Luminant

Mike Blevins
Executive Vice President
& Chief Nuclear Officer
Mike.Blevins@Luminant.com

Luminant Power
P O Box 1002
6322 North FM 56
Glen Rose, TX 76043

T 254 897 5209
C 817 559 9085
F 254 897 6652

CP-200700089
Log # TXX-07163

Ref. # 10CFR50.90

November 15, 2007

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NOS. 50-445 AND 50-446
SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) 07-003
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO
LICENSE AMENDMENT REQUEST ASSOCIATED WITH METHODOLOGY
USED TO ESTABLISH CORE OPERATING LIMITS
(TAC Nos. MD5243 and MD5244)

REFERENCE:
1. Letter logged TXX-07063 dated April 10, 2007 submitting License Amendment Request (LAR) 07-003 revision to Technical Specification 3.1, "REACTIVITY CONTROL SYSTEMS," 3.2, "POWER DISTRIBUTION LIMITS," 3.3, "INSTRUMENTATION," and 5.6.5b, "CORE OPERATING LIMITS REPORT (COLR)," from Mike Blevins to the NRC.
2. Letter logged TXX-07126 dated August 16, 2007 supplementing License Amendment Request (LAR) 07-003, from Mike Blevins to the NRC.

Dear Sir or Madam:

Per Reference 1 as supplemented by Reference 2, Luminant Generation Company, LLC (Luminant Power) submitted proposed changes to the Comanche Peak Steam Electric Station, herein referred to as Comanche Peak Nuclear Power Plant (CPNPP), Unit 1 and Unit 2 Technical Specifications to allow the use of several Nuclear Regulatory Commission (NRC) approved accident analysis methodologies to be used to establish core operating limits.

On November 2, 2007, the NRC provided Luminant Power with a request for additional information regarding the methodology used to establish core operating limits. The response to these questions is provided in an attachment to this letter.

In accordance with 10CFR50.91(b), Luminant Power is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new licensing basis commitments regarding Comanche Peak Units 1 and 2.

A001
NRR

Should you have any questions, please contact Mr. J. D. Seawright at (254) 897-0140.

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

Executed on November 15, 2007.

Sincerely,

Luminant Generation Company LLC

Mike Blevins

By: Rafael Flores

Rafael Flores
Site Vice President

Attachment - Request for Additional Information Related to License Amendment Request associated with Methodologies Used to Establish Core Operating Limits

c - E. E. Collins, Region IV
B. K. Singal, NRR
Resident Inspectors, Comanche Peak

Alice Rogers
Environmental & Consumer Safety Section
Texas Department of State Health Services
1100 West 49th Street
Austin, Texas 78756-3189

REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST ASSOCIATED WITH METHODOLOGIES USED TO ESTABLISH CORE OPERATING LIMITS

1. **(Neutronic Design)**
Are you proposing any changes in the lattice physics and/or core design methods (e.g. cross section generation, peaking factors, power distribution uncertainties, reactivity coefficients) as a result of its transition to Westinghouse methods? If so, please discuss the applicability of these methods to the projected Comanche Peak Steam Electric Station, Units 1 and 2 (CPSES), operating conditions.

Response:
In conjunction with the adoption of standard Westinghouse accident analysis methods, the standard Westinghouse reactor physics methods are also being adopted. The standard Westinghouse reactor physics methods include the Phoenix-P/ANC nodal code system; the qualification of these computer codes and methods was approved by the NRC in their safety evaluation of WCAP-11596-P-A. These methods are used for the calculation of reactor core and fuel assembly steady-state reactivity characteristics and power distributions for use in such areas as reload core design, operation follow of cores, and safety and licensing activities. As noted in the WCAP-11596-P-A and the NRC's safety evaluation, these methods are applicable to all Westinghouse-designed reactor cores, which include Comanche Peak Units 1 and 2.

2. **(General)**
It is not clear which updated Final Safety Analysis Report (FSAR) Chapter 15 events are being analyzed with the new licensing basis methodology. Some events (e.g. steam generator tube rupture, Control Rod Misoperation, Boron Dilution, Rod Ejection, anticipated transient without scram) appear to be analyzed with the existing licensing methodology approved for CPSES. Please discuss which event analyses are applying the new methodology for CPSES (i.e. change relative to current) and which events are maintaining the existing methodology (i.e. no change relative to current).

Response:
Table 1 distinguishes between which FSAR Chapter 15 events were analyzed with new methodology versus old methodology.

3. **(General)**
What are the projected equipment out-of-service options for the CPSES Unit 2 Cycle 11 (spring 2008) and Unit 1 Cycle 14 (fall 2008)? Please verify that the non-loss-of-coolant accident analyses provided support these projected equipment out-of-service options for both Units.

Response:
A key assumption in the accident analyses is that the Limiting Conditions for Operation identified in the plant Technical Specifications are met at the initiation of the transient or accident. This assumption precludes the need to assume any equipment is "out of

service" at the start of an analysis. During power operation, regardless of cycle, the Limiting Conditions for Operation identified in the plant Technical Specifications are required to be met. If equipment is temporarily out of service, the Limiting Condition for Operation would not be met, and the required mitigative actions and completion times identified in the corresponding Technical Specification would be satisfied. Failure to return required equipment to service in the required completion time would typically require a plant shutdown. In summary, there is no "equipment out of service" option considered in the accident analyses because there is no such option.

4. **(General)**
Please verify that the methodology transition will not change the current limiting events for the FSAR Chapter 15 event categories, or that the new limiting events are appropriately analyzed for projected CPSES operation.

Response:
The limiting events for the FSAR Chapter 15 event categories did not change. For example, the turbine trip analysis continues to bound the inadvertent closure of a MSIV analysis, the loss of condenser vacuum analysis, and the loss of electrical load analysis with respect to the peak reactor coolant system pressure.

Each of the Chapter 15 events considered the entire range of projected Comanche Peak conditions of operation.

5. **(Non-equilibrium $F_Q(z)$)**
Please provide the cycle specific W(z) analysis consistent with methodology transition for CPSES Unit 2 Cycle 11 (spring 2008) which demonstrates that, under the proposed Relaxed Axial Offset Control, the $F_Q(z)$ limit will not be violated during non-equilibrium conditions.

Response:
Using approved methods and cycle specific FQ(z) analyses, the Axial Flux Difference (AFD) operating space is defined each cycle to ensure that the FQ(z) limit is met. The cycle specific FQ(z) analyses are used to define W(z) FQ surveillance factors to ensure that a valid surveillance of FQ can be accomplished during the required surveillance intervals. The AFD operating envelope and W(z) surveillance factors are provided to the NRC each cycle via the Core Operating Limits Report (COLR). The U2C11 COLR will be provided to the NRC in accordance with the requirements of the Technical Specifications.

6. **(N-16 Trips)**
Please verify that the methodology transition will continue to provide conservative trip timing for events requiring N-16 reactor trips.

Response:
The RETRAN model of the N-16 reactor trip functions, including time constants and trip response times, are unchanged from the RETRAN N-16 reactor trip model developed and used by Luminant Power (formerly TXU Power) for the current licensing basis analyses; therefore, the trip timing remains conservative.

7. (BEACON Core Monitoring)
Will BEACON be credited for any reduction in assumed power distribution uncertainties? Please discuss any effects on core design margins due to the implementation of BEACON and its improved ability to assess operating margins.

Response:
There will be no changes in the actual core design philosophy as a result of the use of BEACON to monitor the core power distributions. Peaking factors will be established through the core design process such that all specified fuel design limits are met. The uncertainties associated with monitoring the peaking factors are then applied when verifying that the measured peaking factors meet the analytical limits. These measurement uncertainties are calculated dynamically by BEACON and used for power distribution measurements when BEACON is operable. Standard power distribution uncertainties are used for flux map power distribution measurements when BEACON is not operable.

Although not credited in the Technical Specification surveillances, much of the improved capability to assess operating margins is derived from the continuous monitoring of the core power distributions, rather than the "snapshot" obtained when performing flux maps with the moveable incore detector system.

8. (LOL/TT)
Please justify that the lower initial reactor coolant system (RCS) pressure (-30 psi) the for pressurization cases (RCS and main steam system) is conservative. In addition, please verify that a conservative cycle exposure point was assumed for the analysis.

Response:
Consistent with approved methodology, a negative uncertainty is used in defining the initial pressurizer pressure for the peak RCS and main steam system (MSS) pressure cases because it conservatively delays (peak RCS pressure case) or even eliminates (peak MSS pressure case) actuation of the high pressurizer pressure reactor trip, i.e., it maximizes the duration of the primary-to-secondary power mismatch.

Also consistent with approved methodology is the use of bounding, minimum reactivity feedback conditions (beginning of core life) in the analysis. This includes a least-negative Doppler-only power coefficient and a 0 pcm/°F moderator temperature coefficient. Minimum reactivity feedback conditions are conservative because reactor power is maintained, maximizing the primary-to-secondary power mismatch, until the time of reactor trip.

9. (LOL/TT)
Please discuss why the Main Steam Isolation Valve closure and Loss of Condenser Vacuum events are bounded by the Turbine Trip event?

Response:
Regarding the loss of condenser vacuum event, it is an event that can cause a turbine trip. In addition, a loss of the condenser would lead to a loss of main feedwater flow

and preclude the use of steam dump to the condenser. To bound the loss of condenser vacuum scenario in the analysis of the turbine trip event, main feedwater flow is assumed to be isolated coincident with the turbine trip, and steam dump is assumed to be unavailable. As such, the results and conclusions of the turbine trip event analysis are applicable to the loss of condenser vacuum event.

Regarding the inadvertent closure of the main steam isolation valves (MSIV) event, it is bounded by the turbine trip event primarily because the main feedwater flow would be maintained for some time following the closure of the MSIVs. As noted above, in the analysis of the turbine trip event, main feedwater flow is conservatively assumed to be isolated coincident with the turbine trip, which is simulated by rapidly (0.1 second) closing the turbine control valve (TCV). The fact that the MSIV closure time can be as much as 50 times longer than the assumed 0.1 second TCV closure time further supports the claim that the turbine trip analysis is bounding. With the longer closure time of the MSIVs, there is steam flow to the turbine for a longer period following event initiation, which lessens the severity of the event because it reduces the primary-to-secondary power mismatch. Note, however, that rapid (0.1-second) MSIV closure is modeled coincident with the initiating event in the turbine trip analysis case that maximizes the peak secondary-side pressure. MSIV closure is conservative in this case because it minimizes the secondary-side volume available for pressurization.

10. **(Loss of Nonemergency Alternating Current Power to Station Auxiliaries) Section 15.2.6.1 of the CPSES FSAR states the following for this event, "This transient is more severe than the turbine trip event analyzed in Section 15.2.3. For this case, the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core." Please justify that this event is indeed non-limiting as indicated in Reference 1 with respect to departure from nucleate boiling (DNB) and overpressure concerns.**

Response:
With respect to primary- and secondary-side overpressurization concerns, the Loss of Nonemergency AC Power to the Station Auxiliaries (LOAC) event is bounded by the Turbine Trip event because of the difference in the timing of the turbine trip in each event. For the LOAC event, the reactor coolant system (RCS) heats up gradually as the steam generators boil down to the low-low level trip setpoint, at which time reactor trip occurs, and is immediately followed by turbine trip. With nuclear power dropping at nearly the same rate that steam flow drops following an LOAC event, there is very little power mismatch between the primary and secondary systems to force an RCS heatup. With the turbine trip initiating the Turbine Trip event, the power mismatch between the primary and secondary systems is much more severe. Thus, the resultant RCS and MSS heatup and pressurization transients are always more severe for the Turbine Trip event compared to the LOAC event.

With respect to DNB concerns, the LOAC event is bounded by the Complete Loss of Forced Reactor Coolant Flow (CLOF) event because of the difference in the timing of reactor coolant pump (RCP) coastdown in each event. Whereas RCP coastdown is assumed to occur coincident with the reactor trip in the LOAC event, RCP coastdown is the initiating fault in the CLOF event, and the reactor trip occurs after the core flow has already degraded. Since the power-to-flow ratio is much greater for the CLOF event, it is bounding with respect to DNB consequences.

11. (Loss of Nonemergency AC Power to Station Auxiliaries)
Please discuss how the 1600.4 cubic feet maximum pressurizer volume was determined for this event if there was no explicit calculation performed.

Response:
The 1600.4 ft^3 pressurizer water volume value presented for the "Loss of Nonemergency AC Power to the Station Auxiliaries" (LOAC) event in Table 2.1-1 corresponds to the limiting case of the "Loss of Normal Feedwater Without Offsite Power" event discussed in detail in Subsection 2.3.3. As indicated in Subsection 2.3.2, the analysis of the LOAC event would be bounded by the analysis of the Loss of Normal Feedwater Without Offsite Power event with respect to long term heat removal. With respect to the LOAC inputs and results summaries presented in Tables 2.1-1, 2.1-4, and 2.1-6, they are reflective of the Loss of Normal Feedwater Without Offsite Power event.

12. (Locked Rotor)
Please describe how the peak centerline temperature (PCT) was calculated for the Locked Rotor event. The NRC staff's generic review of the VIPRE code did not extend to post-critical heat flux (CHF) conditions. Please provide additional details associated with the analysis to ensure VIPRE was applied in conservative manner where post-CHF conditions were entered.

Response:
As described in Attachment 1 to TXX-07126, the supplemental information provided for WCAP-14565-P-A (VIPRE) contains the explanation of Locked Rotor PCT calculation as well as the use of VIPRE for the post-critical heat flux conditions.

13. (Locked Rotor)
What was the calculated fraction of rods in DNB for the Locked Rotor event?

Response:
The calculated percentage of rods-in-DNB for the Locked Rotor event is 0.135%, which is well below the 10% value assumed in the dose consequences calculations.

14. (Dropped rod cluster control assembly (RCCA), Statically Misaligned RCCA, and Single RCCA Withdrawal)
Section 2.5.3 of Reference 1 seems to indicate that these events were analyzed; however, no calculated results are presented. Please provide the key parameter results, including DNB ratio for these events.

Response:
As described in detail in WCAP-11394-P-A, the DNB design basis for the Dropped RCCA event is confirmed to be met by verifying that there is no $F_{\Delta H}$ condition possible that will lead to a violation of the applicable DNB ratio (DNBR) limit following the limiting dropped RCCA scenario. As such, there is no limiting DNBR value explicitly calculated for the Dropped RCCA event. For the anticipated operating conditions

specific to the CPNPP units, the analysis of the dropped RCCA event showed that the limiting FΔH for the dropped RCCA scenario was 1.569. Since this FΔH value is greater than the FΔH limit of 1.480, it is ensured that the DNB design basis is met for the dropped RCCA event. The maximum clad stress was calculated to be 60268 psi versus a design limit 68342 psi.

For the Statically Misaligned RCCA event, there were no explicit DNBR calculations performed. Rather, an allowable FΔH limit was calculated (1.97) that would result in the safety analysis DNBR limit being reached, which was then compared to actual calculated FΔH values corresponding to misaligned RCCA conditions. The maximum calculated FΔH for one RCCA fully inserted with the control bank inserted at the insertion limit is 1.604 (including uncertainty). The maximum calculated FΔH for one RCCA fully inserted with all other rods out is 1.540 (including uncertainty). Since these values are less than the FΔH limit of 1.97, it is demonstrated that the minimum DNBR for the statically misaligned RCCA event is above the applicable safety analysis limit.

For the Single RCCA Withdrawal event, the calculated rods-in-DNB is less than or equal to 0.3%.

15. (Dropped RCCA Bank)
For this event, Section 2.5.3 of Reference 1 states that the WCAP-11394 analysis is bounding. Please verify the applicability of the generic analysis to the projected operating conditions at CPSES.

Response:
WCAP-11394-P-A describes the methodology that was used to analyze the dropped RCCA(s) and dropped RCCA bank events, collectively refer to as the dropped rod event, for CPNPP. The application of the methodology involves three analysis areas:

1) Transient Analysis,
2) Thermal-Hydraulic Analysis, and
3) Nuclear Analysis.

The Transient Analysis portion of the process consists of defining a bounding set of dropped rod statepoints of reactor power, temperature, and pressure at the most limiting time for various dropped rod scenarios. The Thermal-Hydraulic Analysis portion of the process consists of calculating DNBR-based FΔH limit values for each statepoint. The Nuclear Analysis portion of the process consists of verifying that the potential combinations of dropped rods, over the entire core life, will not result in exceeding the limiting DNBR-based FΔH conditions.

While the thermal-hydraulic and nuclear analyses are performed at CPNPP-specific (and cycle-dependent) conditions, the transient analysis statepoints are generic in nature, i.e., plant- and cycle-independent. The generic statepoints, which were generated as part of the development of WCAP-11394-P-A, cover ranges of dropped rod worths, moderator temperature coefficients, and control bank worths that were confirmed to bound those applicable to CPNPP. As the automatic function of the rod control system can have a significant effect on the dropped rod results (statepoints), key parameters used to model the response of the rod control system in the generic statepoint analysis were confirmed to be consistent with those of the CPNPP units. Thus, it has been confirmed that the generic statepoints are applicable to CPNPP.

16. (RCCA Ejection)
Section 2.5.6 of Reference 1 states that the percentage of fuel rods entering DNB is limited to 10 percent of the fuel rods in the core based on generic analysis. Please justify the applicability of the generic analysis to the projected operating conditions at CPSES.

Response:
The Westinghouse RCCA Ejection Topical Report, WCAP-7588 Rev. 1-A, which was reviewed and approved by the NRC in January of 1975, describes the analyses performed to generically demonstrate that the number of fuel rods reaching DNB is less than 10% of the core. In the WCAP, a worst case was selected consisting of a very conservative hot full power (HFP) ejected rod worth of $1 and post ejection peaking factor of 8.3 at beginning of life (BOL) conditions. The HFP case was selected since it has the lowest initial margin to DNB, and at BOL since it results in minimum moderator feedback. The results showed less than 10% of the fuel rods reached DNB.

The HFP ejected rod worths and post-ejection peaking factors shown in the new CPNPP RCCA ejection analysis are well under the values used in the generic analysis. The beginning of cycle case has an ejected rod worth of $0.436 (0.24%Δk/k with a delayed neutron fraction, Beta, of .0055), and a peaking factor of 5.5. Although the end of cycle values are slightly higher with a worth of $0.568 (0.25%Δk/k with a Beta of 0.0044), and a peaking factor of 6.0, the peak fuel temperatures and peak fuel enthalpy are lower than the beginning of cycle case.

Since the ejected rod worths and peaking factors in the new CPNPP RCCA Ejection analysis are bounded by the values used in the WCAP-7588 Rev. 1-A rods-in-DNB study, the generic result applies to CPNPP.

Reference:

1. Transition of Methods Safety Analyses (Attachment 2 to licensee letter dated August 16, 2007, TXX-07126)

Table 1: FSAR Chapter 15 Analysis Methodology

FSAR Section	Event Description	Analysis Methodology (New or Existing)
15.1.1	Decrease in Feedwater Temperature	New
15.1.2	Increase in Feedwater Flow	New
15.1.3	Excessive Increase in Secondary Steam Flow	New
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	(1)
15.1.5	Steam System Piping Failure	New
15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	(2)
15.2.2	Loss of External Electrical Load	(3)
15.2.3	Turbine Trip	New
15.2.4	Inadvertent Closure of Main Steam Isolation Valves	(3)
15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	(3)
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	New
15.2.7	Loss of Normal Feedwater	New
15.2.8	Feedwater System Pipe Break	New
15.3.1	Partial Loss of Forced Reactor Coolant Flow	New
15.3.2	Complete Loss of Forced Reactor Coolant Flow	New
15.3.3/ 15.3.4	Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Shaft Break	New
15.4.1	Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Condition	New
15.4.2	Uncontrolled RCCA Withdrawal at Power	New
15.4.3	RCCA Misalignment	New
15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	(4)
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	Not applicable to CPNPP
15.4.6	Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution)	New (Modes 1 & 2) and Existing (Modes 3, 4 & 5)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Existing
15.4.8	Spectrum of RCCA Ejection Accidents	New
15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	New
15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	(5)
15.5.3	A Number of BWR Transients	Not applicable to CPNPP
15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	New
15.6.3	Steam Generator Tube Rupture	New
15.6.5	Loss of Coolant Accidents	New

(1) Event consequences are bounded by those of the Steam System Piping Failure event (FSAR 15.1.5).
(2) There are no steam pressure regulators at CPNPP that could fail or malfunction and cause a steam flow transient.
(3) Event consequences are bounded by those of the Turbine Trip event (FSAR 15.2.3).
(4) Event is precluded by the CPNPP Technical Specifications.
(5) Event is covered by the analyses of the Boron Dilution event (FSAR 15.4.6) and the Inadvertent Operation of the Emergency Core Cooling System During Power Operation event (FSAR 15.5.1).