TXU

TXU Electric
Comanche Peak
Steam Electric Station
P.O. Box 1002
Glen Rose, TX 76043
Tel: 254 897 8920
Fax: 254 897 6652
lterry1@txu.com

C. Lance Terry
Senior Vice President & Principal Nuclear Officer

Ref: 10CFR50.90

CPSES-2000001843
Log # TXX-00150
File # 00236, 10010

August 10, 2000

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 00-06
REVISION TO TECHNICAL SPECIFICATION (TS) 5.6.5
REVISION OF LARGE BREAK LOCA METHODOLOGY

REF: TXU Electric Letter, logged TXX-00076, from C.L. Terry to the NRC dated April 13, 2000.

Gentlemen:

Pursuant to 10CFR50.90, TXU Electric hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. This change request applies to both units.

The proposed change will revise TS 5.6.5 entitled CORE OPERATING LIMITS REPORT. TXU Electric proposes to revise the Large Break LOCA methodology used at CPSES Units 1 and 2. The revised methodology for the Large Break LOCA Methodology is currently under review by the NRC (see Referenced letter). It is anticipated that NRC review will be complete in early September, 2000.

Attachment 1 is the required affidavit. Attachment 2 provides a detailed description of the proposed changes, a safety analysis of the proposed changes, TXU Electric's determination that the proposed changes do not involve a significant hazard consideration, a regulatory analysis of the proposed changes and an environmental evaluation. Attachment 3 provides the affected Technical Specification pages marked-up to reflect the proposed changes. Attachment 4 provides retyped Technical

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Specification pages which incorporate the requested changes. Attachment 5 provides marked-up pages of the Final Safety Analysis Report (for information only) to reflect the proposed changes to the FSAR.

TXU Electric requests approval of the proposed License Amendment by September 29, 2000, to be implemented within 30 days of the issuance of the license amendment, consistent with the planned Unit 2 refueling outage.

In accordance with 10CFR50.91(b), TXU Electric is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new or revised commitments.

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

Sincerely,

C. L. Terry

JDS/jds

Attachments
1. Affidavit
2. Description and Assessment
3. Markup of Technical Specifications pages
4. Retyped Technical Specification Pages
5. Proposed FSAR changes (for information)

c - E. W. Merschoff, Region IV
J. I. Tapia, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Authur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)		
	)		
TXU Electric	)	Docket Nos.	50-445
	)		50-446
(Comanche Peak Steam Electric Station,	)	License Nos.	NPF-87
Units 1 & 2)	)		NPF-89

AFFIDAVIT

C. L. Terry being duly sworn, hereby deposes and says that he is Senior Vice President and Principal Nuclear Officer of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 00-06; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

C. L. Terry
Senior Vice President and
Principal Nuclear Officer

STATE OF TEXAS)
)
COUNTY OF Somervell)

Subscribed and sworn to before me, on this 10th day of August, 2000.

SUSAN C. GRAVATT
NOTARY PUBLIC
STATE OF TEXAS
My Commission Expires 3-23-2001

Susan C Gravatt
Notary Public

ATTACHMENT 2 to TXX-00150

DESCRIPTION AND ASSESSMENT

Description and Assessment

1.0 INTRODUCTION

1.1 Proposed change LAR-00-06 is a request to revise Technical Specifications (TS) 5.6.5, "Core Operating Limits Report" for Comanche Peak Steam Electric Station (CPSES) Units 1 and 2.

1.2 FINAL SAFETY ANALYSIS REPORT (FSAR) SECTION

Conforming changes will be made to the Reference Sections of CPSES Final Safety Analysis Report, Chapter 15 and Appendixes 4A and 4B upon approval of this License Amendment Request.

2.0 DESCRIPTION

The proposed change would revise Technical Specification 5.6.5, Item 15) by replacing the reference to the methodology used at CPSES to analyze Large Break Loss of Coolant Accidents (LOCAs). The revised Large Break LOCA analysis methodology is described in the TXU Topical Report ERX-2000-002-P, which has been submitted to the NRC for approval.

Specifically, the proposed change would replace Technical Specification 5.6.5, Item 15) with "ERX-2000-002-P, 'Revised Large Break Loss of Coolant Accident Analysis Methodology,' March, 2000." This revised methodology is currently under review by the NRC with approval expected in early September, 2000.

3.0 BACKGROUND

The Core Operating Limits Report (COLR) was established through implementation of the Generic Letter (GL) 88-16, which provides guidelines for the removal of cycle-specific parameter limits from the Technical Specifications. The limits presented in the COLR may be modified without prior NRC approval provided the requirements of Technical Specification 5.6.5 are met (i.e., the modifications are determined using specific NRC-approved methodologies and meet all applicable limits of the plant safety analysis).

The Large Break Loss of Coolant Accident analysis methodology currently specified in Technical Specification 5.6.5, Item 15) contains an identified error that, if corrected, would reduce the peak clad temperature by an amount which would exceed the requirements of

10CFR50, Appendix K Article I.C.5.c. TXU Electric proposed to the NRC to continue to use the methodology until the NRC approved the upgraded large break Loss of Coolant Accident analysis methodology. When approved, this methodology will be employed in preparing the analysis of record for the currently operating Unit 1, Cycle 8 and will be applied to the upcoming Unit 2, Cycle 6.

Upon approval and implementation of the Revised Large Break LOCA analysis methodology, the analysis methodology presented in Technical Specification 5.6.5, Item 15) will no longer be valid and must be replaced with the approved methodology. Analyses based on the revised methodology are intended to be used to support the startup of Unit 2, Cycle 6; therefore, TXU Electric requests that the NRC review and approve the proposed Technical Specification change prior to September 29, 2000.

4.0 TECHNICAL ANALYSIS

The proposed change is solely administrative in nature. The proposed change is being submitted to maintain the accuracy and utility of the Core Operating Limits Report Technical Specification. The validity of the Revised Large Break Loss of Coolant Accident Analysis methodology will be affirmed through the NRC staff's review of the methodology topical report.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

TXU Electric has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92 as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change involves an administrative change only. Designation of the Revised Large Break Loss of Coolant Accident analysis methodology, described in ERX-2000-002-P, as the approved Large Break Loss of Coolant Accident analysis methodology is required to maintain the accuracy of the Technical Specification 5.6.5 (Core Operating Limits Report) and to maintain consistency with the resolution of issues as prescribed in 10CFR50.46.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change involves an administrative change only. Technical Specification 5.6.5, Item 15) is being changed to reference the revised Large Break Loss of Coolant Accident analysis methodology currently under NRC review. No actual plant equipment will be affected by the proposed change. An analysis for Unit 1, Cycle 8, is imbedded in the referenced Topical Report, from which it is concluded that no failure modes, not bounded by previously evaluated accidents, will be created.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   Margin of safety is associated with the confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. This request involves an administrative change (subject to NRC approval of the revised Large Break Loss of Coolant Accident Analysis methodology) only to incorporate the revised Large Break Loss of Coolant Accident analysis methodology into the allowable analysis methodologies specified in Technical Specification 5.6.5.

   No actual plant equipment will be affected by the proposed change. The compliance of the revised methodology with the requirements of 10CFR50.46 and Appendix K will be addressed through the NRC staff's review of the topical report. Therefore, it is concluded that the use of the proposed methodology will not degrade the confidence in the ability of the fission product barriers to limit the level of radiation dose to the public.

   Therefore the proposed change does not involve a reduction in a margin of safety.

Based on the above evaluations, TXU Electric concludes that the activities associated with the above described changes present no significant hazards consideration under the

standards set forth in 10CFR50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Regulatory Safety Analysis

The description of the Large Break Loss of Coolant Accident methodology is provided in topical report ERX-2000-002-P , "Revised Large Break Loss of Coolant Accident Analysis Methodology," March 2000. Subsequent to NRC approval, the revised methodology will be used to demonstrate compliance with 10CFR50.46 (Acceptance Criteria for emergency core cooling systems for light-water nuclear power reactors) criteria and 10CFR50, Appendix K (ECCS Evaluation Models) requirements.

6.0 ENVIRONMENTAL EVALUATION

TXU Electric has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. Specifically, the proposed change revises the Large Break Loss of Coolant Accident analysis methodology used in preparation of the Core Operating Limits Report (COLR). This revised reference could impact operating limits, specified in the COLR, used in the performance of surveillance requirements. TXU Electric has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

7.0. REFERENCES

TXU Electric letter from C.L. Terry to the NRC logged TXX-00076, dated April 13, 2000, submitting topical report ERX-2000-002-P , "Revised Large Break Loss of Coolant Accident Analysis Methodology."

ATTACHMENT 3 to TXX-00150

MARKUP OF TECHNICAL SPECIFICATION PAGE

Page 5.0-34

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

10) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

11) RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1", December 1990.

12) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

13) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

14) RXE-91-002-A, "Reactivity Anomaly Events Methodology", October 1993.

15) ~~RXE-90-007-A, "Large Break Loss of Coolant Accident Analysis Methodology", April 1993.~~ ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology," March 2000.

16) TXX-88306, "Steam Generator Tube Rupture Analysis", March 15, 1988.

17) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

18) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3,4, and 5," February 1994.

19) RXE-95-001-P-A, "Small Break Loss of Coolant Accident Analysis Methodology," September 1996.

20) Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√ System," Revision 0, March 1997.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 4 to TXX-00150

RETYPED TECHNICAL SPECIFICATION PAGES

Page 5-0-34

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (continued)

10) RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

11) RXE-88-102-P, Sup. 1, "TUE-1 DNB Correlation - Supplement 1", December 1990.

12) RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

13) RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

14) RXE-91-002-A, "Reactivity Anomaly Events Methodology", October 1993.

15) ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology," March 2000.

16) TXX-88306, "Steam Generator Tube Rupture Analysis", March 15, 1988.

17) RXE-91-005-A, "Methodology for Reactor Core Response to Steamline Break Events," February 1994.

18) RXE-94-001-A, "Safety Analysis of Postulated Inadvertent Boron Dilution Event in Modes 3,4, and 5," February 1994.

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d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 5 to TXX-00150

PROPOSED FSAR CHANGES (FOR INFORMATION)

Table 1.6-1 Sheet 9
Page 4A-29
Page 4B-29
Page 15.6-40

TOPICAL REPORTS INCORPORATED BY REFERENCE

Report	Reference Section(s)	Review Status
"Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973	1A(N), 5.4	U
"Calculation Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), June 1974	15.6	A
"Design Bases Analysis of a Postulated Steam Generator Tube Rupture Event for Comanche Peak Station Electric Station, Unit 1," RXE-88-101-P (Proprietary) and RXE-88-101-NP (Non-Proprietary) March, 1988	15.0, 15.6	U
"TUE-1 Departure from Nucleate Boiling Correlation," RXE-88-102-P, January 1989.	4.4, 15.0	A
"VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," RXE-89-002, June 1989.	4.4, 15.0	A
"Steady-State Reactor Physics Methodology," RXE-89-003, July 1989.	4.3	A
"Control Rod Worth Analysis," RXE-09-005, December 1990.	4.3	A
"**Revised** Large Break Loss of Coolant Accident Analysis Methodology," ~~RXE-90-007~~ **ERX-2000-002-P**, ~~January 1991~~ **March 2000.**	15.0, 15.6	A

10. RXE-89-003-P-A, "Steady State Reactor Physics Methodology," June 1994.

11. RXE-90-006-P-A, "Power Distribution Control Analysis and Overtemperature N-16 and Overpower N-16 Trip Setpoint Methodology," June 1994 ~~(Technical Specification 6.9.1.6b. Item 9)~~.

12. ~~RXE-90-007-A~~ ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology," ~~April 2, 1993 (Technical Specification 6.9.1.6b. Item 15)~~ March 2000.

13. RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications," October 1993 ~~(Technical Specification 6.9.1.6b. Item 13)~~.

14. RXE-91-002-A, "Reactivity Anomaly Events Methodology for Comanche Peak Steam Electric Station Licensing Applications," October 1993 ~~(Technical Specification 6.9.1.6b. Item 14)~~.

15. RXE-91-005-A, "Methodology for Reactor Core Response to Steam line Break Events," February 1994 ~~(Technical Specification 6.9.1.6b. Item 17)~~.

16. RXE-94-001-A, "Safety Analysis of the Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5 (Appendix A to TXX-93098)," 02/94 ~~(Technical Specification 6.9.1.6b. Item 19)~~.

17. RXE-95-001-P, “Small Break Loss of Coolant Accident Analysis Methodology,” December 1995.

7. RXE-91-002-A, "Reactivity Anomaly Events Methodology for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

8. RXE-91-005-A, "Methodology for Reactor Core Response to Steam line Break Events", February 1994.

9. RXE-88-102-P-A, "TUE-1 Departure from Nucleate Boiling Correlation", July 1992.

10. RXE-89-002-A, "VIPRE-01 Core Thermal-Hydraulic Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", September 1993.

11. ~~RXE-90-007-A~~ ERX-2000-002-P, "Revised Large Break Loss of Coolant Accident Analysis Methodology", ~~April 2, 1993~~ March 2000.

12. RXE-91-001-A, "Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications", October 1993.

13. RXE-95-001-P, “Small Break Loss of Coolant Accident Analysis Methodology,” December 1995.

14. RXE-88-101-P, Design Basis Analysis of a Postulated Steam Generator Tube Rupture Event for Comanche Peak Steam Electric Station, Unit 1", March 1988.

15. RXE-94-001-A, "Safety Analysis of the Postulated Inadvertent Boron Dilution Event in Modes 3, 4, and 5 (Appendix A to TXX-93098)", 02/94.

REFERENCES

1. Deleted

2. Deleted

3. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3, January 4, 1974.

4. "Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

5. Deleted

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13. ~~Brozak, D. E., Tajbakhsh, A. E.~~ Salim, P., and da Silva, H. C., "Revised Large Break Loss of Coolant Accident Analysis Methodology," ~~RXE-90-007-A, April 2, 1993~~ ERX-02000-002-P, March 2000.