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Ref. # 10CFR50.59
10CFR72.48

February 24, 2016

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

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT
DOCKET NOS. 50-445 AND 50-446 AND 72-74
10CFR50.59 EVALUATION SUMMARY REPORT 019,
10CFR72.48 EVALUATION SUMMARY REPORT 004, AND
COMMITMENT MATERIAL CHANGE EVALUATION REPORT 013

Dear Sir or Madam:

Please find attached the report (Attachment 1) required by 10CFR50.59(d)(2) for those evaluations which were completed at Comanche Peak Nuclear Power Plant (CPNPP) Units 1 and 2 between February 2, 2014, and August 1, 2015, and which were not reported to the Nuclear Regulatory Commission (NRC) in a previous submittal. This report contains a brief description of the changes, tests and experiments implemented or performed pursuant to 10CFR50.59(c), including a summary of the evaluations for each. Items in this report are referenced by their 10CFR50.59 Evaluation Numbers. This report also includes evaluation summaries for certain activities completed pursuant to 10CFR50.59 after August 1, 2015.

Luminant Generation Company LLC (Luminant Power) did complete (or partially complete) one evaluation at CPNPP required by 10CFR72.48 between February 2, 2014, and August 1, 2015, and which were not reported to the Nuclear Regulatory Commission (NRC) in a previous submittal. Therefore, one evaluation summary is required in this report per 10CFR72.48(d)(2) and is included in Attachment 1.

Luminant Power did not make commitment material changes which require reporting for Comanche Peak Units 1 and 2 per the recommendations of Nuclear Energy Institute (NEI) document NEI 95-07, "Guideline for Managing NRC Commitments," Revision 2. Therefore, no descriptions are provided for Commitment Material Change Evaluation Report 013.

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This communication contains no new commitments regarding CPNPP Units 1 and 2.

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

Sincerely,

Luminant Generation Company LLC

Kenneth J. Peters

By:

Timothy A. Hope
Manager, Regulatory Affairs

Attachment: 10CFR50.59 (Report 019) and 10CFR72.48 (Report 004) Evaluation Summaries

c - Marc L. Dapas, Region IV
Balwant K. Singal, NRR
Resident Inspectors, Comanche Peak

10CFR50.59 (Report 019) and 10CFR72.48 (Report 004)
Evaluation Summaries

10CFR50.59 Evaluations:

EV-CR-2010-004331-100
EV-CR-2013-002449-3
EV-CR-2014-005590-2
EV-CR-2014-010354-18
EV-CR-2015-000483-3
EV-CR-2015-009401-2

10CFR72.48 Evaluations:

EV-CR-2013-009020-2

10CFR50.59 (Report 019) and 10CFR72.48 (Report 004) Evaluation Summaries

50.59 Evaluation No. - EV-CR-2010-004331-100 **Units 1 and 2**

Title:

Design modification to de-energize three Auxiliary Feedwater (AFW) pump suction isolation valves for each unit in the closed position by opening their respective breaker to support Rim Pull Forces "Manual Operation Analysis Group 5".

Activity Description:

The scope of this design modification is to position AFW pump suction isolation valves 1-HV-2480, 1-HV-2481 and 1-HV-2482 for Unit 1 and 2-HV-2480, 2-HV-2481 and 2-HV-2482 for Unit 2 in the closed position and then open their respective breaker, thus preventing a fire induced hot-short from causing a mal-operation of the valve(s). This will remove valve position indication via lights on the main control board. Therefore, the valves are being added to the locked valve program such as a manual valve would be for positive position indication. These valves are never opened in normal operation except for testing. Several procedures are being changed to control power to these valves for testing and for abnormal conditions.

Summary of Evaluation:

The current design is for the subject valves to be normally closed with power connected and position indication in the control room. The scope of this design modification package is to close (and verify closed) the Station Service Water (SSW)-AFW pump suction isolation valves in the AFW System and then open their respective breakers. The valves are then "locked closed" to provide positive indication of valve position since valve position is not available in the control room with power removed. These valves are not opened during normal operation or design basis accidents and the SSW supply to AFW is not included in the operating procedure for the AFW System. The Condensate Storage Tank (CST) is the design basis source of auxiliary feedwater for all design basis events. The required volume in the Nuclear Safety Related Condensate Storage Tanks and the normal make-up to the CST bounds all design basis events.

No FSAR design function for AFW or the CST is affected by this change to procedures controlling these valves. The SSW cross-tie was provided for a "beyond the design basis" failure or depletion of the CST (Nuclear Safety Related, seismic Category I).

The only other credit for the SSW-AFW cross-tie is in TS LCO 3.7.6, Action A.1. if the level in the CST would fall below the TS minimum. The LCO requires that the operator "Verify by administrative means OPERABILITY of backup water supply." To support this Action, the SSW-AFW cross-tie valves have been classified as Active (FSAR, Table 3.9-10) and included in the Inservice Testing (IST) program. This condition is only allowed for seven days due to the quality of water from SSW.

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Cross connecting SSW to AFW requires local operator actions. This change adds one step to those local actions. The change being made is for the Operator to re-energize the motor operated valves. The increase in operator burden for aligning the cross-tie is insignificant. There are no response time requirements for completing this alignment.

It should be noted that the SSW-AFW cross-tie cannot be credited for the response to any accident since it could not meet the response time requirements or the water quality requirements in the FSAR. Its credit is only allowed when in the LCO action. Therefore, there must be time to complete the cross-tie without affecting the short term response. The additional step in the cross tie procedure is not a significant change.

Therefore, maintaining the breaker for the cross-tie valve locked normally open does not result in more than a minimal increase in the likelihood of a malfunction (i.e., the inability to open the valves when required).

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50.59 Evaluation No. - EV-CR-2013-002449-3 **Units 1 and 2**

Title:

Reperformance of the 10CFR50.59 review for the SSPS board replacement to consider new 10CFR50.59 guidance for digital changes.

Activity Description:

Replace Westinghouse Solid State Protection System (SSPS) original design printed circuit boards on both Unit 1 and Unit 2 with the Westinghouse new design boards in SSPS Train A and Train B.

The SSPS boards were replaced due to obsolescence issues of the original design boards and to improve reliability.

Specifically, the following boards were replaced in Train A and Train B Solid State Protection System (SSPS) Cabinets of Unit 1 and Unit 2:

- Universal Logic (ULB)
- Safeguards Driver (SGD)
- Under Voltage Driver (UVD)
- Semi-Automatic Tester (SAT)

Summary of Evaluation:

The SSPS old design boards were replaced by new design boards prior to the issue of EGM 14-001. This 10 CFR 50.59 Evaluation for the change is being performed to address EGM 14-001. The evaluation is based on information provided in Topical Report (TR) WCAP-17867-P-A, Revision 1, and NRC Final Safety Evaluation (FSE) for the TR. The review of the TR has determined that no specific site action is needed in the form of appropriate mitigation measure. The evaluation has addressed the plant specific items identified in the NRC FSE of the TR. The evaluation concludes that the activity:

- Does not result in more than a minimal increase in the accidents and malfunctions previously evaluated in the FSAR.
- Does not create a possibility of a new type of event not previously evaluated in the FSAR.
- Does not impact design basis limit for a fission product barrier described in the FSAR.
- Does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses result than any previously evaluated in the FSAR.

Therefore, this activity may be implemented per plant procedures.

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50.59 Evaluation No. - EV-CR-2014-005590-2 **Units 1 and 2**

Title:

Adoption of ANSYS Version 14.0

Activity Description:

Seismic and structural analyses were performed on ancillary equipment provided under 10CFR72 in order to demonstrate conformance with FSAR requirements. Those ancillary equipment that are "special lifting devices" required analyses to demonstrate conformance with the FSAR Heavy Loads Program per the CoC Condition 5. Other ancillary equipment were required to demonstrate that they would not impact a safety related SSC or function through conformance with the FSAR requirements for compliance with Position C.2 of RG 1.29 as Seismic Category II components. The required analyses were performed with ANSYS version 14.0. FSAR Appendix 3.7B(A) described ANSYS versions up to 13.0 as a general purpose finite element program used for seismic analyses. The seismic and structural analyses performed on the ancillary equipment were performed to the most current versions of the ANSYS program. To use ANSYS version 14.0 for seismic and structural analyses is a change in a methodology described in the FSAR.

Summary of Evaluation:

The verification problem solutions generated by Holtec using ANSYS version 14.0 was compared to those obtained from ANSYS version 13.0. A total of 26 different verification problems were used in this comparison. The initial comparison of results found the solutions to be identical for 24 of the verification problems.

An evaluation of the 2 problems that did not match was performed. Additional analyses were performed and input was obtained from ANSYS. The result was that the differences in the problem solutions were determined to be within the expected tolerance of ANSYS.

It is the conclusion of this Evaluation that the use of ANSYS version 14.0 as a computer aided method for performing analysis on a wide range of engineering problems will generate results that are essentially the same as those obtained from ANSYS version 13.0 and ANSYS version 14.0 is not considered a departure from a method of evaluation described in the FSAR.

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Evaluation Summaries

50.59 Evaluation No. - EV-CR-2014-010354-18 **Units 1 and 2**

Title:

Revision of Post Fire Safe Shutdown Procedures ABN-S04A-REV. 6, ABN-S04B-REV. 4. ABN-S05A-REV. S, ABN-S05B-REV. 7, ABN-S06A-REV. S, ABN-S06B-REV. 6

Activity Description:

The CPNPP Fire Safe Shutdown Analysis (FSSA) is being revised, based on implementation of a group of modifications, to remove Operator Manual Actions (OMAs) on components in the protected train which were corrected by the implemented modifications. The manual actions being removed from the FSSA are also being removed from the corresponding ABN series procedures.

Summary of Evaluation:

The procedure changes described herein are changes to the steps taken in the event of a fire to achieve post fire safe shutdown. The changes are made due to the implementation of various modifications and analyses such that specific components are functional due to protection from the effects of the fire, have been modified so that the effects of a fire cannot cause the mal-operation of the component or new components are added and other components no longer have a negative effect on achieving post fire safe shutdown conditions.

These procedure changes are changes to how FSSA components are controlled in the event of a fire and do not increase the probability of an accident or a malfunction of an SSC already analyzed, does not affect radioactive releases in the event of an accident or malfunction, and they do not create the potential for a new accident or a different type of malfunction of the SSCs. These procedure changes do not have any effect on design basis limit for fission product barriers and are unrelated to any methods of evaluation used in establishing design bases or accident analysis.

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50.59 Evaluation No. - EV-CR-2015-000483-3 **Units 1 and 2**

Title:

Westinghouse SHIELD Generation III Shutdown Seal (SDS) design

Activity Description:

Luminant Power has elected to implement the Westinghouse SHIELD Generation III Shutdown Seal (SDS) design as part of its compliance with mitigation strategies, Order EA-12-049.

The Westinghouse SHIELD Shutdown Seal (SDS) limits reactor coolant system (RCS) inventory losses to very low leak rates during a plant event that results in the loss of all reactor coolant pump (RCP) seal cooling.

The SDS is designed to function only when exposed to an elevated fluid temperature downstream of the RCP number 1 seal, such as would occur as a result of the coincident loss of all thermal barrier heat exchanger cooling and number 1 seal injection cooling. SDS activation occurs over the temperature range outside the bounds of normal operation. In its installed and non-activated state, the SDS resides completely out of the normal seal injection and shaft seal leakage flow paths. When activated as designed, on a stationary shaft, the SDS limits RCP shaft leakoff to 1 gallon per minute (gpm) per pump.

If the SDS were to fail (i.e., actuate when not required), the design function of the RCP Seal water leakoff is identified as being adversely impacted during the screen of design changes for Unit 1and Unit 2. The RCP design function, to provide core cooling flow during normal operating conditions, was screened-in for further evaluation given it could be adversely impacted if the SDS were to inadvertently actuate and cause the operators to manually trip the plant.

Also, if the SDS seals were to actuate as designed during a fire, the ability to provide adequate boration from the Refueling Water Storage Tank would be adversely affected. Therefore, the Boric Acid Tanks with their higher boric acid concentration will be utilized for Alternate Shutdown (e.g. fire in the control room).

Summary of Evaluation:

Evaluation of the proposed Westinghouse SHIELD Generation III Shutdown Seal (SDS) design change has demonstrated that actuation, inadvertent actuation or failure to actuate on the part of the SDS results in a less than minimal increase in the likelihood of an accident or a malfunction. There is no potential for creation of a new type of event not previously evaluated. The SDS is not part of the RCS pressure boundary and cannot affect any design basis limit for a fission product barrier. There is no change or departure from a methodology described in the FSAR.

The evaluation addressed the undesired actuation of the SDS. It was concluded that such actuation could not initiate a decrease in reactor coolant system flow rate accident. It was also

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concluded that the likelihood of a malfunction of an SSC important to safety previously evaluated was less than minimal. Actuation of the SDS on a running pump would result in frictional wear of the polymer and metal components of the SDS against the rotating shaft sleeve which could generate small particles and threads of polymer material. This debris would not affect the operation of the number 1, 2, or 3 seals. The suspended particles would be flushed out of the number 1 seal leakoff line and be captured in the seal water return filter. It is unlikely that the small amount of debris would foul the filter. However, in the unlikely event that the filter was heavily fouled, the flow in the number 1 seal leakoff line may be restricted. High filter differential pressure, high number 2 seal leakoff, and a slight decrease in number 1 seal leakoff could occur. This could lead to the operator tripping the RCP and manually shutting down the plant. Although not desired, such a shutdown is within the design and licensing basis of the plant.

Also, the effect on FSSA mitigation strategies to address reduced RCS letdown through the RCP seals due to implementation of SDS does not affect achieving fire safe shutdown performance goals as described in the Fire Protection Report (FPR) since the Boric Acid Tanks can be used to provide the higher concentration of borated water if required. Furthermore, the CPNPP FSSA fire safe shutdown strategies ensure compliance with the approved CPNPP Fire Protection Licensing Basis.

Based on the conclusions described above, prior NRC approval of the proposed change is not required.

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50.59 Evaluation No. - EV-CR-2015-009401-2 **Unit 2**

Title:

Temporary Modification for removal of small right angle tab on bottom of the Fuel Transfer System Fuel Container and Car Assembly.

Activity Description:

The CPNPP Unit 2 fuel transfer car has a physical defect which precludes movement of the car and hence fuel movement. A physical modification to the car recommended by the OEM, Westinghouse was implemented. This Evaluation addresses the 10CFR50.59 evaluation of said change relative to the FSAR-described design of the car in general and the single-failure proof design characteristics which preclude car movement with a fuel assembly not in the horizontal position.

Summary of Evaluation:

The Unit 2 fuel transfer car has a bent tab on the Fuel Transfer System (FTS) Upender Horizontal Mechanical Interlock. The tab was bent to the extent the car was non-functional. The tab is part of the anti-traverse mechanism which is identified in the FSAR as one of several design features of the FTS which preclude car motion when the upender basket is not in the horizontal position. The proposed fix described in the temporary modification would eliminate said interference by cutting/breaking off the bent portion of the affected tab. The resultant configuration, while different from that described in the FSAR, remains single failure-proof through the continued availability of both a mechanical and an electro-mechanical feature to preclude undesirable car motion.

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72.48 Evaluation No. - EV-CR-2013-009020-2 **ISFSI**

Title:

Site Specific Fire Hazard Evaluation (13769701-R-M-00002, Rev. 3, "Comanche Peak ISFSI Project Evaluation of Fire Hazards")

Activity Description:

The site-specific fire hazard at CPNPP is not bounded by the generic fire hazard described in the HI-STORM 100 Cask System FSAR. The activity being evaluated is the site-specific fire hazard analysis, which represents a deviation from the cask FSAR. The fire analysis at CPNPP considers, in addition to the generically analyzed 50 gallons of diesel fuel from the Vertical Cask Transporter (VCT), but also the heat input from combustion of the tires, polyurethane elastomer in the tires, lubricating oils, and hydraulic fluids.

Summary of Evaluation:

This deviation from the generic cask FSAR does not require prior NRC approval. There are no changes to the operating procedures in FSAR Chapter 8 for carrying a loaded cask with a VCT or the design of the VCT or cask system that would make a fire accident more likely. Therefore the frequency of the accident and the likelihood of a malfunction with a different result are not changed from those contemplated in the cask FSAR. The results of the site-specific fire accident show that the MPC confinement boundary remains intact and the potential loss of shielding due to fire damage to the overpack remains small. Therefore, the consequences of a fire accident or malfunction remain the same. Because the fire accident is already described in the cask FSAR, there is no new accident and no malfunction with a different result. The fuel cladding and MPC confinement boundary continue to meet the existing numerical design basis limits for performance. Therefore, a design basis limit for a fission product barrier as described in the FSAR is not exceeded or altered. The analysis for the site-specific fire conditions was performed using the same methodology as described in the cask FSAR; therefore there is no departure from a method of evaluation in the cask FSAR.