March 5, 2004

Mr. M. R. Blevins
Senior Vice President
& Principal Nuclear Officer
TXU Energy
ATTN: Regulatory Affairs
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO TECHNICAL SPECIFICATION FOR CONTAINMENT ISOLATION VALVES (TAC NOS. MB8185 AND MB8186)

Dear Mr. Blevins:

The Commission has issued the enclosed Amendment No. 111 to Facility Operating License No. NPF-87 and Amendment No. 111 to Facility Operating License No. NPF-89 for CPSES, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 18, 2003, as supplemented by your letter dated August 14, 2003.

The amendments revise TS requirements to permanently except seven containment isolation valves in each unit, in the residual heat removal and the containment spray systems, from the local leakage rate testing requirements of 10 CFR Part 50, Appendix J.

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,

/RA/

Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 111 to NPF-87
2. Amendment No. 111 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
License No. NPF-87

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

   A. The application for amendment by TXU Generation Company LP dated March 18, 2003, as supplemented by letter dated August 14, 2003, 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. 111, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Generation Company LP shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 5, 2004

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
License No. NPF-89

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

   A. The application for amendment by TXU Generation Company LP dated March 18, 2003, as supplemented by letter dated August 14, 2003, 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. 111, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Generation Company LP 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 5, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 111

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 111

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3.6-15	3.6-15

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 111 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 111 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated March 18, 2003, as supplemented by letter dated August 14, 2003, TXU Energy (the licensee) requested a technical specification (TS) change for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. Specifically, the change would permanently except seven containment isolation valves (CIVs) in each unit, in the residual heat removal (RHR) safety injection (valves 8809A, 8809B, and 8840) and containment spray (valves HV-4776, HV-4777, CT-142, and CT-145) systems, from the local leakage rate testing requirements of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix J (Reference 1). This would be done by deleting TS surveillance requirements (SRs) 3.6.3.12 and 3.6.3.13.

The August 14, 2003, supplemental letter provided clarifying information and did not change the scope of the original *Federal Register* notice or staff's original no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

There are two Options contained in Reference 1: Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements." Licensees must choose which option to use; CPSES uses Option B.

Option B requires that Type C pneumatic tests (local leakage rate tests of CIVs) be conducted periodically at intervals based on the safety significance and historical performance of each CIV to ensure the integrity of the overall containment system as a barrier to fission product release to reduce the risk from reactor accidents. However, unlike Option A, Option B does not contain criteria for determining which CIVs must be tested; instead, NRC-approved guidelines, discussed below, contain criteria which allow certain CIVs to be excepted from local leakage rate testing. In order to make guidelines such as these become, in effect, requirements, Option B states the following at section V.B.3.:

> The regulatory guide or other implementation document used by a licensee, or applicant for an operating license, to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications.

CPSES, Units 1 and 2, TS 5.5.16, "Containment Leakage Rate Testing Program," requires that leakage rate testing be performed as required by Reference 1, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 2), with certain exceptions listed in the TS. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) report NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995 (Reference 3).

Section 6.0, "General Requirements," of Reference 3 states in part that a local leakage rate test (LLRT) is not required for containment boundaries that do not constitute potential containment atmospheric pathways during and following a design basis accident, or for containment boundaries sealed with a qualified seal system. It further states:

> Primary containment barriers sealed with a qualified seal system shall be periodically tested to demonstrate their functionality in accordance with the plant Technical Specifications. Specific details of the testing methodology and requirements are contained in ANSI/ANS [American National Standards Institute/American Nuclear Society] 56.8-1994 (Reference 4) and should be adopted by licensees with applicable systems. Test frequency may be set using a performance basis in a manner similar to that described in this guideline for Type B and Type C test intervals. Leakage from containment isolation valves that are sealed with a qualified seal system may be excluded when determining the combined leakage rate provided that:
>
> - Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the technical specifications or associated bases, and
>
> - The installed isolation valve seal-water system fluid inventory is sufficient to assume the sealing function for at least 30 days at a pressure of 1.1 P_a.

Current TS 3.6.3, "Containment Isolation Valves," contains SRs 3.6.3.12 and 3.6.3.13, which read as follows:

> SR 3.6.3.12 Safety injection valves 8809A, 8809B, and 8840 shall be leak tested to be within limits with a gas at a pressure not less than P_a, 48.3 psig [pounds per square inch, gauge], or with water at a pressure not less than 1.1 P_a.
>
> SR 3.6.3.13 Containment spray valves HV-4776, HV-4777, CT-142, and CT-145 shall be leak tested to be within limits with water at a pressure not less than 1.1 P_a.

For both SRs, the testing frequency states: "In accordance with the Containment Leakage Rate Testing Program."

These SRs resulted from an earlier staff review. Supplement 22 of the CPSES operating license Safety Evaluation Report (Reference 5) contained the staff's evaluation and acceptance

of the CPSES program for LLRT of CIVs in accordance with Type C testing requirements in Appendix J. This pre-dated the development of Option B of Appendix J, which was promulgated in 1995.

Reference 5, Section 6.2.3.1, "Elimination of Type C Leakage Tests for Certain Containment Isolation Valves," item (3), discussed the testing for the following CIVs: HV-4776, HV-4777, CT-142, CT-145, 1-8840, 1-8809A, and 1-8809B.

CIVs HV-4776, HV-4777, CT-142, and CT-145 on the spray systems were to be leak rate tested with water at a pressure of not less than 1.1 P_a. The justification was that these penetrations have a water-filled loop seal on the containment side of the valves for more than 30 days following the accident. It was noted that the SRs and acceptance criteria should be included in the plant's TS.

In addition, the following three valves were added to be leak rate tested with water: 1-8840, 1-8809A, and 1-8809B of penetrations MIII-23, MIII-4, and MIII-5. Reference 5 stated:

> These valves are outboard containment isolation valves in the RHR discharge lines, which satisfy the following design criteria:
>
> - The systems are protected against missiles and pipe whip.
> - The systems are designed seismic category 1.
> - The systems are classified as ASME [American Society of Mechanical Engineers] safety class 2.
>
> At the penetrations, a pressurized water seal will be maintained throughout the entire 30-day accident-mitigation period. The water seal is on the containment side so that the accident pressure will push the water seal against the valves from inside containment towards outside containment. In accordance with paragraph III.C.3 of Appendix J to 10 CFR Part 50, these valves are not required to be Type C tested. Furthermore, [plant TS] include surveillance requirements and acceptance criteria for leak testing. Therefore, the proposed leak testing with water for the valves listed above is acceptable.

In the current submittal, the licensee asserts that no LLRT is necessary for these CIVs.

3.0 TECHNICAL EVALUATION

3.1 Historical Perspective

Despite the revision of Appendix J in 1995, the regulatory requirements as to whether or not a particular CIV must be locally leakage rate tested are essentially unchanged. The staff's 1990 safety evaluation remains valid. To be excluded from Type C testing, a licensee must typically show that a containment boundary does not constitute a potential containment atmospheric pathway during and following a design basis accident, or that it is sealed with a qualified seal system (Reference 3, Section 6.0; Reference 4, Section 3.3.1). The staff's review of a boundary, to see if it is a potential containment atmospheric pathway, uses conservative deterministic assumptions, such as giving no credit for a component that is not seismic category 1, safety class 2, electrical class 1E, and so on. For example, to establish that a CIV is sealed with water at a pressure of at least 1.1 P_a (calculated peak containment internal

pressure from the design basis loss-of-coolant accident), and thus not a potential containment atmospheric pathway, the seal must be assured by systems and components that meet all the safety pedigrees and with an unlimited supply of water, despite single active failures of pumps, valves, diesel generators, or other equipment. In some cases, an open cross-tie or common header must exist between the two redundant trains of a system so that a water seal will be assured at the containment penetrations of both trains, in case one train's pump failed to operate during an accident. Occasionally, there is a water seal, but the supply of sealing water is limited, so the licensee must periodically perform leakage rate tests with water as the test medium, to demonstrate that the water supply would last for at least 30 days into an accident, as is the case for the seven CIVs in question.

With this application, the licensee has provided detailed analyses of the systems potentially supplying water seals to the subject CIVs during an accident. Because the RHR and containment spray systems operate in different modes during various stages of an accident, these analyses address the cold leg injection, cold leg recirculation, and hot leg injection phases of a postulated loss-of-coolant accident. Although a proper water seal can be assured during most phases, and with most single active failures, the staff finds that there is, for each CIV, at least one combination where it cannot. For these situations, the licensee's analyses would depend on closed systems outside containment to contain CIV leakage and prevent it from entering the environment.

One of the licensee's stated reasons for the current application is that a number of other, very similar plants do not have to perform LLRTs for their congruent CIVs. The licensee stated, in its supplemental letter dated August 14, 2003, that they had confirmed that at each of eight plants, a) they are not required to Type C LLRT the penetrations in the discharge of the RHR pumps to the cold and hot legs, b) their licensing basis is simply the closed system outside containment, and c) this was the original licensing basis at the time the operating license was received.

The staff checked the record for one of the cited plants, Seabrook Station (Seabrook). The staff found that, in Seabrook Safety Evaluation Report, Supplement 5 (Reference 6), the RHR discharge valves had not been excepted from testing, but in Supplement 6 (Reference 7), they had. Reference 7 stated that the reason that the staff accepted this was that the applicant had revised their emergency operating procedures to leave open the valves in the discharge cross-tie line between the two RHR trains during the switchover from cold leg injection to cold leg recirculation. Then, during the cold leg recirculation phase, a single RHR pump could supply a water seal to both containment penetrations. CPSES is different in that their RHR cross-tie valves are closed during the cold leg recirculation phase, so one of the penetrations would not be water sealed with the postulated single active failure of an RHR pump. Also, the staff did not credit a closed system outside containment in this evaluation.

3.2 Closed Systems Outside Containment

Despite the history, the staff has reviewed the licensee's justification for crediting a closed system outside containment as a mechanism for precluding the leakage of containment atmosphere to the external environment, for the subject CIVs.

The licensee asserts that their radioactive system leakage inspection (RSLI) program, and various improvements made since the original review and licensing of the plant, assure that the

pertinent closed systems outside containment will reliably and effectively preclude radioactive releases, so that the CIVs in question need no longer be tested.

The RHR and containment spray systems are designed to be in-service recirculating reactor coolant after a loss-of-coolant accident, and are designed as closed systems outside containment. The licensee states that they meet the following requirements for a closed system outside containment:

- missile protected (from both internal and external missiles),
- seismic category 1,
- safety class 2,
- design temperature and pressure at least equal to containment, and
- tested per the requirements of NUREG-0737 (Reference 8), Section III.D.1.1.

In addition to the NUREG-0737 testing (i.e., RSLI program), the closed systems outside containment are tested and inspected in accordance with ASME Section XI (Reference 9).

All containment piping penetrations, including the closed systems outside containment, are located in radiation controlled areas of the auxiliary, fuel, and safeguards buildings which are monitored by radiation monitors for containment leakage after a loss-of-coolant accident, as described in the CPSES Final Safety Analysis Report (FSAR), Section 7.5. This is consistent with 10 CFR Part 50, Appendix A, General Design Criterion 54 (Reference 10) requirements for leak detection.

The licensee states that leakage from the closed systems outside containment is minimized by the RSLI and maintenance programs. Operating experience and work history demonstrate that the closed systems are suitable barriers and ensures the containment isolation function is maintained. The licensee provided detailed descriptions of the RSLI and maintenance programs, and their operating experience and work history; summaries are provided in the sections below.

3.2.1 The Radioactive System Leakage Inspection Program

The overall objective of the RSLI program is to monitor and reduce leakage from those portions of systems outside containment that contain highly radioactive fluids during post accident operation to as-low-as-reasonably achievable levels. Leakage from radioactive systems outside containment are monitored to meet the commitments in the CPSES FSAR, Section III.D.1.1 (Response to the NRC Action Plan for the Three Mile Island Nuclear Plant Accident) and the requirements of TS Section 5.5.2, “Primary Coolant Sources Outside Containment.”

The RSLI program includes the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

The leakage criteria for the RSLI program are as follows: The limiting leakage value based on a cumulative amount from all liquid systems tested under the RSLI Program is 1.0 gallons per

minute (gpm) per unit. An additional criterion for liquid leakage on individual systems is administratively applied.

The 1.0 gpm criterion is based on accident analysis assumptions for radiological consequences of engineered safety features equipment leakage outside containment (FSAR Section 15.6.5.4). All abnormal leakage is evaluated and corrected under the 10 CFR Part 50, Appendix B (Reference 11) corrective action program in accordance with NRC Generic Letter 91-18, Revision 1.

Each RSLI system is inspected at intervals not to exceed each refueling cycle. Testing is performed at normal system operating pressures. In order to have appropriate portions of systems pressurized, inspections of the containment spray, RHR, and safety injection systems are scheduled to coincide with the operability tests of those systems, when possible.

The test method for the RSLI inspection tests specifies that floor drains are to be observed in every space for the presence of liquid and boron. Additionally, the procedure specifies inspection of the system for leaks at packing leakoffs, flanges, fittings, valves, pumps, etc. With the inspection process checking for leaks at the floor drains where the stem leakoff piping is directed, and checking flanges, fittings, etc., while the system is pressurized, significant leakage would be detected during the test.

During leak testing of the RHR system, the system pressure is greater than 200 psig; therefore, any leakage during an accident (at a lower pressure) would be less. Historical leakage for this test has been very low, less than 0.1 gpm. Currently, the leakage for both units/trains of RHR is 0 gpm.

During leak testing of the containment spray system, the system pressure is greater than 150 psig; therefore, any leakage during an accident (at a lower pressure) would be less. Historical leakage for this test has been very low, less than 0.1 gpm. Currently, the only noted containment spray system leakage is for Unit 2 train B and is 0.002 gpm.

Based on the present maximum observed values of 0.002 gpm and the procedures used to collect that data, it can reasonably be shown that stem leakoff of the 8809, 8840, and containment spray system isolation valves is minimal with relation to containment leakage.

Operations, engineering, and maintenance personnel perform tests, walkdowns, and inspections on a frequent basis, and identify/quantify leakage and initiate corrective actions as necessary.

System engineers review RSLI test data and other significant leakage data and applicable corrective action documents on RSLI system components to maintain a RSLI program leakage table for each unit. This will ensure that the unit's cumulative leakage for portions of systems covered by this program remain within the leakage criteria.

Maintenance personnel implement corrective actions as soon as reasonably possible on leakage identified by RSLI tests or other inspections. These corrective actions include adjusting packing or replacement of seals, gaskets, o-rings, etc., on RSLI system components.

3.2.2 Changes in Design and Maintenance Since Original Licensing

There have been significant changes to the plant design and maintenance since original licensing which directly affect this request. In addition, there is significant maintenance history which is relevant to the subject isolation valves.

During the original licensing process, there was a technical issue about the potential for leaking from the valve stem packing of the subject CIVs. Briefly, if the CIVs were not sealed with water, containment atmosphere could leak out through the packing. At that time, the valves had traditional packing with leak-off connections, and packing leaks were not uncommon.

The licensee's changes to the plant design and maintenance process implemented since original licensing, meant to improve valve stem packing performance, include:

- Issuance of Design Specification CPES-M-1070, "Alternate Valve Stem Packing Replacement;"

- Revisions to Maintenance Procedures MSM-CO-8803, "Borg-Warner Bolted Bonnet Gate Valves" and MSM-CO-8824, "Westinghouse Gate Valves;"

- Formation of a maintenance department "Valve Team."

The Alternate Valve Stem Packing Replacement Program has improved valve packing performance by use of alternate packing configurations and materials in safety and non-safety applications. The program is based on the Electric Power Research Institute (EPRI) Report NP-5697, "Valve Stem Packing Improvements," and is consistent with current valve Original Equipment Manufacturer (OEM) methodologies for stem sealing. The program implements the recommended corrective actions from the EPRI report to:

- Reduce the packing stack height to decrease in-service consolidation and maintain gland load more effectively

- Employ improved stem packing configuration to transmit axial gland loads to radial sealing forces more effectively

- Provide a means of continual gland adjustment through live-loading to maintain sufficient pressure on the gland follower for effective long-term sealing.

In general, a large reduction in packing leaks was realized through implementation of this program. As for the subject CIVs, all but one has had its packing replaced. The only CIV that has not been re-worked with new packing, 1-8840, has never had an identified packing leak. None of the subject CIVs has ever had a significant packing leak.

The revised maintenance procedures, cited above, have addressed such issues as body-to-bonnet surface defect repair, disc and seat sealing surface repair, stem scoring, etc. The licensee has incorporated additional guidance obtained from the valves' original equipment manufacturers, such as repair guidance, dimensional information, and re-assembly procedures. The goal was to provide effective maintenance on service-induced defects without impacting the valve design function.

The Valve Team is a specialized group focusing only on valve maintenance. The team is a multi-discipline organization dedicated only to valves and actuators, and includes mechanics, electricians, and instrumentation and control technicians. This group receives specialized training on valve repair techniques from a variety of sources.

The Valve Team methodology uses a philosophy of returning a valve to a "like new" condition. The detailed maintenance activities involve close dimensional verifications and inspection activities in order to ensure that the work performed restores the design sealing functions.

3.3 Evaluation

The licensee is taking the position that, with assured water seals at most times and with reliable closed systems outside containment to preclude off-site releases for those times when a water seal is not assured, the subject testing is not necessary. The staff finds that, despite the divergence from historical reviews, it is in this case acceptable to credit the closed systems outside containment for those periods when water seals are not assured, due to the licensee's considerable efforts to reduce potential leakage with its RSLI program and its other design and maintenance changes, described in the sections above. The staff finds that continued testing of the subject CIVs, as currently required by TS SRs 3.6.3.12 and 3.6.3.13, does not provide a significant additional safety benefit to the public. Therefore, public health and safety will remain adequately protected without the subject testing.

Based on the foregoing evaluation, the staff finds that the combination of assured water seals and closed systems outside containment will preclude leakage of containment atmosphere to the environment through the subject CIVs (8809A, 8809B, 8840, HV-4776, HV-4777, CT-142, and CT-145) during a loss-of-coolant accident. Therefore, the staff finds that these CIVs do not constitute potential containment atmospheric pathways during and following a design basis accident and, in accordance with Section 6.0 of Reference 3, local leakage rate tests under Reference 1, are not required. Thus, the staff finds that SRs 3.6.3.12 and 3.6.3.13 may be deleted from the CPSES, Units 1 and 2, TSs.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change SRs. 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 considerations, and there has been no public comment on the finding (68 FR 8289 published on April 15, 2003). 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 REFERENCES

1. Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

2. Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” September 1995.

3. Nuclear Energy Institute Report NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” July 26, 1995.

4. American National Standard ANSI/ANS-56.8-1994, “American National Standard for Containment System Leakage Testing Requirements,” American Nuclear Society, August 4, 1994.

5. CPSES SSER 22, “Safety Evaluation Report related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2,” NUREG-0797, Supplement No. 22, January 1990.

6. Seabrook SSER 5, “Safety Evaluation Report related to the operation of Seabrook Station, Units 1 and 2,” NUREG-0896, Supplement No. 5, July 1986.

7. Seabrook SSER 6, “Safety Evaluation Report related to the operation of Seabrook Station, Units 1 and 2,” NUREG-0896, Supplement No. 6, October 1986.

8. NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.

9. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI

10. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion 54, "Piping Systems Penetrating Containment."

11. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

7.0 CONCLUSION

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

Principal Contributor: J. Pulsipher

Date: March 5, 2004

cc:

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