March 15, 2007

Mr. M. R. Blevins
Senior Vice President &
Chief Nuclear Officer
TXU Power
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - SAFETY EVALUATION REGARDING TXU GENERATION COMPANY LP REQUEST FOR REVIEW OF TOPICAL REPORTS (TAC NO. MC6899)

Dear Mr. Blevins:

By letter dated February 17, 2005, as supplemented by letters dated July 17, 2006, and February 22, 2007, TXU Generation Company LP (the licensee) requested Nuclear Regulatory Commission (NRC) review of Comanche Peak Steam Electric Station (CPSES) topical report ERX-04-005, “Application of TXU Power’s Non-LOCA [Non-Loss-of-Coolant Accident] Transient Analysis Methodologies to a Feed Ring Steam Generator Design,” and CPSES topical report ERX-04-004, “Replacement Steam Generator Supplement to TXU Power’s Large and Small Break Loss of Coolant Accident Analysis Methodologies.” The topical reports were submitted under separate correspondence, both dated January 25, 2005.

The current CPSES steam generators are of the preheat design (designated as the Westinghouse D-4 for Unit 1 and D-5 for Unit 2), where approximately 90 percent of the main feedwater is injected directly into the cold-leg side of the steam generator tube bundle. Baffles direct this main feedwater across the cold-leg tube bundle five times before it exits the preheater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected above the tube bundle through the auxiliary feedwater nozzle. In CPSES, Unit 1, the original steam generators (OSGs), with preheaters, are to be replaced with replacement steam generators (RSGs), which do not contain integral preheaters. The primary difference between the OSG and the RSG, designated as the Westinghouse Δ76 design, is the use of a feed ring in the RSG to distribute the main feedwater above the tube bundle in the upper-downcomer regions of the steam generators where it mixes with the entirety of the recirculating fluid before entering the tube bundle region.

The NRC staff has reviewed the licensee’s request, and has determined that the modeling and methodology changes described by CPSES topical reports ERX-04-005 and ERX-04-004, as supplemented by TXU Generation Company LP’s letters dated July 17, 2006, and February 22, 2007, are adequate to perform the safety analysis for CPSES, Unit 1, with RSGs, provided the following conditions are met:

1. TXU Generation Company LP will perform a benchmark analysis to the first large-scale transient that would provide sufficient information for the benchmarking analysis.

2. TXU Generation Company LP will apply a 250 °F penalty on the peak clad temperature for the limiting 4-inch diameter break in the cold leg for the CPSES, Unit 1, Cycle 13 SBLOCA [small-break loss-of-coolant accident] analysis.

3. TXU Generation Company LP will submit a license amendment request to revise Technical Specification 5.6.5 to allow the use of the Westinghouse NOTRUMP-based SBLOCA methodology by April 30, 2007.

4. TXU Generation Company LP will submit a unit-specific evaluation model by July 31, 2007, to be applied to CPSES, Unit 1, beginning with Cycle 14 operation in the fall of 2008.

The NRC staff's safety evaluation of the above discussed topical reports is enclosed. If you have any questions, please contact me at (301) 415-1476.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosure: Safety Evaluation

cc w/encl: See next page

2007, are adequate to perform the safety analysis for CPSES, Unit 1, with RSGs, provided the following conditions are met:

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO TOPICAL REPORTS SUPPORTING THE

STEAM GENERATOR REPLACEMENT

FACILITY OPERATING LICENSE NO. NPF-87 AND

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 letter dated February 17, 2005 (Reference 1), as supplemented by letters dated July 17, 2006, and February 22, 2007 (References 2 and 3, respectively), TXU Generation Company LP (the licensee) requested the Nuclear Regulatory Commission's (NRC's) review of Comanche Peak Steam Electric Station (CPSES) topical report ERX-04-005, "Application of TXU Power's Non-LOCA [Non-Loss-of-Coolant Accident] Transient Analysis Methodologies to a Feed Ring Steam Generator Design," and CPSES topical report ERX-04-004, "Replacement Steam Generator Supplement to TXU Power's Large and Small Break Loss of Coolant Accident Analysis Methodologies." The topical reports were submitted under separate correspondence, both dated January 25, 2005 (References 4 and 5, respectively).

The current CPSES steam generators are of the preheat design (designated as the Westinghouse D-4 for Unit 1 and D-5 for Unit 2), where approximately 90 percent of the main feedwater is injected directly into the cold-leg side of the steam generator tube bundle. Baffles direct this main feedwater across the cold-leg tube bundle five times before it exits the preheater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected above the tube bundle through the auxiliary feedwater nozzle. In CPSES, Unit 1, the original steam generators (OSGs), with preheaters, are to be replaced with replacement steam generators (RSG), which do not contain integral preheaters. The primary difference in the RSG, designated as the Westinghouse Δ76 design, is the use of a feed ring to distribute the main feedwater above the tube bundle in the upper-downcomer regions of the steam generators where it mixes with the entirety of the recirculating fluid before entering the tube bundle region.

The purpose of topical report ERX-04-005 is to describe the effects of the Δ76 feed ring steam generators on the non-LOCA transient and accident analysis methodologies currently used for CPSES, Units 1 and 2. Significant changes to the applications of those methodologies, made necessary by the replacement of the original D-4 preheat steam generator design in Unit 1, are also described. Topical report ERX-04-005 is a supplement to the current methodologies. Those methodologies will continue to be used to support both CPSES units. ERX-04-005 will apply only to CPSES, Unit 1.

The purpose of topical report ERX-04-004 is to describe the effects of the Δ76 feed ring steam generators on the LOCA transient and accident analysis methodologies currently used for CPSES, Units 1 and 2. Significant changes to the applications of those methodologies, made necessary by the replacement of the original D-4 preheat steam generator design in Unit 1, are also described. Topical report ERX-04-004 is a supplement to the current methodologies. Those methodologies will continue to be used to support both CPSES units. ERX-04-004 will apply only to CPSES, Unit 1.

2.0 TECHNICAL EVALUATION

The current CPSES steam generators are of the preheat design (designated as the Westinghouse D-4 for Unit 1 and D-5 for Unit 2), where approximately 90 percent of the main feedwater is injected directly into the cold-leg side of the steam generator tube bundle. Baffles direct this main feedwater across the cold-leg tube bundle five times before it exits the preheater region and is allowed to mix with the recirculating fluid and continue to flow through the tube bundle. The remainder of the main feedwater flow is injected above the tube bundle through the auxiliary feedwater nozzle. In CPSES, Unit 1, the original steam generators (OSG), with preheaters, are to be replaced with replacement steam generators (RSG), which do not contain integral preheaters. The primary difference in the RSG, designated as the Westinghouse Δ76 design, is the use of a feed ring to distribute the main feedwater above the tube bundle in the upper-downcomer regions of the steam generators where it mixes with the entirety of the recirculating fluid before entering the tube bundle region. Significant other differences are identified in Table 1 of Reference 1 and include an increase in the number of steam generator tubes, a reduced tube pitch, change from a square tube alignment to a triangular alignment, change in tube material to Alloy 690, increase in the secondary side-heat transfer area, reduction in the secondary side volume, increase in the reactor coolant system (RCS) volume, increase in the nominal circulation ratio, increase in the narrow-range instrument span, and increase in the secondary side mass.

All of the changes associated with the RSG have the potential to affect the safety analysis at CPSES. However, the focus of this evaluation is the impact those changes have on the currently approved methodologies and the actions taken by the licensee to accommodate those changes. The analyses provided by the licensee were only provided as demonstration analyses for the purpose of demonstrating the acceptability of the methodologies and have been used by the NRC staff for that purpose.

2.1 Non-LOCA Transients

The purpose of topical report ERX-04-005 is to describe the effects of the Δ76 feed ring steam generators on the non-LOCA transient and accident analysis methodologies currently used for CPSES, Units 1 and 2. While the licensee recognizes the potential for changes to the methodologies associated with the RSG, there is no test or operational data available for the Δ76 feed ring steam generators to benchmark the methodologies against. Therefore, the evaluation of the change in methodologies is based on analysis and comparison to other designs. ERX-04-005 will only apply to CPSES, Unit 1.

2.1.1 Modeling

CPSES currently uses a RETRAN-02 model for performing system analysis. Within this model the RSG dictates changes to the steam generator portion. If the OSG and RSG were both of the preheat design then those changes could be apportioned among the existing nodalization of the steam generator. Due to the change from the preheat design to the feed ring design, a change in nodalization is also dictated.

The licensee chose the feed ring steam generator nodalization from WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analysis," for modeling the RSG in the CPSES RETRAN-02 model. In response to questions from the staff, the licensee provided information that the feed ring steam generator nodalization from WCAP-14882-P-A has been used in other recent licensing activities, including steam generator replacement and extended power uprate activities. These examples include non-Westinghouse applications (Reference 2). The licensee also provided information comparing its RSG with those used in the other applications. While the Δ76 feed ring steam generators being used by CPSES do not exactly match the other applications, the information clearly indicates that the feed ring steam generator nodalization from WCAP-14882-P-A is reasonable for modeling the Δ76 feed ring steam generators.

To evaluate the RSG model for steady-state conditions, the licensee provided comparisons between its RSG model and vendor design. In response to the NRC staff questions; the licensee provided comparisons for secondary side total mass, vapor mass, liquid mass, circulation ratio, and steam generator pressure at various power levels. In all cases, the licensee's RSG model showed excellent agreement with the vendor's design (Reference 2).

To evaluate the RSG model for transient conditions, the licensee provided the results of demonstration analyses. In the original submittal, the licensee provided graphs of the parameters associated with a feedline break analysis. The analysis was performed using the revised CPSES RETRAN-02 model for the feed ring steam generator. The information provided by TXU Generation Company LP was compared to that provided in support of the Callaway steam generator replacement (Reference 6). With respect to the information provided in support of the Callaway steam generator replacement, the CPSES feedline break information showed reasonable agreement with the trends associated with RCS temperature, steam generator pressure, and pressurizer pressure. In response to staff questions, the licensee provided graphs of the parameters associated with two additional transients: decrease in main feedwater temperature, and turbine trip. The decrease in main feedwater temperature information provided by TXU Generation Company LP was compared to that provided in support of the Callaway steam generator replacement. The CPSES decrease in main feedwater temperature information showed reasonable agreement with the trends associated with normalized power, pressurizer pressure, and deviation from nucleate boiling ratio. The turbine trip information provided by TXU Generation Company LP was compared to that provided in support of the Callaway steam generator replacement. The CPSES turbine trip information showed reasonable agreement with the trends associated with normalized power, pressurizer pressure, and RCS temperature. Comparison with the Callaway information was chosen as the analysis was recent and information provided by the licensee indicates that the Callaway RSG would be the closest to the Δ76 feed ring steam generators. While the difference in the two plants and the specifics of the transient analyses make a direct comparison problematic, confirmation that the CPSES RSG model generates similar trends provides reasonable assurance that it can reasonably predict transients characteristics.

2.1.2 Input Selection

The method of selecting the inputs for the various transients remains relatively unchanged with the inclusion of the feed ring steam generators. In Reference 4, the licensee states that, "The methodologies for the selection of initial conditions, as described in Reference 4, remain unchanged: for a given parameter, the importance or sensitivity of the results to that parameter are assessed, the nominal value is determined, uncertainty allowances are specified, and, if appropriate, sensitivity studies are used to assess the conservative directions for the applications of the uncertainty allowances. For sensitive parameters, the nominal value ± the uncertainty allowance is specified as an initial condition or performance characteristic." The NRC staff finds that the described method for selecting the inputs is reasonable.

2.1.3 Summary

Based on the above evaluation, the NRC staff finds that there is reasonable assurance that the revised model and methodology will adequately predict the performance of CPSES, Unit 1, when operating with RSGs. To ensure the model is tuned, the licensee has committed in Reference 2 to perform a benchmark analysis of the first large-scale transient that would provide sufficient information for benchmarking the revised model.

2.2 LOCA Transients

The purpose of topical report ERX-04-004 is to describe the effects of the Δ76 feed ring steam generators on the LOCA transient and accident analysis methodologies currently used for CPSES, Units 1 and 2. This evaluation provides a review of the TXU Generation Company LP's requested model changes to the current staff approved methods and models for use in evaluating the emergency core cooling system (ECCS) performance for small-break LOCA (SBLOCA) for CPSES, Unit 1. The model changes were submitted to the staff for review and approval in support of the steam generator replacement activities for Unit 1. No modifications to the large-break LOCA methods were proposed by the licensee.

The modeling change consists of modifying the RELAP5 downcomer nodalization to combine the dual inner and outer-ring volumes into a single upper-downcomer volume. In reviewing the model changes and the small-break spectrum submitted by TXU Generation Company LP, the staff noted several concerns with the model and the results of the Advanced Nuclear Fuel (ANF)-based RELAP5 SBLOCA break spectrum. The staff also found that the ECCS evaluation model, in particular the heat-up methodology, did not conform to the requirements to Appendix K of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46; specifically, paragraph I.C.4 in that the model allowed a return to nucleate boiling before the reflood phase. The staff also noted unrealistic thermal-hydraulic behavior in the ANF-based RELAP5 code predictions that produced a non-conservative heat-up of the fuel for the limiting break and an atypical loop-seal clearing phenomenon. Because the deficiencies in the ANF-based RELAP5 code were not corrected in a modified version of the NRC staff-approved RELAP5 code and attendant methods, the staff assessed the impact of the modeling concerns and issues on ECCS performance for Unit 1 and determined a penalty on the peak cladding temperature (PCT) to be applied to the results of the heat-up analyses for SBLOCA break spectrum analysis submitted by the licensee. These concerns and conformance issues, along with the basis for the PCT penalty, are discussed further in this section.

There were no proposed changes to the currently approved large-break LOCA methods.

2.2.1 Lack of RELAP5 Validation Against Integral Test Data

TXU Generation Company LP has submitted a reanalysis of the SBLOCA break spectrum for Unit 1 in support of the steam generator replacement project. TXU Generation Company LP has requested staff review and approval of modifications to its currently accepted ANF-based RELAP5 nodalization scheme for application to evaluations of ECCS performance for SBLOCA for Unit 1 only. The modification consists of combining the upper-downcomer inner and outer-ring cells into two single, vertically stacked volumes in the RELAP5 input nodalization scheme. This includes combining cell #104 and #106 in the upper downcomer into a single cell and cells #100 and #102 also into a single cell, as illustrated in Figures 2.2 and 2.3 of Reference 7.

This modification to the RELAP5 input nodalization scheme restores the scheme to the original ANF RELAP5 nodalization scheme previously approved by the NRC staff. As such, the validation provided during the earlier review and acceptance of the ANF RELAP5 model applies. However, staff review of the ANF methodology, upon which the TXU Generation Company LP SBLOCA methodology was originally based, raised additional issues and concerns. In particular, the validation of this method against integral SBLOCA experimental data is incomplete. Inspection of Reference 8, describing the ANF RELAP5 methods and code validation efforts, shows the comparison of this model using the TXU Generation Company LP proposed modified nodalization to the SEMISCALE Test S-UT-8 SBLOCA integral test. Review of Appendix E of Reference 8 shows that the comparisons to this SEMISCALE Test are incomplete. Furthermore, the comparison raises the following additional concerns:

2.2.1.1 Test S-UT-8 was conducted to 750 seconds as shown in Reference 9. The licensee presented comparisons of key parameters from RELAP5 to only 300 seconds into this test. The PCT in the test occurs at about 675 seconds and was found to be about 1100 °F. Furthermore, review of the test comparison shows that the RELAP5 model overpredicts the amount of liquid in the vessel during the period of time the loop seal clears, producing an early short duration heat-up of the fuel. Even though the heat up to about 750 °F was predicted by RELAP5, this was due to the longer uncovery time predicted by RELAP5 which compensates for the calculated overprediction of the liquid level in the vessel. Particularly important is the liquid level at the end of the RELAP5 run (300 seconds). Figure E.5.6 of Reference 8 shows that the level in the vessel is increasing and is higher than the test data, demonstrating that there is too much mass in the vessel. Since the comparison was not continued beyond 300 seconds, there was no validation of the code's ability to predict the remainder of this test which displayed the long-term core uncovery period and heat-up of the fuel, typical of small breaks that produce limiting PCTs.

2.2.1.2 At 300 seconds, the RELAP5 code also predicts a faster RCS depressurization rate when compared to the data. The faster depressurization coupled with the higher liquid mass in the vessel undermines the credibility of the RELAP5 model and its ability to predict SBLOCA ECCS response. Both of these parameters are key to successful prediction of SBLOCA ECCS response and represented nonconservative predictions by the RELAP5 code.

2.2.1.3 No other comparisons to integral SBLOCA experimental data were presented in Reference 8. The NRC staff finds that the licensee has not presented a proper validation of the RELAP5 code and the proposed modeling techniques against any integral SBLOCA test data. A few of the notable integral tests for validating SBLOCA codes include SEMISCALE Tests S-LH-1, S-LH-2, S-UT-6, S-UT-7, S-UT-8, and S-07-10D, the ROSA small-break integral test series, and the international integral test series.

The staff notes that the complete lack of any comparisons to integral SBLOCA test data raises concerns about the ability of the code to predict key phenomenological behavior (two-phase level swell, depressurization rate, and fuel rod heat-up) governing SBLOCA ECCS response.

2.2.2 Anomalous RELAP5 Steam Cooling Behavior

The staff also notes that only integer break sizes were presented in the submittal of Reference 5. The staff has noted in the past that analyses of integer break sizes do not assure that the limiting break size can be identified, as PCTs can increase by as much as 200 °F for break diameters between 3 and 4 inches, or between 4 and 5 inches. In response to the NRC staff’s requests for additional information (RAIs) of Reference 2, the licensee performed additional break evaluations between 3 and 4 inches, and between 4 and 5 inches in diameter. However, the NRC staff notes that for the 4-inch break (the reported limiting break) a cooling anomaly at about 900 seconds occurs, which terminated the heat up of the fuel for about 100 seconds during the long-term uncovery period. This cooling period, which de-superheats the steam in the hot channel by as much as 400 °F, occurred also for breaks between 3 and 4 inches and between 4 and 5 inches requested in the NRC staff’s RAIs. Consequently, it is still not clear to the NRC staff that the limiting break has been identified.

After additional RAIs, the licensee identified the cause as mist droplet flow penetrating the top of the hot bundle from the loop piping and upper plenum. The staff notes that mist droplet flow with void fractions 0.97 and higher are not expected to cool the hot bundles in the interior of the core during this time period. In particular, droplets do not cool very hot walls in excess of the minimum film boiling temperature (i.e., about 850 °F). In fact, they have little or no lateral momentum sufficient to penetrate the highly superheated vapor adjacent the rods during these long uncovery periods when the top half of the core can be exposed to steam cooling. Clad temperatures are about 1500 °F when the artificial cooling of the hot rod occurred for the 4-inch break. In any event, mist flow has never been observed in any integral test to cool the hot channel during the late long-term uncovery period following an SBLOCA. The staff notes that the RELAP5 drag model also exhibits excess interfacial friction during long-term uncovery of the core, which tends to artificially entrain liquid at the two-phase surface in the core and upper plenum and populate the upper plenum and hot legs with excess liquid following small breaks, particularly when the steaming rates are well below the entrainment criterion for these conditions. The staff does not agree with the licensee that a credible SBLOCA phenomenological behavior has been predicted by the RELAP5 code.

2.2.3 Nonconformance to 10 CFR 50.46, Appendix K Requirements

Inspection of the hot-rod clad temperature profiles for the 3 and 4-inch break sizes shows that nucleate boiling lockout was not included in the heat-up analyses as required by 10 CFR 50.46, Appendix K. The heat-up due to the clearing of the loop seals produced clad temperatures more than 300 °F above the fluid saturation temperature during the early blowdown for breaks in the 3 to 5-inch break diameter range. Section C.5 b.(3) of Appendix K states, “Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 °F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.” Reflood begins when the accumulators begin to inject into the RCS and typically occurs after the PCT is reached for the more limiting smaller breaks. This nonconformance was not corrected in the ANF RELAP5 code and heat-up methodologies used.

2.2.4 Credit for Hot-Leg Nozzle Gap Leakage Paths in the RELAP5 Model

The NRC staff does not allow credit for the hot-leg nozzle gaps in the SBLOCA model. These flow paths are to be removed in the RELAP5 model when performing SBLOCA ECCS licensing evaluations. The leakage paths from the upper downcomer into the upper heat are also to be removed. There is no assurance that these small leakage paths will remain open during a spectrum of SBLOCAs. These leakage paths represent an additional steam venting path directly to the break from the upper plenum that lowers the pressure at the top of the core and minimizes the depth and duration of uncovery of the core, reducing PCTs during the long term.

2.2.5 Loop Seal Thermal-Hydraulic Behavior

The staff also found that the loop seal clearing predicted by the RELAP5 code was questionable. For the 4-inch break, the loop seal in the broken loop clears of liquid first. Later in the event, the broken loop refills with liquid, which is followed by the three intact loop seals clearing of their liquid inventories. This behavior is not understood by the NRC staff. It is noted, however, that past versions of the RELAP5 allowed slug flow to develop in all control volumes based solely on void fraction. Thus, slug flow could develop in the suction leg piping causing an artificial clearing effect since the pressure loss during slug flow represents an increased interfacial steam-liquid flow resistance when the transition from the bubbly to the slug flow regime occurs. The staff believes that slug flow will not develop in the cold-leg piping of any pressurized-water reactor, due to the large pipe diameter. In response to an RAI, the licensee reran the 4-inch diameter break allowing only the broken loop seal to remain cleared while the other loops remained blocked with liquid. This increases the loop resistance and hence the pressure in the upper plenum, which results in deeper core uncovery and higher PCTs during the long term. This analysis showed that the PCT was very close to that for the original 4-inch break. It is also noted that this case credited the hot-leg nozzle gaps and upper-downcomer leakage paths, which reduces PCT for all breaks.

2.2.6 Staff Assessment and Imposed PCT Penalty

Because the RELAP5 model was not modified to address and correct the above issues and concerns, the staff is imposing a 250 °F penalty on the limiting break when employing this RELAP5 code and attendant methodologies for CPSES, Unit 1, Cycle 13. The NRC staff's evaluation of the penalty was based on information specific to CPSES, Unit 1, Cycle 13. The NRC staff has not evaluated this penalty for future cycles and therefore does not consider it to be bounding for future cycles. Since the PCT for the limiting break was calculated to be 1830 °F, this produces a PCT of 2080 °F for the 4-inch break. The basis for the penalty is discussed below:

2.2.6.1 The penalty for lack of a return to nucleate boiling lockout is estimated by the staff to be about 100 °F. This is based on use of the Hsu and Westwater pool film boiling heat transfer correlation (Reference 10) in lieu of nucleate boiling, at the hot spot following clearing of the loop seal for the 4-inch break. A simple energy balance using the film boiling coefficient instead of the nucleate boiling-heat transfer rate was used to recompute the clad temperature based on a decay heat level at the time the loop seal cleared for the 4-inch break.

2.2.6.2 To address the anomalous cooling of the hot rod by the mist droplet flow from the upper plenum and hot legs; a pool boiling level dependent heat transfer code was employed that allows nucleate boiling in the two-phase region of the core, while only pure steam cooling and thermal radiation were allowed in the exposed portion of the core. This removed the artificial cooling of the hot rod that occurred for about 100 seconds at about 900 seconds for the 4-inch break. Inputs to the pool boiling heat-up code used the RELAP5 calculated pressure and liquid level as boundary conditions to perform the evaluation. The PCT was calculated to increase by 200 °F for the 4-inch break in the staff evaluation. Accordingly, the penalty imposed to address the unrealistic cooling behavior exhibited by the RELAP5 code is therefore determined to be 200 °F.

2.2.6.3 The licensee further argued that if the mist droplet flow is removed from the blowdown cooling, less heat addition to the RCS results. The staff agrees that the RCS would then be expected to depressurize faster and actuate the accumulators earlier terminating the heat-up period earlier. However, this is judged to be a small effect since the small amounts of liquid involved (void fractions of 0.97 and higher) are not anticipated to result in much of a reduction in core heat addition if this heat addition is precluded during blowdown. This would be partially offset by increased wall-heat addition from the metal sources in the vessel. The licensee did not perform RELAP5 calculations to verify the magnitude of the faster depressurizations. However, because some minimal depressurization is expected, the staff will propose a PCT benefit of –50 °F to account for the faster depressurization.

2.2.7 Penalty

To summarize, the penalty for the failure to prevent a return to nucleate boiling is 100 °F while the penalty for the temperature anomaly is 200 °F. Since the benefit for the faster depressurization and earlier termination of the clad temperature rise is –50 °F, this results in an overall penalty of 250 °F. When applied to the 4-inch break, this produces a PCT for the limiting SBLOCA of 2080 °F (1830 °F + 250 °F = 2080 °F). Also, since the clad temperature is above 1500 °F for the limiting 4-inch for a short time period (i.e., about 400 seconds), clad oxidation will be well below the 17-percent peak local clad oxidation limit.

The licensee notes that it will be changing methodologies for Unit 1 from the ANF-based RELAP5 methodology reviewed herein to the Westinghouse NRC-approved NOTRUMP SBLOCA methodology. In Reference 3, the licensee has committed to submitting a license amendment request to revise Technical Specification 5.6.5 to allow the use of the Westinghouse NOTRUMP-based SBLOCA methodology by April 30, 2007. In Reference 3, the licensee has also committed to submitting a unit-specific evaluation model by July 31, 2007, to be applied to CPSES, Unit 1, beginning with Cycle 14 operation in the fall of 2008. Because the licensee is updating the licensed methodology to the NOTRUMP code, the staff believes that the 250 °F penalty is appropriate as an interim measure to address the issues identified during the review of the proposed model changes to the ANF-based methodology using RELAP5. As a consequence, this evaluation is valid, contingent upon the licensee changing to the NOTRUMP methodology and submitting the analysis results by July 31, 2007.

2.2.8 Summary

The NRC staff has reviewed the proposed changes to the licensee's ANF-based RELAP5 methodology and as a result of the previously expressed concerns with the RELAP5 and the noncompliance with the requirements of 10 CFR 50.46, Appendix K, the staff believes it is appropriate to impose a penalty on the PCT for the limiting 4-inch diameter break in the cold leg and to limit the application of the methodology to CPSES, Unit 1, Cycle 13. In Reference 3, the licensee has agreed to accept the PCT penalty as determined by the staff, and to transition to the Westinghouse NOTRUMP code in support of CPSES, Unit 1, Cycle 14. The licensee's acceptance of the penalty and restriction of application the CPSES Cycle 13 addresses the staff concerns.

There were no proposed changes to the currently approved large-break LOCA methods.

3.0 CONCLUSION

The staff has determined that the modeling and methodology changes described by CPSES topical reports ERX-04-005, "Application of TXU Power's Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design," and ERX-04-004, "Replacement Steam Generator Supplement to TXU Power's Large and Small Break Loss of Coolant Accident Analysis Methodologies," as supplemented by TXU Generation Company LP letters dated July 17, 2006, and February 22, 2007, are adequate to perform the safety analysis for CPSES, Unit 1, with RSGs, provided the following conditions are met.

- TXU Generation Company LP will perform a benchmark analysis to the first large-scale transient that would provide sufficient information for the benchmarking analysis.

- TXU Generation Company LP will apply a 250 °F penalty on the PCT for the limiting 4-inch diameter break in the cold leg for the CPSES, Unit 1, Cycle 13 SBLOCA analysis.

- TXU Generation Company LP will submit a license amendment request to revise Technical Specification 5.6.5 to allow the use of the Westinghouse NOTRUMP-based SBLOCA methodology by April 30, 2007.

- TXU Generation Company LP will submit a unit-specific evaluation model by July 31, 2007, to be applied to CPSES, Unit 1, beginning with Cycle 14 operation in the fall of 2008.

4.0 REFERENCES

1. TXU Generation Company LP, letter dated February 17, 2005, from Mike Blevins, Senior Vice President & Chief Nuclear Officer to USNRC, Re: "Comanche Peak Steam Electric Station (CPSES), Docket No. 50-445, Request for Review of Previously Submitted Licensee Topical Reports" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050590178).

2. TXU Generation Company LP, letter dated July 17, 2006, from Mike Blevins, Senior Vice President & Chief Nuclear Officer to USNRC, Re: "Comanche Peak Steam Electric Station (CPSES), Docket No. 50-445 - Response to Request for Additional Information Related to TXU Power's Request for Review of Previously Submitted Licensee Topical Reports (TAC No. MC6899)" (ADAMS Accession Nos. ML062050011, ML062050012, ML062050013, ML062050014, and ML062050015).

3. TXU Generation Company LP, letter dated February 22, 2007, from Mike Blevins, Senior Vice President & Chief Nuclear Officer to USNRC, Re: "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Related to TXU Power's Request for Review of Previously Submitted Licensee Topical Reports (TAC No. MC6899)" (ADAMS Accession No. ML070600170).

4. TXU Generation Company LP, letter dated January 25, 2005, from Mike Blevins, Senior Vice President & Chief Nuclear Officer to USNRC, Re: "Comanche Peak Steam Electric Station (CPSES), Docket No. 50-445, Submittal of TXU Power's Application of Non-LOCA Transient Analysis Methodologies to a Feed Ring Steam Generator Design, Topical Report # ERX-04-005, Revision 0" (ADAMS Accession No. ML050310419).

5. TXU Generation Company LP, letter dated January 25, 2005, from Mike Blevins, Senior Vice President & Chief Nuclear Officer to USNRC, Re: "Comanche Peak Steam Electric Station (CPSES), Docket No. 50-445, Submittal of Supplement to the CPSES Loss of Coolant Accident (LOCA) Analysis Methodologies - Topical Report #ERX-04-004, Revision 0" (ADAMS Accession No. ML050310408).

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8. "USNRC's Safety Evaluation of Advanced Nuclear Fuels' Small Break LOCA Evaluation Model ANF-RELAP and Acceptance for Referencing of Topical Report," XN-NF-82-49 Revision 1, July 1988.

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Date: March 15, 2007