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Ref: 10CFR50.90

CPSES-200502416
Log # TXX-05198
File # 00236

December 16, 2005

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) 05-007
REVISION TO TECHNICAL SPECIFICATION (TS) 5.6.6;
REACTOR COOLANT SYSTEM (RCS) PRESSURE AND
TEMPERATURE LIMITS REPORT (PTLR)

Dear Sir or Madam:

Pursuant to 10CFR50.90, TXU Generation Company LP (TXU Power) 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 (TS). This change request applies to both Units.

The proposed amendment would revise the TS consistent with the NRC-approved Revision 0 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR." The proposed change to reference only the Topical Report number and title in TS 5.6.6 does not alter the analytical methods used to determine the pressure-temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of currently approved Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license.

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Attachment 1 provides a detailed description of the proposed changes, a technical analysis of the proposed changes, TXU Power'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 2 provides the affected Technical Specification (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides retyped Technical Specification pages which incorporate the requested changes. Enclosure 1 contains a sample PTLR revision provided for information to assist the NRC reviewer. Enclosure 2 contains the latest analysis of the Unit 2 Reactor Vessel Radiation Surveillance Program, which is provided for your information as required by Generic Letter 92-01, Revision 1.

TXU Power requests approval of the proposed License Amendment by November 1, 2006, to be implemented within 120 days of the issuance of the license amendment.

This communication contains no new or revised commitments.

Should you have any questions, please contact Mr. Bob Kidwell at (254) 897-5310.

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

I state under penalty of perjury that the foregoing is true and correct.

Executed on 16 December, 2005

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W Madden

Fred W. Madden
Director, Regulatory Affairs

RJK

Attachments
1. Description and Assessment
2. Proposed Technical Specifications Changes
3. Retyped Technical Specification Pages

Enclosures
1. Sample Pressure and Temperature Limits Report (PTLR)
2. WCAP-16277-NP; "Analysis of Capsule X from the TXU Energy Comanche Peak Unit 2 Reactor Vessel Radiation Surveillance Program"

c -
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ATTACHMENT 1 to TXX-05198

DESCRIPTION AND ASSESSMENT

LICENSEE'S EVALUATION

1.0 DESCRIPTION

By this letter, TXU Generation Company LP (TXU Power) requests an amendment to the Comanche Peak Steam Electric Station (CPSES) Unit 1 Operating License (NPF-87) and Unit 2 Operating License (NPF-89) by incorporating the attached changes into the CPSES Unit 1 and 2 Technical Specifications (TS). The proposed license amendment 05-007 revises the TS requirements related to the Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," Revision 0 (Reference 8.1).

2.0 PROPOSED CHANGE

Consistent with Revision 0 of TSTF-419, the proposed TS changes include:

1. The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is revised to delete the reference to the Specifications containing the limits specified in the PTLR.

2. The requirement in TS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to submit any changes in P/T limits or LTOP System Setpoints for NRC staff prior approval is revised to allow the use of an NRC-approved Topical Report which will be identified by number and title only.

TXU Power has reviewed the TSTF traveler (Reference 8.1) and the model safety evaluation (Reference 8.2) and has concluded that the justifications presented in TSTF-419 Revision 0 and the model SE prepared by the NRC staff for the Standard Technical Specifications are applicable to this proposed change.

No changes to the CPSES Final Safety Analysis Report are anticipated at this time as a result of this License Amendment Request.

3.0 BACKGROUND

NRC Generic Letter 96-06, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, allows a licensee to relocate the pressure-temperature (P/T) limit curves from their plant technical specifications (TS) to a Pressure Temperature Limits Report (PTLR) or a similar document. The Low Temperature Overpressure Protection (LTOP) System limits were also allowed to be relocated to the same document. The methodology used to determine the P/T and LTOP System limit parameters must comply with the specific requirements of Appendices G and H to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR), be documented in an NRC approved topical report or in a plant-specific submittal, and be incorporated by reference into the TS. Subsequent changes in the methodology must be approved by a license amendment.

By letter dated May 23, 2002, the Westinghouse Owners Group (WOG) submitted Topical Report (TR) WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for NRC staff review and approval. This TR was developed to define a methodology for reactor pressure vessel (RPV) pressure-temperature (P/T) limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). A prior revision, WCAP-14040, Revision 2, had previously been approved as a PTLR methodology by the NRC staff's safety evaluation dated October 16, 1995. WCAP-14040, Revision 3, was submitted for NRC staff approval to reflect recent changes in the WOG methodology. Given the scope of the changes incorporated in WCAP-14040, Revision 3, and a significant amount of rewriting which was done to improve clarity of some sections, the NRC staff reviewed the TR in its entirety and published the NRC staff's safety evaluation dated February 27, 2004 (Reference 8.3).

The WOG subsequently republished this approved TR as WCAP-14040, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," incorporating the NRC staff's final safety evaluation and adding an Appendix B; "Correspondence with the NRC."

4.0 TECHNICAL ANALYSIS

The current definition of PTLR identifies the specifications in which the pressure and temperature limits are addressed. Specification 5.6.6.a requires that the individual specifications that address RCS pressure and temperature limits be referenced. The proposed changes to the definition will eliminate the duplication between the definition of PTLR and Section 5.6.6.a.

The revision to TS 5.6.6 to allow the Topical Reports to be identified by number and title will allow TXU Generation Company LP (TXU Power) to use a current NRC-approved Topical Report to support limits in the PTLR without having to submit an amendment to facility operating license each time the Topical Report is revised. The PTLR would provide the specific information identifying the particular approved Topical Report(s) used to determine the pressure-temperature (P/T) limits or Low Temperature Overpressure Protection (LTOP) System limits. This arrangement still provides the assurance that only the approved versions of the referenced Topical Reports will be used for the determination of the P/T limits or LTOP System limits since the complete citation will be provided in the PTLR.

The requirement to operate within the limits in the PTLR is specified in and controlled by the Technical Specifications. The figures, values, and parameters associated with the P/T limits and LTOP setpoints are located in the PTLR and the methodology for their development must be reviewed and approved by the NRC. The proposed changes do not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration Determination

TXU Generation Company LP (TXU Power) has reviewed and concurs with the determination of whether or not a significant hazards consideration is involved with the proposed amendment (as provided by Reference 8.1) by focusing on the three standards set forth in 10CFR50.92, Issuance of amendment, 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 changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10CFR50.59 and where required receive NRC review and approval. The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with safety analysis assumptions and resultant consequences.

   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 changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

   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

   The proposed changes to reference the Topical Report number and title do not alter the use of the analytical methods used to determine the P/T limits or LTOP setpoints that have been reviewed and approved by the NRC. This method of referencing Topical Reports would allow the use of current Topical Reports to support limits in the PTLR without having to submit an amendment to the operating license. Implementation of revisions to Topical Reports would still be reviewed in accordance with 10 CFR 50.59 and where required receive NRC review and approval. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The setpoints at which protective actions are initiated are not altered by the proposed changes. Sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event.

   Therefore, it is concluded that this change does not involve a significant reduction in the margin of safety.

Based on the above evaluations, TXU Power concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The acceptability of a facility's proposed PTLR methodology is based on the NRC regulations and guidance as discussed below.

Appendix G to 10 CFR Part 50 requires that facility pressure-temperature (P/T) limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs.

Regulatory Guide 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation.

Generic Letter 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review, and Generic Letter 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code.

The attributes of the vessel fluence methodology are described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Regulatory Guide 1.190 is based on General Design Criteria (GDC) 14, 30 and 31 of Appendix A to 10 CFR Part 50. In this context, GDC-14 relates to an extremely low probability of leakage from the pressure coolant boundary; GDC-30 relates to the design of the reactor coolant boundary; and GDC-31 relates to material embrittlement and the effect of irradiation.

The review requirements for the low temperature overpressure protection (LTOP) transients are described in SRP Section 5.2.2. SRP Section 5.2.2 is based on GDC-15 as it relates to the reactor coolant boundary design margin and GDC-31 as it relates to embrittlement and the effect of irradiation.

Generic Letter 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR. Generic Letter 96-03 establishes the information which should be included in an acceptable PTLR methodology (which will be used to develop the PTLR), and the information which should be included within the PTLR itself. These information criteria are principally addressed in a table contained in Attachment 1 of Generic Letter 96-03 entitled, "Requirements for Methodology and PTLR," and are subdivided into seven technical elements which must be addressed by the licensee. Generic Letter 96-03 also addresses the appropriate modifications to the administrative controls section of a facility's TS which are necessary to implement a PTLR. TSTF-419 provides guidance on an alternative set of TS administrative control section changes which may be made to implement a PTLR in accordance with Attachment 1 of Generic Letter 96-03.

6.0 ENVIRONMENTAL CONSIDERATION

TXU Power has evaluated the proposed change and has determined that the change does 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. PRECEDENTS

This change is generally consistent with the changes to the Improved Technical Specifications described in TSTF-419, Revision 0 (Reference 8.1) and the NRC staff's model SE (Reference 8.2). Plants which have received approval for similar changes, in whole or in part, are listed below:

- Fort Calhoun (ADAMS Accession number ML032300305)
- Sequoyah (ADAMS Accession number ML042600465)
- Diablo Canyon (ADAMS Accession number ML041400243)

8.0 REFERENCES

8.1 Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler, TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," Revision 0 (ADAMS Accession number ML012690234).

8.2 NRC letter of March 21, 2002 to NEI approving traveler TSTF-419 and providing the model safety evaluation (ADAMS Accession number ML020800488).

8.3 NRC letter of February 27, 2004 to NEI; "Final Safety Evaluation For Topical Report WCAP-14040, Revision 3, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints And RCS Heatup And Cooldown Limit Curves" (ADAMS Accession number ML040620297).

ATTACHMENT 2 to TXX-05198

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Pages 1.1-5
5.0-35

1.1 Definitions (continued)

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, the power operated relief valve (PORV) lift settings and the LTOP arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

~~Because plant specific analytic methods have not been approved for CPSES, the P/T Limits and the LTOP System Setpoints shall not be revised without prior NRC approval.~~ 81

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

(continued)

WCAP-14040-NP-A; "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

ATTACHMENT 3 to TXX-05198

RETYPED TECHNICAL SPECIFICATION PAGES

Pages 1.1-5
5.0-35

1.1 Definitions (continued)

MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, the power operated relief valve (PORV) lift settings and the LTOP arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

- WCAP-14040-NP-A; "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

(continued)

ENCLOSURE 1 to TXX-05198

Sample PTLR Revision

ERX-05-00x, Rev. 0

SAMPLE

COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)

PRESSURE AND TEMPERATURE LIMITS REPORT

(APPLICABLE UP TO 36 EFPY)

Month 200x

Prepared: ______________________ Date:______________
Engineer
Safety Analysis

Approved: ______________________ Date:______________
Manager
Safety Analysis Manager

Approved: ______________________ Date:______________
Manager
Technical Programs Manager

Pressure and Temperature Limits Report for Comanche Peak Steam Electric Station (Applicable Up To 36 EFPY)

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Pressure and Temperature Limits Report for Comanche Peak Steam Electric Station (Applicable Up To 36 EFPY)

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Pressure and Temperature Limits Report for Comanche Peak Steam Electric Station (Applicable Up To 36 EFPY)

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 1 and Unit 2 in accordance with the requirements of Technical Specification 5.6.6. A description of the Low Temperature Overpressure Protection (LTOP) System power-operated relief valve (PORV) setpoints 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 analytical methods used to determine the RCS pressure and temperature limits are described in Reference 1. The methods used to develop the LTOP System PORV setpoints are also described in Reference 1.

This report covers CPSES Unit 1 and Unit 2 operation for 36 Effective Full Power Years (EFPY).

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 and hydrostatic testing, and criticality. The P/T limits for both CPSES units are based on the approved methodology presented in Reference 1.

The RCS P/T limits are based on the results of the evaluations of the reactor vessel specimen capsules as presented in References 2 and 3 for Units 1 and 2, respectively. The more limiting material is used to develop RCS P/T limits that bound both CPSES units.

The RCS P/T limits calculated for selected heatup and cooldown rates for CPSES Unit 1 and Unit 2 are extracted from Reference 4. These limits are based on curves without the "Flange Notch" requirement.

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 reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature limits of Appendix G of 10 CFR 50. The reactor vessel is the limiting RCPB component for demonstrating such protection.

The LTOP System provides reduced setpoints for the pressurizer Power-Operated Relief Valves (PORVs) as a function of the RCS temperature. The LTOP System PORV setpoints protect the steady-state cooldown curves of Reference 4 without the "Flange Notch" requirement. The methodology used to select the setpoint pressures is described in Reference 1. Allowances for instrument uncertainties have been included in the development of these setpoints.

The LTOP System PORV setpoints for CPSES Unit 1 (original Model D4 steam generators and Model Δ76 steam generators) and Unit 2 (with original Model D5 steam generators) are presented in Table 2-1.

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 36 EFPY using the methods of WCAP-14040-NP-A, Revision 4 [1]. The predictions showed that the materials in the Unit 1 and Unit 2 reactor vessels responded similarly to neutron irradiation but at 36 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 setpoints.

The reactor vessel specimen capsules are withdrawn when the projected neutron fluence would exceed one-times the projected end-of-life vessel fluence and less than two-times the projected end-of-life vessel fluence, in accordance with the Reference 5.

For Unit 1, the required specimen capsules U and Y have been withdrawn and evaluated [2]. The third required specimen capsule, Capsule X, is scheduled to be withdrawn during 1RF11 in the fall of 2005, with a fluence within the range of one-times to two-times the 52 EFPY Peak Fluence [2]. Two of the standby capsules (Capsules V and W) were withdrawn in 1RF09 and stored for later evaluation, if necessary. The third standby capsule is scheduled to be withdrawn during 1RF11 in the fall of 2005 and stored for later evaluation, if necessary. Because all reactor vessel surveillance capsules have been withdrawn and stored, a capsule removal schedule is not required for Unit 1.

For Unit 2, the required specimen capsules U and X have been withdrawn and evaluated [3]. The third required specimen capsule, Capsule W, is scheduled to be withdrawn during 2RF11 in the spring of 2010, with a fluence within the range of one-times to two-times the 54 EFPY Peak Fluence [3]. The schedule for the third capsule withdrawal differs from the specific recommendations contained in Reference 3, but satisfies the requirements of Reference 5 based on an expected end-of-life fluence corresponding to the 54 EFPY Peak Fluence. Two of the standby capsules (Capsules V and Y) were withdrawn in 2RF07 and stored for later evaluation, if necessary. The third standby capsule is scheduled to be withdrawn during 2RF11 in the spring of 2010 and stored for later evaluation, if necessary. The reactor vessel surveillance capsules for Unit 2 is presented in Table 2-3.

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 nominal PORV setpoints for use with the Low Temperature Overpressure (LTOP) System are shown in Table 2-1. The PORV setpoints in Table 2-1 are applicable to Unit 1 (original Model D4 steam generators and Model Δ76 steam generators) and Unit 2 (with original Model D5 steam generators).

Table 2-1
PORV Setpoints for Low Temperature Overpressure (LTOP) System For Unit 1 and Unit 2 Steam Generators, Applicable Up To 36 EFPY

Adjusted RCS Temperature (°F)	PORV #1 Setpoint (psig)	PORV #2 Setpoint (psig)
70	425	573
380	425	573
470	2335	2335

2.4 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance capsule withdrawal schedule for Unit 2 is provided in Table 2-3. All Unit 1 surveillance capsules have been withdrawn from the reactor vessel.

Table 2-2

Unit 2 Reactor Vessel Material Surveillance Program B Withdrawal Schedule

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME	WITHDRAWAL OUTAGE
U	58.5°	3.93	1st Refueling	1st Refueling
X	238.5°	4.15	8.83 EFPY	2RF07
W	121.5°	4.11	13 EFPY	2RF11
Z	301.5°	4.11	Standby	2RF11
V	61.0°	3.87	Standby	2RF07
Y	241.0°	3.87	Standby	2RF07

Figure 2-1 Reactor Coolant System Heatup Limitations - Applicable for the First 36 EFPY

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Plate R-1107-1 (*from Comanche Peak Unit 1*)
LIMITING ART VALUES AT 36 EFPY: 1/4T, 92°F

3/4T, 80°F

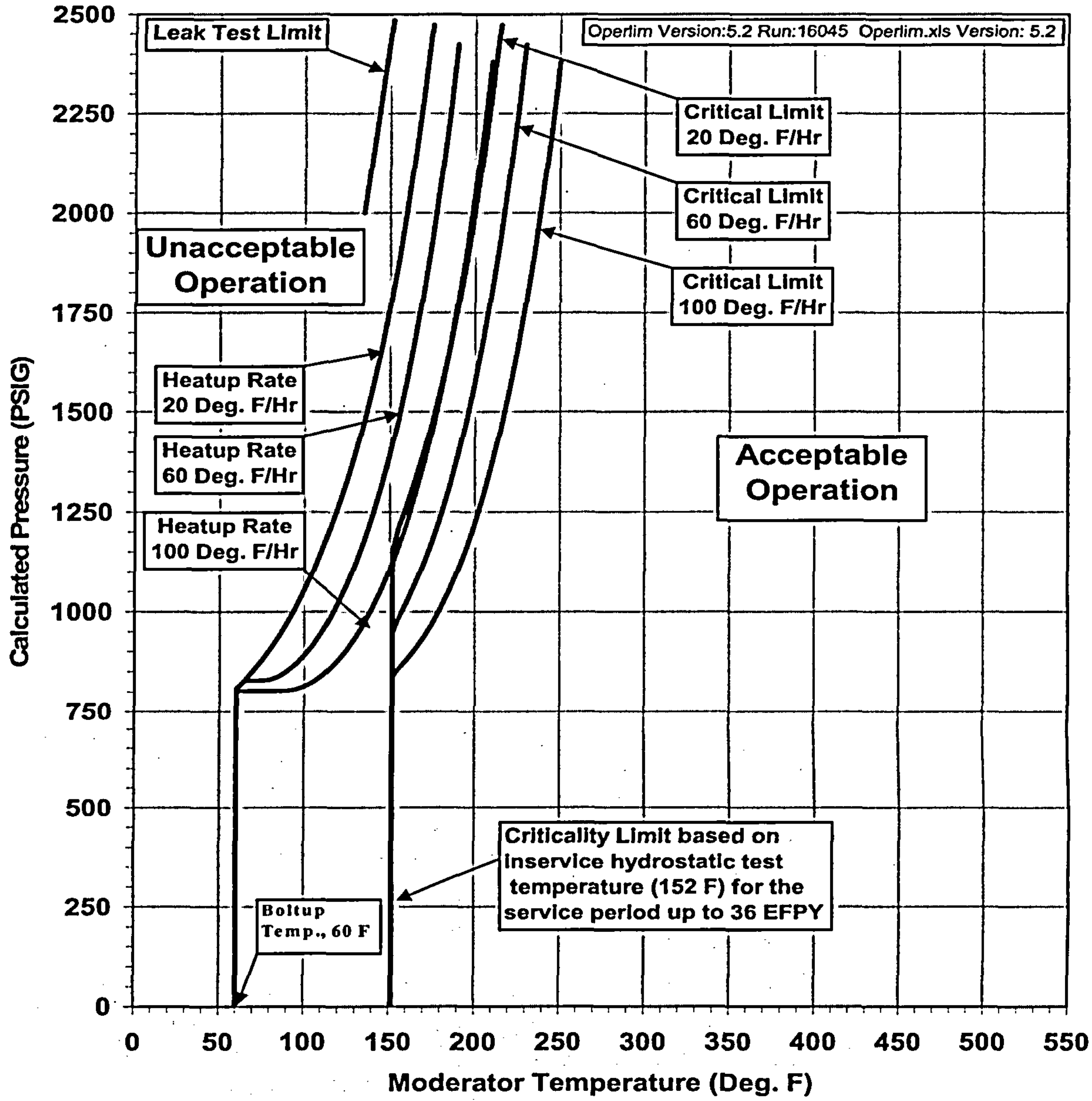
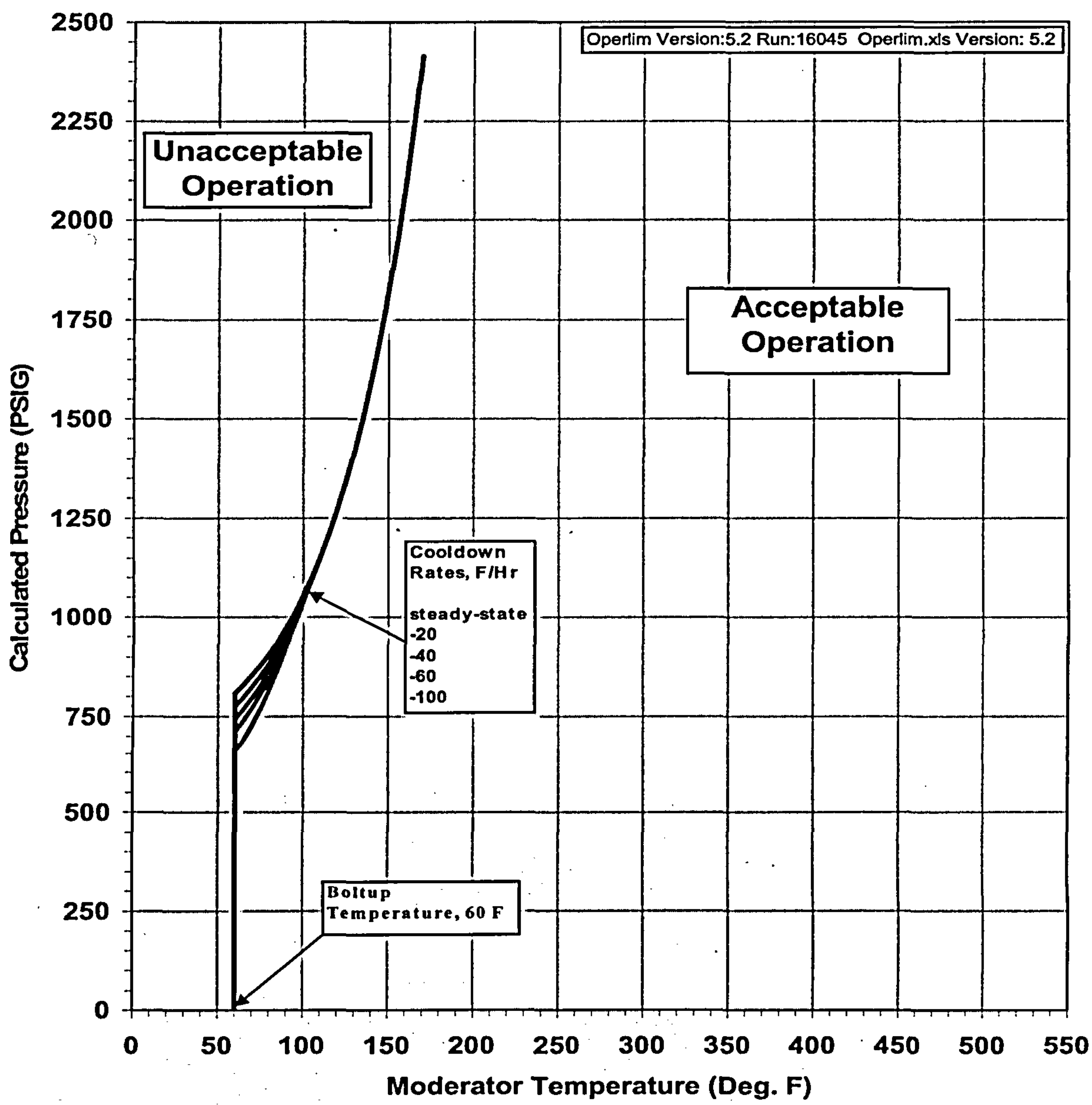


Figure 2-2 Reactor Coolant System Heatup Limitations - Applicable for the First 36 EFPY

<u>MATERIAL PROPERTY BASIS</u>

LIMITING MATERIAL: Intermediate Shell Plate R-1107-1 (*from Comanche Peak Unit 1*)
LIMITING ART VALUES AT 36 EFPY: 1/4T, 92°F
3/4T, 80°F



3.0 REFERENCES

1. "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-NP-A, Revision 4, May, 2004.

2. "Analysis of Capsule Y from the TU Electric Company Comanche Peak Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-15144-NP, Revision 0, January, 1999.

3. "Analysis of Capsule X from the TU Energy Comanche Peak Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-16277-NP, Revision 0, September, 2004.

4. "Comanche Peak Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," WCAP-16346-NP, Revision 0, October 2004.

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

ENCLOSURE 2 to TXX-05198

WCAP-16277-NP; "Analysis of Capsule X from the TXU Energy Comanche Peak Unit 2 Reactor Vessel Radiation Surveillance Program"