February 8, 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 – REQUEST FOR ADDITIONAL INFORMATION, LICENSE AMENDMENT REQUEST 07-004, CHANGES TO TECHNICAL SPECIFICATIONS TO REVISE RATED THERMAL POWER FROM 3458 MWT TO 3612 MWT (TAC NOS. MD6615 AND MD6616)

Dear Mr. Blevins:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated August 28, 2007, as supplemented by letters dated October 24 and December 7, 2007, and January 10 and 31, 2008, TXU Generation Company LP (subsequently renamed Luminant Generation Company LLC) submitted Technical Specification changes to revise the rated thermal power for Comanche Peak Steam Electric Station, Units 1 and 2, from 3458 megawatts thermal (MWt) to 3612 MWt for NRC review in accordance with Section 50.90 of Title 10 of the *Code of Federal Regulations*.

The NRC staff has determined that additional information is required to complete the review of the application. The specific information requested is addressed in the enclosure to this letter. The enclosed request for additional information (RAI) was also sent to Mr. Jimmy Seawright via e-mail on January 14 and 29, and February 4, 2008. You are requested to provide your response to the enclosed RAI by February 22, 2008.

The NRC staff notes that RAIs were previously transmitted to you by letters dated December 11, 2007 and January 7, 2008. Response to these RAI was received on January 11 and 31, 2008.

The NRC staff considers that timely responses to RAIs help ensure sufficient time is available for NRC staff to complete its review and contribute toward the NRC's goal of efficient and effective use of staff resources.

If you have any questions, please contact me at (301) 415-3016.

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

Enclosure: Request for Additional Information

cc w/encl: See next page

If you have any questions, please contact me at (301) 415-3016.

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

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ADAMS Accession No. ML080310869 NRR-088 (*) Input memo dated (**) via followup e-mail

OFFICE	DORL/LPL4/PM	DORL/LPL4/LA	DIRS/IOLB/BC	DE/EEEB/BC	DSS/SRXB/BC	DE/EMCB/BC	DORL/LPL4/BC
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DATE	2/4/08	2/4/08	1/10/08	2/4/08	1/25/08	1/25/08	2/8/08

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REQUEST FOR ADDITIONAL INFORMATION

RELATED TO REVIEW ASSOCIATED WITH

STRETCH POWER UPRATE

LUMINANT GENERATION COMPANY LLC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated August 28, 2007, as supplemented by letters dated October 24 and December 7, 2007, and January 10 and 31, 2008, TXU Generation Company LP (subsequently renamed Luminant Generation Company LLC) submitted Technical Specification (TS) changes to revise the rated thermal power for Comanche Peak Steam Electric Station (CPSES), Units 1 and 2, from 3458 megawatts thermal (MWt) to 3612 MWt for NRC review.

The NRC staff has determined that the additional information, as requested by the following NRC branches, is required to complete the review of the application. It should be noted that even if all of the technical branches have completed their initial review of the application; there may be further requests for additional information (RAI) while NRC staff develops the safety evaluation.

- Operator Licensing and Human Performance Branch
- Mechanical and Civil Engineering Branch
- Electrical Engineering Branch (follow-up request)

This RAI was also sent to Mr. Jimmy Seawright via e-mail on January 14 and 29, and February 4, 2008.

OPERATOR LICENSING AND HUMAN PERFORMANCE BRANCH

1. In Section 2.11.1.2 (page 2.11-2 of WCAP-16840-P, CPSES, Units 1 and 2, Stretch Power Uprate Licensing Report (SPULR)), it is stated that the Emergency Operating Procedures (EOPs) "may" require changes to setpoints. Identify all changes to the EOPs and state whether any of the changes will affect the time required or the time available to perform EOP operator actions.

2. Identify any operator manual actions credited in the Design-Basis Accident analysis that are affected by the Stretch Power Uprate (SPU). Describe any changes to these actions or the associated controls, displays, or alarms. Specifically, address any changes to the time required or the time available for the actions.

3. On page 2.11-5 of the SPULR, it is stated that "…hardware modifications may involve associated control system modifications…" Identify any control systems that will be modified and describe any resultant effects on plant operator actions, timing, or operator interfaces.

4. On page 2.11-5 of the SPULR, it is stated that no changes to the Safety Parameter Display System (SPDS) are anticipated. Determine whether changes to the SPDS will be made and, if so, describe the changes and their effect on operators' ability to monitor critical safety functions.

5. Describe any controls, displays, or alarms that will be upgraded from analog to digital instruments as a result of the SPU. Describe how upgraded instruments will be tested for usability to confirm that operators can use the digital instruments reliably.

MECHANICAL AND CIVIL ENGINEERING BRANCH

1. Section 2.2.1.1 of the SPULR states that that as discussed in Final Safety Analysis Report (FSAR) Sections 3.6B.2.1.1 and 3.6B.2.1.2, the current licensing basis of CPSES, Units 1 and 2, utilizes the NUREG-16061 Volume 3, leak-before-break (LBB) methodology and excludes the dynamic effects of postulated pipe ruptures in the primary coolant loop piping and 10-inch and larger reactor coolant loop (RCL) branch lines from the design basis.

   a. Identify branch line breaks that are used for the reactor coolant system loss-of-coolant accident analysis for CPSES, Units 1 and 2, at the SPU conditions.

   b. Confirm that the pressurizer spray line, the safety-injection line, and the main steam (MS), feedwater (FW), and auxiliary FW line breaks are considered in the analyses for the SPU conditions. If not, provide technical justification for not including these pipe breaks.

   c. Provide justification that the basis for using LBB methodology is still valid under the proposed SPU conditions.

2. The postulated pipe-break acceptance criteria are described in FSAR Section 3.6B. The pipe-rupture protection criteria conform to the guidelines of Branch Technical Position MEB 3-1. Section 2.2.1.2.2 of the SPULR states that “[a]ffected piping systems were evaluated to address revised SPU operating conditions”. Section 2.2.1.2.2 also states that “[t]he evaluations performed for these piping systems did not result in any new or revised pipe break locations. The evaluations performed for these systems, with the exception of the main feedwater system, did not identify any significant increases in operating conditions that would impact existing design basis pipe break, jet impingement, and pipe-whip analyses. The feedwater system will experience an increase in operating pressure due to SPU. Any resulting modifications to existing pipe whip restraints and/or pipe supports will be provided, if required, to accommodate the higher pipe break loadings.”

    a. Provide a summary description of the evaluations, explaining how the evaluations were performed. Include assumptions and load combinations along with summaries of results that show that you meet the FSAR pipe-break acceptance criteria when SPU conditions are included.

    b. Provide a description of any new pipe-whip restraints, pipe supports or modifications to existing installations that are required due to higher SPU conditions and discuss the projected schedule of work completion.

3. Identify the Code of Record utilized in qualifying Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) piping and pipe supports for SPU conditions. If different from the plant Code of Record, provide justification.

4. Tables 1.1-1 and 1.1-2 of the SPULR provide SPU NSSS performance capability working group (PCWG) parameters. Included are pressures and temperatures for reactor coolant and steam generator (SG).

    a. Provide PCWG parameters for licensed power.

    b. Show licensed and SPU FW flow.

    c. Provide maximum operating and design pressures and temperatures for RCL, MS, and FW piping for licensed and SPU power.

5. Section 2.2.2.1.2.2 of the SPULR refers to Table 2.2.2-1 for RCL pipe stresses. Table 2.2.2-1 is not included in the application.

    a. Confirm that the Tables which contain the RCL pipe stresses are Table 2.2.2.1-1 for Unit 1 and Table 2.2.2.1-2 for Unit 2.

    b. Confirm that the values in these Tables are for SPU conditions and provide corresponding values at current conditions.

    c. Provide stresses, cumulative usage factors (CUFs), and allowable values for loop drains and fills.

d. For CUF values that exceed 0.1, verify that these locations are postulated pipe breaks.

6. Section 2.2.2.2.2.2 of the SPULR states that "[t]he two piping systems of most concern with respect to flow rate increases are the main steam and feedwater systems." Section 2.2.2.2.2.3 states that "[a]dditionally, the implementation of the SPU will result in higher flow rates for several piping systems. Piping systems experiencing these higher flow rates will be reviewed for potential vibration issues."

    a. Identify all piping systems that would experience higher flow rates due to the SPU implementation.

    b. Provide a clear description of the planned activities to address flow-induced vibration (FIV) on susceptible systems.

    c. Describe the methodology and provide the acceptance criteria for the evaluation of FIV for these piping systems.

    d. Provide evaluation summaries which show that the acceptance criteria have been met for SPU conditions.

    e. Describe the vibration monitoring program at the startup for the SPU implementation, its basis, and acceptance criteria. Confirm whether it is in accordance with the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants, Part 3.

7. Section 2.2.2.2.2.2 of the SPULR indicates the following:

    The BOP piping and support systems listed in Section 2.2.2.2.2.1 have been evaluated relative to the impact of SPU. Thermal, pressure, and flow change factors equal to the ratio of SPU to actual analyzed value were determined. "For change factors greater than 1.00, an additional evaluation was performed to address the specific increase in temperature, pressure and/or flow rate in order to determine piping and support system acceptability."

    a. List all systems (inside and outside containment) with "change factors" greater than 1.00.

    b. For systems with "change factors" greater than 1.00, provide the method of your evaluation. Provide a quantitative summary of the maximum stresses and fatigue usage factors (if applicable) for original and SPU conditions with a comparison to Code of Record-allowable stresses. Include only maximum stresses and data at critical locations (i.e. nozzles, penetrations, etc.). List all pipe system modifications (for pipe supports see (d) below) required due to SPU and schedule of completion. For affected nozzles and containment penetrations, provide a summary of loads compared to specific allowable values for the nozzles and penetrations.

c. For systems with a thermal change factor greater than 1.00, provide a description of preoperational measures taken to ensure that thermal expansion will not impose an unanalyzed condition that could potentially overstress piping and supports. In addition, confirm that a program will be in place for monitoring thermal expansion at the startup of the SPU.

d. For systems in (b), state the method used for evaluating pipe supports when considering SPU conditions and confirm that the supports on affected piping systems have been evaluated and shown to remain structurally adequate to perform their intended design function. Provide a description of all pipe support modifications needed to meet design basis at SPU conditions. In addition, list the type, size, loading (current and SPU), and location of supports that need to be modified and added due to SPU conditions.

e. Discuss the schedule for completion of all piping and pipe support modifications and additions.

8. Section 2.2.2.2.2.2 of the SPULR states that "an evaluation of the feedwater system was required to address the flow rate increase resulting from the SPU and its impact on fluid transient loads (that is, water hammer loads) resulting from feedwater isolation valve closure/check valve slam/feedwater pump trip events."

a. Provide a discussion of the results and whether the FW piping and supports are capable of withstanding water-hammer loads resulting from the higher SPU flow rates without any modifications. If modifications are required, provide a detailed description of such modifications and projected completion schedule.

b. Confirm whether stress summaries of Tables 2.2.2.2-1 and 2.2.2.2-2 include stresses due to fluid transient loads associated with the SPU. If not, provide a stress summary of the FW system piping evaluation that contains stresses due to SPU higher fluid transient loads. In addition, for FW nozzles and containment penetrations provide, a summary of loads compared to specific allowable values for the nozzles and penetrations.

9. Section 2.2.2.2.2.3 of the SPULR states that "[f]or piping systems that will experience plant modifications (see LR Section 1.0) to address SPU conditions, the piping and support evaluations will be performed as part of the overall design change package associated with the specific plant modification." Note that SPULR Section 1.0 does not specifically list any piping or pipe support modifications but simply refers to Section 2.2.2.2 for pipe support modifications. The next paragraph in Section 2.2.2.2.2.3 states that "[t]he piping and support evaluations performed concluded that all piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from SPU conditions, with

pipe support modifications if required in order to accommodate the revised support loads due to the SPU." These statements are confusing. Since the evaluations for piping and pipe supports have been completed and it has been determined, as implied above, that all piping systems remain acceptable and will continue to satisfy design-basis requirements, it should be known whether plant modifications are required to satisfy design-basis requirements. Provide a list of all piping systems that will experience plant modifications and clearly describe any plant modification to piping and/or pipe supports. Also, provide the schedule of modification completion.

10. Section 2.2.2.2.2.3 of the SPULR states that "[p]iping systems not specifically listed in Tables 2.2.2.2-1 and 2.2.2.2-2 did not require detailed evaluation to reconcile SPU conditions or involve piping and support systems that will experience plant modifications." This statement is not clear as it implies that piping systems not specifically listed in Tables 2.2.2.2-1 and 2.2.2.2-2 involve piping and support systems that will experience plant modifications.

    a. Clarify whether or not these Tables contain piping systems that require modifications. If not, provide a similar summary for all piping systems that will experience plant modifications.

    b. Identify all piping systems that require modifications. Provide descriptions of the modifications and a projected completion schedule.

11. Tables 2.2.2.2-1 and 2.2.2.2-2 of the SPULR show calculated and allowable stress values and refer to equations 9, 13, and 14.

    a. Confirm whether the referred "Equation 9," "Equation 13," and "Equation 14," correspond to stresses due to ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Subsection NC specified "Occasional Loadings," "Thermal Expansion," and "Sustained Loads" plus "Thermal Expansion," respectively.

    b. Provide the basis for the allowable stress values of 48,000 pounds per square inch (psi) and 24,000 psi and explain quantitatively how they were derived. These allowable values are shown in the 2nd and 4th rows, respectively, of Table 2.2.2.2-1.

    c. The allowable value of 22,500 psi is shown on Table 2.2.2.2-2 for the Extraction Steam to Heaters 3A and 3B. Verify whether this allowable value is correct for the indicated loading condition.

12. The SPULR notes that various ASME Code Class 1 components have failed to meet the primary plus secondary stress intensity requirement of 3Sm (ASME Section III, Paragraph NB 3222.2) but have been found acceptable as they have met alternate subparagraphs of ASME Code, Section III, Subsection NB.

    a. For these components, discuss the basis that allows usage of each of the alternate subparagraphs quoted in the SPULR.

b. Provide summaries of the evaluations which show that the special rules and requirements for exceeding 3Sm as provided by the alternate subparagraphs have been met.

c. Show values in Tables where reference to notes is made without the provision of values, including Tables 2.2.2.5-10 and 2.2.2.5-11.

d. For Tables containing structural integrity values only at SPU conditions, include similar values at current licensing conditions.

For the above 12a through 12d requests, include components from the following tables:

- Reactor Pressure Vessel (RPV) and Supports Tables 2.2.2.3-1 and 2.2.2.3-2.
- Control Rod Drive Mechanism (CRDM) Table 2.2.2.4-1.
- SGs and Supports Table 2.2.2.5-10 and Table 2.2.2.5-11.
- Pressurizer and Supports Table 2.2.2.7-2.
- Reactor Pressure Vessel Internals and Core Supports Table 2.2.3-6.

13. For RPV internals, the FIV analyses results are shown in Tables 2.2.3-4 and 2.2.3-5 of the SPULR. Table 2.2.3-6 shows a summary of component stresses and fatigue usage factors.

    a. Verify whether the reported values in Tables 2.2.3-4, 2.2.3-5, and 2.2.3-6 are for both CPSES units and confirm that the reported values are for SPU conditions. Also, provide corresponding values at current conditions.

    b. Table 2.2.3-5 provides a material endurance limit for the guide tubes of 101.5 x10-6 in/in strain. This material endurance limit appears to be very low. Provide the material for the guide tubes and the source that shows this material endurance limit or the source that is used to derive it.

14. Tables 2.2.2.5-5, 2.2.2.5-6, and 2.2.2.5-7 of the SPULR contain summaries of the FIV analyses results for the CPSES, Unit 1 SG tubes.

    a. Provide similar summaries for CPSES, Unit 2.

    b. Include FIV analyses summaries for the steam dryer, dryer supports, and flow-reflector with respect to the fluid-elastic instability, acoustic loads, and vortex shedding due to the SPU higher steam flow for both CPSES units. If FIV analysis for the dryer, supports, and flow reflector has not been performed or FIV is not thought to be a concern for these components, provide an acceptable justification.

15. Tables 2.2.2.5-10 and 2.2.2.5-11 of the SPULR contain CPSES, Unit 2 stress and fatigue evaluation summaries for the SG primary and secondary components, respectively. Provide similar summaries for CPSES, Unit 1. If there are no changes from the original analyses, provide summaries from the original analyses along with an explanation of why the stresses and fatigue usage factors increased for the CPSES, Unit 2 SG primary and secondary components, but remained the same for CPSES, Unit 1. Include stress and fatigue evaluation summaries for the FW ring for both CPSES units.

16. Discuss in detail the method for avoiding adverse flow effects during power ascension and after achieving SPU conditions. Include systems to be monitored, data to be collected, and methods of data collection. Specify hold points and duration, inspections, plant walkdowns, vibration data locations, and planned data evaluation.

17. Discuss the procedure that will be utilized for preparation and response to the potential occurrence of loose parts as a result of the SPU. The evaluations should also include calculations, when applicable, of the fluid-elastic stability ratio, and stresses due to turbulent and vortex shedding.

18. Provide a summary of the evaluation of thermowells and sample probes in the MS, FW and Condensate piping systems for increased vibrations due to the increased SPU flow rate.

19. Section 2.2.2.4 of the SPULR, CRDM, states that “[a] summary of the stress results of the evaluations performed for the SPU is presented in Tables 2.2.2.4-1 and 2.2.2.4-4 through 2.2.2.4-6,” and “[t]he cumulative usage factors that were calculated are given in Tables 2.2.2.4-2 and 2.2.2.4-3.”

    a. In Tables 2.2.2.4-1 through 2.2.2.4-6, provide corresponding values for the current licensed power and material designations.

    b. For CPSES, Unit 1, provide stress and CUF values at same locations for the upper, middle and lower joints as shown for CPSES, Unit 2 in Tables 2.2.2.4-3 through 2.2.2.4-6.

    c. In Table 2.2.2.4-4, the yield stress for the upset condition is shown to be higher than the yield stress of the normal condition. Explain why the temperature in the upset condition would be lower than the temperature in the normal condition.

FOLLOW-UP RAI FROM ELECTRICAL ENGINEERING BRANCH

The Electrical Engineering Branch has reviewed the responses to its RAI dated December 11, 2007 (pages 5-6 referenced below), provided by the licensee in its letter dated January 10, 2008, and has the following additional questions:

1. In response to Question 1 [December 11, 2007, page 5], the licensee stated that the Post Accident Operability Time (PAOT) impact is minor, since there is only a 7 degree Fahrenheit (°F) difference between the SPU loss-of-coolant accident (LOCA) curve (170 °F) and the intersection point of the Equipment Qualification (EQ) profile (163 °F) at the 24-hour mark.

   Provide a justification as to why the small temperature difference between SPU LOCA curve and the intersection point of the EQ profile at the 24-hour mark and later is considered acceptable.

2. In response to NRC staff Question 4 [December 11, 2007, page 5], the licensee stated that Electric Reliability Council of Texas (ERCOT) was requested to perform the necessary studies to accept the uprated plant power output level changes of about 49 MW for CPSES, Unit 1 and 37 MW for CPSES, Unit 2. However, in a meeting held between ERCOT and TXU Electric Delivery (TXUED) on December 14, 2006, ERCOT stated that an additional steady state study and a stability study would not be required for this small addition of 86 MW to the ERCOT grid. Further, a TXUED letter dated April 24, 2007, states that based on a recent review of the ERCOT Steady State Working Group base cases, transmission circuit line capacities were sufficient to handle the proposed increase in generation capability.

   The NRC staff's review focuses on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power to the plant following implementation of the proposed SPU. It is not clear from TXUED letter dated April 24, 2007, whether all aspects of the impact on the grid due to implementation of the proposed SPU have been evaluated. Also, in response to NRC staff Question 5 [December 11, 2007, page 5], the expected increase in power level change for CPSES, Unit 1 is calculated as 56.76 MW, and for CPSES, Unit 2 as 44 MW, a total of 100.76 MW (compared to 86 MW stated in response to the staff Question 4). Provide an evaluation which confirms that all the aspects of the impact due to the maximum increase in power level of 100.76 MW have been studied.

3. In response to NRC staff Question 6 [December 11, 2007, page 6], the licensee stated that it has been decided to replace the main transformers at CPSES, Units 1 and 2 to remove the voltage restriction and add additional margin. The new main transformers for CPSES, Unit 2 will be installed in the fall of 2009, and in the spring of 2010 for CPSES, Unit 1.

   Confirm whether studies have been completed to determine any adverse impact of the proposed main transformer data on the various grid-related studies.