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Ref: TS 5.6.6

CPSES-200200535
Log # TXX-02042

February 27, 2002

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
TRANSMITTAL OF UNIT 2 PRESSURE AND TEMPERATURE LIMITS REPORT, REVISION 3

REF:
1) TXU Electric Letter, logged TXX-01133, from Mr. C. L. Terry to the NRC dated August 16, 2001, "Revision to the Reactor Vessel Material Surveillance Program Withdrawal Schedule."

2) NRC Safety Evaluation, "Comanche Peak Steam Electric Station, Unit 2 - Request to Revise Reactor Pressure Vessel Material Surveillance Program Schedule (TAC NO. MB2761)" dated February 5, 2002.

Gentlemen:

Per reference 1, TXU Generation Company LP (TXU Energy) requested approval for a revision to the reactor vessel material surveillance program withdrawal schedule for CPSES Unit 2. Reference 2 found that the proposed withdrawal schedule for the capsules for the CPSES Unit 2 reactor pressure vessel surveillance program satisfies the requirements of Appendix H to 10 CFR 50 and is acceptable.

CPSES Technical Specification (TS) 5.6.6 requires that the Pressure and Temperature Limits Report (PTLR) be provided to the NRC upon issuance for each reactor vessel fluence period and for each revision or supplement thereto. Enclosed is the CPSES Unit 2 PTLR, Revision 3. This revision to the PTLR does not change the pressure temperature limits or the Low Temperature Overpressure Protection (LTOP) system setpoints.

A008

This communication contains no new licensing basis commitments regarding CPSES Unit 2.

Sincerely,

TXU Generation Company LP
By: TXU Generation Management Company LLC,
Its General Partner

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

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

JCH/jch

Enclosure

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Resident Inspectors, CPSES

ENCLOSURE to TXX-02042

ERX-99-003, Rev. 3

CPSES UNIT 2

PRESSURE AND TEMPERATURE LIMITS REPORT

(APPLICABLE UP TO 16 EFPY)

February 2002

Prepared: Parvez Salim Date: 2/25/02

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Safety Analysis

Reviewed: WJ Boatwright Date: 2/25/02

W. James Boatwright
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Approved: Whee G. Choe Date: 2/25/02

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Safety Analysis Manager

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1.0 INTRODUCTION

This report presents the Reactor Coolant System (RCS) Pressure and Temperature (P/T) limits for Comanche Peak Steam Electric Station (CPSES) Unit 2 in accordance with the requirements of Technical Specification 5.6.6. A description of the Low Temperature Overpressure Protection (LTOP) System is also provided in this report. In addition, the requirements of the reactor vessel material surveillance program are discussed.

The following two Technical Specification Limiting Conditions of Operation (LCO) are addressed in this report:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) System

The P/T Limits (Section 2.2) and the LTOP System Setpoints (Section 2.3) in this report shall not be revised without prior NRC approval.

The analytical methods used to determine the RCS P/T limits are consistent with the following guidance:

10 CFR 50 Appendices G and H	(References 1 & 2)
Regulatory Guide 1.99, Revision 2	(Reference 3)
NUREG-0800, Standard Review Plan Section 5.3.2	(Reference 4)
ASME Code Section III, Division 1, Appendix G	(Reference 5)

The LTOP limits are consistent with the following guidance:

10 CFR 50 Appendix G	(Reference 1)
NUREG-0800, Standard Review Plan Section 5.2.2	(Reference 6)

This report covers CPSES Unit 2 operation for 16 Effective Full Power Years (EFPY), which approximately corresponds to the Cycle 11 operation of the unit.

2.0 OPERATING LIMITS

RCS P/T Limits

The RCS P/T limits presented in this report consist of the RCS (except the pressurizer) temperature rate-of-change limits and P/T limits during heatup, cooldown, inservice leak & hydrostatic testing, and criticality. The P/T limits for CPSES Unit 2 are based on the Westinghouse procedure outlined in Reference 7.

Appendix G of 10 CFR Part 50 [1] establishes specific fracture toughness requirements for ferritic materials in the Reactor Coolant Pressure Boundary (RCPB) in light water nuclear power reactors. An adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests is also required by this reference. Furthermore, Reference 1 mandates the use of Appendix G, Section III of the American Society of Mechanical Engineers (ASME) Code [5] to establish P/T limits.

The RCS P/T limits for CPSES Unit 2 are presented in Reference 8.

LTOP System

The LTOP System acts as a backup to the reactor operators to mitigate RCS pressurization transients at low temperatures so the integrity of RCPB is not compromised by violating the pressure and temperature limits of Appendix G of 10 CFR 50 [1]. The reactor vessel is the limiting RCPB component for demonstrating such protection. The LTOP system provides the maximum allowable actuation logic setpoints for the Power Operated Relief Valves (PORVs) for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The LTOP setpoints for CPSES Unit 2 is presented in Reference 9 (supplemented by Reference 10).

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reduction in toughness that results from neutron radiation is measured as an increase in the Nil Ductility Reference Temperature (RT_{NDT}) and reduction of the upper-shelf energy of reactor vessel beltline materials, including welds. At CPSES, these quantities were predicted at 16 EFPY using the methods of Regulatory Guide 1.99, Revision 2 [3]. The predictions showed that the materials in the Unit 1 and Unit 2 reactor vessels responded similarly to neutron irradiation but at 16 EFPY, the plate material in the Unit 1 beltline was most limiting. Forecast properties of the limiting material were used to establish P/T limits for heatup and cooldown curves and LTOP setopints. For uniformity, the Unit 1 curves were adopted for Unit 2.

The actual neutron-induced shifts in the RT_{NDT} and the upper-shelf energy is periodically measured by withdrawing a surveillance capsule and evaluating the specimens of the limiting beltline material that it contains. The CPSES capsules were built under the ASTM E 185-70 Standard [11]. The evaluation of the irradiated reactor vessel material specimens is conducted in accordance with ASTM E 185-82 [12] and Appendix H of 10 CFR 50 [2]. The operating P/T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, Revision 2 [3].

The analysis of Capsule U from CPSES Unit 2 is presented in Reference 13.

2.1 RCS Temperature Rate-of-Change Limits (LCO 3.4.3)

2.1.1 Maximum Heatup Rate

The RCS heatup rate limit is 100°F in any 1-hour period.

2.1.2 Maximum Cooldown Rate

The RCS cooldown rate limit is 100°F in any 1-hour period.

2.1.3 Maximum Temperature Change During Inservice Leak and Hydrostatic Testing

During inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves, the RCS temperature change limit is 10°F in any 1-hour period.

2.2 P/T Limits for Heatup, Cooldown, Inservice Leak & Hydrostatic Testing, and Criticality (LCO 3.4.3)

2.2.1 P/T Limits for Heatup, Inservice Leak & Hydrostatic Testing, and Criticality

The P/T limits for heatup, inservice leak & hydrostatic testing, and criticality are specified in Figure 2-1.

2.2.2 P/T Limits for Cooldown

The P/T limits for cooldown are specified in Figure 2-2.

2.3 LTOP System Setpoints (LCO 3.4.12)

The maximum allowable PORV pressure setpoints as a function of RCS temperature are shown in Figure 2-3.

2.4 Reactor Vessel Material Surveillance Program

The vessel material surveillance schedule is provided in Table 2-1. This schedule was approved by the NRC in Reference 14.

Table 2-1

Reactor Vessel Material Surveillance Program – Withdrawal Schedule

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
U	58.5°	4.10	0.9 EFPY*
X	238.5°	4.10	9 EFPY
W	121.5°	4.10	14 EFPY
Z	301.5°	4.10	Standby
V	61.0°	3.74	Standby
Y	241.0°	3.74	Standby

* Capsule withdrawn and analyzed

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE R3807-2
INITIAL RT_{NDT}:	10°F
ART AT 16 EFPY:	1/4T : 81°F
	3/4T : 62°F

APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 110 PSIG FOR POSSIBLE INSTRUMENTATION ERRORS.

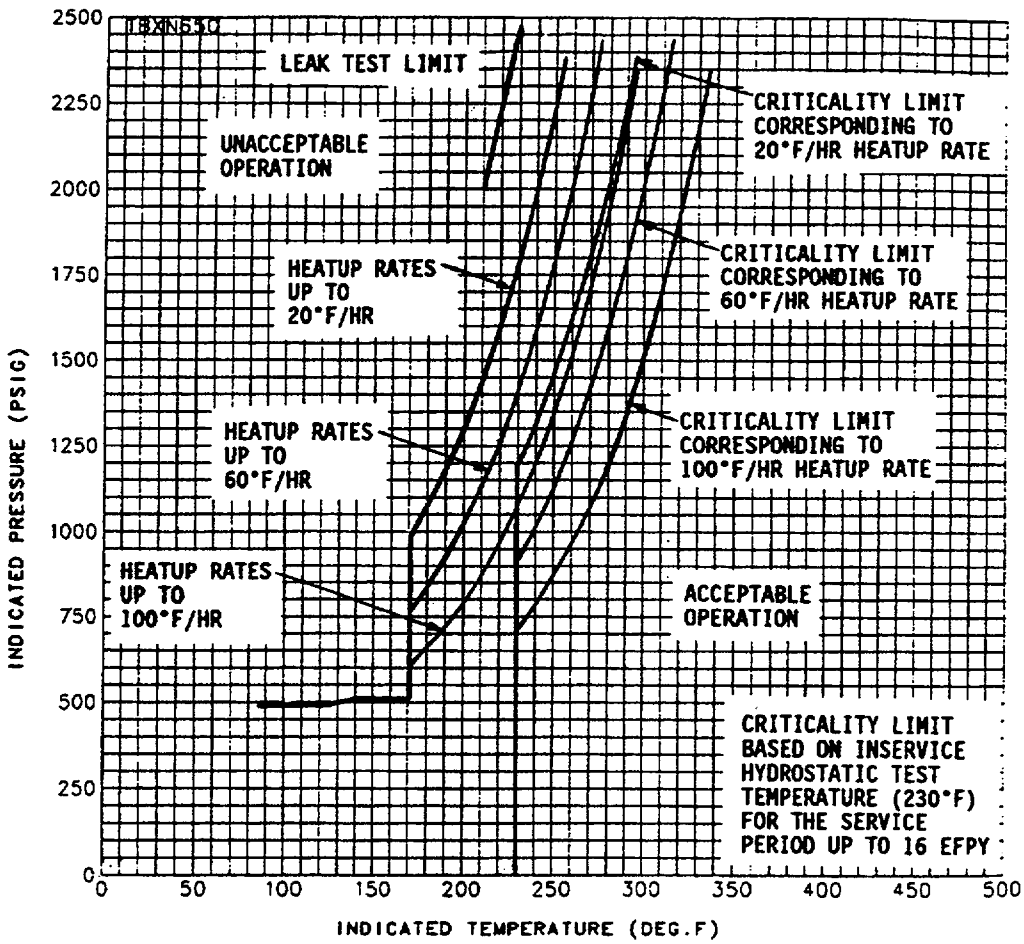

Figure 2-1

RCS Heatup Limitations – Applicable Up To 16 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE R3807-2
INITIAL RT_{NDT}:	10°F
ART AT 16 EFPY:	1/4T : 81°F
	3/4T : 62°F

APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY. CONTAINS MARGINS OF 10°F AND 110 PSIG FOR POSSIBLE INSTRUMENTATION ERRORS.

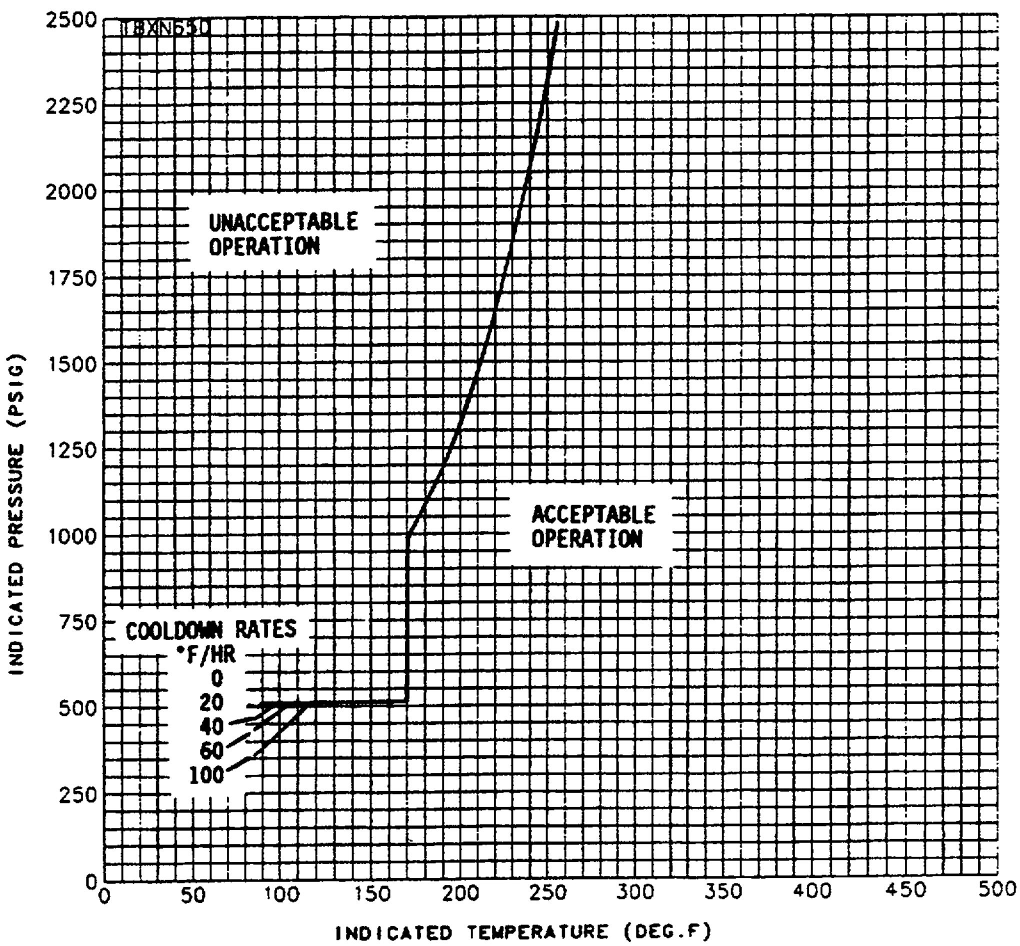

Figure 2-2

RCS Cooldown Limitations – Applicable Up To 16 EFPY

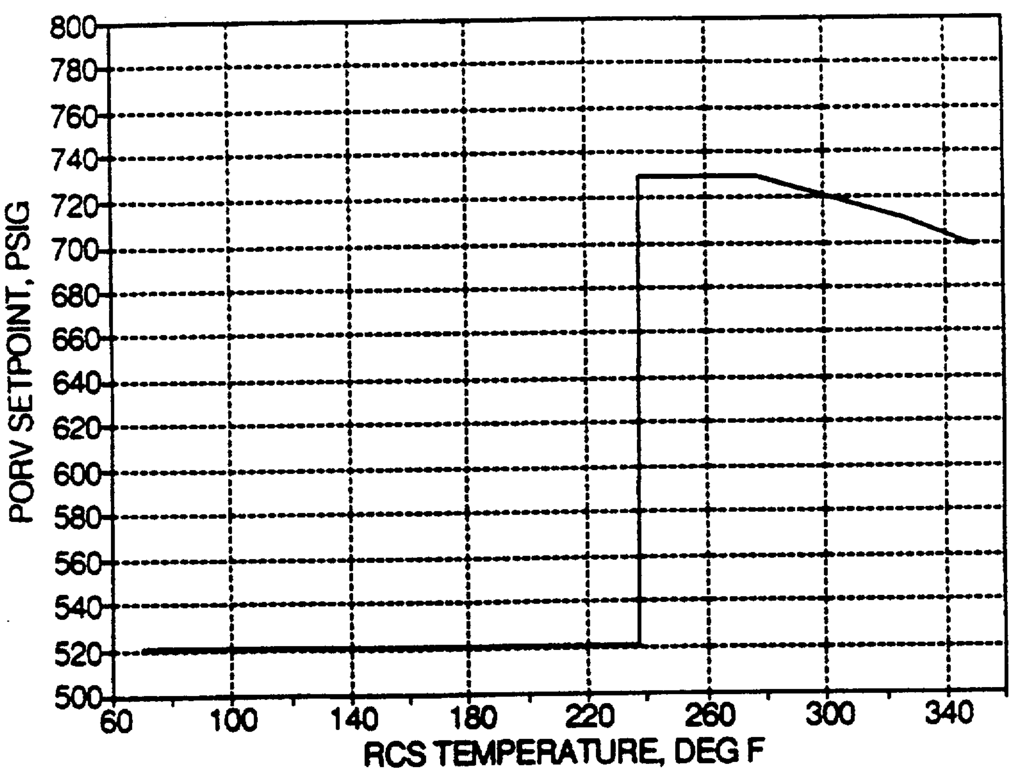

Figure 2-3

PORV Setpoints for Overpressure Mitigation – Applicable Up To 16 EFPY

3.0 REFERENCES

1. "Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C.

2. "Code of Federal Regulations, 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C.

3. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Washington, D.C., May 1988.

4. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1 – Appendices, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure," pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

6. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

7. "Procedure for Developing Heatup and Cooldown Curves," Westinghouse Electric Corporation, Generation Technology Systems Division Procedure GTSD-A-1.12 (Rev. 0), July 13, 1988.

8. "Comanche Peak Steam Electric Station Unit Number 1 & 2, Heatup and Cooldown Curves Using 110 psig and 10°F Margins for Instrument Error," Westinghouse Letter WPT-14868, October 12, 1992.

9. "Setpoint Program Determination for the Westinghouse Overpressure Mitigation System in Comanche Peak Units 1 and 2," Westinghouse Letter Report transmitted via Letter WPT-14228, January 3, 1992.

10. "Results of Review of Comanche Peak Cold Overpressure Mitigation System Nonconservatism," Westinghouse Letter Report transmitted via Letter WPT-14881, September 2, 1992.

11. ASTM E 185-70, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels."

12. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E706 (IF)."

13. "Analysis of Capsule U from the Texas Utilities Electric Company Comanche Peak Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Report WCAP-14315, July 1995.

14. "Comanche Peak Steam Electric Station, Unit 2 - Re Request to Revise Reactor Pressure Vessel Material Surveillance Program Schedule," Letter from David H. Jaffe (NRC) to C. Lance Terry (TXU), February 5, 2002.