April 3, 2008

Mr. M. R. Blevins
Executive Vice President
& Chief Nuclear Officer
Luminant Generation Company LLC
ATTN: Regulatory Affairs
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGES TO TECHNICAL SPECIFICATIONS TO REFLECT CYCLE-SPECIFIC SAFETY ANALYSIS ASSUMPTIONS AND RESULTS OF ADOPTION OF WESTINGHOUSE METHODOLOGIES (TAC NOS. MD6561 AND MD6562)

Dear Mr. Blevins:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 145 to Facility Operating License No. NPF-87 and Amendment No. 145 to Facility Operating License No. NPF-89 for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 16, 2007, as supplemented by letter dated December 13, 2007.

The amendments revise TS 3.1.4, "Rod Group Alignment Limits," Table 3.3.1-1, "Reactor Trip System Instrumentation," Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," TS 3.4.10, "Pressurizer Safety Valves," TS 3.7.1, "Main Steam Safety Valves (MSSVs)," and Table 3.7.1-1, "Operable Main Steam Safety Valves Versus Maximum Allowable Power." The purpose of the TS changes is to reflect the cycle-specific safety analysis assumptions and results associated with the adoption of Westinghouse accident analyses methodologies.

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/

Balwant K. Singal, 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. 145 to NPF-87
2. Amendment No. 145 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
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BMarcus, NRR/DSS/EICB
GHill, OIS (4)

(**) See previous concurrence

ADAMS Accession Nos.: Pkg ML080580003 (AMD ML080580004, License/TS Pgs ML080580005) (*) SE input memo

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DSS/SRXB/BC	DE/EICB/BC	DIRS/ITSB/BC	OGC – NLO w/comments	NRR/LPL4/BC
NAME	BSingal	JBurkhardt	GCranston (*)	WKemper(*)	GWaig (**)	JRund (**)	THiltz
DATE		4/2/08	1/28/08	1/28/08	3/7/08	3/12/08	4/2/08

OFFICIAL AGENCY RECORD

cc:
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 2159
Glen Rose, TX 76403-2159

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mr. Fred W. Madden, Director
Regulatory Affairs
Luminant Generation Company LLC
P.O. Box 1002
Glen Rose, TX 76043

Timothy P. Matthews, Esq.
Morgan Lewis
1111 Pennsylvania Avenue, NW
Washington, DC 20004

County Judge
P.O. Box 851
Glen Rose, TX 76043

Environmental and Natural
Resources Policy Director
Office of the Governor
P.O. Box 12428
Austin, TX 78711-3189

Mr. Richard A. Ratliff, Chief
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Mr. Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, TX 78701-3326

Ms. Susan M. Jablonski
Office of Permitting, Remediation
and Registration
Texas Commission on Environmental
Quality
MC-122
P.O. Box 13087
Austin, TX 78711-3087

Anthony P. Jones
Chief Boiler Inspector
Texas Department of Licensing
and Regulation
Boiler Division
E.O. Thompson State Office Building
P.O. Box 12157
Austin, TX 78711

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. NPF-87

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

   A. The application for amendment by Luminant Generation Company LLC dated August 16, 2007, as supplemented by letter dated December 13, 2007, 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 as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 145 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan 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 implementation prior to startup from the fall 2008 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License No. NPF-87
and Technical Specifications

Date of Issuance: April 3, 2008

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. NPF-89

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

   A. The application for amendment by Luminant Generation Company LLC dated August 16, 2007, as supplemented by letter dated December 13, 2007, 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 as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 145 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. Luminant Generation Company LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented implementation prior to startup from the spring 2008 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Thomas G. Hiltz, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License No. NPF-89
and Technical Specifications

Date of Issuance: April 3, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 145

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 145

TO FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Facility Operating License 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
3.1-10	3.1-10
3.3-15	3.3-15
3.3-16	3.3-16
3.3-17	3.3-17
3.3-18	3.3-18
3.3-19	3.3-19
3.3-20	3.3-20
--	3.3-20a
3.3-29	3.3-29
3.3-30	3.3-30
3.3-31	3.3-31
3.3-32	3.3-32
3.3-33	3.3-33
3.3-34	3.3-34
3.4-21	3.4-21
3.7-1	3.7-1
3.7-4	3.7-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 145 TO

FACILITY OPERATING LICENSE NO. NPF-89

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1. INTRODUCTION

By letter dated August 16, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML072330512), supplemented by letter dated December 13, 2007 (ADAMS Accession No. ML073550859), TXU Generation Company LP (subsequently renamed Luminant Generation Company LLC, the licensee) requested changes to the technical specifications (TSs) for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The supplemental letter dated December 13, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 25, 2007 (72 FR 54482).

The proposed license amendment request (LAR) would revise the following TSs and TS Tables for CPSES, Units 1 and 2, to reflect cycle-specific safety assumptions and results associated with the proposed adoption of the Westinghouse accident analyses methodologies and revise the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation Allowable Value (AV) trip functions:

- TS 3.1.4, “Rod Group Alignment Limits”
- Table 3.3.1-1, “Reactor Trip System Instrumentation”
- Table 3.3.2-1, “Engineered Safety Feature Actuation System Instrumentation”
- TS 3.4.10, “Pressurizer Safety Valves”

- TS 3.7.1, “Main Steam Safety Valves”

- Table 3.7.1-1, “Operable Main Steam Safety Valves versus Maximum Allowable Power”

The Westinghouse accident analysis methodology was submitted by the licensee to the NRC April 10, 2007 (ADAMS Accession No. ML071070307), and supplemented by letters dated July 31, August 16, November 15 (two letters), and November 19, 2007 (ADAMS Accession Nos. ML072200641, ML072340228, ML073241097, ML073241095, ML073330059, respectively). The Westinghouse accident analysis methodology has been approved by the NRC staff by letter dated April 2, 2008 (Reference 1).

This change request applies to CPSES, Units 1 and 2. The CPSES, Unit 2 amendment is planned for implementation prior to startup from the spring 2008 refueling outage and the CPSES, Unit 1 amendment is planned for implementation prior to startup from the fall 2008 refueling outage.

2.0 REGULATORY EVALUATION

The following regulatory bases and guidance documents are applicable to the proposed TS changes and were considered by the NRC staff in its review of the application:

1. Title 10 of the *Code of Federal Regulations* (10 CFR), Appendix A, General Design Criteria (GDC) 10 requires that protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). GDC 15 requires that protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

2. Section 50.36 of 10 CFR, provides the regulatory requirements for the content required in a licensee’s TSs. Section 50.36 states, in part, that the TSs will include surveillance requirements (SRs) to assure that the quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation (LCOs) will be met. Further 10 CFR 50.36(d)(1)(ii)(A) states: “[l]imiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.”

3. Regulatory Guide (RG) 1.105, “Setpoints for Safety-Related Instrumentation,” describes a method acceptable to the NRC staff for complying with the NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

4. Regulatory Issue Summary (RIS) 2006-17, NRC Staff Position on the Requirements of 10 CFR 50.36, “Technical Specifications,” Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels.

3.0 TECHNICAL EVALUATION

3.1 TS 3.1.4, Rod Group Alignment Limits

This change would revise TS SR 3.1.4.3 to reflect a different value for the Rod Cluster Control Assembly (RCCA) drop time. The current value is ≤ 2.4 seconds and the proposed value is ≤ 2.7 seconds. The proposed maximum rod drop time of 2.7 seconds was used in each of the revised transient and accident analyses.

The staff reviewed the non-loss-of-coolant accident (non-LOCA) analyses using the proposed Westinghouse accident analyses methodologies which have been summarized in NRC safety evaluation dated, April 2, 2008 (Reference 1). The staff concluded that: (1) the licensee used the approved codes and methodologies to perform transient analyses; (2) the values used for the input parameters are conservative in predicting the worst consequences; (3) the computer codes are reliable in calculating core power, pressure, temperature, reactor coolant system (RCS) flow and pressurizer water volume during the transient; and (4) the results of the analyses show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Hence, the NRC staff concludes that the change of the maximum rod drop time of 2.7 seconds is acceptable.

3.2 TS Table 3.3.1-1, RTS Instrumentation

The licensee proposed to revise CPSES, Units 1 and 2, RTS instrumentation to reflect a new Safety Analysis Limit developed with the revised accident analysis and updated uncertainty analyses. The basic uncertainty algorithm used to determine the overall instrument uncertainty for the RTS trip functions is the square-root-sum-of-the-squares of the applicable uncertainty terms. This approach is consistent with NRC RG 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” and Part 1 Instrument Society of America Standard 67.04.01-2006, “Setpoints for Nuclear Safety-Related Instrumentation.”

The staff has reviewed the licensee’s setpoint methodology and calculations, and concludes that the methodology demonstrates that the proposed AVs are reasonable. The licensee has defined the AV as the Nominal Trip Setpoint (NTSP) plus or minus the calibration accuracy in the non-conservative direction which is closer to the Safety Analysis Limit. The licensee has included footnotes that state: (1) if the as-found channel setpoint is conservative with respect to the AV, but outside its predefined as-found acceptance band, the channel shall be evaluated to verify it is functioning as required before returning the channel to service, and (2) the instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the NTSP or a value that is more conservative than the trip setpoint or the channel shall be declared inoperable. The staff has determined that the licensee’s setpoint calculation meets the guidance provided in RIS 2006-17.

3.2.1 Power Range Neutron Flux High and Low Reactor Trip Function

The proposed change would revise TS Table 3.3.1-1, Function 2, to reflect different values of 109.6 percent and 25.6 percent of span for the Allowable Values for the Power Range Neutron Flux High and Low Trip Functions, respectively. The original values of the Safety Analysis Limits (SALs) for these trip functions were based on a Rated Thermal Power of 3411 megawatts thermal (MWt). When both CPSES units were previously uprated by approximately 1.4 percent power, the SALs were retained, but renormalized to the new Rated Thermal Power of 3458 MWt. With the proposed adoption of the Westinghouse accident analyses methodologies, the SALs for these functions were returned to previous values based on the assumed core power level of 3612 MWt.

Based on Reference 1, the NRC staff's review confirmed that the licensee and its vendor have processes to assure that the CPSES-specific input parameter values and ranges and operator action times (where appropriate) that were used to conduct the analyses will assure that 10 CFR 50.46 limits are not exceeded, and that long- term cooling can be assured for all break sizes by providing the means to remove decay heat for extended periods, while also preventing the precipitation of boric acid for all break sizes and locations. Furthermore, the NRC staff finds that the analyses were conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM BE-LOCA and NOTRUMP SBLOCA methodologies, and that the results satisfied the requirements of 10 CFR 50.46(b), based on the proposed stretch power uprate (SPU) conditions. From Reference 1, the staff finds that the values used for the input parameters are conservative in predicting the worst consequences and the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Based on the findings from Reference 1, the staff concludes that the changes to the Power Range Neutron Flux High and Low Trip Functions are acceptable.

3.2.2 Overtemperature N-16 Reactor Trip Function

The overtemperature ΔT [change in temperature] trip function is designed to provide protection against departure from nucleate boiling (DNB) during AOOs. The setpoint is calculated such that a reactor trip will be initiated before the core safety limits are exceeded. The operation of the Main Steam Safety Valves (MSSVs) also limit the power/temperature range over which the overtemperature trip setpoint must provide DNB protection. If an event results in an axial power shape which is more severe, from a DNB standpoint, than the Reference axial power shape, the overtemperature N-16 trip setpoint is automatically reduced in order to assure that the DNB protection afforded by the trip setpoint remains adequate.

As part of the analyses performed to support the implementation of the Westinghouse accident analyses, revised values of the overtemperature N-16 setpoint were selected. From Reference 1, the staff concluded that the licensee used the ΔT trip model and recalculated the reactor trip setpoints as previously approved by References 2 and 3. The adequacy of these setpoints was confirmed by showing that the DNB design basis is met in the non-LOCA analyses that credit these functions for mitigation. Based on the findings from Reference 1, the staff concludes that the change to the overtemperature N-16 reactor trip setpoint is acceptable.

3.2.3 Overpower N-16 Reactor Trip Function

The overpower ΔT trip function is designed to provide protection against fuel centerline melting during AOOs. As part of the analyses performed to support the implementation of the Westinghouse accident analyses, revised values of the overpower N-16 setpoint were selected. From Reference 1, the staff concluded that the licensee used the ΔT trip model and recalculated the reactor trip setpoints as previously approved by References 2 and 3. The adequacy of these setpoints was confirmed by showing that the DNB design basis is met in the non-LOCA analyses that credit these functions for mitigation. Based on the findings from Reference 1, the staff concludes that the change to the overpower N-16 reactor trip setpoint is acceptable.

3.2.4 Steam Generator (SG) Water Level Low-Low Reactor Trip Function

The proposed change would revise TS Table 3.3.1-1, Functional Unit 14 to reflect a different value for the Allowable Value for the Steam Generator Water Level – Low-Low trip function. In the new analyses performed in accordance with the Westinghouse methodologies, different SALs are assumed for some of the analyses, in which this trip function is credited. Because of potential adverse containment environments, the feedwater line break analysis is most limiting. In Reference 1, the staff reviewed the licensee's results using RETRAN-02 to perform analyses of various events, which included loss of normal feedwater. The staff found that the code is reliable in calculating the system responses during transients and the results of the transient analysis meet the acceptance criteria for the analysis of record. Furthermore, the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, are met. Based on the findings, the staff concludes the changes are acceptable.

3.3 TS Table 3.3.2-1, ESFAS Instrumentation

The licensee proposed to revise CPSES, Units 1 and 2, ESFAS instrumentation to reflect a new Safety Analysis Limit developed with the revised accident analysis and updated uncertainty analyses. The basic uncertainty algorithm used to determine the overall instrument uncertainty for the ESFAS trip functions is the square-root-sum-of-the-squares (SRSS) of the applicable uncertainty terms. This approach is consistent with NRC RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," and Part 1 Instrument Society of America Standard 67.04.01-2006, "Setpoints for Nuclear Safety-Related Instrumentation."

The staff has reviewed the licensee's setpoint methodology and calculations, and concludes that the methodology demonstrates that the proposed AVs are reasonable. The licensee has defined the AV as the NTSP plus or minus the calibration accuracy in the non-conservative direction which is closer to the Safety Analysis Limit. The licensee has included footnotes that state (1) if the as-found channel setpoint is conservative with respect to the AV, but outside its predefined as-found acceptance band, the channel shall be evaluated to verify it is functioning as required before returning the channel to service and (2) the instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the NTSP or a value that is more conservative than the trip setpoint or the channel shall be declared inoperable. The staff has determined that the licensee's setpoint calculation meets the guidance provided in RIS 2006-17.

3.3.1 SG Water Level (P14) – High-High ESFAS Instrumentation

The proposed change would revise TS Table 3.3.2-1, Functional Unit 5b, to reflect a different value for the Steam Generator Water Level – High-High (P-14) trip function. The licensee performed an analysis of an increase in feedwater flow to validate the new SAL using the proposed Westinghouse accident analyses methodologies. Specifically, the methodology is described in WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis," April 1999.

In Reference 1, the staff reviewed the results of the increase in feedwater flow analysis using RETRAN-02. The staff concluded the values used for the input parameters are conservative in predicting the worst consequences and the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Based on the findings, the staff concludes the changes are acceptable.

3.3.2 Steam Generator Water Level Low-Low ESFAS Instrumentation

The proposed change would revise TS Table 3.3.2-1, Functional Unit 6c to reflect a different value for the Allowable Value for the Steam Generator Water Level Low-Low ESFAS trip function. In the new analyses performed in accordance with the Westinghouse methodologies, different SALs are assumed for some of the analyses in which this trip function is credited. Because of potential adverse containment environments, the feedwater line break analysis is most limiting. This change is consistent with the change made with Functional Unit 14, Steam Generator Water Level – Low-Low trip function (Section 3.2.4). Based on the findings from Reference 1, the staff concludes that the change to the Steam Generator Water Level Low-Low ESFAS trip function is acceptable.

3.3.3 Steam Line Pressure Low – ESFAS Instrumentation, Footnote (c)

The proposed change would revise TS Table 3.3.2-1, footnote (c) to reflect new time constants for the compensated steam pressure low – ESFAS trip function. The current time constants used in the steamline pressure – low compensated circuits greatly amplify a sharp pressure decrease. Therefore, in practice, a compensated steamline pressure – low isolation signal could be initiated when the actual steam pressure is significantly higher than the trip setpoint. During the proposed transition to Westinghouse methodologies, the compensation circuitry time constants were reduced from 50/5 to 10/5, resulting in an approximately five-fold decrease in the sensitivity of this circuitry to a decrease in the steamline pressure. The licensee stated that this change is expected to improve operating margins when the reactor operators open the steam dump valves or the atmospheric relief valves to stabilize RCS temperatures or to initiate a plant cool down. By avoiding steam line isolation, thereby keeping the condenser available as the primary heat sink, nuclear safety is also improved.

In Reference 1, the staff reviewed the results of the decrease in steamline analysis. The staff concluded the values used for the input parameters are conservative in predicting the worst consequences and the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Based on the findings, the staff concludes the changes are acceptable.

3.4 TS 3.4.10, Pressurizer Safety Valves

The proposed change would revise the LCO for TS 3.4.10 to be consistent with a pressurizer safety valve (PSV) set pressure of 2460 pounds per square inch gauge (psig) and as-found tolerances of +1 percent to 2 percent of the nominal set pressure. The current TS LCO is equivalent to a nominal set pressure of 2485 psig with an as-found tolerance of ±1 percent. With the proposed adoption of the Westinghouse methodologies at the bounding power level of 3612 MWt, based on the results of the analyses, the licensee found it necessary to reduce the PSV set pressure from 2485 psig to 2460 psig in order to meet the maximum RCS pressure relevant event acceptance criterion of 110 percent of the design pressure. The limiting transient for this function is the loss of load/turbine trip analysis.

In Reference 1, the staff reviewed the results of the turbine trip analysis. The staff concluded the values used for the input parameters are conservative in predicting the worst consequences and the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Based on the findings, the staff concludes the changes are acceptable.

3.5 TS 3.7.1, Main Steam Safety Valves

The proposed change would revise the Required Action for TS 3.7.1, Condition A, to reflect a different maximum power level for continued operation with an inoperable MSSV. The change would also revise TS Table 3.7.1-1 to reflect different maximum Allowable Power levels for operation with inoperable MSSVs.

Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 94-001 which described a potential concern with TS Table 3.7.1-1. The recommended changes from this NSAL were eventually incorporated into the Improved Standard Technical Specifications (ISTS) through Technical Specification Task Force (TSTF) 235, Revision 1. These documents described a simplified heat balance calculation methodology for determining the maximum allowable power level for continued operation with inoperable MSSVs. An alternate approach identified in the NSAL required the performance of specific system thermal hydraulic analyses. These analyses were previously performed for CPSES, Units 1 and 2, but are not considered necessary in the future. With the proposed adoption of the Westinghouse methodologies, the simplified heat balance calculation method will be adopted.

From Reference 1, the staff reviewed the results of the inadvertent closure of MSSVs. The staff concluded the values used for the input parameters are conservative in predicting the worst consequences and the results of the analysis show that the acceptance criteria, satisfying the GDC 10 and 15 requirements regarding the fuel integrity and RCS pressure boundary integrity, for the analysis of record are met. Based on its findings and from review of TSTF-235 and the ISTS, the staff concludes the changes are acceptable.

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. 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 in the *Federal Register* on September 25, 2007 (72 FR 54482). 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. NRC letter dated April 2, 2008, transmitting the safety evaluation and Technical Specification changes to allow the use of Westinghouse developed and NRC approved analytical methods to establish core operating limits for CPSES, Units 1 and 2.

2. WCAP-8745-P-A, “Design Bases for Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions,” September 1986.

3. RXE-91-001-A, “Transient Analysis Methods for Comanche Peak Steam Electric Station Licensing Applications,” dated October 1993.

Principal Contributors: Kerby Scales
Barry Marcus
Jason Paige

Date: April 3, 2008