February 20, 2007

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATION CHANGES RELATED TO CPSES UNIT 1 STEAM GENERATOR REPLACEMENT PROJECT (TAC NOS. MD0187 AND MD0188)

Dear Mr. Blevins:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 131 to Facility Operating License No. NPF-87 and Amendment No. 131 to Facility Operating License No. NPF-89 for CPSES, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 21, 2006, as supplemented by letters dated September 12 and December 14, 2006.

The amendments increase the allowable values (AVs) for steam generator (SG) water level trip setpoints and the required minimum SG secondary side water inventory in shutdown modes for the replacement SGs in CPSES Unit 1. For CPSES Unit 2, the corresponding AVs and the SG secondary water inventory in the current TSs remain unchanged since the existing SGs in Unit 2 will continue to be used.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

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

Docket Nos. 50-445 and 50-446

Enclosures:
1. Amendment No. 131 to NPF-87
2. Amendment No. 131 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

February 20, 2007

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* See SE dated 1/9/07.

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/SPWB/BC*	NRR/ITSB/BC	OGC - NLO	NRR/LPL4/BC
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DATE	2/7/07	2/7/07	1/9/07	2/15/07	2/15/07	2/16/07

OFFICIAL RECORD COPY

Comanche Peak Steam Electric Station

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TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. NPF-87

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

   A. The application for amendment by TXU Generation Company LP dated February 21, 2006, as supplemented by letters dated September 12 and December 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications and paragraph 2.C.(2) of Facility Operating License No. NPF-87 as indicated in the attachment to this license amendment.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: February 20, 2007

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. NPF-89

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

   A. The application for amendment by TXU Generation Company LP dated February 21, 2006, as supplemented by letters dated September 12 and December 14, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications and paragraph 2.C.(2) of Facility Operating License No. NPF-89 as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: February 20, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 131

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 131

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Facility Operating Licenses, Nos. NPF-87 and NPF-89, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-87

REMOVE	INSERT
- 3 -	- 3 -

Facility Operating License No. NPF-89

REMOVE	INSERT
- 3 -	- 3 -

Technical Specifications

REMOVE	INSERT
3.3-17	3.3-17
3.3-32	3.3-32
3.3-33	3.3-33
3.4-10	3.4-10
3.4-13	3.4-13
3.4-14	3.4-14
3.4-16	3.4-16

(3) TXU Generation Company LP, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

TXU Generation Company LP is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 131 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Generation Company LP shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TXU Generation Company LP, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, and described in the Final Safety Analysis Report, as supplemented and amended;

(4) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use, at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source, and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) TXU Generation Company LP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

TXU Generation Company LP is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 131 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Generation Company LP shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DELETED

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 131 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU GENERATION COMPANY LP

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated February 21, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060580206), as supplemented by letters dated September 12 and December 14, 2006 (ADAMS Accession Nos. ML062640237 and ML070030515, respectively), TXU Generation Company LP (the licensee), requested changes to the Technical Specifications (TSs) for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2.

The proposed changes would increase the allowable values (AVs) for steam generator (SG) water level trip setpoints and the required minimum SG secondary side water inventory in shutdown modes for the replacement SGs in CPSES Unit 1. For CPSES Unit 2, the corresponding AVs and the SG secondary water inventory in the current TSs remain unchanged since the existing SGs in Unit 2 will continue to be used.

The current SGs in Unit 1 are the Westinghouse Model D4 design. The licensee would replace these existing SGs with the Model Δ76 SGs during the Unit 1 Cycle 12 refueling outage. The U-tube heat transfer area for the Δ76 SG design is increased over the D4 SG design by about 60 percent, to 76,000 ft^2. The top of the U-tubes, approximately 8 feet higher than that in the D4 SG, is well above the bottom of the SG water level instrument span. In the D4 SG, the top of the U-tubes is at approximately the same elevation as the bottom of the level instrument span. Therefore, the SG water level indication corresponding to the top of the U-tubes is significantly greater in the Δ76 SG. The licensee proposed TS changes to reflect new SG water level setpoints and inventory requirements for the replacement SGs in Unit 1. Specifically, the changes include: (1) an increase in the AV for the SG water level low-low reactor trip setpoint and auxiliary feedwater (AFW) actuation setpoint from 23.1 percent narrow-range span to 36.0 percent narrow-range span, based on the nominal trip setpoint (NTS) of 38.0 percent narrow-range span; (2) an increase in the AV for the SG water level high-high trip setpoint from 84.3 percent narrow-range span to 86 percent narrow-range span, based on the NTS of 86.0 percent narrow-range span; and (3) an increase in the required SG secondary side water level from 10.0 percent to 38.0 percent in shutdown modes for decay heat removal. The SG water level trip setpoints affected by this proposed change are in TS Table 3.3.1-1, "Reactor Trip System Instrumentation," and TS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation." The SG secondary water inventory requirements affected by this proposed change are Surveillance Requirements (SRs) 3.4.5.2, "RCS [Reactor Coolant System] Loops - Mode 3" SR 3.4.6.2, "RCS Loops - Mode 4" and 3.4.7.2, "RCS Loops - Mode 5 Loops Filled," and Limiting Condition for Operation (LCO) 3.4.7.b.

The supplements dated September 12 and December 14, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 6, 2006 (71 FR 32609).

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission's (NRC or the Commission) regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. Additionally, Criterion 2 of 10 CFR 50.36(c)(2)(ii) requires an LCO to be established for a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. Accordingly, the limits for instrument channels that initiate protective functions must be included in the TSs.

In accordance with General Design Criterion (GDC) 20, "Protection system functions," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The acceptable fuel design limits are specified in NUREG-0800, Standard Review Plan.

Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

3.0 TECHNICAL EVALUATION

The NRC issued Information Notice (IN) 2002-10 on March 7, 2002 (Reference 3), to alert holders of operating licenses to the potential for non-conservative setpoints in the SG water level. The IN was issued as a result of a February 9, 2002, occurrence at Diablo Canyon Power Plant, Unit 2, where the SG narrow-range water level instrumentation did not respond as expected to initiate an automatic reactor trip and AFW actuation on the SG water level low-low signal during a plant trip. This event prompted Westinghouse, the SG manufacturer, to issue various Nuclear Safety Advisory Letters (NSALs).

As discussed in NSAL-02-3 and its revision, Westinghouse attributed the water level uncertainties mainly to a differential pressure (ΔP), previously unaccounted for, created by steam flow past the mid-deck plate in the moisture separator section of the SG. Westinghouse-designed SGs incorporate a mid-deck plate at the top of the primary separator assembly between the upper and lower taps used for the SG narrow-range water level instruments. The installation of the mid-deck plate was to reduce moisture carryover. When some of the steam flows through the separator downcomer, instead of the primary separator orifice, this steam with some entrained moisture will flow upwards through the flow area in the mid-deck plate, creating a pressure differential. The mid-deck plate ΔP, which is a function of steam flow, causes the SG narrow-range instrumentation to read higher than the actual water level, and adversely affects the SG level low-low trip with an uncertainty bias in a non-conservative direction. Therefore, the SG water level instrumentation without accounting for this ΔP phenomenon could be non-conservative during certain transients.

NSAL-02-4 deals with uncertainties in the measurement created because the void content of the two-phase mixture above the mid-deck plate is not reflected in the calculation. The uncertainties may adversely affect the SG water level high-high trip signal for actuation of turbine trip and feedwater system isolation in a non-conservative direction.

NSAL-03-09 indicates that Westinghouse has developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items. These items include the mid-deck plate, feedwater ring and feedwater ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, as well as transient conditions due to events such as the loss of normal feedwater, or a steamline break outside containment. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties in terms of the SG water level control system, and the SG water level low-low and high-high trip functions.

The NRC staff evaluated the licensee's proposed TS changes for compliance with the regulatory requirements and guidance discussed in Section 2.0 of this Safety Evaluation and verified that the licensee had appropriately addressed the issues discussed in the IN and NSALs.

SG water level low-low channels are part of the reactor protection system (RPS) and engineered safety features actuation system (ESFAS). They are designed to trip the reactor (as an RPS function) and start the AFW pumps (as an ESFAS function) for protection of the reactor core from a loss of heat sink in the event of a sustained steam and feedwater flow mismatch.

SG water level high-high channels are also part of the ESFAS. They are designed to trip the turbine and isolate feedwater to the SGs. The SG level high-high signal functions to prevent the SGs from overfilling with water and water entering the main steam piping, during an excessive feedwater flow event.

The proposed TSs would change the AV for the SG narrow-range water level low-low reactor trip and AFW actuation setpoint from 23.1 percent narrow-range span to 36.0 percent narrow-range span, and also change the AV for the SG water level high-high trip setpoint from 84.3 percent narrow-range span to 86.0 percent narrow-range span. The changes are necessary to reflect the replacement SGs in Unit 1.

In References 2 and 5, the licensee provided its request for additional information responses that discussed the methods used to calculate the NTSs and AVs of SG water level setpoints for Unit 1, including the equation used to determine the channel statistical allowance. The calculation of the channel statistical allowance includes process effects and instrumentation loop uncertainty. The allowance for process effects accounts for non-instrument related effects such as process pressure variation and process measurement accuracy (PMA). The PMAs, as described in NSAL-03-09, as well as allowances for adverse containment environments and transient effects, were treated as biases and were combined algebraically. Instrumentation loop uncertainties address the accuracies of instruments, such as transmitter and rack, which are independent and random accuracies. The instrumentation loop uncertainties were statistically combined using the square-root-of-the-sum-of-squares technique.

As indicated in the NSALs, the SG level setpoint uncertainties may be caused by several PMA terms that were not previously accounted for in many plants with Westinghouse-designed SGs. The licensee considered in its Unit 1 setpoint calculations the following PMA items: (1) process pressure variations; (2) reference leg temperature variations; (3) fluid velocity effects; (4) downcomer subcooling effects; (5) mid-deck plate ΔP effects; (6) intermediate deck plate ΔP effects; (7) feedring ΔP effects; (8) lower deck plate and deck support ΔP effects; and (9) non-recoverable losses due to carry-under into the lower downcomer. In addressing the physical effects of the design-transient conditions, the licensee identified the events that credited a particular SG water trip function for consequence mitigation, and determined the physical effects of the transient conditions with respect to PMA terms identified in NSALs, and overall impact on the SG level setpoint uncertainties for those affected events. The licensee presented the results of its calculation in Table 1 of Reference 2.

The NRC staff's evaluation of this methodology is detailed in the following sections.

3.1 SG Water Level Setpoints

3.1.1 Setpoint Methodology

The Westinghouse method used for the Unit 1 TSs determines a performance-based AV. According to the licensee's setpoint methodology, the AV satisfied by verification that the channel "as-left" and "as-found" conditions about the NTS are within the total rack uncertainty allowance of the rack calibration accuracy, rack drift and rack temperature effects.

In a letter dated March 31, 2005 (Reference 4), to the Nuclear Energy Institute, the NRC staff stated that the RCS pressure and thermal safety limits (SLs) are protected by the practice of setting the instrument trip setpoint at, or more conservative than, the calculated trip setpoint (TSP) that accounts for credible uncertainties. However, existing AV-based TSs for the RPS and ESFAS do not require that licensees control the instrument setting-based TSP. To resolve this issue, the TSs should include a requirement to return the as-left instrument setting to the TSP established to protect the SLs. In the proposed TS, the licensee added to the TSs associated with the RPS and ESFAS functions the following two notes to function 14 of

TS Table 3.3.1-1 for the SG level low-low reactor trip, and functions 5.b and 6 of TS Table 3.3.2-1 for the turbine trip and AFW isolation based on an SG level high-high actuation setpoint and the AFW actuation based on an SG level low-low actuation setpoint, respectively:

> Note q - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel should be evaluated to verify that it is functioning as required before returning the channel to service.
>
> Note r - The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Nominal Trip Setpoint, or a value that is more conservative than the Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, the methodology used to determine the as-found tolerance and the methodology used to determine the as-left tolerance shall be specified in the Technical Specifications Bases.

The above notes satisfy the intent of the regulations to protect the SLs by adjusting the trip setpoint within the as-left tolerance, and follow the guidance of the March 31, 2005, letter. Therefore, the NRC staff determines that the added notes are acceptable. Also, the NRC staff has evaluated the licensee's setpoint methodology (discussed in References 1 and 2) and concluded that the methodology demonstrates that the trip setpoint and the as-left and as-found tolerances are established and held within specified limits to protect the SLs. Therefore, the methodology is acceptable.

3.1.2 SG Level Low-Low Trip and AFW Actuation Setpoint (TS Tables 3.3.1-1 and 3.3.2-1)

Following the guidance in NSAL-03-09 and its associated analyses, the licensee calculated the SG water level low-low uncertainties (References 2 and 5) based on transient conditions of analyses for the events of a single loop loss of normal feedwater and small steamline break outside containment. The licensee presented the calculated NTSs and AVs in Reference 1, and the channel statistical allowance and total allowance (TA) in Reference 2 for Unit 1. The NTS was determined by adding the TA to the safety analysis limit (SAL) used in the transient analysis of design-basis events. The SAL (References 1 and 2) used to determine the SG level low-low trip and AFW actuation setpoint is the same value used in the analyses of record for events of a turbine trip, loss of non-emergency alternating current power, loss of normal feedwater, feedwater line break and small steamline breaks. The SAL will be used also for reload analyses to demonstrate compliance with the GDC 20 requirements that require the protection systems be designed to trip the reactor to assure that the specified acceptable fuel design limits are not exceeded. Therefore, the NRC staff concluded that the SAL is acceptable for use in determining the NTS for the SG level low-low reactor trip and AFW actuation setpoints.

The licensee indicated that the SAL used (References 1 and 2) is 0.0 percent and the calculated TA is 38.0 percent of the SG narrow-range span for Unit 1. The NTS of 38.0 percent was determined by adding TA of 38.0 percent to the SAL of 0.0 percent. This higher NTS would result in an earlier reactor trip and AFW actuation during decreased SG water inventory transients such as a loss of normal feedwater event, and thus, would provide a greater margin to protect the RCS from exceeding pressure limit and the specified acceptable fuel-design limits. Therefore, the NRC staff concluded that the calculated NTS was conservative and acceptable.

The TS AV is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is declared inoperable and corrective action must be taken. The AVs were determined by adding (or subtracting toward a conservative direction) the rack uncertainty allowances for: (1) instrument calibration uncertainties, (2) instrument uncertainties during normal operation, and (3) instrument drift to the NTS. The calculated total rack uncertainty allowance is 2.0 percent based on the sum of the rack calibration accuracy, drift and temperature effect tolerances. The proposed AV of 36 percent was calculated by subtracting the total rack uncertainty allowance of 2.0 percent from the NTS of 38.0 percent. The NRC staff determined that the AV was acceptable, as this value was calculated based on the methods consistent with the acceptable setpoint methodologies used in determining setpoint AVs included in the current CPSES TSs. The NRC staff also found that the calculated maximum AV of 36.0 percent narrow-range span was correctly reflected in TS Table 3.3.1-1 and Table 3.3.2-1 for the proposed AV of the SG water level low-low reactor trip setpoint and AFW actuation setpoint. Therefore, the NRC staff concluded that proposed AVs in TS Table 3.3.1-1 and Table 3.3.2-1 were acceptable for Unit 1.

3.1.3 SG Level High-High Trip Setpoint (TS Table 3.3.2-1)

The SG level high-high trip is used in the design-basis analysis to terminate a feedwater malfunction or excessive feedwater flow event. Following the trip signal, the turbine will trip and the AFW will isolate. NSAL-02-4 indicated that the void content of the two-phase mixture above the mid-deck plate was not reflected in the SG level setpoint calculations. This results in an actuation of the SG water level high-high trip signal for the turbine trip and AFW isolation in a non-conservative direction. In addressing the NSAL-02-4 issue, the licensee calculated the total channel uncertainty including the effects of void content above the mid-deck plate.

The licensee presented the calculated values of NTS, TA and AV for the SG level high-high trip setpoint in References 1 and 2. The NTS was determined by subtracting the TA from the SAL that was used in the transient analysis of an excessive feedwater flow event to demonstrate compliance with the GDC 20 requirements that require the protection systems be designed to trip the reactor to assure that the specified acceptable fuel design limits are not exceeded.

For the determination of the SG level high-high NTS, the SAL used (References 1 and 2) was 97.9 percent and the TA was 13.9 percent of the SG narrow-range span for Unit 1. The SAL of 97.9 percent span (Reference 2) considered the effects of the void content discussed in NSAL-02-4 and was bounded by that (100 percent) used in the existing transient analysis for the excessive feedwater flow event. In addition, the calculated SG level high-high NTS of 84.0 percent narrow-range span for Unit 1 was calculated by subtracting the TA of 13.9 percent from the void-content adjusted SAL of 97.9 percent. This lower NTS would result in an earlier turbine trip and feedwater isolation during an excessive feedwater flow event, and thus, would provide a greater margin to protect the SG from overfilling with water. Therefore, the NRC staff concluded that the calculated NTS was conservative and acceptable.

The licensee used the same method discussed in Section 3.1.2 above to calculate the AV. The proposed AV of 86.0 percent was calculated by adding the total rack uncertainty allowance of 2.0 percent to the 84.0 percent. The NRC staff determined that the proposed AV of 86 percent is acceptable, since this value was calculated based on the methods consistent with the acceptable setpoint methodologies used in determining setpoint AVs included in the current CPSES TSs. In addition, inclusion of the notes discussed in Section 3.1.1 above to the TS instrumentation provides reasonable assurance that the plant will operate in accordance with the safety analyses and operability of the instrumentation is assured. The NRC staff also found that the proposed AV of 86.0 percent narrow-range span was correctly reflected in TS Table 3.3.2-1 for the proposed AV of the SG water level high-high trip setpoint. Therefore, the NRC staff concluded that proposed AV in TS Table 3.3.2-1 is acceptable.

3.1.4 Minimum Water Inventory in the Required SGs (TS SRs 3.4.5.2, 3.4.6.2, 3.4.7.2, and LCO 3.4.7.b)

The licensee proposed to increase the secondary side water level from 10.0 percent to 38.0 percent narrow-range span in required SGs in the following TSs: SR 3.4.5.2, “RCS Loops - Mode 3,” SR 3.4.6.2, “RCS Loops - Mode 4,” SR 3.4.7.2, “RCS Loops - Mode 5 Loops Filled;” and LCO 3.4.7.b.

In the shutdown modes (Modes 3 and 4, and Mode 5 with the RCS loops filled), the current TSs require one or more SGs be available for cooling the RCS with natural circulation. To satisfy this requirement, the corresponding SR requires verification of SG operability by ensuring that the minimum SG secondary side narrow-range water level is 10.0 percent for the existing SGs with the Westinghouse Model D4 design. As indicated in the current TS bases, the indicated 10.0 percent narrow-range span water inventory is required to completely cover the top of the SG U-tubes to ensure that the SG can function as an effective heat sink.

In the replacement Δ76 SGs for Unit 1, the top of U-tubes corresponds to approximately 31.0 percent narrow-range span. The licensee proposed a value of 38.0 percent narrow-range span for the required minimum SG inventory. This value includes the SG water level indication uncertainty of 6.0 percent span for normal operation. Since the value of 38.0 percent narrow-range span proposed in SRs 3.4.5.2, 3.4.6.2, 3.4.7.2, and LCO 3.4.7.b was determined to completely cover the top of the SG U-tubes, the NRC staff determined that it met the intent of the current TS requirements to ensure that the replacement Δ76 SGs were effective for decay heat removal during shutdown modes. Therefore, the NRC staff concluded that the proposed TS SR changes were acceptable.

3.2 Technical Conclusion

The staff has reviewed the licensee’s proposed changes to the CPSES Units 1 and 2 TSs to increase for Unit 1 (1) the AVs for the SG water level low-low reactor trip reactor and AFW actuation setpoint from 23.1 percent to 36.0 percent narrow-range span in TS Table 3.3.1-1 and Table 3.3.2-1, (2) the AV for the SG water level high-high trip setpoint from 84.3 percent narrow-range span to 86.0 percent narrow-range span in TS Table 3.3.2-1, and (3) the required SG secondary side water level from 10.0 percent to 38.0 percent narrow-range span in SRs 3.4.5.2, 3.4.6.2, 3.4.7.2, and LCO 3.4.7.b. Based on the evaluation discussed in Section 3.1, the NRC staff finds that the proposed TSs appropriately address the issues specified in the NSALs, and the proposed AVs are based on acceptable setpoint methodologies. Therefore, the NRC staff concludes that the licensee's proposed TS changes comply with the GDC 20 requirements and are acceptable. For Unit 2, the corresponding AVs and the SG secondary water inventory in the current TS remain unchanged since the existing SGs in Unit 2 will continue to be used.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published June 6, 2006 (71 FR 32609). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from M. Blevins (TXU Power) to USNRC, "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, License Amendment Request (LAR) 05-08, Revision to Technical Specification 3.3.1, "Reactor Trip System Instrumentation," 3.3.2, "ESFAS Intrumentation," 3.4.5, "RCS LOOPS - Mode 3," 3.4.6, "RCS LOOPS - Mode 4," and 3.4.7, "RCS LOOPS - Mode 5, Loops Filled," dated February 21, 2006 (ADAMS Accession No. ML060580206).

2. Letter from M. Blevins (TXU Power) to USNRC, "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Related to License Amendment Request 05-08, Revision to Technical Specifications for Nominal Trip Setpoints (NTS) and Allowable Value (AV) Setpoints for SG Water Level Low-Low and High-High, TAC Nos. MD0187 and MD0188," dated September 12, 2006 (ADAMS Accession No. ML062640237).

3. NRC Information Notice 2002-10, “Nonconservative Water Level Setpoint on Steam Generators,” dated March 7, 2002.

4. Letter from J. A. Lyons (NRC) to A. Marion (NEI), “Instrumentation, Systems, and Automation Society S67.04 Methods for Determining Trip Setpoints and Allowable Values for Safety-Related Instrumentation,” dated March 31, 2005 (ADAMS Accession No. ML050870008).

5. Letter from M. Blevins (TXU Power) to USNRC, “Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and 50-446, Response to Request for Additional Information Related to License Amendment Request 05-08, Revision to Technical Specifications for Nominal Trip Setpoints (NTS) and Allowable Value (AV) Setpoints for SG Water Level Low-Low and High-High, TAC Nos. MD0187 and MD0188,” dated December 14, 2006 (ADAMS Accession No. ML070030515).

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