TXU Power

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Ref: 10CFR50.90

CPSES-200600041
Log # TXX-06003

January 6, 2006

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) DOCKET NOS. 50-445 AND 50-446 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE AMENDMENT REQUEST (LAR) 04-002: REVISION TO TECHNICAL SPECIFICATION (TS) 3.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION (TAC NO. MB2620/2621)

REF:

1) TXU Energy letter logged TXX-04049 from Mike Blevins to the NRC dated April 13, 2004
2) NRC letter from Mohan C. Thadani to Michael R. Blevins dated February 16, 2005
3) Letter from Alexander Marion, Nuclear Energy Institute, to James E. Lyons, Deputy Director, Division of Licensing Project Management dated March 18, 2005
4) TXU Power letter logged TXX-05067 from Mike Blevins to the NRC dated March 18, 2005
5) Letter from James A. Lyons, Deputy Director, Division of Licensing Project Management to Alexander Marion, Nuclear Energy Institute dated March 31, 2005.
6) Facsimile transmission from David H. Jaffe, NRC, to Rob Slough, TXU Power dated April 11, 2005.
7) TXU Power letter logged TXX-05157 from Mike Blevins to the NRC dated August 31, 2005.

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Gentlemen:

In Reference 1 above, TXU Generation Company LP (TXU Energy) transmitted an application for amendment to Facility Operating License Number NPF-87 and NPF-89 for CPSES Unit 1 and Unit 2. The proposed amendment would revise the trip setpoint allowable value for Refueling Water Storage Tank (RWST) Level Low-Low (ESFAS function 7.b) for Unit 2 to be the same as for Unit 1. This change would also revise the frequency for calibration of the RWST water level transmitters for both units from 9 months to 18 months. Reference 2 forwarded a request for additional information to support the amendment application.

As part of a proposed generic resolution to the issues pertaining to the use of the Instrumentation, Systems, and Automation Society (ISA) Standard, ISA 67.04, Part II, Method 3, Reference 3 forwarded a request from the Nuclear Energy Institute for the NRC staff to withdraw Requests for Additional Information (RAIs) concerning license amendment requests (LARs) involving instrument setpoints that are based on ISA Method 3.

Reference 4 provided TXU Power's response to the Staff's request for additional information in Reference 2 and stated TXU Power's intention to conform to the industry resolution of this issue in any future submittals involving setpoints.

On March 31, 2005, Reference 5 provided NRC's response to the Nuclear Energy Institute's letter of March 18, 2005 (Reference 3). Reference 6 forwarded the NRC letter of March 31, 2005 (Reference 5) to TXU Power for action. Reference 7 provided TXU Power's response to the request for additional information forwarded in Reference 6.

On October 27, 2005, TXU Power received via email an additional request for information as shown in the Attachment to this letter. TXU Power provided a response to Questions 1, 3, 5, and 6, also via email, on November 25, 2005. The attachment to this letter provides TXU Power's response to Questions 2 and 4.

The additional information provided in this letter and attachment does not impact the conclusions of the No Significant Hazards Consideration provided in Reference 1. In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Texas State official.

This communication contains no new commitments.

Should you have any questions, please contact Robert A. Slough at (254) 897-5727.

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

Executed on January 5, 2006.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

By: Fred W. Madden
Fred W. Madden
Director, Regulatory Affairs

RAS
Attachment

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

Ms. Alice Rogers
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78756-3189

REQUEST FOR ADDITIONAL INFORMATION

RE: THE REVIEW OF REQUEST FOR REVISION OF TECHNICAL SPECIFICATION 3.3.2 (TAC NOS. MB2620 AND MB2621)

Question #1

Describe the instrumentation setpoint methodology used at Comanche Peak Units 1 and 2 for establishing TS limits.

> This discussion should include acceptable As-Found band, acceptable As-Left band, setting tolerance, and reset criteria used to determine the acceptability of the instrumentation.

Question #1 Response

This question was previously answered via Reference #4 (TXU Power letterTXX-05067).

Question #2:

The NRC staff's concerns are limited to the limiting safety system setting (LSSS) for variables upon which a safety limit has been placed, as discussed in 10 CFR 50.36(c)(ii)(A). For each setpoint to be changed, clarify whether it is an LSSS related to a variable upon which a safety limit has been placed. If you determined that it is not, explain why not.

The staff will generally use the following criteria to determine whether the instrument setpoint being changed falls within the scope of the LSSS issue or not:

(A) Instrument setpoints are for TS functions in the Reactor Trip System.

(B) Instrument setpoints are for TS functions that protect a safety limit (whether or not the Bases designate the function as an LSSS).

(C) Setpoints that are not in Instrumentation LCOs but whose function protects a safety limit (whether or not the Bases designate the function as an LSSS).

Question #2 Response:

The Bases for the CPSES Units 1 and 2 Technical Specification 2.1.1, "Reactor Core SLs," states the following:

> "BACKGROUND
>
> GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational

occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level(the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System Allowable Values in Table 3.3.1-1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Protection for these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS Allowable Values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature N-16 reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation.

Appropriate functioning of the RPS and the steam generator safety valves ensure that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs. Limits on process variables are developed both to protect the reactor core SLs and for compliance with the additional restrictions on hot leg enthalpy and vessel exit quality. The Reactor Core Safety Limit figures, provided in the COLR, reflect these process variable limits."

RWST Level Low-Low (ESFAS function 7.b) is not a function provided by the Reactor Protection System, does not provide any signal or input which is used to generate a protection signal provided by the Reactor Protection System, and does not protect any reactor Safety Limit. Therefore, RWST Level is not a variable for which a Limiting Safety System Setting has been specified for protection of a reactor safety limit.

Question #3

10 CFR 50.36(c)(ii)(A) requires that if it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action. Describe how the surveillance test results and the associated TS limits as determined by the methodology are used to establish the operability of the instrument channel. Include a discussion of plant processes for evaluating channels identified to be operable but degraded. If the requirements for determining operability of the instrumentation being tested are located in a document other that the TS (e.g., plant test procedure), discuss how the requirements of 10 CFR 50.36 are met.

Question #3 Response

This question was previously answered via Reference #4 (TXU Power letterTXX-05067).

Question #4:

10 CFR 50.36(c)(ii)(A) requires that an LSSS be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded. Discuss how TS limits established by the methodology ensure that the SL will not be exceeded. Include in your discussion information on the controls you employ to ensure that the trip set point established after completing periodic surveillance is consistent with your methodology. If the controls are located in a document other than the TS, discuss how those controls satisfy the requirements of 10 CFR 50.36.

Question #4 Response:

See response to Question #2 above. Since the RWST Level Low-Low function is not a Limiting Safety System Setting which has been specified for the protection of a reactor safety limit, this question is not applicable.

Question #5

Enclosed in Reference (2) are draft changes to plant TS that are acceptable to the NRC staff for implementing the concepts in the reference (1) letter related to setpoint allowable values for safety related instrumentation. Specifically, Part A provides two notes that apply to setpoint verification surveillance needed to address instrument trip setpoint allowable value issues, and Part B is a check list that provides the TS Bases content for the two notes in Part A. The staff believes that the TS Notes and the discussion of the content for the related TS Bases will satisfactorily address both the NRC staff's and industry's concerns with instrument settings, and ensure compliance with 10 CFR 50.36, "Technical Specification." Discuss your intent in supporting the requested changes in the LAR.

Question #5 Response

This question was previously answered via Reference #4 (TXU Power letterTXX-05067) and Reference #7 (TXU Power letter TXX-05157). However, based on the determination in the response to Question #2 above, TXU Power wishes to withdraw the proposed changes to the CPSES Technical Specifications provided in Reference #7 and revert back to the changes as initially proposed in Reference #1, i.e., without the requested footnote.

Question #6

In April 13, 2004 submittal, Attachment 1 Section 4.0, Technical Analysis stated that the Rosemount transmitter has displayed significantly better performance than the Veritrak transmitter. A new uncertainty analysis, based on the use of Rosemount transmitter has been developed. Please provide the related documents that support the calibration frequency change from 9 months to 18 months for staff review.

Question #6 Response

This question was previously answered via Reference #4 (TXU Power letterTXX-05067). The requested documents were subsequently made available to the NRC Staff for their review by Mr. Rich Luckett from the Nuclear Energy Institute (NEI). Mr. Luckett has subsequently been replaced at NEI by Mr. Tony Harris.