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CP-201100619 Ref. # 10 CFR 52
Log # TXNB-11030

May 2, 2011

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555
ATTN: David B. Matthews, Director
Division of New Reactor Licensing

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 3 AND 4
DOCKET NUMBERS 52-034 AND 52-035
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION NO. 5069 (SECTION 19) AND NO. 5683 (SECTION 5.2.3)

Dear Sir:

Luminant Generation Company LLC (Luminant) submits herein the response to Request for Additional Information (RAI) No. 5609 (CP RAI #210) and No. 5683 (CP RAI #215) for the Combined License Application for Comanche Peak Nuclear Power Plant Units 3 and 4. The RAIs address tornadoes during shutdown and the reactor coolant chemistry control program, respectively. The response to CP RAI #210 was initially submitted in TXNB-11025 dated April 19, 2011 and is resubmitted herein to assure that all interested parties receive the information.

Should you have any questions regarding these responses, please contact Don Woodlan (254-897-6887, Donald.Woodlan@luminant.com) or me.

The only commitment in this letter is captured on page 2 of the letter.

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

Executed on May 2, 2011.

Sincerely,

Luminant Generation Company LLC

Donald R. Woodlan for

Rafael Flores

Attachments: 1. Response to Request for Additional Information No. 5069 (CP RAI #210)
2. Response to Request for Additional Information No. 5683 (CP RAI #215)

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NRO

Regulatory Commitments in this Letter

This communication contains the following new or revised commitments which will be completed or incorporated into the CPNPP licensing basis as noted:

Number	Commitment	Due Date/Event
8267	The response to CP RAI #210 Question 19-14 states: The truck bay entrance is designed to withstand a design basis tornado when closed and an administrative control will be in place to ensure that the truck bay entrance is closed when a tornado is nearby or forecast for the immediate area.	Prior to fuel load

Electronic distribution w/attachments:

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Comanche Peak, Units 3 and 4

Luminant Generation Company LLC

Docket Nos. 52-034 and 52-035

RAI NO.: 5069 (CP RAI #210)

SRP SECTION: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation

QUESTIONS for PRA and Severe Accidents Branch (SPRA)

DATE OF RAI ISSUE: 3/15/2011

QUESTION NO.: 19-14

NUREG-0800, Standard Review Plan, Chapter 19 establishes criteria that the NRC staff intends to use to evaluate whether an applicant meets the NRC's regulations.

The staff reviewed the high winds and tornadoes risk assessment and noted that tornadoes were not assessed for shutdown conditions, particularly modes 5 and 6. Please provide an assessment that confirms that tornadoes do not contribute more than ten percent of the total shutdown core damage frequency and total shutdown large release frequency compared to the USAPWR DC PRA. In this assessment, please consider that the containment could be open and that the capability to reclose containment during/following a high wind event could be impacted.

ANSWER:

The risk from tornadoes during modes 5 and 6 does not contribute more than ten percent of the total shutdown core damage frequency and total shutdown large release frequency compared to the US-APWR DCD PRA. FSAR Subsection 19.1.5 has been revised to include this information.

When a tornado strikes the plant during modes 5 and 6, there is a possibility that a tornado-initiated accident scenario may be induced, with some mitigation functions made inoperable due to damage from the tornado strike. SSCs that provide decay heat removal from the reactor and spent fuel are located in seismic category I structures and will not be affected by a design basis tornado strike. The accident scenarios and the quantification results obtained from the tornado risk analysis for plant shutdown are described below.

1) Accident scenarios

Since the SSCs that provide decay heat removal from the reactor and spent fuel are located in seismic category I structures, loss of offsite power (LOOP) is the only initiating event that could be caused by a design basis tornado strike. If a tornado greater than the design basis tornado strikes the plant, degradation and failure of safety-related SSCs may occur that can threaten the ability to continue decay heat removal.

There is a possibility that the containment hatch could be open during plant shutdown. In the US-APWR design, access to the containment hatch is provided through the reactor building, which is designed to withstand a design basis tornado. The containment hatch being open does not affect the vulnerabilities of SSCs located in the containment. The containment can communicate with the outdoors through the equipment hatch and the truck bay entrance in the reactor building. The truck bay entrance is designed to withstand a design basis tornado when closed and an administrative control will be in place to ensure that the truck bay entrance is closed when a tornado is nearby or forecast for the immediate area.

When a tornado strikes the plant, there is a possibility that a loss of decay heat removal may be induced, with some mitigation functions made inoperable due to damage from the tornado. The shutdown PRA was reviewed to identify possible degradation of mitigation functions that may be caused by a tornado strike. The following mitigation and support systems may be degraded by tornado-induced failures from a design basis tornado strike as discussed in FSAR Subsection 19.1.5:

- Alternate CCW utilizing the fire protection water supply system
- Alternate CCW utilizing the non-essential chilled water system
- Non-safety electric power system
- Alternate ac power supply system (this is a mitigation system for LOOP events)

Based on the results of the plant vulnerability analysis and the discussion above, tornado-induced accident scenarios were categorized into three scenarios as shown in Table 1.

2) Quantification

The initiating event frequency F_I [/Y] for each accident scenario was estimated by applying the annual frequency of tornado wind of interest F_T [Y] based on NUREG/CR-4461 Revision 1, duration of shutdown per one refueling outage t_S [hr] , and the refueling outage cycle T_R[Y] using the equation below:

$$F_I = F_T \times (t_s/8760) \times (1/T_R)$$

The conditional core damage probability (CCDP) was calculated based on the PRA model for plant operating state (POS) 8-1, which represents mid-loop operation after refueling. The CCDP for POS 8-1 was considered to be a representative value for other POSs for the reasons below:

- For LOOP events with no alternate CCW available (Scenario 1 in Table 1), the CCDP is dominated by an accident sequence involving failure to restore the CCW function after recovery of emergency power by the gas turbine generators. This accident sequence is common to all POSs, and considering the impact of loss of CCW function on mitigation systems, this sequence will be the dominating accident scenario for all other POSs.
- For LOOP events with no alternate CCW available and no alternate ac (AAC) power available (Scenario 2 in Table 1), the CCDP is dominated by the accident sequence involving failures of the emergency power source that leads to total loss of AC power. This accident sequence is common to all POSs, and considering the impact of loss of AC power on mitigation systems, this sequence will be the dominating accident scenario for all other POSs.
- POS 8-1 has the least number of CCW trains and Class 1E power available. This condition will lead to higher CCDP values for the above mentioned accident scenarios and is considered to result in bounding CCDP values.

The core damage frequency from tornado strikes during plant shutdown is estimated to be approximately $3x10^{-9}$ per year. This value is two magnitudes lower than the low power shutdown risk from internal events reported in the US-APWR DCD PRA. Therefore, it can be concluded that the core damage frequency (CDF) and large release frequency (LRF) from tornado strikes during plant shutdown will not contribute more than ten percent of the shutdown risk determined in the US-APWR DCD PRA.

The dominant core damage scenarios from tornado strike during shutdown are as follows:

- An F-scale 1 or F-scale 2 tornado strikes the plant and the plant switchyard is damaged, resulting in a LOOP event that cannot be recovered within 24 hours. The fire protection water supply system and the non-essential chilled water system are also damaged by the tornado strike, resulting in unavailability of the alternate CCW function. Emergency power is restored by the Class 1E power sources or the AACs, but the CCW cannot be restored due to failures in either the CCW system, ESW system or the safety-related UHS cooling tower fans. Loss of CCW results in loss of residual heat removal (RHR) function and mitigation functions to inject water to the reactor. Water level in the reactor coolant system (RCS) decreases due to evaporation, and eventually the core will be uncovered. The frequency of this scenario is 1.2×10^{-9} per year.

- A tornado with an intensity greater than F-scale 3 but below the design basis tornado strikes the plant and the plant switchyard is damaged resulting in a LOOP event that cannot be recovered within 24 hours. The fire protection water supply system and the non-essential chilled water system are also damaged by the tornado strike, resulting in unavailability of the alternate CCW function. The turbine building is also damaged and the AAC is unavailable. The Class 1E power source fails to operate and total loss of AC power occurs. RHR function and mitigation functions to inject water to the reactor are lost, water level in the RCS decreases due to evaporation, and eventually the core will be uncovered. If the Class 1E power source is available but the CCW cannot be restored, the core will be damaged as described in the loss of CCW sequence above. The frequency of this scenario is 8.5×10^{-10} per year.

- A beyond design basis tornado strikes the plant and all safety systems are damaged. This event leads directly to core damage. The frequency of this scenario is 8.0×10^{-10} per year.

Table 1 - Tornado Accident Scenarios for Plant Shutdown

Scenario	Wind speed	Assumed impact on plant	Initiating Event Frequency [/Y]	CCDP	CDF [/Y]
1	86-135 (F1 and F2 scale)	· LOOP (Initiating Event) · Loss of alternate CCW	4.4E-06	2.8E-04	1.2E-09
2	135-230 (F3, F4 and F5 scale)	· LOOP (Initiating Event) · Loss of alternate CCW, and · Loss of AAC power supply	4.5E-07	1.9E-03	8.5E-10
3	more than 230 mph (F5 scale)	Failure of safety related systems Assumed core damage	8.0E-10	1	8.0E-10

Impact on R-COLA

See attached marked-up FSAR Revision 1 pages 19.1-8, 19.1-9 and 19.1-51.

Impact on S-COLA

None; this response is site-specific.

Impact on DCD

There is no impact on the DCD.

- Alternative ac power supply system (this is a mitigation system for LOOP events, which is initiating event potentially caused by a tornado strike)

LOOP is the most severe initiating event for tornado strikes with enhanced F-scale intensity of F3 or greater and dominates the plant risk profile. LOOP event is applied to the tornado PRA as the most limiting case. RCOL2_19-9

Based on the results of the plant vulnerability analysis and the discussion above, tornado-induced accident scenarios were categorized into three scenarios as shown in Table 19.1-203. The frequency of each scenario derived from the hazard fragility analysis of the T/B is also shown. RCOL2_19-9

- Quantification

For the tornado induced accident scenarios, the CDF was calculated based on the internal event PRA results. The dominant core damage scenarios were the following:

- Enhanced F-scale intensity F1 and F2 tornado strike-induced LOOP and plant switchyard damaged combined with failure of all four CCW or ESW pumps. RCOL2_19-9

The plant switchyard is assumed to be damaged by the tornado strike of enhanced F-scale intensity F1 and F2. A LOOP occurs and CCW or ESW pumps fail to re-start due to common cause failure. Since there is no function to cool reactor coolant pump (RCP), RCP seal loss-of-coolant accident (LOCA) occurs, which results in the core damage. The CDF for this scenario is 2.1E-08/RY.

- ~~Tornado strike induced~~Enhanced F-scale intensity of F3, F4 and F5 tornado strike-induced LOOP and T/B damage combined with failure of all four emergency gas turbine generators.

The plant switchyard and the T/B are assumed to be damaged by the tornado strike with wind speed between 136 mph and 230 mph. A LOOP occurs and the emergency gas turbine generators fail to operate due to common cause failure. The alternative power source is unavailable since the T/B is damaged and total loss of ac power occurs. Offsite power cannot be recovered due to damage of the T/B. ~~Reactor coolant pump (RCP) seal loss of coolant accident (LOCA)~~RCP seal LOCA occurs and eventually the core is damaged. The CDF for this scenario is 2.2E-08/RY. RCOL2_19-9 RCOL2_19-9

- Failure of all safety systems by a beyond design basis tornado. This event leads directly to core damage. This CDF for this scenario is 2.5E-08/RY.

The total CDF caused by a tornado strike during power operation is less than 7E-08/RY. Tornado induced CDF is one order of magnitude lower RCOL2_19-14 than the total CDF for internal events and internal flood and internal fire events.

The CDF from tornadoes during low-power and shutdown (LPSD) does not contribute more than ten percent of the total shutdown CDF and total shutdown LRF compared to the US-APWR DCD PRA. A tornado event during LPSD does not have a significant contribution to risk. RCOL2_19-14

External Flooding

Subsection 2.4.2 systematically considers the various factors that can contribute to the incident of external flooding. Based on the discussions in this section, the contribution of such events to the total CDF is considered insignificant. These events meet the preliminary screening criteria of ANSI/ANS-58.21-2007 (Reference 19.1-8).

Transportation and Nearby Facility Accidents

These events consist of the following:

- Hazards associated with nearby industrial activities, such as manufacturing, processing, or storage facilities
- Hazards associated with nearby military activities, such as military bases, training areas, or aircraft flights
- Hazards associated with nearby transportation routes (aircraft routes, highways, railways, navigable waters, and pipelines)

In Subsection 2.2.3.1, design basis events internal and external to the nuclear power plant are defined as those events that have a probability of occurrence on the order of about 10^{-7}/RY or greater and potential consequences serious enough to affect the safety of the plant to the extent that the guidelines in 10 CFR Part 100 could be exceeded. The following categories are considered for the determination of design basis events: explosions, flammable vapor clouds with a delayed ignition, toxic chemicals, fires, collisions with the intake structure, and liquid spills.

The effects of these events on the safety-related components of the plant are insignificant as discussed in Subsection 2.2.3.1. These events meet the preliminary screening criteria of ANSI/ANS-58.21-2007 (Reference 19.1-8).

Aircraft Crash

As described in Subsection 3.5.1.6, the probability of aircraft-related accidents for CPNPP Units 3 and 4 is less than the order of 10^{-7} per year for aircraft, airway, and airport information reflected in Subsection 2.2. Thus, this event is not addressed further.

CP COL 19.3(4)

Table 19.1-206

Site-specific Key Assumptions

Key Insights and Assumptions	Disposition	
Site-Specific Design Features and Assumptions Design features and assumptions that contribute to high reliability of continuous operation after the 24 hour mission time are the followings.		
- The normal makeup water to the UHS inventory is from Lake Granbury via the circulating water system.	FSAR 9.2.5.2.2	
- UHS transfer pumps and the ESW pumps located in each basin are powered by the different Class 1 E buses. UHS transfer pump operates to permit the use of three of the four basin water volumes.	FSAR 9.2.5.2.2, 9.2.5.3	
- The transfer line is a high integrity line, regularly tested and inspected for corrosion.	FSAR 9.2.1.2.1, 9.2.5.4	
- There are adequate low-level and high-level alarms to provide rapid control room annunciation of a level problem and to allow adequate time to confirm the level and take effective action to address it.	FSAR 9.2.5.5	
- Two basins contain enough water to supply water to remove decay heat for at least 24 hours after plant trip.	FSAR 9.2.5	
Overfill protection will be provided to prevent overfilling the basin and failing the pump(s). This feature is important to prevent degradation of the ESWS when the basin is overfilled due to failure in the transfer pump or circulation system.	FSAR 13.5 Prepare operational procedures to monitor the water level of basin at main control room.	RCOL2_19-11
Backup actions can avoid excessive room heat up in the event of loss of ESW room ventilation. Based on this assumption, loss of ESW room ventilation is not modeled in the PRA model. Operational procedures to avoid excessive room heat up will be prepared.	FSAR 13.5 Prepare operational procedures to monitor the water level of basin at main control room.	RCOL2_19-11
Plant specific SSCs that potentially impact plant safety are seismically designed and thus will not impact the plant HCLPF. HCLPF values for the plant specific SSCs, such as cooling towers, will be confirmed with calculation using EPRI TR-103959 methodology after completion of seismic design and stress analysis of the SSCs.	FSAR 19.1.2.4 FSAR 19.1.5.1 DCD Tier 1 ITAAC #24	RCOL2_19-11
NFPA 1144 minimum setback distance in the Owner Controlled Area will be procedurally maintained. Also, the Owner Controlled Area adjacent to the isolation zone will be cleared of any concentration of vegetation for security reasons.	FSAR 9.5 NFPA 1144 minimum setback distance will be procedurally maintained	RCOL2_19-10
Administrative controls are in place to ensure that the truck bay entrance of the reactor building is closed when a tornado is nearby or source of high wind is forecast for the immediate area.	FSAR 13.5	RCOL2_19-14

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Comanche Peak, Units 3 and 4

Luminant Generation Company LLC

Docket Nos. 52-034 and 52-035

RAI NO.: 5683 (CP RAI #215)

SRP SECTION: 05.02.03 - Reactor Coolant Pressure Boundary Materials

QUESTIONS for Component Integrity, Performance, and Testing Branch 1 (AP1000/EPR Projects) (CIB1)

DATE OF RAI ISSUE: 4/4/2011

QUESTION NO.: 05.02.03-1

Comanche Peak Nuclear Power Plant (CPNPP) FSAR Tier 2, Section 5.2.3.2.1, "Reactor Coolant Chemistry," provides standard supplemental information in STD COL 5.2(12) to address COL 5.2 (12). The applicant replaced a sentence in the US-APWR DCD stating that the COL applicant will meet the latest version of the EPRI water chemistry guidelines in effect at the time of COLA submittal. However, the Staff has determined that the modification of the second sentence of the third paragraph to the USAPWR DCD doesn't properly address the COL Item. It appears that the applicant is trying to address the COL Item by modifying the description of the COL Item as it appears in the US APWR DCD Section 5.2.3.2.1.

Therefore, the staff requests that the applicant address COL item 5.2(12) by providing the version of the EPRI "Primary Water Chemistry Guidelines" that the applicant will use. In addition, the staff requests that the applicant modify the CPNPP FSAR to provide supplemental information to address the COL Item and not simply modify the COL Item as it appears in the US APWR DCD Section 5.2.3.2.1.

ANSWER:

The EPRI Primary Water Chemistry Guidelines in effect at the time of COLA submittal (September 2008) was Revision 6. However, the reactor coolant chemistry control program is continually updated and is based on the latest guidance available. The FSAR has been revised to reflect this response.

Impact on R-COLA

See attached marked-up FSAR Revision 1 page 5.2-1.

Impact on S-COLA

This response is considered standard.

Impact on DCD

None.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.2.1.1 Compliance with 10 CFR 50, Section 50.55a

~~CP~~STD COL 5.2(11)

Replace the third sentence of the second paragraph in DCD Subsection 5.2.1.1 with the following. RCOL2_05.02.01.01-1 CTS-01140

~~Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4~~The licensee uses ASME Code editions and addenda that is the same as those specified in the US-APWR DCD Table 5.2.1-1 and DCD Subsection 3.9.10, Reference 3.9-13. CTS-01140 RCOL2_05.02.01.01-1

5.2.1.2 Compliance with Applicable Code Cases

Replace the third paragraph in DCD Subsection 5.2.1.2 with the following.

~~CP~~STD COL 5.2(1)
~~CP~~STD COL 5.2(2)
~~CP~~STD COL 5.2(3)

~~Comanche Peak Nuclear Power Plant (CPNPP) Units 3 and 4~~The licensee uses no Code Cases listed in Regulatory Guide (RG) 1.84 beyond those listed in the referenced DCD. The use of Code Cases including those listed in RG 1.147 is identified in the inservice inspection (ISI) program (Subsection 5.2.4 and Section 6.6). The use of Code Cases including those listed in RG 1.192 is identified in the inservice testing (IST) program (Subsection 3.9.6 and 5.2.4). CTS-01140

5.2.3.2.1 Chemistry with Reactor Coolant

STD COL 5.2(12) Replace the second sentence of the third paragraph with the following.

~~Water chemistry of the US-APWR reactor coolant will meet the latest version of the EPRI Water Chemistry Guidelines in effect at the time of COLA submittal.~~The reactor coolant chemistry control program is based on the latest effective version of the EPRI Water Chemistry Guidelines. RCOL2_05.02.03-1
