SSER 1
I.C.1.1	Procedures for Transients and Accidents, Short-Term; Small-Break LOCA	SSERs 6, 12, & 22 [3/11/93; 50-446/92-60]
I.C.1.2.A	Procedures for Transients and Accidents, Short-Term; Inadequate Core Cooling; Reanalyze Guidelines	SSERs 6, 12, & 22
I.C.1.2.B	Procedures for Transients and Accidents, Short-Term; Inadequate Core Cooling; Revise Procedures	SSERs 6, 12, & 22 [3/11/93; 50-446/92-60]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
I.C.1.3.A	Procedures for Transients and Accidents, Short-Term; Transients and Accidents; Reanalyze Guidelines	SSERs 6, 12, & 22
I.C.1.3.B	Procedures for Transients and Accidents; Short-Term; Transients and Accidents	SSERs 6, 12, & 22 [3/11/93; 50-446/92-60]
I.C.2	Shift Relief and Turnover Procedures	SSER 6 [11/14/90; 50-446/90-40]
I.C.3	Shift Supervisor Responsibilities	SSER 1 [11/14/90; 50-446/90-40]
I.C.4	Control Room Access	SSER 1 [11/14/90; 50-446/90-40]
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	SSERs 6, & 23 [11/14/90; 50-446/90-40]
I.C.6	Procedures for Verification of Current Performance of Operating Activities	SSER 1 [11/14/90; 50-446/90-40]
I.C.7.1	NSSS Vendor Review of Procedures; Low Power Test Program	SER & SSER 23 [11/14/.90; 50-446/90-40]
I.C.7.2	NSSS Vendor Review of Procedures; Low Power, Power Ascension and Emergency Procedures	SER & SSER 23 [11/14/90; 50-446/90-40]
I.C.8	Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants	SSER 6
I.D.1	Control Room Design Reviews	SSER 22 (Unit 1) SSER 26 (Unit 2)

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
I.D.2.1	Plant Safety Parameter Display Console; Description	SSER 22
I.D.2.2	Plant Safety Parameter Display Console; Installed	SSER 22 (Unit 1) SSER 26 (Unit 2) [3/11/93; 50-446/92-60]
I.D.2.3	Plant Safety Parameter Display Console; Fully Implemented	SSER 22 (Unit 1) SSER 26 (Unit 2) [3/11/93; 50-446/92-60, final verification tracked as insp. item]
I.G.1.1	Training During Low-Power Testing; Proposed Tests	SSER 6
I.G.1.2	Training During Low-Power Testing; Submit Analysis and Procedures	SSER 6
I.G.1.3	Training During Low-Power Testing; Training and Results	SSER 6
II.B.1.1	Reactor Coolant System Vents; Design	SSER 6
II.B.1.2	Reactor Coolant System Vents Install	SSER 6 [3/11/93; 50-446/92-60]
II.B.1.3	Reactor Coolant System Vents; Procedures	SSER 6 & TS 3.6.1.7 [3/11/93; 50-446/92-60]
II.B.2.1	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation; Design	SSERs 2 & 22
II.B.2.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation; Corrective Actions	SSERs 2 & 22 [Incorp. in II.B.2.3]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.B.2.3	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation; Modifications	SSERs 2 & 22 [8/22/91; 50-446/91-21] [1/27/93; 50-446/93-05]
II.B.2.4	Superseded by 10 CFR 50.49	
II.B.3.1	Post-Accident Sampling; Interim System	SSERs 6, 22, & 23 [11/14/90; 50-446/90-40]
II.B.3.2	Post-Accident Sampling; Corrective Actions	SSERs 6 & 22 [Incorp. in II.B.3.4]
II.B.3.3	Post-Accident Sampling; Procedures	SSERs 6 & 22 [3/11/93; 50-446/92-60]
II.B.3.4	Post-Accident Sampling; Plant Modifications	SSERs 6, 22, & 23 [3/11/93; 50-446/92-60]
II.B.4.1	Training for Mitigating Core Damage; Develop Training	SER & SSER 23
II.B.4.2.A	Training for Mitigating Core Damage; Initial	SER & SSER 23 [11/14/90; 50-446/90-40]
II.B.4.2.B	Training for Mitigating Core Damage; Complete	SER & SSER 23 [11/14/90; 50-446/90-40]
II.D.1.1	Relief and Safety Valve Testing Requirements; Submit	SSER 21 (Unit 1) SSER 26 (Unit 2)
II.D.1.2.B	Relief and Safety Valve Testing Requirements; Plant-Specific Report	SSER 21 (Unit 1) SSER 26 (Unit 2)
II.D.1.3	Relief and Safety Valve Testing Requirements; Block Valve Testing	SSER 21 (Unit 1) SSER 26 (Unit 2)
II.D.3.1	Valve Position Indication; Install Direct Indicators of Valve Position	SER [3/11/93; 50-446/92-60]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.D.3.2	Valve Position Indication; Technical Specifications	SER & TS 3.4.4 & 4.0.5
II.E.1.1.1	Auxiliary Feedwater System; Analysis	SER & SSER 21
II.E.1.1.2	Auxiliary Feedwater System Evaluation; Short-Term Modifications	SER & SSER 21 [3/11/93; 50-446/92-60, Item II.E.1.2]
II.E.1.1.3	Auxiliary Feedwater System Evaluation; Long-Term Modifications	SER & SSER 21 [3/11/93; 50-446/92-60, Item II.E.1.3]
II.E.1.2.1.A	Auxiliary Feedwater System Initiation and Flow; Control Grade	SSER 21 [4/24/89; 50-446/89-17]
II.E.1.2.1.B	Auxiliary Feedwater System Initiation and Flow; Safety Grade	SER [4/24/89; 50-446/89-17]
II.E.1.2.2.A	Auxiliary Feedwater System Flow Indication; Control Grade	SER [4/24/89; 50-446/89-17]
II.E.1.2.2.B	Auxiliary Feedwater System Flow Indication; LL Cat A Technical Specifications	SER & TS 3.7.1.2
II.E.1.2.2.C	Auxiliary Feedwater System Flow Indication; Safety Grade	SER [3/11/93; 50-446/92-60]
II.E.3.1.1	Emergency Power for Pressurizer Heaters; Upgrade Power Supply	SER & SSER 22 [3/11/93; 50-446/92-60]
II.E.3.1.2	Emergency Power for Pressurizer Heaters; Technical Specifications	SER & TS 3.4.3
II.E.4.2.1-4	Containment Isolation Dependability; Diverse Isolation	SSER 23 [4/24/89; 50-446/89-17]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.E.4.2.5.A	Containment Isolation Dependability; Containment Pressure Setpoint; Specify Pressure	SSER 22
II.E.4.2.5.B	Containment Isolation Dependability; Containment Pressure	SSER 22 [8/21/91; 50-446/91-46]
II.E.4.2.6	Containment Isolation Dependability; Containment Purge Valves	SSER 23 [12/31/92; 50-446/92-51]
II.E.4.2.7	Containment Isolation Dependability; Radiation Signal on Purge Valves	SSER 23 & TS 3.6.1.7 [4/24/89; 50-446/89-17]
II.E.4.2.8	Containment Isolation Dependability; Technical Specifications	SSER 23 & TS T-3.3-4.2
II.F.1.1	Accident Monitoring; Procedures	SSER 3 [Refer to II.F.1 items below]
II.F.1.2.a	Accident Monitoring; Noble Gas Monitor	SSER 3 & TS T-3.3-4.1.b [4/24/89; 50-446/89-17; (interim) [9/20/89; 50-446/89-67 (Long term); 3/11/93; 50-446/92-60]
II.F.1.2.b	Accident Monitoring Particulate Sampling	SSER 3 & TS T-3.3-4.1.a [5/18/89; 50-446/89-24 (long term); 2/1/93; 50-446/92-54]
II.F.1.2.c	Accident Monitoring; Containment High-Range Monitors	SSER 3 & TS T-3.3-6.10 [3/11/93; 50-446/92-60]
II.F.1.2.d	Accident Monitoring; Containment Pressure	SSER 3 & TS T-3.3-6.1 [3/11/93; 50-446/92-60]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.F.1.2.e	Accident Monitoring; Containment Water Level	SER & TS T-3.3-6.8 [3/11/93; 50-446/92-60]
II.F.1.2.f	Accident Monitoring; Containment Hydrogen	SER & TS T-3.3-7 & 3.6.4.1 [3/11/93; 50-446/92-60]
II.F.2.1	Instrumentation for Detection of Inadequate Core Cooling; Procedure	SSERs 6, 21, & 23 [Incorp. in II.F.2.2]
II.F.2.2	Instrumentation for Detection of Inadequate Core Cooling; Subcool Meter; Install	SSERs 6, 21, 23 [3/11/93; 50-446/92-60]
II.F.2.4	Instrumentation for Detection of Inadequate Core Cooling; Additional Instruments	SSER 6, 21, & 23 & TS T-3.3-6 [3/11/93; 50-446/92-60]
II.G.1.1	Power Supply for Pressurizer Relief Block Valves and Level Indication; Upgrade	SER [3/11/93; 50-446/92-60]
II.G.1.2	Power Supply for Pressurizer Relief Block Valves and Level Indication; Technical Specifications	SER & TS 3.4.4
II.K.1.5	Measures to Mitigate Small-Break LOCA and Loss-of-Feedwater Accidents; IE Bulletins; Review ESF Valves	SER
II.K.1.10	Measures to Mitigate Small-Break LOCA and Loss-of-Feedwater Accidents; IE Bulletins; Operability Status	SER
II.K.1.17	Measures to Mitigate Small-Break LOCA and Loss-of-Feedwater Accidents; IE Bulletins; Trip per Pressurizer Low Level	SER

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.K.2.13	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA with no AFW	SSER 6
II.K.2.17	Analysis of Potential Voiding in RCS During Anticipated Transients	SSER 6
II.K.3.1.A	Automatic PORV Isolation System; Design	SSER 6
II.K.3.1.B	Automatic PORV Isolation System; Test/Install	SSER 6 [5/18/89; 50-446/89-24]
II.K.3.10	Anticipatory Trip H: Power	SSER 25 [5/18/89; 50-446/89-24]
II.K.3.11	Justification for Use of Certain PORVs	SSER 6
II.K.3.12.A	Confirm Existence of Anticipatory Trip Upon Turbine Trip; Proposed Modifications	SER & SSER 22 [5/18/89; 50-446/89-24]
II.K.3.12.B	Confirm Existence of Anticipatory Trip Upon Turbine Trip; Modify	SER & TS T-3.3.1-16 [5/18/89; 50-446/89-24]
II.K.3.17	Report on Outage of ECCS	SER
II.K.3.2	Report on Overall Safety PORV Isolation System	SER
II.K.3.25.A	Effect of Loss of AC Power on Pump Seals; Proposed Modifications	SER
II.K.3.25.B	Effect of Loss of AC Power on Pump Seals; Modifications	SER [5/18/89; 50-446/89-24]

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
II.K.3.3	Report Safety and Relief Valve Failures Promptly and Challenges Annually	SER & TS 6.9.1.2 & 6.9.1.5
II.K.3.30.A	Schedule for Outline of Small-Break LOCA Model	SSERs 6 & 12
II.K.3.30.B	Small-Break LOCA Model; Justification	SSERs 6 & 12
II.K.3.30.C	Small-Break LOCA Model; New Analysis	SSERs 6 & 12
II.K.3.31	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	SSERs 6, 12, & 21
II.K.3.5.A	Automatic Trips of Reactor Coolant Pumps; Proposed Modifications	SSER 22
II.K.3.5.B	Automatic Trips of Reactor Coolant Pumps; Modifications	SSER 22 [9/6/91; 50-446/91-38]
II.K.3.9	Proportional Integral Derivative Controller Modification	SER & SSER 22 [12/31/92; 50-446/92-51]
III.A.1.1	Improve Emergency Preparedness	SSERs 6, 12, 22, & 24
III.A.1.2.1	Upgrade Emergency Support Facilities; Interim TSC, OSC, and EOF	SSERs 3 & 22 [11/14/90; 50-446/90-40]
III.A.1.2.2	Upgrade Emergency Support Facilities; Design (Superseded by MPAs F063, F064 and F065)	SSERs 3 & 22
III.A.1.2.3	Upgrade Emergency Support Facilities; Modifications (Superseded by MPAs F063, F064 and F065)	SSERs 3 & 22 [11/14/90; 50-446/90-40]

TMI ACTION PLAN ISSUES (Continued)

ITEM		SER/SSER resolved [Verif. rpt. no., If Applicable]
III.A.2.1	Upgrade Preparedness; Emergency Plans	SSERs 3 & 6 (App. G., Sec. 4)
III.A.2.2	Upgrade Preparedness; Meteorological Data	SSERs 3 & 6 (App. G., Sec. 4)
III.A.2.3	Upgrade Preparedness; Implement Plans	SSERs 3 & 6 (App. G., Sec. 4)
III.D.1.1.1	Integrity of Systems Outside Containment; Leak Reduction	SSERs 4, 22, & 23 [10/21/91; 50-446/91-46]
III.D.1.1.2	Integrity of Systems Outside Containment; Technical Specifications	SSERs 4 & 23 & TS 6.8.3
III.D.3.3.1	Improved Plant Iodine Instrumentation Under Accident Conditions; Determine Presence of Radioiodine	SER & SSERs 6 & 22 [4/1/91; 50-446/91-07]
III.D.3.3.2	Improved Plant Iodine Instrumentation Under Accident Conditions; Modification to Accurately Measure Iodine	SER & SSERs 6 & 22 [4/1/91; 50-446/91-07]
III.D.3.4.1	Control Room Habitability; Review	SER [11/14/90; 50-446/90-40]
III.D.3.4.2	Control Room Habitability; Schedule Modifications	SER [11/14/90; 50-446/90-40]
III.D.3.4.3	Control Room Habitability; Implement Modifications	SER & TS T-3.3-4 & 3.7.7 [11/14/90; 50-446/90-40]

APPENDIX D

LIST OF PRINCIPAL CONTRIBUTORS

Contributor	Organization
E. Baker	Office of Nuclear Reactor Regulation Project Directorate IV-2
D. Graves	Senior Resident Inspector Region IV
B. Holian	Office of Nuclear Reactor Regulation Project Directorate IV-2
R. Jenkins	Office of Nuclear Reactor Regulation Electrical Engineering Branch
P. Madden	Office of Nuclear Reactor Regulation Plant Systems Branch
I. Miller	Office of Nuclear Reactor Regulation Plant Systems Branch
E. Peyton	Office of Nuclear Reactor Regulation Project Directorate IV-2
R. Schaaf	Office of Nuclear Reactor Regulation Project Directorate IV-2
D. Skay	Office of Nuclear Reactor Regulation Project Directorate IV-2
S. West	Office of Nuclear Reactor Regulation Plant Systems Branch

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

April 2, 1993

MEMORANDUM FOR: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: COMANCHE PEAK UNIT 2 FULL-POWER LICENSING (REFERENCE THE MARCH 30 MEMORANDUM FROM THE EDO TO THE COMMISSION)

Enclosure 3 to the subject memorandum discussed information regarding thermo-lag testing primarily from a generic perspective. The staff believes it also important for the Commission to understand the staff's rationale for accepting the thermo-lag installation at Comanche Peak. Enclosed is a discussion of the staff's rationale for finding the thermo-lag installation at Comanche Peak acceptable.

James H Snezek / for
James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

cc: SECY
OGC
OCA
OPA
PDR

**ADDITIONAL INFORMATION ON THERMO-LAG
HOSE STREAM TESTING
COMANCHE PEAK**

In May of 1992, TU Electric (TU) advised the NRC staff that they were going to conduct a comprehensive fire endurance testing program to qualify their Comanche Peak, Unit 2, Thermo-Lag fire barrier designs. The initial testing methodology was based on the acceptance criteria established by American Nuclear Insurers (ANI) Information Bulletin #5 (79), "ANI/MAERP Standard Fire Endurance Test Method to Qualify a Protective Envelope for Class IE electrical Circuits," July 1979. As a result of the initial tests conducted in June and August 1992, the staff had concerns regarding the use of these criteria in that the criteria allowed the fire barrier test specimen to exhibit signs of barrier burn through, cable damage to occur, and allowed the barrier to be breached during the standard hose (solid) stream test.

Based on the staff's concerns, TU revised their fire testing methodology to eliminate the above concerns. In a letter dated October 29, 1992, the staff concurred that TU's revised methodology was acceptable.

TU's revised hose stream test applies water through a 1½-inch fog nozzle set at a discharge angle of 30 degrees with a nozzle pressure of 75 psi and minimum discharge flow rate of 75 gpm. The fog stream is applied to the specimen with the nozzle positioned 5 feet away from the center of the specimen at a pressure of 75 psi for 5-minutes.

NUREG-0800, Standard Review Plan (SRP) 9.5.1, "Fire Protection for Nuclear Power Plants," provides fire protection guidance which the staff finds acceptable with regard to meeting NRC fire protection regulations. An applicant can either follow this guidance or can propose to the staff an alternative method for achieving the level of fire safety required by the regulations. TU proposed an alternative method for hose stream testing. This method and the fire endurance testing acceptance criteria and methodology was found technically correct. The following presents the basis for the staff's acceptance of the TU alternative hose stream testing method:

1. TU's revised fire endurance testing methodology and acceptance criteria adopted the fog hose stream testing criteria established by SRP 9.5.1 for penetration seals.

2. The Comanche Peak, Unit 2, fire protection program is based on a "defense-in-depth concept." The implementation of this program required the establishment of a fire prevention program; controls on ignition sources; fire protection features which provide fire barrier separation between safe shutdown trains; rapid detection of a fire and smoke condition; automatic and manual fire suppression and control methods; and, fire resistive structural building features.

3. Comanche Peak, Unit 2, used fire resistive building construction techniques for their safety-related buildings. In addition, the combustible fire loads associated with safety-related and safe shutdown plant areas at Comanche Peak are generally low.

4. If a fire were to occur in a safe shutdown area at Comanche Peak, structural building collapse is unlikely due to the fire resistive construction techniques used in the design of these structures. Therefore, directional loads imposed on these barriers by falling structural objects during a fire is not expected. Further, these areas are generally protected by automatic sprinkler protection and would actuate in a timely manner to control the fire.

5. The fire brigade at Comanche Peak is trained in the application of water through fog streams for controlling energized high voltage equipment fires.

In addition, it should be noted that standpipe systems in nuclear power plants follow the design guidance of NFPA 14. This standard requires that the flow pressures at the hose valve where the hose station's manual fire fighting hose (100 feet) connects to the standpipe be restricted to 100 psi. Note that the friction loss in 100 feet of 1½-inch quality fire hose is approximately 25 psi at a nozzle flow rate of 86 gpm. This would yield a nozzle pressure at the hose stations of 75 psi. During the November 16-20, 1992, fire protection inspection of CPSES Unit 2, representative interior hose stations were inspected. No playpipe type solid stream nozzles were found. The type of nozzles used were the electrically-safe fog type.

It should be noted, the fog stream application (hose stream test) at 75 psi with a nozzle flow of a minimum of 75 gpm, when compared to the actual in plant hose station pressure and nozzle flow conditions, yields a 11 gpm variance and no variance in pressure. Since there was no variance in pressure, the variance in flow is considered to have a minimal affect on the eroding and cooling affects of water on the test assembly. The staff concluded that the hose stream testing performed by CPSES in their fire barrier testing program was acceptable.

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

March 30, 1993

MEMORANDUM FOR: The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: COMANCHE PEAK UNIT 2 FULL-POWER LICENSING (SRM M930316)

On March 16, 1993, the staff briefed the Commission on the status of the Comanche Peak Unit 2 full-power licensing review. On March 24, 1993, the plant achieved initial criticality and all low-power testing was successfully completed on March 28, 1993. An NRC inspection team observed initial criticality and low-power testing. In a letter of March 28, 1993 (Enclosure 1) TU Electric notified the NRC that it is ready for operation above 5% power.

Enclosures 2 and 3 provide additional information regarding current 10 CFR 2.206 petitions related to Comanche Peak and technical information regarding Thermo-Lag. The staff is also responding on March 30, 1993, to the Commission's Order of March 26, 1993, in the Construction Permit Extension Proceeding.

The staff has assessed the status of the issues that the Office of Investigations (OI) is reviewing pertaining to the Comanche Peak facility. There is no change to OI's conclusion, stated in a memorandum of February 23, 1993, that the subject issues "would not preclude the Commission's consideration for a Full Power License."

On March 29, 1993, the Regional Administrator, Region IV, recommended to the Director, Office of Nuclear Reactor Regulation, that a full-power license be issued to Comanche Peak Unit 2. This recommendation is enclosed for your information (Enclosure 4). On the basis of this recommendation, as well as on additional staff review, the Director of the Office of Nuclear Reactor Regulation has determined that the plant meets the Commission's regulations,

Contact:
Brian Holian, NRR
504-1334

and the activities authorized by a full-power license can be conducted without endangering the public health and safety. Accordingly, I recommend that the Commission vote to authorize the Director of the Office of Nuclear Reactor Regulation to issue a full-power license for Comanche Peak Unit 2.

Original signed by
James H. Sniezek for

James M. Taylor
Executive Director for Operations

Enclosures:
1. Letter from TU Electric to NRC dtd. 3/28/93
2. 10 CFR 2.206 Petitions
3. Additional Information on Thermo-Lag
4. Memorandum from Region IV to NRR dtd. 3/29/93

cc w/enclosures:
SECY
OGC
OPA
OCA

*See Previous Sheet for Concurrence

Office	PDIV-2/LA	PDIV-2/PM	PDIV-2/D	TECH EDITOR	OGC*	ADR4/5*
Name	EPeyton	BHolian	SBlack	RSanders	LChandler	MVirgilio
Date	3/30/93	3/30/93	3/30/93	3/29/93	3/30/93	3/30/93
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

Office	DRPW:D*	ADP	NRR:DSSA*	NRR:D	EDO	
Name	JRoe	JPartlow	AThadani	TMurley	JTaylor	
Date	3/30/93	3/30/93	3/30/93	3/30/93	3/30/93	/ /
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

Document Name: B:\WITS51.CP

DISTRIBUTION
Docket File
PDIV-2 R/F
TMurley
JPartlow
JRoe
MVirgilio
SBlack
BHolian
EPeyton
NRR Mail Room (WITS 9300051) 12/G/18
MBridgers (WITS 9300051)
JTaylor
JSniezek
HThompson
JBlaha
JMilhoan, Region IV
WRussell
FGillespie
JKnubel, EDO

Log # TXX-93158
File # 10010
Ref. # 10CFR.50.57

TUELECTRIC

March 28, 1993

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) - UNIT 2
DOCKET NO. 50-446
READINESS FOR ISSUANCE OF THE UNIT 2
FULL POWER OPERATING LICENSING

REF:
1) TU Electric letter logged TXX-93093 from William J. Cahill, Jr. to the NRC dated February 19, 1993
2) TU Electric letter logged TXX-93011 from William J. Cahill, Jr. to the NRC dated January 8, 1993
3) TU Electric letter logged TXX-93051 from William J. Cahill, Jr. to the NRC dated January 25, 1993
4) TU Electric letter logged TXX-93140 from William J. Cahill, Jr. to the NRC dated March 22, 1993

Gentlemen:

TU Electric has completed and evaluated the low power physics testing and the additional testing that can be completed prior to proceeding above 5% reactor power. Enclosure 1 provides a listing of the testing that is described in Chapter 14 of the Final Safety Analysis Report (FSAR) which was conducted during Mode 6 through Mode 2 since the issuance of the CPSES Unit 2 low power operating license.

Additionally, TU Electric has performed a self-assessment of the readiness of CPSES Unit 2 for proceeding above 5% power in accordance with the description provided in Reference 1. This self-assessment has been reviewed and evaluated by the Station Operations Review Committee (SORC). The SORC has concluded that CPSES Unit 2 is ready for operation above 5% power.

TU Electric is at this time ready to receive an operating license for CPSES Unit 2 which authorizes operation up to 100% reactor power.

Sincerely,

William J. Cahill, Jr.

William J. Cahill, Jr.

Enclosure

c - Mr. J. L. Milhoan, Region IV
Resident Inspectors, CPSES (2)
Mr. T. A. Bergman, NRR
Mr. B. E. Holian, NRR
Mr. L. A. Yandell, Region IV

A. Deferred Preoperational Tests and Retests conducted during Mode 6 through Mode 2. (Reference 2, Reference 3, and Reference 4).

1. Pressurizer Spray Valve Leak Tightness
2. Power Operated Relief Valve (PORV) Leak Tightness
3. Reactor Cavity Humidity Detectors
4. Steam Dump Valve Stroke Verification
5. Public Address and Emergency Evacuation Alarm System
6. Main Stream Isolation Valve (MSIV) Stroke Timing
7. Plant Computer Flux Mapping Module
8. Plant Computer Data Archive Capability
9. Plant Computer Delta I Module
10. Heat Ventilating and Air Conditioning (HVAC) System Flow Balance

Test results for the tests listed above have been evaluated by the Test Review Group as acceptable for proceeding to full power.

B. Initial Startup Tests conducted during Mode 6 through Mode 2 (Reference FSAR Chapter 14, Table 14.2-3 and Figure 14.2-4B).

1. Reactor Trip System
2. Boron Reactivity Worths
3. Rod Drop Tests
* 4. Reactor Coolant Flow Test
5. Reactor Coolant Flow Coastdown
* 6. Control Rod Drive Tests
7. Rod Position Indicators
8. Moderator Temperature Reactivity Coefficient
9. Control Rods Reactivity Worths
10. Auxiliary Startup Instrumentation
* 11. Chemical Tests
* 12. Core Performance Evaluations
* 13. Calibration of Nuclear Instrumentation
* 14. Radiation Survey
* 15. Core Reactivity Balance
* 16. Incore Nuclear Instrumentation
17. Reactor Coolant Leak Test
18. Rod Control System Test

Test results for the testing listed above have been evaluated by the Test Review Group as acceptable to proceed above 5% power.

---

* Those items identified with an asterisk are complete for Mode 6 through Mode 2 but additional testing will continue as a normal part of power ascension testing. Those items not asterisked are complete.

Enclosure 2

10 CFR 2.206 PETITIONS

The purpose of this enclosure is to provide the basis for the staff's recommendation that the full power license be issued for TU Electric's Comanche Peak Steam Electric Station (CPSES) Unit 2 with 10 CFR 2.206 petitions not finalized.

The two unresolved Comanche Peak 10 CFR 2.206 petitions are discussed below.

Comanche Peak Specific

Michael Kohn, on behalf of Messrs. Macktal and Hasan, submitted a 10 CFR 2.206 petition on June 11, 1992. The petitioner alleges that the purchase agreement for Tex-La's minority interest in CPSES by TU Electric violates NRC regulations on restrictive settlement agreements. The NRC acknowledgement letter, sent to the petitioner on August 12, 1992, stated that staff review has determined that the agreements do not appear to violate the provisions of the Energy Reorganization Act or 10 CFR 50.7. Notwithstanding, letters to TU Electric and the former co-owners of CPSES were issued on January 12, 1993, that requested information pertaining to the settlement agreements. The responses to these letters have been received and are being evaluated by the staff. The Director's Decision is expected to be issued in April 1993. The staff does not believe these issues affect issuance of a full power license for Unit 2. Although there may have been the potential for safety information to have been withheld, the petitioner did not identify any issues with respect to which he believed information had, in fact, been withheld.

On the basis of the following, the staff concludes that there is no safety significance associated with the issues currently identified in this petition:

(1) No violation of the regulations has been identified with respect to the settlement agreements, nor were any safety issues identified,

(2) About 12,000 hours of direct inspection (since the resumption of Unit 2 construction) has been conducted at CPSES Unit 2 by NRC personnel, and

(3) A recent NRC review of the licensee's SAFETEAM program (Inspection Report 50-446/92-60) concluded that the program provides both: (a) a means for employees to bring concerns to management, and (b) plant management with a mechanism for the early identification of issues that could impact the safety of the plant.

The Commission requested additional information regarding this issue by an Order issued in the context of the construction permit amendment proceeding, dated March 26, 1993. The staff response is being filed on March 30, 1993.

Generic Thermo-Lag

The Nuclear Information and Resource Service (NIRS) submitted a 10 CFR 2.206 petition on July 21, 1992, as supplemented by addendum of August 12, 1992. On February 1, 1993, a Partial Director's Decision (DD-93-03) was issued regarding this petition.

On December 15, 1992, NIRS filed another petition pursuant to 10 CFR 2.206 regarding Thermo-Lag. This petition, which addresses numerous plants, including CPSES, Units 1 and 2, is being considered by the staff as a supplement to the petition filed on July 21, 1992. The acknowledgement letter, issued on February 4, 1993, denied the requested action to shut down all plants with Thermo-Lag, stating that no immediate safety concerns were raised. The review of the petition, which reiterates items from a previous petition, and includes issues regarding Thermo-Lag voiding and stapling is expected to be complete by May 1993. The staff's evaluation of CPSES Unit 2 fire barrier acceptability is presented in SSER 26, in which the staff concluded that with several commitments, TU Electric's fire barrier program is acceptable. The staff specifically requested information on the stapling issue and although not explicitly discussed in SSER 26, was considered in the discussion on page 9-20 in support of this conclusion.

Enclosure 3

ADDITIONAL INFORMATION ON THERMO-LAG

This enclosure provides additional information regarding two issues relating to Thermo-Lag: hose stream testing and seismic concerns.

HOSE STREAM TEST

The American Society for Testing and Materials (ASTM), as part of their testing methods for determining the fire endurance for various types of building construction, adopted the current method for hose stream testing in 1933. Currently, fire testing standards are focused on building columns, walls and partitions, and floor construction.

The intent of the hose stream test is to impose a cooling, impact, and erosion effect on the building construction being evaluated. The weaknesses of a structural building system after being subjected to the hose stream test, fall into the following categories: structural failures, thermal (brittle) failures and erosion failures.

Focusing on raceway fire barrier systems, the staff, as part of their acceptance criteria development, examined the applicability of the ASTM standard hose stream test to these barrier systems. Staff consensus, during this examination, identified the need for some form of hose stream test. The staff elected to adopt approved hose stream testing methods identified by Position 5.a to Standard Review Plan (SRP) 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants." This SRP position established the fire endurance and hose stream testing acceptance criteria for other non-structural fire resistive barrier components (penetration seals) and allowed the use of the fog nozzle method for hose stream testing.

The staff's rationale for allowing the use of the fog method is based on the following:

- Structural

  Walls, fire doors and dampers are structural building components, whose failure could either contribute to the structural failure of the building or fire growth within the building.

  Fire barrier systems used to separate safe shutdown functions within the same fire area are not considered to aid in the prevention of structural building collapse.

  Fire resistive construction techniques are used in the design of nuclear power facilities to prevent such structural failures. In addition, the combustible fire loads are generally low.

  Under actual in-plant fire conditions, structural collapse is unlikely.

Therefore, directional loads (simulated by the standard hose stream test) imposed on these barriers by falling structural objects during a fire is not expected.

Fire fighting activities, resulting from manual suppression can cause some level of barrier impact.

Manual fire fighting suppression operations in the areas of energized electrical equipment require the use of fog streams.

The fog nozzle test method (Pressure and Flow) simulates in-plant manual fire suppression techniques which would be employed by the fire brigade.

- Cooling

  The fog stream method applies more water to the test specimen over a greater duration of time.

  (Fog - 375 gallons vs. standard - 210 gallons) (Duration of application: standard - 1 minute vs. Fog - 5 minutes).

  The amount of water applied and the duration of application is sufficient to determine the thermal fragility of the barrier system under simulated fire fighting hose stream applications.

- Erosion

  The Fog method is sufficient to demonstrate erosion conditions which would be encountered by the implementation of in-plant fire fighting techniques.

  Experience (TU Electric tests) has demonstrated the ability of this method to impact the fire barrier material char layer, seams and joints.

  This method is capable of eroding the char layer and has made openings in the areas of joints and seams. (Scheme 12-2, 24-inch wide tray with T-Section, hose stream test damage)

Additionally, the staff considered the fact that nuclear power plant fire protection programs are based on a "defense-in-depth concept," (e.g., prevention of fires; control of ignition sources; fire protection features which provide fire barrier separation between safe shutdown trains; rapid detection of a fire and smoke condition; automatic and manual fire suppression and control methods) in support of using the fog stream test method.

SEISMIC ISSUES

The NRC does not require that fire protection systems, including features such as fire barriers, be formally qualified for seismic events. Such qualification is required for safety-related systems that are used to mitigate design basis events such as large pipe breaks. The NRC is requiring licensees to address the potential consequences of events beyond the design basis as part of a systematic review of plant vulnerabilities (Individual Plant Examinations for External Events). One area specifically to be examined is fires caused by earthquakes.

The staff has, however, specifically examined the potential for Thermo-Lag panels to break up during a seismic event, thereby creating a threat to nearby safety-related equipment. A 10 CFR 2.206 petition from Nuclear Information Research Service postulated such a breakup, with the panels acting as a shear (severing cables and shattering cable trays), thus jeopardizing safe shutdown. To the NRC staff's knowledge, the Thermo-Lag vendor has not performed seismic tests of prefabricated panels. The staff has reviewed a seismic analysis (performed by a consultant to the Thermo-Lag vendor) of such panels attached to cable trays and conduit sections. In addition, the staff visited a plant to understand more about Thermo-Lag usage, installation details, and material properties. It is the staff's judgement after this review, that Thermo-Lag panels are not likely to get detached from raceways during a safe shutdown earthquake. Although the material may crumble and crack, the staff concluded that considering the material properties of Thermo-Lag and the design of raceways, shattering of raceways and severing of cables are not credible scenarios, and that the safe shutdown capability would be maintained. As discussed above, the beyond design basis accident scenarios of earthquake induced fires will be considered under the Individual Plant Examinations for External Events.

Enclosure 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

MAR 29 1993

Docket No. 50-446

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: James L. Milhoan, Regional Administrator

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION UNIT 2
FULL-POWER LICENSE RECOMMENDATION

The Region IV staff and I recommend the issuance of a full-power operating license for Comanche Peak Steam Electric Station (CPSES) Unit 2. The bases for this recommendation are contained in a readiness assessment which was provided to you on February 1, 1993, prior to the issuance of the low-power license and the additional information provided below.

1. The construction and preoperational test inspection programs have been completed.

2. The start-up test and operations inspection programs have been initiated and an NRC master inspection plan for Unit 2 has been developed and approved. The NRC resident inspectors, augmented by Regional staff, provided around-the-clock coverage of fuel loading, initial approach to criticality, and power ascension below 5 percent of rated power. NRC inspectors also observed mode changes, zero power start-up tests, normal day-to-day operations, and maintenance activities. Two issues were identified during initial licensed operations as discussed in Attachment 1 to this memorandum. Those issues have been satisfactorily resolved. In addition, a regional readiness assessment team inspection was conducted March 25-28, 1993, to evaluate the licensee's performance during initial critical operations. The team concluded that TU Electric has implemented appropriate measures including management oversight, corrective actions, and self-assessments to support plant operations. The operations, engineering and maintenance organizations demonstrated a common resolve for the safe conduct of plant operations. Based on direct observation and evaluation of licensee performance, we believe that TU Electric's performance to date demonstrates their readiness to operate the plant above 5 percent power.

3. The CPSES Oversight Panel met on March 11 and 28, 1993, to perform an evaluation of licensee performance, and to review the results of the readiness assessment team inspection and the around-the-clock inspection coverage of the low power testing. The panel concluded that TU Electric is ready to conduct dual unit operations, including power operation of Unit 2.

4. Region IV staff were requested by memorandum dated December 3, 1992, to identify any issues that could impact the issuance of a license for CPSES Unit 2. There were no concerns expressed by any Region IV personnel relative to the issuance of a low-power license for Unit 2. Since the issuance of the low-power license, no concerns have been expressed to Region IV management.

5. Acceptable progress has been made in addressing those items identified in my February 1, 1993, memorandum as being incomplete at the time of low power licensing. Resolution of the remaining open items is not deemed necessary to support issuance of a full-power license. Attachment 2 provides the status of those items.

NRC inspectors will continue to implement an augmented inspection and evaluation program to monitor the licensee's performance during the period of power escalation to full-power. This will include:

- Around-the-clock inspection coverage to monitor the licensee's performance during important tests and during parts of power escalation.

- Witnessing selected start-up tests and reviewing the start-up testing results.

- Reviewing the licensee's self-assessment to be performed at the 50 percent power plateau.

With respect to the staff requirements memorandum concerning the March 16, 1993, discussion on full-power operating licensing for Comanche Peak (Unit 2), it is our understanding that your staff is responding to the Commission's request for additional information. Region IV has reviewed and concurred in the response.

In conclusion, we find that CPSES, Unit 2, construction has been substantially completed in accordance with Construction Permit CPPR-127, the FSAR, and NRC regulations; that fuel loading, initial start-up, and operations to date have been conducted safely; and that TU Electric is ready to safely operate Unit 2 above 5 percent power.

James L. Milhoan

James L. Milhoan
Regional Administrator

Attachments: As stated

cc w/attachments: (see next page)

cc w/attachments:
J. Taylor, (MS 17G21)
J. Sniezek, (MS 17G21)
RIV Coordinator, OEDO (MS 17G21)
R. Bernaro, (MS 6E6)
J. Partlow, (MS 12G18)
W. Russell, (MS 12G18)
J. Roe, (MS 13E4)
M. Virgilio, (MS 13A2)
S. Black, (MS 13H15)
B. Holian, (MS 13H15)
C. Rossi, (MS 9A2)
C. Grimes, (MS 11E22)
J. Montgomery
L. Callan
A. Beach
S. Collins
L. Yandell
D. Graves

ATTACHMENT 1

ISSUES IDENTIFIED SINCE LOW-POWER LICENSING

Two issues were identified after the issuance of the low-power operating license that were of some potential significance and which were discussed in a staff note for the Commission on March 10, 1993. The first issue involved the identification of a Borg-Warner pressure seal check valve in the auxiliary feedwater system (Valve AF-106), that failed a surveillance test for back flow. The second issue involved the identification of noise on the reactor coolant system loose parts monitor that indicated the potential presence of a loose part in the reactor pressure vessel. The status of those issues is described below:

BORG-WARNER CHECK VALVE AF-106 STATUS

On March 8, 1993, auxiliary feedwater check valve AF-106 failed a back flow test. The valve was disassembled for maintenance and found to be rotated 10 degrees with respect to the disk seating surface. This resulted in the disk hanging approximately 1/2" off its seat at the lower surface. The valve was reworked and reassembled, and satisfactorily back flow tested on March 11, 1993. TU's investigation indicated that a satisfactory back flow test was completed last fall. After the test was completed, work instructions called for the installation of a block and key device that was intended to aid maintenance personnel in properly aligning the valve during future maintenance activities. However, it appears that some action during the installation of the block and key device caused the misalignment of the valve. At that time, no post work testing was required. As an immediate corrective action, the licensee determined that 25 of these check valves had the block and key device installed. Fourteen of this number are safety-related, and all were back flow tested satisfactorily after installation of the device. The remaining 11 check valves were evaluated through the use of temperature monitors, radiographs, or back flow testing and determined to be satisfactory. All other Borg-Warner check valves installed in Unit 2 are bolted bonnet valves that are not susceptible to misalignment during reassembly. The licensee completed a root cause analysis of this failure on March 23, 1993. It was concluded that the cause for the event was uncertain, however, a contributing factor was the lack of verification of alignment during the process to install the block and key devices. The NRC readiness assessment team reviewed the issue and concluded that TU Electric had conducted an appropriate root cause analysis. The corrective actions implemented were found to address the condition in which the valve was found and should prevent recurrence. This same corrective action should also be effective in preventing similar problems on the Borg-Warner pressure seal check valves where the block and key devices are not installed.

LOOSE PARTS MONITOR ALARM

On March 7, 1993, with Unit 2 in Mode 4, operations personnel identified a high noise alarm on the reactor coolant system loose parts monitor (LPM). Subsequent evaluation of this condition by the licensee's plant engineering organization, in consultation with the LPM manufacturer (Babcock and Wilcox), and the Nuclear Steam Supply System vendor (Westinghouse), concluded that the LPM high noise alarm was attributable to flux thimble tube vibrations. Based on the technical analysis provided by Westinghouse, the licensee concluded

that there was no loose part present, that there were no short term safety concerns related to the flux thimble tube vibrations, and that the LPM system was acceptable for continued operation. NRR and Region IV staff agreed with the licensee's conclusions. The licensee has an inspection and maintenance program, as required by NRC Bulletin 88-09, that will address vibration of the bottom mounted incore flux thimbles.

ATTACHMENT 2

STATUS OF OPEN ITEMS FOR COMANCHE PEAK UNIT 2

The low-power license recommendation letter dated February 1, 1993, provided the status of items that were not required to be completed prior to the issuance of the low-power operating license. The current status of those items is discussed below, and none of the issues remaining open needs to be resolved prior to full-power licensing.

- The Operational Readiness Assessment Team (ORAT) identified weaknesses in the areas of configuration control (system status control), procedures, and corrective actions which the licensee indicated would receive necessary attention prior to fuel load. In the area of system status control, the licensee reverified all safety system lineups prior to Mode 6 and revised the controlling procedure to ensure positive system status control. Region IV inspectors confirmed by procedure reviews and system walkdowns that the licensee had taken appropriate corrective actions. In the procedure area, the licensee revised the procedures in question and committed to perform a complete procedure review (approximately 700 procedures) over the next two years. Region IV inspectors confirmed that the specific procedure deficiencies identified were corrected by the licensee. The contract auxiliary operators who had not received training required by procedures were removed from plant duties. Also, the inspectors verified that the work performed by the contract auxiliary operators was reviewed by the licensee, and that the operators were reassigned to tasks authorized by management and commensurate with their current training. With regard to corrective actions, the licensee completed evaluation of the four items identified by the ORAT. Region IV inspectors determined that post-test requirements on a feedwater check valve were appropriately applied and that an extensive field verification of abnormal procedures was being completed by the licensee on the committed schedule. Overall, Region IV is satisfied that the ORAT concerns have been satisfactorily addressed.

- All TMI items, except the Safety Parameter Display System (SPDS), were verified complete prior to low-power licensing. The SPDS was operational, but a required assessment after 30 days of operation had not been completed. By letter dated March 22, 1993, the licensee indicated that no specific date for the start of 30-day assessment had been established, but that the assessment would start within 60 days after fuel load (April 7, 1993). This is the same approach used for Unit 1, and the Region finds it acceptable.

- Thermo-Lag fire barrier material was verified as installed prior to low-power licensing; however, not all Thermo-Lag installations had completed a 30-day cure time. In addition, 13 (box enclosure) installations that were configured differently from tested configurations had not been adequately justified by analysis or testing. These installations resulted in the implementation of fire watches as a compensatory measure pending completion of the cure time and configuration upgrades. Currently, the 30-day cure time has passed for all of the installations completed prior to the issuance of the low-power license. Also, as discussed in draft Supplemental Safety Evaluation Report 27 (to be published concurrent with full-power license), testing performed on March 4, 1993, established a seven-day cure time as acceptable for

all future installations. With regard to the box enclosure configurations, all were subsequently found acceptable after the staff's review, and these upgrades were certified complete by TU Electric on March 10, 1993. Fire watches are no longer required with regard to these two specific issues.

- Identified weaknesses regarding the control of temporary modifications as discussed in Inspection Report 50-445/92-62; 50-446/92-62, have been effectively addressed by interim actions taken by the licensee.

- At the time of low-power licensing, the Region IV staff perceived the lack of a sense of ownership by the Unit 2 operations staff that was being rapidly dispelled as the two units were brought together under a single operating organization. Since issuance of the low-power license, two special inspections were conducted that looked, in part, at this issue. The inspection of the licensee's personnel error reduction program in February 1993, and the augmented inspection of licensee performance during initial low-power operations during March 25-28, 1993, determined that the plant staff has exhibited an increased sense of ownership of Unit 2 commensurate with that seen on Unit 1.

- Test and retest deferrals discussed in Attachment 13.2 of the February 1, 1993, low-power license recommendation letter have been successfully completed in accordance with the licensee's schedule. The remaining deferred tests are scheduled to be completed consistent with plant operational requirements. Region IV continues to track the completion of these deferred tasks.

- All NRC items required to be completed prior to exceeding 5 percent power have been completed. The remaining open items identified in the Inspection Followup System are being tracked for completion at the appropriate time.

- At this time, there are four outstanding allegations pertaining to the construction and operation of the CPSES facility. The Region IV Allegations Review Panel convened on March 11, 1993, and determined that the resolution of the pending allegations was not necessary to support the issuance of a full-power license. No additional allegations have been received for Comanche Peak since March 11, 1993.