March 31, 1993

Docket No. 50-446

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

Dear Mr. Cahill:

SUBJECT: COMANCHE PEAK UNIT 2 FULL-POWER LICENSING

Enclosed for your information are two documents that the staff filed with the Commission this week. The first enclosure is a Commission Information Paper which contained responses to questions which arose at the March 16, 1993 Commission meeting relating to Comanche Peak Unit 2 full-power licensing. This paper also provided information on the plant's status and readiness for a full-power license. The second enclosure is the staff's recommendation to the Commission that it vote to authorize issuance of the full-power license.

Sincerely,

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

Enclosures:
1. SECY 93-81
2. Commission Memo dtd. 3/30/93

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ENCLOSURE NO. 1

POLICY ISSUE
(Information)

March 29, 1993

SECY-93-081

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: STATUS OF COMANCHE PEAK UNIT 2 LICENSING

PURPOSE:

To respond to Commissioners' request for more information from the March 16, 1993, discussion on Comanche Peak, Unit 2 full-power operating license (SRM M930316) and to provide an update on plant status and readiness for a full-power license.

DISCUSSION:

During a briefing by representatives of public interest groups, the licensee, and the NRC staff, the Commissioners requested further information on the following items:

- the proposed license exemption for criticality monitoring,
- licensee use of a computer-based configuration management system,
- cable ampacity derating values and existing margins, and
- requirements for laboratories that perform fire protection tests.

The staff responded to these items in the enclosure.

Contact: Brian Holian, NRR
504-1334

Donna Skay, NRR
504-1322

NOTE: TO BE MADE PUBLICLY AVAILABLE IMMEDIATELY

9304010379 XA 7pp.

After receiving a low-power operating license on February 2, 1993, TU Electric completed fuel loading of Comanche Peak, Unit 2 on February 7, 1993. Mode change milestones were successfully completed, culminating in entering Mode 2 at 4:25 a.m. (CST) on March 24, 1993. At 8:46 p.m. (CST) on March 24, 1993, Comanche Peak, Unit 2 achieved initial criticality. The licensee completed its low-power physics testing by March 26 and plans to complete test reviews on March 28. The licensee expects to be ready to proceed above 5-percent power by March 29, 1993.

Region IV began around-the-clock coverage of control room and plant activities on March 23, 1993. In addition, a six-person team from Region IV and NRR began an augmented inspection of Unit 2 critical operations on March 24, 1993. The NRR/Region IV Comanche Peak Oversight Panel will review the findings of these two efforts on March 28, 1993. Should the Comanche Peak Oversight Panel find the low-power operations satisfactory, the Regional Administrator's letter to the Director, NRR recommending full-power license issuance would be expected on March 29, 1993. If the licensee's operations proceed satisfactorily, the staff request that a full-power license be authorized could be sent to the Commission as early as the afternoon of March 29, 1993.

After receipt of the full-power license, the licensee forecasts achieving Mode 1 operations within 2 days, synchronizing to the electrical grid within 5 days, performing required startup testing throughout April and May, performing a planned outage in early June, and beginning commercial operations on June 17, 1993.

James H Sniezek /for

James M. Taylor
Executive Director for Operations

Enclosure:
Response to Commissioners' Questions

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ENCLOSURE

RESPONSE TO COMMISSIONERS' QUESTIONS

Item 1: The proposed license exemption for criticality monitoring.

Response

This license exemption, which was also approved for Unit 1, is related to the monitoring requirements of 10 CFR 70.24 for the movement and storage of fuel in the fuel handling building. The requirements state that in any area in which special nuclear material is handled, two gamma- or neutron-sensitive radiation detectors shall be used to audibly signal if accidental criticality should occur. The regulation does not require underwater monitoring when the fuel is stored beneath a water shield or monitoring of packaged fuel assemblies. The exemption was approved for Unit 2 based on the fact that procedural controls are in place that will preclude criticality during receipt, inspection, or storage of new fuel. These procedural controls allow only two fuel assemblies to be outside of a shipping container, storage rack, or transfer tube at one time, and require a minimum 12-inch distance between such assemblies.

Invariably, applicants for the Part 70 cold fuel license have requested an exemption from the criticality alarm requirements of 10 CFR 70.24. This exemption has been routinely granted for all licensees of power reactors when the request is submitted with specific procedural controls for the handling of new fuel. In order to reduce the regulatory burden of seeking routinely granted exemptions, a change to 10 CFR 70.24(e) has been proposed that would codify both the exemption and the related conditions. The amendment to Part 70 would exempt power reactors from monitoring the dry storage of unirradiated fuel if conditions are present that will ensure that the fuel will not become critical. The primary basis for the rule change is that the monitors do not perform a safety function to prevent an accident but simply alert personnel that one has occurred. The conditions imposed on applicants as part of the 10 CFR Part 70 licensing review are sufficient to prevent a criticality accident from occurring.

Item 2: Licensee use of a computer-based configuration management system.

Response

This issue was raised during a discussion of a weakness discovered by the Operational Readiness Assessment Team (ORAT) when it inspected the area of configuration control. The Regional Administrator explained that the weakness dealt with valve misalignments found during the inspection. The ORAT report stated that the misalignments were the result of the failure of operators to document valve manipulations. The staff found the corrective actions, which included revision of the system status control procedure, acceptable; through followup inspections, the staff confirmed that TU Electric has adequately addressed these issues.

Subsequent to this discussion, a question was raised as to whether the licensee had a computerized system to aid in configuration management. Discussions with TU Electric revealed that a computer aided design (CAD) system is used as a drafting tool to store and retrieve plant drawings. The licensee stated that the scope of the computerized system has been limited to vital station drawings (i.e., system flow diagrams) of critical systems to make the best use of resources and to maintain system efficiency and speed. This system assists in making timely updates to important drawings, which enhances the plant's configuration management.

Item 3: The specific cable ampacity derating values and existing margins.

Response

Cables enclosed in electrical raceways that are protected with fire barrier materials are derated because of the insulating effect of the fire barrier material. Among the other factors that affect ampacity derating are 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 concerns it has 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 documented TU Electric's interim ampacity derating factors, and acknowledged that TU Electric had performed a calculation to evaluate the acceptability of a 40-percent cable tray derating factor.

Accordingly, the staff: (1) documented TU Electric's commitment to complete plant-specific ampacity derating testing by the completion of the first refueling outage and (2) concluded that the use of TU Electric's interim ampacity derating factors was acceptable.

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. NRC staff observed test preparation and testing from March 2-7, 1993. The first test group, conducted from March 2 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"-wide x 4"-deep x 12-feet-long 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 #750 MCM volt copper cable and a free air drop (large) made of three 1/C # 750 MCM volt copper cable. The ampacity derating factor test results are summarized below.

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) 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%
#2-EE-CA-0008-3038	From 480 V	Cable tray - 38% Conduit - 23%
#2-EE-053	All other	Cable tray - 40% Conduit - 35%
#16345-EE(B)-140	Air drops	Cable tray - 39% Conduit - 35%

On March 10 and March 23, 1993, TU Electric sent letters to NRC containing preliminary information about both TU Electric's calculated excess ampacity margin and the test results for the plant-specific ampacity derating tests.

The following table summarizes the preliminary test data, and provides the ampacity derate margin (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 submits the final test reports (consistent with the schedule published in SSER 26).

Item 4: Clarification on requirements for laboratories that perform fire protection tests and how such requirements are determined to be met.

Response

Section 9.5-1 of the Standard Review Plan uses the phrase "nationally recognized testing laboratory," which was adopted from terminology used in the past by the National Fire Protection Association (NFPA). The SRP does not define the phrase and the NFPA subsequently dropped the phrase from all of its published documents. In its *Fire Protection Handbook*, the NFPA stated that it dropped the phrase because there was always a doubt about exactly what constituted a "nationally recognized testing laboratory."

The NRC does not have codified requirements or guidance for evaluating the acceptability of fire testing laboratories to perform fire protection tests. The staff does, however, apply its fire protection engineering expertise and judgment and ensures that such laboratories have the facilities, equipment, personnel, and experience needed to conduct such tests.

Early in its review of the Thermo-Lag issues, the staff asked the National Institute of Standards and Technology (NIST) to identify attributes for evaluating fire testing laboratories. NIST stated that suitable laboratories would have the following attributes:

- The facilities and equipment needed to conduct tests in accordance with national consensus standards such as American Society of Testing and Materials (ASTM) and the National Fire Protection Association (NFPA). For the purposes of testing raceway fire barriers, this includes a full-scale floor furnace that conforms to all of the requirements of ASTM E-119.

- Knowledgeable and experienced staff such as full-time permanent staff trained in running ASTM E-119-type tests and a responsible test officer authorized to sign the test reports. (Although not essential, NIST stated that participation on ASTM or NFPA fire testing standards committees offers additional evidence of laboratory competence. Omega Point Laboratories participates on these committees.)

- Experience in conducting fire endurance tests for more than one client.

- Acceptance of the laboratory's ASTM E-119 test results by the U.S. model building code organizations and other authorities having jurisdiction over fire protection, such as insurance companies and Federal, State, and local agencies responsible for fire safety.

NIST independently reviewed Omega Point Laboratories (OPL) against these attributes and found that it was suitable for conducting raceway fire barrier tests. This conclusion is documented in a letter from NIST to K.S. West, NRC, dated November 5, 1991.

As a fundamental part of its review of the Comanche Peak Thermo-Lag fire barrier program, the staff evaluated OPL during visits to the laboratory. The staff observed all aspects of the laboratory's work on full-scale fire barrier tests and reviewed its fire test reports. The staff concluded that OPL was equipped and qualified to conduct raceway fire barrier tests.

An additional inspection of Omega Point Laboratory was conducted in March 1993 by the Vendor Inspection Branch (VIP). Again, no specific criteria were used to qualify the laboratory. The inspection report is not complete, however, the initial findings did not reveal any significant technical deficiencies. Discussions with the VIB indicate that none of the inspections findings affect the NRC's assessment that OPL is qualified to perform raceway fire barrier tests.

ENCLOSURE NO. 2

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
PDR

Enclosure 1

TU ELECTRIC

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

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.

9303310209 3pp

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.

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* 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.

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

Enclosure 4

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.

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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)

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.