Mr. C. Lance Terry
Group Vice President, Nuclear
TU Electric
400 North Olive Street, L.B. 81
Dallas, TX 75201

June 13, 1996

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - AMENDMENT NOS. 51 AND 37 TO FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89 (TAC NOS. M94992 AND M94993)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment Nos. 51 and 37 to Facility Operating License Nos. NPF-87 and NPF-89 for the Comanche Peak Steam Electric Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 12, 1996 (TXX-96008).

The amendments revise the Technical Specifications (TSs) to reflect the approval for use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50, Appendix J, Option B for CPSES Units 1 and 2. Implementation of the new performance based leakage rate testing program will be based on the guidance provided by Regulatory Guide 1.163, September 1995.

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

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

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 51 to NPF-87
2. Amendment No. 37 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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

June 13, 1996

Mr. C. Lance Terry
Group Vice President, Nuclear
TU Electric
Energy Plaza
1601 Bryan Street, 12th Floor
Dallas, TX 75201-3411

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - AMENDMENT NOS. 51 AND 37 TO FACILITY OPERATING LICENSE NOS. NPF-87 AND NPF-89 (TAC NOS. M94992 AND M94993)

Dear Mr. Terry:

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The amendments revise the Technical Specifications (TSs) to reflect the approval for use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50, Appendix J, Option B for CPSES Units 1 and 2. Implementation of the new performance based leakage rate testing program will be based on the guidance provided by Regulatory Guide 1.163, September 1995.

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

Sincerely,

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

Docket Nos. 50-445 and 50-446

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1. Amendment No. 51 to NPF-87
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cc w/encls: See next page

Mr. C. Lance Terry
TU Electric Company

Comanche Peak, Units 1 and 2

cc:
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 1029
Granbury, TX 76048

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

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

Mr. Roger D. Walker, Manager
Regulatory Affairs for Nuclear
Engineering Organization
Texas Utilities Electric Company
1601 Bryan Street, 12th Floor
Dallas, TX 75201-3411

Texas Utilities Electric Company
c/o Bethesda Licensing
3 Metro Center, Suite 610
Bethesda, MD 20814

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

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

Office of the Governor
ATTN: Susan Rieff, Director
Environmental Policy
P. O. Box 12428
Austin, TX 78711

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

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

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. NPF-87

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Texas Utilities Electric Company (TU Electric, the licensee) dated March 12, 1996 (TXX-96008), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 51, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip M Ray

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

Attachment: Changes to the Technical Specifications

Date of Issuance: June 13, 1996

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

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. NPF-89

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Texas Utilities Electric Company (TU Electric, the licensee) dated March 12, 1996 (TXX-96008), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 37, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Phillip M Ray

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

Attachment: Changes to the Technical
Specifications

Date of Issuance: June 13, 1996

ATTACHMENT TO LICENSE AMENDMENT NOS. 51 AND 37

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

DOCKET NOS. 50-445 AND 50-446

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

REMOVE	INSERT
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-8	3/4 6-8
B 3/4 6-1	B 3/4 6-1
6-7	6-7
-	6-7a

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 2.1.1 of the Technical Requirements Manual; and

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days. The blind flange on the fuel transfer canal need not be verified closed except after each drainage of the canal.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g).

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a, within 1 hour initiate action to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than or equal to 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g), except as noted below:

SURVEILLANCE REQUIREMENTS (Continued)

a. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

b. Containment ventilation isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2 or 4.6.1.7.3, as applicable;

c. Safety injection valves 8809A, 8809B, and 8840 shall be leak tested with a gas at a pressure not less than P_a, 48.3 psig, or with water at a pressure not less than 1.1 P_a, at intervals defined in the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g);

d. Containment spray valves HV-4776, HV-4777, CT-142, and CT-145 shall be leak tested with water at a pressure not less than 1.1 P_a, at intervals defined in the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g); and

e. The provisions of Specification 4.0.2 are applicable only to b.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

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

ACTION:

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

   3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

   4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g).

b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal indicated pressure shall be maintained between -0.3 psig and 1.3 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the adjusted average of two temperatures at or above the following containment locations of which at least one temperature is from location a. or above and shall be determined at least once per 24 hours:

Location

a. Dome, El. 1000'-6"

b. Floor, El. 860'-0"

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

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

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1. Containment Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during shutdown by a visual inspection of these surfaces. This inspection shall be performed in accordance with the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g) to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports. Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the EXCLUSION AREA BOUNDARY radiation doses to within the dose guideline values of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates, as specified in the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g), ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. The overall containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

For specific system configurations, credit may be taken for a 30-day water seal that will be maintained to prevent containment atmosphere leakage through the penetrations to the environment. The following is a list of the containment isolation valves that meet this system configuration and the Maximum Allowed Leakage Rate (MALR) required to maintain the water seal for 30 days.

Valve No.	MALR (cc/hr)
1-8809A	77
1-8809B	77
2-8809A	75
2-8809B	73
1-8840	2577
2-8840	2382
CT-142	4734
CT-145	4734
HV-4776	4734
HV-4777	4734

The surveillance testing for measuring leakage rates, as specified in the Containment Leakage Rate Testing Program (refer to Technical Specification Section 6.8.3g), is consistent with Reg. Guide 1.163, 1995 and the requirements of Option B of 10 CFR 50 Appendix J.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential of 5 psid with respect to the outside atmosphere, and (2) the containment peak pressure does not exceed the design pressure of 50 psig during a LOCA.

The indicated containment pressure values of -0.3 psig and 1.3 psig correspond to analytical limits of -0.5 psig and 1.5 psig, respectively, with allowance for measurement uncertainty.

The maximum peak pressure expected to be obtained from a LOCA event is 48.3 psig, which is less than design pressure and is consistent with the safety analyses. This value includes the limit of 1.5 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. The average temperature shall be by an adjusted averaging of at least 2 of the measurements made at the listed locations, by fixed or portable instruments with allowance for temperature measurement uncertainty.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 48.3 psig in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch and 12-inch containment and hydrogen purge supply and exhaust isolation valves are required to be locked closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves locked closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Ventilation System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the Containment Ventilation System during operations is restricted to the 18-inch pressure relief discharge isolation valves (with an effective diameter of 3 inches) since, these venting valves are capable of closing during a LOCA or steam line break accident. Therefore, the Exclusion Area dose guideline of 10 CFR 100 would not be exceeded in the event of an accident during containment venting operation.

PROCEDURES AND PROGRAMS (Continued)

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

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

f. Containment Leakage Rate Testing Program

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

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

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

Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

   a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

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

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

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

6.9.1.1 Not used.

ANNUAL REPORTS*

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

Reports required on an annual basis shall include:

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

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

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

ANNUAL REPORTS (Continued)

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM, and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR 50.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves,

*A single submittal may be made for a multiple unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 51 AND 37 TO

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

TEXAS UTILITIES ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated March 12, 1996 (TXX-96008), Texas Utilities Electric Company (TU Electric/the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. NPF-87 and NPF-89) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed changes would revise the Technical Specifications (TSs) to reflect the approval for the licensee to use of the new Containment Leakage Rate Testing Program as required by 10 CFR Part 50, Appendix J, Option B for CPSES, Units 1 and 2. Implementation of the new performance based leakage rate testing program will be based on the guidance provided by Regulatory Guide 1.163, September 1995.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate values specified in the TSs and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors" to 10 CFR Part 50, was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program".

Based on the results of this study, the staff developed a performance based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the staff for implementing Option B. This Regulatory Guide states that the Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the staff for complying with Option B with four exceptions, which are described therein.

Option B requires that the Regulatory Guide or other implementation document used by a licensee to develop a performance based leakage rate testing program must be included, by general reference, in the plant TSs. The licensee has referenced Regulatory Guide 1.163 in the CPSES TSs.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs for implementing Option B. After some discussion, the staff and NEI agreed on a set of model TSs which were transmitted to NEI in a letter dated November 2, 1995. These TSs are to serve as a model for licensees to develop plant specific TSs in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component factors that are indicative of or affect performance such as an administrative leakage limit must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to ensure that they are selected in a reasonable manner, they are not TSs requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's March 12, 1996, letter to the NRC proposes to establish a "Primary Containment Leakage Rate Program" and proposes to add this program to the TSs. The program references Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program" dated September, 1995 which specifies methods acceptable to the staff for complying with Option B. This requires a change to TS 3/4.6.1.1, "Containment Integrity," 3/4.6.1.2, "Containment Leakage," 3/4.6.1.3, "Containment Air Locks," and 3/4.6.1.6, "Containment Structural Integrity" and the addition of Specification 6.8.3g, "Containment Leakage Rate Testing Program," to implement the new performance based leakage rate testing program as permitted by 10 CFR Part 50, Appendix J, rather than paraphrasing the requirements of the existing regulation.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B, and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The staff finds that the TSs changes proposed by the licensee meet the requirements of 10 CFR Part 50, Appendix J, Option B and, are consistent with the model TSs included in the staff's November 2, 1995, letter to NEI and are therefore acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 15999). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributor: T. Polich

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