UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

November 19, 2010

EA-10-144

Rafael Flores, Senior Vice President
and Chief Nuclear Officer
Luminant Generation Company LLC
Comanche Peak Nuclear Power Plant
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000445/2010006; 05000446/2010006; PRELIMINARY WHITE FINDING

Dear Mr. Flores:

On June 18, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed the onsite portion of a component design bases team inspection at the Comanche Peak Nuclear Power Plant. The enclosed inspection report documents the inspection findings. The team discussed the preliminary findings on June 18, 2010, with Mr. Ben Mays, Vice President, Nuclear Engineering and Support and other members of your staff. After additional in-office inspection, the team leader conducted a final telephonic exit on November 4, 2010, with Mr. Ben Mays, and other members of your staff.

The inspection examined activities conducted under the conditions of your license as they relate to safety and compliance with the Commission's rules and regulations. The team reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

The report discusses preliminary results of the inspection including a finding, which involves the failure to evaluate and then incorporate relevant operating experience information into station instructions, procedures, or drawings. This resulted in a condition where failure of the condensate storage tank diaphragm could block the suction to the auxiliary feedwater pumps. The finding associated with this condition was assessed based on the best available information, including influential assumptions and vendor information, using the applicable significance determination process. The preliminary significance determination was based on Inspection Manual Chapter 0609, Appendix M, “Significance Determination Process Using Qualitative Criteria,” and indicated that the finding was of low to moderate safety significance (White). Additional details of the primary assumptions associated with the preliminary significance determination are documented in Attachment 2 of the enclosure.

The finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC’s Web site at http://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html.

Before we make a final decision on this matter, we are providing you an opportunity to (1) to attend a Regulatory Conference where you can present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and its significance, at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation sections of Attachment 2 of IMC 0609.

In accordance with NRC Inspection Manual Chapter 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 calendar days of the date of this letter. The Significance Determination Process encourages an open dialogue between the NRC staff and the licensee. However, the dialogue should not impact the timeliness of the staff's final determination.

Because the NRC has not made a final determination in this matter, a Notice of Violation is not being issued for the inspection finding at this time. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

This report also documents four NRC identified findings of very low safety significance (Green) and one NRC-identified Severity Level IV violation. The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with the NRC Enforcement Policy. If you contest the noncited violations or the significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Comanche Peak Nuclear Power Plant. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Comanche Peak Nuclear Power Plant.

Please contact Mr. Thomas Farnholtz at (817) 860-8243 and in writing within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Roy J. Caniano, Director
Division of Reactor Safety

Dockets: 50-445; 50-446
Licenses: NPF-87; NPF-89

Enclosure:
NRC Inspection: Report 05000445/2010006; 05000446/2010006
w/Attachments: Attachment 1: Supplemental Information
Attachment 2: Phase 3 Analysis
Attachment 3: Appendix M Analysis
Attachment 4: Vendor Letter

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Inspection Reports/MidCycle and EOC Letters to the following:
ROPreports

Only inspection reports to the following:
OEDO RIV Coordinator (Geoffrey.Miller@nrc.gov)

R:\ ADAMS ML

ADAMS: □ No X Yes	X SUNSI Review Complete	Reviewer Initials: WCS
	X Publicly Available	X Non-Sensitive
	□ Non-publicly Available	□ Sensitive

SRI/EB1	RI/EB1	C:EB1	RI/EB2	SOE/OB	RA/ACES
WCSifre	PAGoldberg	TRFarnholtz	JLWatkins	KDClayton	RLKellar
/RA/	**/RA/**	**/RA/**	**/RA/**	**/RA/**	**/RA/**
11/17/2010	11/17/2010	11/18/2010	11/18/2010	11/18/2010	11/18/2010
SRI/DRP	**C:DRP**	**SRA/DRS**	**RI/TSB**	**D:DRS**	
JGKramer	WCWalker	MFRunyan	BBRice	RJCaniano	
/RA/	**/RA/**	**/RA/**	**/RA/**	**/RA/**	
11/17/2010	11/18/2010	11/18/2010	11/18/2010	11/19/2010	

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-445, 50-446
License:	NPF-87, NPF-89
Report:	05000445/2010006 and 05000446/2010006
Licensee:	Luminant Generation Company LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	FM-56, Glen Rose, Texas
Dates:	May 24 - 28, 2010 Onsite June 1 - 4, 2010 In-Office June 7 - 18, 2010 Onsite
Team Leader:	W. Sifre, Senior Reactor Inspector
Inspectors:	K. Clayton, Senior Operations Engineer P. Goldberg, Reactor Inspector J. Watkins, Reactor Inspector B. Rice, Reactor Inspector J. Leivo, NRC Contractor, Beckman and Associates M. Yeminy, NRC Contractor, Beckman and Associates
Approved By:	Roy J. Caniano, Director, Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000445/2010006, 05000446/2010006; May 24 - 28, 2010 and June 7 – 18, 2010; In-office June 1 – 4, 2010, Comanche Peak Nuclear Power Plant, Units 1 and 2: baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Bases Inspection."

The report covers an announced inspection by a team of four regional inspectors and two contractors. One Apparent Violation, one Severity Level IV violation, and four violations of very low safety significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process," and the crosscutting aspects were determined by using Inspection Manual Chapter 0310, "Components within the Crosscutting Areas." Findings for which the Significance Determination Process does not apply may be Green or be assigned a Severity Level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649*,* "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC Identified Findings

Cornerstone: Mitigating Systems

- Apparent Violation. The team identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, involving the failure of personnel to initiate a SmartForm to enter actual or potential adverse conditions into the corrective action program following receipt of operating experience. Specifically, in July 2002, the licensee received relevant information provided by the manufacturer of the Unit 1 and 2 condensate storage tank diaphragms to ensure the diaphragm integrity would be maintained but failed to enter the issue into the corrective action program as required by Comanche Peak Station Procedure STA-206, "Review of Vendor Documents and Vendor Technical Manuals," Revision 20. In addition, in November 2007, the licensee received industry-operating experience regarding a condensate storage tank diaphragm failure at the Farley Nuclear Plant but failed to enter this issue into the corrective action program as required by Procedure STA-426, "Industry Operating Experience Program," Revision 1. Because actions were not taken in response to the vendor and operating experience information, the diaphragm was susceptible to failure, which could cause a loss of suction to all three auxiliary feedwater pumps. This finding was entered into the licensee's corrective action program as Condition Reports CR-2010-005508, CR-2010-005581 and CR-2010-005962.

  The team determined that the failure to incorporate relevant operating experience information into station instructions, procedures, or drawings to maintain the condensate storage tank diaphragm in a configuration where its performance during accident conditions would preclude blockage of the suction pipes to the auxiliary feedwater pumps was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening, in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding represented the degradation of equipment and functions specifically designed to mitigate the loss of feedwater and that during an event the loss would degrade or make inoperable all three of the auxiliary feedwater pumps. Therefore, the finding was potentially risk significant and a Phase 3 analysis was required. The preliminary significance determination was based on Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," and indicated that the finding was of low to moderate safety significance (White). This finding has a crosscutting aspect in the area of human performance, work practices, because the licensee did not define and effectively communicate expectations regarding procedural compliance and personnel following procedures involving evaluation of operating experience [H.4(b)](Section1R21.2.2).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, which states, in part, that all testing required to demonstrate that structures, systems, components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, as of June 18, 2010, the licensee failed to complete pre-operational testing required to demonstrate that the emergency diesel generator air start system receivers satisfied the requirements and acceptance limits contained in applicable design documents. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005924.

  The team determined that the failure to ensure that the testing required to demonstrate that the Unit 1 emergency diesel generator air start systems will perform satisfactorily in service and in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents was a performance deficiency. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.4).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of June 18, 2010, the licensee failed to properly translate technical specification allowable diesel generator frequency range to design documents. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005563.

  The team determined that the failure to analyze the emergency diesel generators for operation over the entire range of allowed frequency was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding has a crosscutting aspect in the area of problem identification and resolution because the licensee did not effectively incorporate operating experience into the preventive maintenance program for the emergency diesel generators. Specifically, the licensee failed to incorporate information provided in Information Notice 2008-02, which could have affected the capability of equipment such as safety related motor operated pumps to perform their safety function under the most limiting conditions [P.2(a)](Section 1R21.2.5).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of June 18, 2010, the licensee failed to perform an adequate hydrogen evolution calculation, for the safety-related and nonsafety-related batteries, using the most limiting expected condition of forcing maximum current into a fully charged battery which led to a ventilation system design that did not limit hydrogen accumulation to less than two percent of the total volume of the battery areas during all conditions. This finding was entered into the licensee's corrective action program as condition reports CR-2010-005941, CR-2010-005941, and CR-2010-006561.

  The team determined that the failure to adequately perform the hydrogen evolution calculation for the safety-related battery, using the most limiting condition, was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone attribute of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.2.10).

- Severity Level IV. The team identified a noncited violation of 10 CFR 50.9, Completeness and Accuracy of Information, which states, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects. Specifically, on June 20, 2007, the licensee asserted in their response to Generic Letter 2007-01, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," Request 2, that Comanche Peak "periodically performs visual inspection for corrosion and degradation of cable tray supports and a preventive maintenance program for inspection/removal of water from manholes." The team determined the licensee had no preventive maintenance program or procedures in place to govern the inspection or preventive maintenance activities described in their response, and there was no evidence that these manholes, raceways, and supports had ever been inspected prior to November 2009. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005784.

The team determined that the failure to provide accurate information in the licensee's response to Generic Letter 2007-01 was a performance deficiency. The finding is more than minor because the information was material to the NRC's decision-making processes. Specifically, the information requested by Generic Letter 2007-01 was to enable NRC staff to determine whether the applicable regulatory requirements identified in the generic letter (10 CFR Part 50, Appendix A, General Design Criteria 4, 17, and 18; 10 CFR 50.65(a)(1); 10 CFR Part 50, Appendix B, Criterion XI), were being met with regard to the operational readiness of critical systems that could cause a plant transient or mitigate accidents, and to obtain further information on cable failures (Section 1R21.3.2).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of June 18, 2010, the underground duct banks connecting the safeguards buildings to the service water intake structure had installed conduit seals at a low point in the cable manholes, thereby defeating the design requirement to avoid or minimize the accumulation of water in the duct banks. This configuration could result in long-term submergence of safety related medium voltage cables and long-term degradation or failure of the cables. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005843.

  The team determined that the failure to implement a design requirement to avoid or minimize accumulation of water in the underground duct banks was a performance deficiency. The finding is more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance (Section 1R21.3.2).

REPORT DETAILS

1 REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 Component Design Bases Inspection (71111.21)

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than two or a Birnbaum value greater than 1E-6. The items selected included components in both safety-related and nonsafety related systems including pumps, circuit breakers, heat exchangers, transformers, and valves. The team selected the risk significant operating experience to be inspected based on its collective past experience.

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed calculations to independently verify the licensee's conclusions. The team also verified that the condition of the components were consistent with the design bases and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry-operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For the review of operator actions, the team observed operators during simulator scenarios, as well as during simulated actions in the plant.

The team performed a margin assessment and detailed review of the selected risk significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions because of modifications, and margin reductions identified because of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 20 to 30 total samples that include 10 to 20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 15 components, 4 operator actions, and 5 operating experience items.

The selected inspection items supported risk significant functions as follows:

(1) Electrical power to mitigation systems: The team selected several components in the offsite and onsite electrical power distribution systems to verify operability to supply alternating current (ac) and direct current (dc) power to risk significant and safety-related loads in support of safety system operation in response to initiating events such as loss of offsite power, station blackout, and a loss-of-coolant accident with offsite power available. The team also reviewed the licensee's response to Information Notice 2002-12, "Submerged Safety-Related Electrical Cables" and Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients." As such the team selected:

  (a) The emergency diesel generator air start system

  (b) The 6.9 kV engineered safety features switchgear to determine the adequacy of the loading margins available for accident conditions. Response to Information Notice 2007-34, "Operating Experience Regarding Electrical Circuit Breakers"

  (c) The time delay relays for the 6.9Kv bus diesel generator start function

  (d) The emergency diesel generator jacket water heat exchanger systems

  (e) The 125 Vdc safety-related station batteries

  (f) The 6.9 kV safeguard bus undervoltage relays

  (g) The preferred feeder circuit Breakers T1EB2 / T1EB3 to the 480 Vac switchgear 1EB2

  (h) The 125 Vdc Distribution Panel 1ED1 (1-5) fused disconnect switch

(2) Initiating events minimization:

  (a) The condensate storage tank and the water volume available for auxiliary feedwater

  (b) The auxiliary feedwater system flow control valves. Operator actions to isolate auxiliary feedwater flow to a steam generator fault inside containment within the required 10 minutes

  (c) The component cooling water heat exchangers

(d) The residual heat removal recirculation isolation valves

(3) Decay heat removal:

(a) The safety chilled water chillers

(b) The service water pumps and motor operated Valves 1-HV-4286 / 4287 provide cooling water flow to remove decay heat. Response to Generic Letter 1989-13, "Service Water System Problems Affecting Safety-Related Equipment"

.2 Results of Detailed Reviews for Components:

.2.1 Safety Chilled Water Chillers:

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report; design bases documents, calculations, and recent corrective and preventive maintenance of the safety chilled water chillers. These reviews were conducted to verify the adequacy of design for the room coolers, and to verify that heat will be adequately removed during operation of the equipment in the rooms. The team also conducted walkdowns of the room cooler areas to ensure adequate equipment physical condition. Specifically, the team reviewed:

- Heat load and heat removal calculations, including service water temperature and flow requirement calculations for the room coolers.
- Recent thermal performance test results, which included measurement of air and water flow rates, and a calculation of as-found heat exchanger fouling factors.
- Piping and instrumentation diagrams, vendor manual, and a sample of condition reports for the room cooler.

b. Findings

No findings were identified.

.2.2 Condensate Storage Tank

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations, recent corrective action documents, and technical specifications for the condensate storage tank including the water volume available for the auxiliary feedwater system. The inspection included a walkdown of the Unit 1 condensate storage tank and the suction piping for the auxiliary feedwater pumps. In addition, the inspection included a special observation of the condensate storage tank diaphragm, located inside the tank, once without and once with the use of a special camera. Specifically the team reviewed:

- The lowest level at which water is added to the condensate storage tank to increase inventory and the level at which water addition is stopped. The inspectors reviewed these levels with respect to instrument uncertainties and the effect of the nitrogen pressure under the tank diaphragm on the level indication and the setpoints of the level instruments.

- The maintenance and operating history and practices associated with the condensate storage tank and diaphragm.

- The seismic analysis of the condensate storage tank with special emphasis on the seismic analysis of the ring supporting the diaphragm.

- The condensate storage tank diaphragm, the possibility of its failure and the consequences of such a failure. The inspectors paid special attention to the diaphragm's density (specific gravity) and possible modes of failure. The review included discussions with the diaphragm manufacturer and with other industry users of this type of diaphragm.

b. Findings

Failure to Incorporate Relevant Operating Experience Information into Station Procedures Regarding the Condensate Storage Tank and Diaphragm

Introduction. The team identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, for the failure to ensure that vendor information and operating experience were properly evaluated. The failure to properly assess operating experience for the Unit 1 condensate storage tank resulted in a condition where failure of the diaphragm could result in all three auxiliary feedwater pump suction lines from the condensate storage tank being blocked. This finding has a preliminary safety significance of low to moderate (White).

Description. The team reviewed the condensate storage tank with special attention paid to the diaphragm (also referred to as a bladder) installed inside the tank including the possibility of its failure and the consequences of such a failure. The purpose of the diaphragm is to maintain water chemistry, specifically dissolved oxygen, by ensuring separation between the water in the tank and the atmosphere. The team reviewed the diaphragm's physical properties, operating parameters, and searched for possible modes of failure of the diaphragm that could affect safety related functions of the condensate storage tank. The search included discussions with the diaphragm manufacturer and with other users of this type of diaphragm.

The original diaphragm installed in the Unit 1 condensate storage tank was made of a rubber type material that was lighter than water. It was replaced in 1995 with a diaphragm made of a thermoplastic elastomer material which is heavier than water (specific gravity of 1.15 +/- 0.1). The new diaphragm is equipped with four floaters that keep the diaphragm from sinking to the bottom of the tank.

The team reviewed the licensee's practice of adding nitrogen to the condensate storage tank from a nozzle under the water's surface including the effect on the water's surface, the status of the resultant bubble between the water surface and the diaphragm, and whether the nitrogen addition reached the space between the vertical section of the diaphragm and the tank's inner wall. This was important to ascertain whether the gas volume on the water surface communicated with the volume between the diaphragm and the condensate storage tank wall.

The licensee received an operating experience notification, dated July 29, 2002, in the form of a letter from the diaphragm vendor (see Attachment 4). The letter states, in part, that the diaphragm, to function freely without undue stress, must have some air remaining between the tank wall and the diaphragm material on the waterside. The letter also states that, "Our concerns are greater for the absence of gases since we have observed the diaphragm material sticking tighter than wallpaper to the tank wall." The team reviewed procedure STA-206, "Review of Vendor Documents and Vendor Technical Manuals," Revision 20, which was in place at the time the July 2002 letter was received. Section 6.2.2 of this procedure required that for vendor documents that impact site procedures or activities, the reviewer should ensure that an update document for the affected procedure or activity is issued. In addition, this section required that, if anytime during the vendor document review process it is discovered that actual or potential adverse conditions exist, the issue should be entered into the corrective action program. The team determined that the reviewer failed to recognize the potential significance of this vendor information to Comanche Peak, failed to enter this condition into the corrective action program, and failed to initiate appropriate procedure changes that would ensure that the condensate storage tank diaphragm was not placed in a condition that could result in failure of the diaphragm.

Because of a very high nitrogen bubble, the licensee ceased nitrogen injection to the Unit 1 condensate storage tank on March 15, 2010, while continuing to evacuate air from the tank. The team noted that the gas between the diaphragm and condensate storage tank wall was evacuated. The nitrogen bubble between the underside of the diaphragm and the water surface indicated that there was little or no communication between the area above the water surface and the area between the diaphragm and the tank wall.

The team noted that the regularly conducted inspections of the diaphragm done by the licensee were performed by removing an inspection port located at the top of the condensate storage tank and observing the top of the diaphragm. However, due to the tank configuration, the vertical sides of the diaphragm were not visible. On June 11, 2010, the team and the licensee inspected the diaphragm with a camera that could be angled such that the entire diaphragm's vertical section could be observed. The video produced during the inspection revealed that the vertical section of the diaphragm was tightly adhered to the condensate storage tank wall with a vacuum immediately under the top ring from which the diaphragm was hung. The vacuum was created because the licensee ceased nitrogen injection on March 15, 2010, but continued evacuation of the space between the condensate storage tank wall and the diaphragm in an attempt to reduce the size of the 48-inch high nitrogen bubble. As a result, the licensee declared the Unit 1 condensate storage tank inoperable on June 11, 2010, and injected 275 cubic feet of nitrogen into the space between the tank wall and the diaphragm. Following the nitrogen injection the licensee declared the Unit 1 condensate storage tank operable.

The team noted that a condensate storage tank diaphragm failure had occurred at the Farley Nuclear Plant, on October 29, 2007. The failed diaphragm at Farley Nuclear Plant was the same type as that used in the condensate storage tank at the Comanche Peak facility. The modes of failure in that case were identified as (1) the diaphragm was tightly stuck to the wall of the tank which prevented its free decent and ascent with the water level and (2) three of the four floaters that kept the heavier-than-water diaphragm from sinking to the bottom of the tank were found dislodged from the pockets in the diaphragm and floating on the water surface. The floaters that were ejected were sealed on only three sides instead of the required four. The team determined that the floaters of the Comanche Peak condensate storage tank diaphragm were susceptible to the same failure because they were only sealed on three sides. The diaphragm that failed at Farley Nuclear Plant sank to the bottom of the tank resulting in the blocking of pump suction piping.

The team requested that the licensee search for operating experience identifying the failure at Farley Nuclear Plant. The licensee located an operating experience report notifying Comanche Peak of this failure. The notification was received in November 2007. The licensee had assigned notification number OE25829, and "Level 2", which did not require a condition report and only required notification to the appropriate personnel. The operating experience was sent to the appropriate personnel for review, but no condition report was written and there was no documentation of their review. The team reviewed Procedure STA-426, "Industry Operating Experience Program," Revision 1, which was in place at the time the November 2007 operating experience was received. Section 6.2.5 of this procedure required that individuals who receive distribution of operating experience should carefully examine the information for applicability to Comanche Peak programs, procedures, processes, and/or systems, structures, and components. If they determine that further evaluation is necessary or specific improvements need to be made to preclude the event from occurring at Comanche Peak, then the responsible individual should enter the issue into the corrective action program. The team determined that the reviewer failed to recognize the potential significance of this operating experience to Comanche Peak, failed to enter this condition into the corrective action program, and failed to initiate appropriate procedure changes that would ensure that the condensate storage tank diaphragm was not placed in a condition that could result in a similar failure as that seen at the Farley Nuclear Plant.

The action taken on March 15, 2010, to discontinue nitrogen injection and continue evacuation was performed in accordance Procedure COP-303A, "Condensate," Revision 11, Procedure Change Notice 5. The team reviewed this procedure and noted that it contained no specific cautions or other information regarding the concerns specified in the July 2002 vendor letter or the November 2007 operating experience notification for the Farley Nuclear Plant diaphragm failure. The lack of specific instructions regarding nitrogen injection into the condensate storage tank combined with less than complete inspections of the diaphragm resulted in placing the diaphragm in a condition that could potentially result in its failure.

A failure of the diaphragm, similar to the failure that had occurred at the Farley Nuclear Plant, would most likely occur during tank drawdown or during an accident when the auxiliary feedwater pumps are required to operate. The two suction nozzles of the three auxiliary feedwater pumps are located less than three feet apart near the bottom of the tank.

Because of the inspection, the licensee took corrective actions to change the way the diaphragm bubble was inspected, increased the frequency of material inspections, and changed the method of adding nitrogen to the condensate storage tank.

Analysis. The team determined that the failure to incorporate relevant operating experience information into station instructions, procedures, or drawings to maintain the condensate storage tank diaphragm in a configuration where its performance during accident conditions would preclude blockage of the suction pipes to the auxiliary feedwater pumps was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening, in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding represented the degradation of equipment and functions specifically designed to mitigate the loss of feedwater and that during an event the loss would degrade or make inoperable all three of the auxiliary feedwater pumps. Therefore, the finding was potentially risk significant and a Phase 3 analysis was required (see Attachment 2). The preliminary significance determination was based on Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process using Qualitative Criteria," and indicated that the finding was of low to moderate safety significance (White) (see Attachment 3). This finding has a crosscutting aspect in the area of human performance, work practices, because the licensee did not define and effectively communicate expectations regarding procedural compliance and personnel following procedures involving evaluation of operating experience [H.4(b)].

Enforcement. The team identified an apparent violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, which states, in part, that "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Comanche Peak Station Procedure STA-206, "Review of Vendor Documents and Vendor Technical Manuals," Revision 20, Section 6.2.2 stated, in part, "If anytime during the vendor document review process it is discovered that actual or potential adverse conditions exist, a SmartForm shall be initiated." Comanche Peak Station Procedure STA-426, "Industry Operating Experience Program," Revision 1, Section 6.2.5 stated, in part, "Individuals who receive distribution of industry operating experience should carefully examine the information for applicability to Comanche Peak programs, procedures, processes, and/or systems, structures, and components. If they determine that further evaluation is necessary or specific improvements need to be made to preclude the event from occurring at Comanche Peak, then the responsible individual should generate a SmartForm." Contrary to the above, on two occasions, the licensee failed to initiate a SmartForm to enter actual or potential adverse conditions into the corrective action program. Specifically, in July 2002, the licensee received relevant information provided by the manufacturer of the Unit 1 and 2 condensate storage tank diaphragms but failed to enter this issue into the corrective action program or to incorporate this information into station procedures. In addition, in November 2007, the licensee received industry-operating experience regarding a condensate storage tank diaphragm failure at the Farley Nuclear Plant but failed to enter this issue into the corrective action program or to incorporate this information into station procedures governing the operation of the Unit 1 condensate storage tank diaphragm. The purpose of establishing these measures was to avoid damage to the diaphragm, which could then sink to the bottom of the condensate storage tank and potentially cause a loss of suction to all three auxiliary feedwater pumps. This finding was entered into the licensee's corrective action program as Condition Reports CR-2010-005508, CR-2010-005581 and CR-2010-005962. Pending completion of a final significance determination, the performance deficiency will be considered an apparent violation, AV 05000445/2010006-01, "Failure to Incorporate Relevant Operating Experience Information into Station Procedures Regarding the Condensate Storage Tank and Diaphragm."

.2.3 Residual Heat Removal Isolation Valves 8701A, 8701B, 8702A, 8702B

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the residual heat removal isolation Valves 8701A, 8701B, 8702A, and 8702B. Specifically, the team reviewed:

- The valve modifications, safety analyses, system drawings, specifications, test data, system health reports, and operating surveillance procedures.
- The valve vendor manual and related vendor correspondence and system drawings.
- The valve maintenance, and operational requirements related to valve design pressure, torque and stem thrust requirements, and permissive set points for system pressure.
- The design calculations and documentation of periodic surveillance tests were reviewed to verify that design performance requirements were satisfied.
- Maintenance, in-service testing, corrective actions and design change histories were reviewed to assess the potential for component degradation and resulting impact on design margins and performance.

b. Findings

No findings were identified.

.2.4 Emergency Diesel Generator Air Start System

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, maintenance history, operational requirements, modifications, system drawings, specifications, test data, system health, as well as operating and surveillance procedures. The team concentrated its efforts on the air start system's capability of performing its safety function, i.e., delivering a motive force necessary to start the emergency diesel generator and having the capacity to provide 5 start attempts without recharging as required by license basis documents. The team also conducted walkdowns of portions of the emergency diesel generator air start system to verify that the installed configuration was consistent with design basis information and visually inspected the material condition of the air start systems. Specifically, the team reviewed:

- The diesel generator vendor manual, related vendor correspondence, and system drawings related to the air start system.
- Design calculations and documentation of periodic surveillance tests and pre-operational tests to verify that design performance requirements were satisfied.
- Maintenance, in-service testing, corrective actions, and design change histories to assess the potential for component degradation and resulting impact on design margins and performance.

b. Findings

Inadequate Test Control of the Diesel Generator Air Starting System

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, related to the air start systems for both Unit 1 emergency diesel generators. Specifically, the inspectors identified that the pre-operational test for each diesel generator starting air system was not properly designed and implemented to demonstrate that the system as-built configuration satisfied the requirements described in the Updated Final Safety Analysis Report. This resulted in the failure to ensure each diesel generator air receiver is capable of starting the diesel engine five consecutive times without recharging the receivers.

Description. The design basis for the air start system, as stated in the Updated Final Safety Analysis Report, Section 9.5.6.2, "System Description," Revision 102, is that each diesel generator has two 100 percent capacity air systems. Each system includes an air receiver that is sized to store enough air for five starts with an initial nominal air receiver pressure between 220 psig and 250 psig. Additionally, the Updated Final Safety Analysis Report states in Section 9.5.4.4, "Inspection and Testing Requirements," that prior to plant initial operation, the diesel generators are installed and thoroughly tested to demonstrate their ability to perform as designed.

The inspectors reviewed factory test results and pre-operational test results that the licensee performed to demonstrate that the design basis was satisfied at the time the plant was constructed. During the review of the factory test data, the inspectors noted that there was no documentation verifying that the factory tests were an accurate representation of the as-built configuration of the Unit 1 emergency diesel generators or that the test was performed in a manner that was in accordance with the licensee's license basis.

The inspectors reviewed pre-operational test procedures and results and determined that the test did not adequately demonstrate that the Unit 1 emergency diesel generator air start systems met the design requirement of having the capability of cranking a cold diesel five times without recharging the receivers. The test procedure was inadequate in that it allowed the use of the air compressor in between start attempts. Specifically, in Procedure OPT-214A, "Diesel Generator Operability Test," Revision 19, steps 7.1.16 and 7.2.16 stated, in part, "If the air pressure has fallen, place the Diesel Generator Air Compressor 1 switch…in the "Hand" position and allow the pressure to return to the value recorded on Data Sheet 1."

The inspectors could not determine from the test data whether or not the air compressor was used to adjust the air pressure between start attempts. However, since the procedure allowed such actions the test results were unreliable. Furthermore, a review of the pre-operational test results revealed that the test start point of 244 psig and 248 psig for DG 1-01 and DG 1-02 respectively was nonconservative in that it did not bound the normal operating pressure band of 220 psig to 250 psig.

The pre-operational testing results were as follows:

Table 1: Unit 1, Diesel Generator 1-01

Start Attempt	Starting Air Pressure (psig)	Remaining Air Pressure (psig)
1	244	220
2	220	202
3	202	184
4	184	168
5	168	155

Table 2: Unit 1, Diesel Generator 1-02

Start Attempt	Starting Air Pressure (psig)	Remaining Air Pressure (psig)
1	248	224
2	224	202
3	202	184
4	184	169
5	169	154

During a review of past results from surveillance procedure OPT-214A, the inspectors found that the receiver pressure was consistently below the pre-operational test starting pressures of DG 1-01 and DG 1-02. Specifically, in 50 percent of the surveillance tests reviewed, the air receiver pressure was below 240 psig. The starting pressures used in the Unit 1 pre-operational test for diesel generators do not bound the normal operating conditions of the emergency diesel air start systems. Upon questioning, the licensee stated that it was their understanding that the five-start capability for each receiver was a sizing criteria used to purchase the receivers and that demonstrating the capability for five starts per receiver was not required.

The inspectors reviewed applicable documentation that described the initial licensing basis for the emergency diesel generator air start systems. Design Basis Document DBD-ME-011, "Diesel Generator Sets," Revision 30, states, in part, that "the diesel generator sets air start system shall be designed to start the diesel generator set when required and shall meet the requirements of Standard Review Plan Section 9.5.6." Acceptance Criteria 4g in Section 9.5.6 of the Standard Review Plan requires that as a minimum, the air starting system should be capable of cranking a cold diesel engine five times without recharging the receiver(s). The air starting system capacity should be determined as follows: (1) each cranking cycle duration should be approximately three seconds; (2) consist of two to three engine revolutions; or (3) air start requirements per engine start provided by the engine manufacturer, whichever air start requirement is larger.

NRC Safety Evaluation Report (NUREG – 0787) related to the operation of Comanche Peak, Section 9.5.6, states that each emergency diesel generator has an independent and redundant air-starting system consisting of two separate full-capacity air-starting subsystems each with sufficient storage capacity to provide a minimum of five consecutive cold engine starts. Thus, the requirements of 10 CFR 50, Appendix A, General Design Criteria 17 were not met.

The inspectors determined that the five-start capability for each receiver was an initial design requirement and was required to be demonstrated via appropriate testing in order to satisfy 10 CFR 50, Appendix A, General Design Criteria 17. The inspectors determined that the failure to ensure that all testing required to demonstrate that the emergency diesel air start system will perform satisfactorily in service was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents was a violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control.

<u>Analysis</u>. The team determined that the failure to ensure that the testing required to demonstrate that Unit 1 emergency diesel generator sets air start systems will perform satisfactorily in service and in accordance with written test procedures which incorporated the requirements and acceptance limits contained in applicable design documents was a performance deficiency. The finding was more than minor because it was associated with the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was

of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, which states, in part, that "all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Contrary to the above, the licensee failed to ensure that all testing required to demonstrate that structures, systems, and components would perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents. Specifically, as of June 2010, the licensee failed to ensure that pre-operational testing required to demonstrate that the emergency diesel generator air start system receivers satisfied the requirements and acceptance limits contained in applicable design documents. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005924. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 05000445/2010006-002, "Inadequate Test Control of the Diesel Generator Air Starting System."

.2.5 Service Water Pumps

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design bases documents, calculations, corrective maintenance, and post-maintenance tests of the service water pumps to ensure that the equipment was capable of meeting design requirements. The inspectors reviewed calculations related to pump flow, head, and net positive suction head and compared them to requirements to ensure that the pumps were capable of functioning as required especially under loss of offsite power with electrical power supply from the emergency diesel generators. This included the range of emergency diesel generator frequency allowed by technical specifications for unrestricted plant operation. Specifically the team reviewed:

- Piping and instrumentation diagrams and pump alignment requirements
- Pump capacity and number of pumps required for accident mitigation
- Correlation between calculated requirements, test acceptance criteria, and test results

b. Findings

Inadequate Analysis of Emergency Diesel Generator Frequency

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, because the licensee's safety analysis failed to account for the full range of emergency diesel generator frequency allowed by Technical Specifications. Specifically, the licensee analyzed the performance of the service water pumps and all safety related pumps assuming operation at a frequency of 60 Hertz.

Description. The Comanche Peak Technical Specifications allow unrestricted plant operation with emergency diesel generator frequency between 58.8 and 61.2 Hertz (60 Hertz ±2 percent). This frequency range is not accounted for in the safety analysis. The performance of motor operated pumps varies with the speed of the pump, which is directly affected by the frequency of the emergency diesel generator's alternating current. Low frequency will result in a lower flow rate and lower developed head while high frequency will result in a greater flow rate and a higher developed head. The inspectors determined that the licensee's system calculations and safety analyses used a specific diesel frequency of 60 Hertz. Comanche Peak Engineering Report ER-ME-109, "Evaluation of Safety Related Pump Degradation Issues," Revision 1 stated that a frequency other than 60 Hertz "may cause accident or consequences to be outside the bounding limits of the accident analyses. The same is true for the systems [that directly support accident mitigation] such as the Component Cooling Water and Station Service Water pumps." Nevertheless, the licensee did not take steps to correct the condition by using the bounding ±2 percent frequency for all safety related centrifugal pumps.

The team determined that the failure to include the allowable diesel generator frequency of 58.8 Hertz (60 Hertz -2 percent) is nonconservative because the pumps will be operating at a two percent lower flow rate and a lower developed head of about four percent. The overall effect is equivalent to a pump degradation of 4.5 percent as documented in Section 7.3 of Engineering Report ER-ME-109.

The team also determined that the failure to include the allowable diesel generator frequency of 61.2 Hertz (60 Hertz +2 percent) is nonconservative because it will cause the pumps to operate at a higher flow rate and pressure. A two percent higher flow rate will cause the centrifugal pumps to require greater net positive suction head than originally assumed. Operating at a higher frequency could cause vortex formation to occur earlier (at a higher tank water level) than assumed, resulting in the water supply being available for a shorter duration. In addition, diesel fuel would be consumed by the emergency diesel generator at a greater rate making the available fuel last for a shorter duration.

The licensee issued Condition Report CR-2008-000934-00, to address NRC Information Notice 2008-02 "Findings Identified During Component Design Bases Inspections." One of the issues addressed in the condition report is emergency diesel generator frequency, but the licensee failed to note the vulnerability where the safety analysis did not account for the frequency range allowed by Technical Specifications. Moreover, the licensee reviewed the issue of emergency diesel generator frequency in their self-assessment in preparation for this inspection. The issue was identified at other nuclear power plants, but the licensee's self-assessment failed to identify it as a concern at Comanche Peak.

Analysis. The team determined that the failure to analyze the emergency diesel generators for operation over the entire range of allowed frequency was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. Specifically, the licensee has procedures in place that require operators to take specific action to manually maintain the proper frequency range. This finding has a crosscutting aspect in the area of Problem Identification and Resolution because the licensee did not effectively incorporate pertinent operating experience into the preventive maintenance program for the emergency diesel generators. Specifically the licensee failed to incorporate industry-operating experience, which could have affected the capability of equipment to perform their safety function under the most limiting conditions [P.2(a)].

Enforcement. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states, in part, that "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, the licensee failed to ensure that measures were established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of June 18, 2010, the licensee failed to properly translate Technical Specification allowable frequency range to design documents. This finding was entered into the licensee's corrective action program as Condition Report CR-2010-005563. Because this finding was determined to be of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 05000445, 05000446/2010006-03, "Inadequate Analysis of Emergency Diesel Generator Frequency."

.2.6 Component Cooling Water Heat Exchanger 'A'

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design bases documents, calculations, and recent corrective and preventive maintenance of the Train A component cooling water heat exchanger. Specifically, the team reviewed:

- Accuracy of test results, impact of instrument calibration, instrument uncertainties, tube plugging, water temperature (tube and shell sides), and fouling factor.
- Design basis heat load sizing analysis to verify the capability to meet design basis heat removal requirements.
- Heat exchanger design documentation, including specifications, data sheets, and applicable design calculations for agreement with the design basis, safety analysis, and testing requirements.
- Vendor manual requirements for agreement with operating and maintenance procedures and records.
- Current system health report, trend data, inspection frequency, applicable operating experience, as well as significant corrective action documents and their impact on design basis margin.

b. Findings

No findings were identified.

.2.7 Emergency Diesel Generator Jacket Water System

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design bases documents, calculations, corrective maintenance and post-maintenance tests of the emergency diesel generator jacket water heat exchangers to ensure that the equipment was capable of meeting design requirements. The team also performed walkdowns of the heat exchanger areas. Specifically the team reviewed:

- Calculations for heat exchanger fouling, and the minimum allowable flow.
- Design calculations and documentation of periodic surveillance tests to verify that design performance requirements were satisfied.
- Maintenance, in-service testing, corrective actions, and design change histories to assess the potential for component degradation and the resulting impact on design margins and performance.

b. Findings

No findings were identified.

.2.8 6.9 Kv Switchgear

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design bases documents, calculations, corrective maintenance and post-maintenance tests of the nonsafety-related and safety-related portions of the 6.9 Kv switchgear to verify the capability to supply electrical power to safety-related loads. The team performed a visual non-intrusive inspection to assess the installation configuration, material condition, and potential vulnerability to hazards. Specifically, the team reviewed:

- Selected calculations of record that established the electrical loading for the 6.9 Kv switchgear for design basis events to assess the adequacy of the loading margins available for accident conditions.

- Preventive maintenance procedures and the results of the most recent preventive maintenance and refurbishment activities for circuit breakers T1EB2 (serves 480 Vac switchgear 1EB2), T1EB3 (serves 480 Vac switchgear 1EB2), 1APSWS (service water pump motor feeder), and 1EA2-1 (preferred source breaker), to confirm that the activities were consistent with selected vendor manual requirements and that as-found conditions were being properly dispositioned.

- Recent system health reports and a selected sample of condition reports.

b. Findings

No findings were identified.

.2.9 6.9 kV Safeguard Bus Diesel Start Time Delay Relays

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the 6.9kV safeguard bus diesel start time delay relays. The team performed non-intrusive visual inspections of selected sequencer cabinets to identify and evaluate material condition and potential vulnerability to external hazards, such as seismic interactions, and flooding. Specifically, the team reviewed:

- Potential vulnerabilities to common cause failures and their consequences. This included a review of selected portions of schematic diagrams to identify potential common cause failure modes resulting from power supply failures or other circuit failures.

- Associated health reports, component replacement status and history.

- Surveillance test procedures and records.

- Selected condition reports associated with the relays, to assess the reliability of the components.

b. Findings

No findings were identified.

.2.10 125 Vdc Safety-Related Station Batteries

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, and calculations for the 125 Vdc safety-related station batteries. The team performed non-intrusive visual inspections, and witnessed a weekly surveillance test performed by the licensee. The team also walked down the battery room areas to evaluate potential vulnerability to external hazards such as hydrogen accumulation, seismic interactions, and flooding. The team interviewed the system engineer regarding equipment history and conditions. Specifically, the team reviewed:

- Methodology, assumptions, and selected design inputs and results for the battery sizing and 125 Vdc panel loading and voltage drop calculations, to confirm that the batteries would have sufficient capability for supporting design basis events and station blackout events.

- Hydrogen evolution calculations and heating, ventilation, and air conditioning calculations for the battery rooms, to evaluate the capability for controlling hydrogen concentration below acceptable levels under design basis conditions.

- Surveillance procedures and selected results for the weekly, monthly, and quarterly surveillance tests; the 18-month surveillance tests, the service discharge tests, performance discharge tests; and modified performance discharge tests.

- Recent system health reports and a selected sample of condition reports.

b. Findings

Inadequate Evaluation of Hydrogen Generation for Safety-Related and Nonsafety-Related Batteries

Introduction. The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, because the licensee failed to perform an adequate hydrogen evolution calculation, for the safety-related and nonsafety-related batteries, using the most limiting expected condition of forcing maximum current into a fully charged battery which led to a ventilation system design that did not limit hydrogen accumulation to less than 2 percent of the total volume of the battery areas during all conditions.

Description. The inspectors reviewed the calculations associated with the hydrogen generation associated with the Unit 1 and Unit 2 safety and nonsafety-related batteries and battery room ventilation. The team identified a nonconservative calculation, which led to a ventilation system design that did not limit hydrogen accumulation to less than 2 percent of the total volume of the battery areas during all conditions as described in their design basis documents.

The inspectors determined that all of the safety-related and nonsafety-related battery rooms are connected via a common corridor air space through held open fire doors into each battery room. Licensee calculation number X-EB-HV-15, "Hydrogen Level in Battery Rooms, Units 1 and 2," determined hydrogen evolution in the 125Vdc safety-related and nonsafety-related battery rooms. This calculation contained several nonconservative assumptions or design inputs. Specifically, during a loss of offsite power event and loss of coolant accident event when temperatures in the battery rooms can reach 120 degrees Fahrenheit, the hydrogen accumulation in the battery rooms will exceed two percent of the total volume of the battery area when forcing maximum available current into a fully charged battery, which can occur due to a failure of the current limiting feature of the battery charger. The nonconservatism assumptions in this calculation were as follows:

(1) The licensee assumed the equalized current for the charger was the maximum current from the charger. This assumption results in a comparatively small value of hydrogen generation. However, both calculation X-EB-HV-15 and IEEE Standard 484 (referenced in the calculation), state that the worst-case condition exists when forcing maximum current into a fully charged battery such as during a charger failure. This condition would result in a much higher hydrogen evolution rate.

(2) Calculation X-EB-HV-15 did not properly account for the increase in hydrogen evolution rates at the design basis ambient temperature of 120 degrees Fahrenheit for a loss of coolant accident with a loss of offsite power.

(3) Calculation X-EB-HV-15 assumed that the equipment room supply fans would be providing suction to the battery room exhaust fans during a loss of offsite power. However, these fans are nonsafety class and may not be available during a loss of offsite power. This results in a substantially lower airflow through the battery rooms.

(4) Calculation X-EB-HV-15 did not evaluate the airflows, heat loading, and projected room temperatures using as built and design bases conditions of airflows.

(5) The licensee did not consider the hazards introduced to the Class 1E system batteries by the non-Class 1E batteries with similar hydrogen evolution issues that share the same air space due to the open doors.

Analysis. The team determined that the failure to adequately perform the hydrogen evolution calculation for the safety-related and nonsafety-related batteries, using the most limiting condition, was a performance deficiency. This finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone attribute of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Inspection Manual Chapter 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings, and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control which states, in part, that "measures shall be established to assure that applicable regulatory requirements and the design basis, as specified in the license application, for those structures, systems, and components to which this Appendix applies are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. Specifically, as of June 18, 2010, the licensee failed to perform an adequate hydrogen evolution calculation, for the safety-related and nonsafety-related batteries, using the most limiting expected condition of forcing maximum current into a fully charged battery which led to a ventilation system design that did not limit hydrogen accumulation to less than 2 percent of the total volume of the battery areas during all conditions. This finding was entered into the licensee's corrective action program as condition reports CR-2010-005941, CR-2010-005941, and CR-2010-006561. Because this finding was determined to be of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation consistent with the NRC Enforcement Policy: NCV 05000445, 05000446/2010006-04, "Inadequate Evaluation of Hydrogen Generation for Safety-Related and Nonsafety-Related Batteries."

.2.11 Protective Undervoltage Relays 27-2A/1EA2, 27-2A/1EA1, 27-2B/1EA1

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the selected protective undervoltage relays. The team performed non-intrusive visual inspections to identify and evaluate external material condition as well as potential vulnerability to external hazards, such as vulnerability to post-accident radiation dose effects on protective relays, seismic interactions, and flooding. Specifically, the team reviewed:

- Selected schematic diagrams and calculations of record for establishing the setpoints for the 6.9 kV Safeguard Bus undervoltage relays used for motor trip (dead bus status), alternate source breaker closure, and starting the emergency diesel generator, to confirm that the relays would drop out at low voltage and perform their safety functions in accordance with the design basis.

- A sample of recent surveillance test results to confirm implementation of the setpoints in accordance with the calculations and to assess the condition of the relays.

- Recent system health reports and associated actions.

b. Findings

No findings were identified.

.2.12 Preferred Feeder Breakers T1EB2 / T1EB3 to 480 Vac Switchgear 1EB2

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent condition reports for the selected breakers. Specifically, the team reviewed:

- Preventive maintenance procedures and the results of the most recent preventive maintenance and refurbishment activities, to confirm that they were consistent with selected vendor manual requirements and that, as-found conditions were being properly dispositioned.

- Recent system health reports and associated actions.

b. Findings

No findings were identified.

.2.13 Service Water Motor Operated Valves 1-HV-4286 / 4287 (electrical inspection only)

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the selected motor operated valves. The team performed visual inspections of the motor operated valves to identify and evaluate visible material condition as well as potential vulnerability to external hazards, such as seismic interactions, and flooding. Specifically, the team reviewed:

- Electrical calculations to confirm that adequate voltage would be available at the motor terminals for design basis conditions.

- Schematic diagrams to evaluate potential vulnerability to common cause failures and to evaluate testability of circuit functions as evidenced by surveillance procedures.

- Recent system health reports associated with the motor operated valves.

b. Findings

No findings were identified.

.2.14 125 Vdc Distribution Panel 1ED1 Fused Disconnect Switch

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the selected 125Vdc fused disconnect switch. Specifically, the team reviewed:

- Calculations that established the ratings of the fuse and disconnect switch as well as the design and qualification documentation for the cable reduction splice installed within the 125 Vdc distribution panel.

- Schematic diagrams and alarm response procedures to confirm that a blown fuse or misaligned disconnect switch would be alarmed and identifiable in the control room.

- Recent 125 Vdc system health reports.

- Recent work orders governing the preventive maintenance of these components, and performed a non-intrusive visual inspection of a corresponding fused disconnect switch and splice on distribution panel 2ED2, to assess material condition, consistency of the configuration with design and qualification basis, cable supports, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.15 Auxiliary Feedwater System Flow Control Valves

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report, design basis documents, calculations and recent corrective action documents for the auxiliary feedwater system flow control valves. Specifically, the team reviewed:

- Calculations for sizing the valve air accumulators to provide sufficient air and time to hold a valve closed when there is a faulted steam generator.

- Work orders for replacement of parts, and testing in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

- Stroke times and vendor information.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience:

.3.1 NRC Generic Letter 1989-13 "Service Water System Problems Affecting Safety Related Equipment"

a. Inspection Scope

The team reviewed the licensee's responses to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and its compliance with the commitments specified in the responses. The team reviewed the program document and methodology, the validity of practicing frequent testing in lieu of using the design service water temperature when projecting performance to accident conditions, and the validity of the specified frequency of testing with respect to the available margin in fouling factor. The inspectors also reviewed the practice of cleaning the heat exchanger when the margin is low and also every refueling outage. The team reviewed the chemical treatment of the water, scheduled inspections and tests, as well as trending of the fouling factor, trending of service water temperature, and trending of available margin. The review of the generic letter was performed with respect to the programs and actions taken affecting the component cooling water heat exchanger.

b. Findings

No findings were identified.

.3.2 NRC Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"

a. Inspection Scope

The team reviewed the licensee's responses to Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," and its compliance with the commitments specified in the responses. The licensee's response to the generic letter reported one such cable failure of undetermined cause and location, the 'C' phase feeder to the motor for service water pump 1-01, and described the licensee's inspection, testing, and monitoring programs. To assess the licensee's disposition of issues identified in the generic letter, the team selected the service water pump motor feeder cables and reviewed associated documents, including:

manhole, duct bank, and raceway drawings; available medium voltage cable specifications; available documentation associated with dewatering and inspection of manholes; available megger test data and procedures for cable and motor testing; and corrective action history associated with any cable degradation or failures since 2005. The team reviewed the type of insulation systems for the replacement cable, to assess vulnerability to prolonged submergence. The team also visually inspected the condition of two of the fourteen manholes (MH1EB1, MH1EB2) and associated cables and raceways, and interviewed cognizant licensee staff regarding operating history and past conditions.

b. Findings

(1) Failure to Provide Accurate Information in Response to Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"

Introduction. The team identified a Severity Level IV noncited violation of 10 CFR 50.9, Completeness and Accuracy of Information, because the licensee's June 20, 2007 response to Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," did not accurately describe the licensee's programs, procedures, or practices for inspection, testing, and monitoring programs to detect the degradation of inaccessible or underground power cables that support emergency diesel generators, offsite power, essential service water, service water, component cooling water, and other systems that are in the scope of 10 CFR 50.65 (the Maintenance Rule).

Description. The licensee's June 20, 2007 response to Generic Letter 2007-01, Request 2, stated that Comanche Peak, "periodically performs visual inspection for corrosion and degradation of cable tray supports and a preventive maintenance program for inspection/removal of water from manholes. These actions help to eliminate or minimize conditions known to impact cable degradation rates for cables that are within the scope of 10 CFR 50.65."

The team identified the following:

The licensee had no preventive maintenance program or procedures in place to govern the inspection or preventive maintenance activities described in their response, and there was no evidence that these manholes, raceways, and supports had ever been inspected prior to November 2009, as the licensee indicated in their response to Generic Letter 2007-01. During these recent inspections and dewatering activities (using portable pumps), the licensee identified evidence that medium voltage safety-related cables (6900 Vac service water pump motor feeders) had been completely submerged in one Train A manhole and in four Train B manholes.

The licensee also stated in their response to Request 2, that "The underground ductwork at Comanche Peak is designed to slope toward manholes to avoid accumulation of water in the duct banks. The conduits embedded in concrete floors/walls inside the plant are sealed to prevent intrusion of water inside the conduits. These features eliminate or minimize cable exposure to environment of concern known to impact cable degradation rates identified in this generic letter."

The team determined from visual inspection of the conduits entering manholes 1EB1 and 1EB2; review of the results of the licensee's November 2009 inspections; and review of design and construction documents, that water could enter the underground conduit and accumulate in the duct banks, because of the conduit and conduit seal configuration. The licensee concluded from inspections completed in November, 2009, that the metallic conduit that encloses the cables is not watertight, and allows water to enter and flood the underground conduits, particularly from water entering from a sandy filler zone between the duct bank and cable vault structures, or between the duct banks and building structures (for example, the safeguards buildings and the service water intake structure). In addition, the team concluded from visual inspections of MH1EB1 and MH1EB2 and from design documents that the conduits that slope downward to the manholes from the service water intake structure and from the Unit 1 safeguards building were sealed at the manhole, which could result in prolonged submergence of cables within underground conduit. The team determined that this conduit and seal configuration was a design deficiency from original construction.

Analysis. The team determined that the failure to provide accurate information in the licensee's response to Generic Letter 2007-01 was a performance deficiency. The finding is more than minor because the information was material to the NRC's decision-making processes. Specifically, the information requested by Generic Letter 2007-01 was to enable NRC staff to determine whether the applicable regulatory requirements identified in the generic letter (10 CFR Part 50, Appendix A, General Design Criteria 4, 17, and 18; 10 CFR 50.65(a)(1); 10 CFR Part 50, Appendix B, Criterion XI), were being met with regard to the operational readiness of critical systems that could cause a plant transient or mitigate accidents, and to obtain further information on cable failures.

Enforcement. The team identified a Severity Level IV noncited violation of 10 CFR 50.9, Completeness and Accuracy of Information, which states, in part, that "information provided to the Commission be complete and accurate in all material respects." Contrary to the above, the licensee failed to provide information that was complete and accurate in all respects. Specifically, on June 20, 2007, the licensee's response to Generic Letter 2007-01, Request 2, specified that Comanche Peak "periodically performs visual inspection for corrosion and degradation of cable tray supports and a preventive maintenance program for inspection/removal of water from manholes." The licensee had no preventive maintenance program or procedures in place to govern the inspection or preventive maintenance activities described in their response, and there was no evidence that these manholes, raceways, and supports had ever been inspected prior to November 2009. The licensee has entered this violation into their corrective action program as Condition Report CR-2010-005784. The finding was characterized as a Severity Level IV violation in accordance with the NRC Enforcement Policy. Because this finding was determined to be of Severity Level IV safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000445; 05000446/2010006-05, "Failure to Provide Accurate Information in Response to Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"."

(2) Failure to Implement Design Features for Precluding or Minimizing Long-Term Accumulation of Water in Underground Conduits Containing Medium Voltage Safety Related Cables

Introduction. The team identified a Green noncited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, because the licensee installed conduit seals at a low point in the safety related cable manholes, thereby defeating the design requirement to avoid or minimize the accumulation of water in the duct banks. This configuration could result in long-term submergence of safety-related medium voltage cables and long-term degradation or failure of the cables.

Description. The licensee's June 20, 2007 response to Generic Letter 2007-01, Request 2, stated that "The underground ductwork at Comanche Peak is designed to slope toward manholes to avoid accumulation of water in the duct banks. The conduits embedded in concrete floors/walls inside the plant are sealed to prevent intrusion of water inside the conduits. These features eliminate or minimize cable exposure to environment of concern known to impact cable degradation rates identified in this generic letter." More recently, while evaluating flooding conditions observed in the manholes during licensee inspections in November, 2009, the licensee concluded in the corrective action plan for Evaluation EVAL-2009-005076, performed November 19, 2009 to support resolution of Condition Report CR-2009-005076-00, and Evaluation EVAL-2009-006801, performed January 7, 2010, that the design of the duct banks encourages drainage from the conduits to the cable vaults. On that basis, the licensee determined that no corrective action was required.

The team performed visual inspection of the conduits entering manholes MH1EB1 and MH1EB2. The team also reviewed the design and construction documents associated with the manholes and the results of the licensee's November 2009 inspections of the manholes. The team determined that contrary to the licensee's assertion water could enter the underground conduit and accumulate in the duct banks, because of a deficient conduit and conduit seal configuration. The team concluded that the conduits, which slope downward to the manholes from the service water intake structure and from the Unit 1 safeguards building, were sealed at the manhole entrance, which could result in prolonged submergence of cables within the underground conduits. The team determined this conduit and seal configuration was a design deficiency from original construction.

Analysis. The team determined that the failure to implement a design requirement to avoid or minimize accumulation of water in the underground duct banks was a performance deficiency. The finding is more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening in accordance with Manual Chapter 0609, Attachment 4, Phase1 – Initial Screening and Characterization of Findings, and determined that the finding was of very low safety significance (Green) because it was a design or qualification issue confirmed not to result in a loss of operability or functionality, it did not result in the loss of a system safety function, it did not represent the loss of a single train for greater than technical specification allowed outage time, it did not represent a loss of one or more non-technical specification risk significant equipment for greater than 24 hours, and it did not screen as potentially risk significant due to seismic, flooding, or severe weather. This finding did not have a crosscutting aspect because the most significant contributor did not reflect current licensee performance.

Enforcement. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, which states in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Contrary to the above, the licensee failed to establish measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions. Specifically, as of June 18, 2010, the underground duct banks connecting the safeguards buildings to the service water intake structure had installed conduit seals at a low point in the cable manholes, thereby defeating the design requirement to avoid or minimize the accumulation of water in the duct banks as specified in NRC Generic Letter 2007-01. This configuration could result in long-term submergence of safety-related medium voltage cables and long-term degradation or failure of the cables. This finding was entered into the licensee’s corrective action program as Condition Report CR-2010-005843. Because this finding was determined to be of very low safety significance and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation, consistent with the NRC Enforcement Policy: NCV 05000445; 05000446/2010006-06, “Failure to Implement Design Features for Precluding or Minimizing Long-Term Accumulation of Water in Underground Conduits Containing Medium Voltage Safety-Related Cables.”

.3.3 NRC Information Notice 2002-12, “Submerged Safety-Related Electrical Cables”

a. Inspection Scope

The team reviewed the licensee’s evaluation and disposition of NRC Information Notice 2002-12, “Submerged Safety-Related Electric Cables.” This activity was conducted under the same program and reviewed in conjunction with Generic Letter 2007-01, discussed in Section 1R21.3.2 above.

b. Findings

No findings were identified.

.3.4 NRC Information Notice 2007-34, “Operating Experience Regarding Electric Circuit Breakers”

a. Inspection Scope

The team reviewed the licensee’s documented evaluation and disposition of NRC Information Notice 2007-34, “Operating Experience Regarding Electric Circuit Breakers” under their operating experience program for each of the issues identified in this information notice. The team selectively reviewed condition reports identified by the licensee’s queries of their corrective action database for the 6900 Vac switchgear, to determine whether the licensee responses were effective in avoiding the problems discussed in the information notice. The team also interviewed the 6900 Vac system engineer to identify and discuss equipment repair history and refurbishment.

b. Findings

No findings were identified.

.3.5 NRC Information Notice 2008-02, “Findings Identified During Component Design Bases Inspections”

a. Inspection Scope

The team reviewed the licensee’s documented evaluation and disposition of NRC Information Notice 2008-02, “Findings Identified During Component Design Bases Inspections” under their operating experience program for each of the issues identified in this information notice.

The team selectively reviewed condition reports identified by the licensee’s queries of their corrective action database for the issues discussed in the information notice, to determine whether the licensee’s responses were effective in avoiding the problems discussed in the information notice.

b. Findings

No findings were identified.

.4 Results of Reviews for Operator Actions

a. Inspection Scope

The team reviewed four risk significance operator actions as follows:

- Isolate Auxiliary Feedwater Flow to a Steam Generator Fault Inside Containment within Ten Minutes as Required by the Finals Safety Analysis Report. The team observed a simulator job performance measure to isolate auxiliary feedwater flow to a faulted steam generator. The activity was satisfactorily performed within the required ten minutes as described in the final safety analysis report.

- Isolate a Ruptured Steam Generator within Thirteen Minutes of Event Initiation as Required by the Final Safety Analysis Report. The team observed a simulator job performance measure to isolate ruptured steam generator. The activity was satisfactorily performed within the required thirteen minutes as described in the final safety analysis report.

- Initiate a Cool Down within Five Minutes of Isolating a Ruptured Steam Generator as Required by the Final Safety Analysis Report. The team observed a simulator job performance measure to initiate a cool down within five minutes of isolating a ruptured steam generator. The activity was satisfactorily performed within the required five minutes as described in the final safety analysis report.

- During a Station Blackout, One Emergency Diesel Generator is Running but not on the Bus Due to a Low Voltage/Frequency Condition. The team observed a simulator job performance measure to address one diesel generator running but not connected to the bus due to a low voltage/frequency condition during a station blackout.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On June 18, 2010, the team leader presented the preliminary inspection results to Mr. B. Mays, Vice President, Nuclear Engineering and Support, and other members of the licensee’s staff.

On November 4, 2010, the team leader conducted a telephonic final exit meeting with Mr. B. Mays, Vice President, Nuclear Engineering and Support and other members of the licensee’s staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee-Identified Violations

None.

Attachments: Supplemental Information

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

I. Ahmad, Consulting Engineer
J. Back, Operating Experience Coordinator
D. Davis, Projects Manager
C. Feist, Consulting Engineer
D. Goodwin, Director Engineering Support
A. Hall, Operations Support Manager
J. Henderson, Engineering Smart Team Manager
J. Hicks, Regulatory Affairs
T. Hope, Nuclear Licensing Manager
H. Joiner, Operating Experience Supervisor
D. Kross, Plant Manager
F. Madden, Director Nuclear Oversight and Regulatory Affairs
S. Maier, Alliance Manager
A. Martin, Consulting Engineer
B. Mays, Vice President Nuclear Engineering and Support
G. Merka, Regulatory Affairs
J. Meyer, Technical Support Manager
D. Moore, Director Shaw Engineering and Technical Support
B. Patrick, Director Maintenance
W. Reppa, System Engineering Manager
S. Sewell, Director Nuclear Operations
R. Smith, Director Nuclear Training
S. Smith, Plant Manager
J. Smith, System Engineer
G. Techentine, System Engineer
T. Terryah, System Engineering Manager, Balance of Plant
T. Tigner, CAP Supervisor
L. Windham, Consulting Engineer
L. Yeager, Design Engineering Analysis Manager

NRC Personnel

J. Kramer, Senior Resident Inspector
B. Tindell, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000445/2010006-001	AV	Failure to Incorporate Relevant Operating Experience Information into Station Procedures Regarding the Condensate Storage Tank and Diaphragm (Section 1R21.2.2)

Opened and Closed

05000445/2010006-002	NCV	Inadequate Test Control of the Diesel Generator Air Starting System (Section 1R21.2.4)
05000445;05000446/2010006-003	NCV	Inadequate Analysis of Emergency Diesel Generator Frequency (Section 1R21.2.5)
05000445;05000446/2010006-004	NCV	Inadequate Evaluation of Hydrogen Generation for Safety-Related and NonSafety-Related Batteries (Section 1R21.2.10)
05000445;05000446/2010006-005	NCV	Failure to Provide Accurate Information in Response to Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients" (Section 1R21.3.2)
05000445;05000446/2010006-006	NCV	Failure to Implement Design Features for Precluding or Minimizing Long-Term Accumulation of Water in Underground Conduits Containing Medium Voltage Safety Related Cables (Section 1R21.3.2)

LIST OF DOCUMENTS REVIEWED

PROCEDURES

NUMBER	TITLE	REVISION/DATE
1-ALB-11B	Alarm Procedure - Battery Room Exhaust Fan(s) 7,8,9, and 10 Fail	5
1CP-PT-29-5	Diesel Generator Reliability Test	0
2CP-PT-14-03	Preoperational Test Procedure Loss of Instrument Air	1
2CP-PT-30-01A	Emergency Diesel Generator "Train A"	2
2CP-PT-30-01B	Emergency Diesel Generator "Train B"	1
ABN-305	Auxiliary Feedwater System Malfunction	6
ABN-601	Response to a 138/345 KV System Malfunction	10
COP-303A	Condensate	11
COP-609A	Chemistry Operating Procedures Manual for Diesel	9
COP-815A	Chemistry Operating Procedures Manual for Safety Chilled Water	3

PROCEDURES

NUMBER	TITLE	REVISION/DATE
ECA-0.1A	Loss of All AC Power Recovery Without SI Required	8
ECE-6.02-02	Engineering Review of Procurement Documents	11
EL-2	In-service Testing Plan for Pumps & Valves	10
EOP-0.0A/B	Reactor Trip or Safety Injection	8
EOP-1.0A/B	Loss of Reactor or Secondary Coolant	8
EOP-2.0A/B	Faulted Steam Generator Isolation	8
EOP-3.0A/B	Steam Generator Tube Rupture	8
EOS-1.2A	Post LOCA Cooldown and Depressurization	8
INC04052A-R1	Chanel Calibration - Safety Chilled Water System Condenser Pressure Control Channel 4552	1
INC-2050	Calibration of Temperature Devices	3
INC-2060	Calibration of Pressure Switches	5
IST-302	In-service Testing of Power-Operated Valves	4
MSE-C0-6305	6.9 kV 7.5 HK Circuit Breaker Enhanced Maintenance	2
MSE-C0-6311	6.9 kV Switchgear Auxiliary Switch Maintenance	1
MSE-G0-4003	Motor Insulation Resistance Testing	3
MSE-G0-4201	Megger Testing of Power Cables, Motors, and Generators	6
MSE-P0-7333	Centrifugal Water Chiller Maintenance	2
MSE-P1-5003	Unit 1 Class 1E Station Batteries 18 Month Inspection	0
MSE-P2-5003	Unit 1 Class 1E Station Batteries 18 Month Inspection	0
MSE-PO-4318	Service Water Pump Motor Inspection	9
MSE-S0-5000	Class 1E Station Batteries Weekly-Monthly-Quarterly Surveillance Tests	4
MSE-S0-5702	Class 1E Station Batteries Service Discharge Test	9
MSE-S0-5710	Battery Performance Discharge Test, Class 1E Station Batteries	6

PROCEDURES

NUMBER	TITLE	REVISION/DATE
MSE-S0-5715	Class 1E Station Batteries Modified Performance Discharge Test	2
MSE-S0-6301	6.9 kV Air Circuit Breaker Inspection and Cleaning	6
MSE-S1-0602A	Unit 1 Train A Electrical Undervoltage Relay Test, Response Time Test, and Bus Transfer Test	1
MSE-S1-0602B	Unit 1 Train B Electrical Undervoltage Relay Test, Response Time Test, and Bus Transfer Test	1
MSE-S1-0603A	Unit 1 Train A Undervoltage Relay Calibration and Response Time Surveillance Test	6
MSE-S2-0602A	Unit 2 Train A Electrical Undervoltage Relay Test, Response Time Test, and Bus Transfer Test	2
MSE-S2-0603A	Unit 2 Train A Undervoltage Relay Calibration and Response Time Surveillance Test	3
MSE-S2-0603B	Unit 2 Train B Undervoltage Relay Calibration and Response Time Surveillance Test	4
MSM-C0-8864	Crosby Safety Valve Maintenance	2
MSM-G0-0204	Safety Valve and Relief Valve Bench Testing	6
MSM-P0-3357	Emergency Diesel Engine Jacket Water Cooler Cleaning	1
NDE-4-02	ASME Section XI Visual Examination VT-2	6
ODA-407	Guideline on Use of Procedures	12
OP51-SYS.AFI	Auxiliary Feedwater System	March 31, 2008
OPT-206	Auxiliary Feedwater System	28
OPT-207A	Service Water System	13
OPT-207B	Service Water System	15
OPT-209B	Safety Chilled Water System	10
OPT-216A	Remote Shutdown Operability Test	11
OPT-216A	Remote Shutdown Operability Test	11
OPT-530A	AFW Check Valve Reverse Flow Test	2

PROCEDURES

NUMBER	TITLE	REVISION/DATE
OPT-601A	Train A Motor Driven Auxiliary Feedwater Accumulator Check Valve Test	4
OPT-601B	Train B Motor Driven Auxiliary Feedwater Accumulator Check Valve Test	4
OTP-214A	Diesel Generator Operability Test	19
PPT-S0-6004	Motor Operated Rising Stem Valve Risk-Informed IST Testing	4
STA-421	Initiating Condition Reports	16
STA-422	Processing Condition Reports	21
STA-426	Industry Operating Experience Program	4
STA-677	Preventive Maintenance program	10
STA-702	Surveillance Procedure	18
STA-734	Service Water System Monitoring Program	3
STA-744	Maintenance Effective Monitoring Program	4
TSP-509	Predictive Maintenance Thermographic Analysis Program	6
VL-06-002932	Fisher Type 657 Diaphragm Actuator	March, 2006
WCI-606	Work Control Process	14
WCI-677	Database Change Processing	3

CONDITION REPORTS

2007-000861	2009-008923	2009-004885	2005-004220	2010-005838
2007-003124	2010-000073	2004-003225	2006-002208	2010-005843
2007-003192	2010-000080	2006-004133	2008-003002	2010-005943
2008-000975	2010-000084	2010-001731	2009-005076	2010-006022
2008-001146	2010-000277	2009-002395	2009-006702	2010-006028
2008-001224	2010-000624	2008-002730	2009-006754	2010-006031
2008-001308	2010-001027	2006-002647	2009-006801	2000-002081
2008-003035	2010-001854	2010-002198	2010-000407	2009-006268
2008-003421	2010-002349	2010-001255	2010-000453	2007-002254
2009-000696	2010-002469	2010-005508	2008-003180	2009-002817
2009-002453	2010-003000	2010-005923	2010-004213	2009-008286
2009-003767	2010-003318	2010-005822	2010-004263	2010-004431
2009-003821	2010-003419	2008-000934	2010-004530	2010-001662

CONDITION REPORTS

2009-003917	2010-004438	2009-000867	2010-004698	2010-001287
2009-005347	2010-005840	2010-003305	2010-005584	2010-000789
2009-005677	2010-005884	2007-000728	2010-005585	2008-001141
2009-008129	2010-005886	2010-005563	2010-005667	2008-003299
2009-008769	2008-000676	2010-005581	2010-005784	2008-002171
2009-008851	2009-004528	2010-005962	2010-005790	2009-008566
2007-002411	2007-003077	2007-000875	2009-002752	2009-003479
2009-006137	2008-001103	2010-005086	2008-000948	2010-000704
2009-006299	2010-000976	2010-000084	2009-002763	2010-000074
2010-004269	2009-008603	2009-006199	2010-000145	2080-001125
2010-004148	2010-003517	2009-004952	2010-001659	2000-001957
2005-002219	2010-000638	2008-002171	2010-003853	2008-001754
2007-002290	2009-008489	2010-003318	2010-001736	2010-005553
2010-006071	2009-004952	2010-003255	2010-004009	2010-004059
2008-003553	2007-001609	2007-000803	2009-002232	2009-008489
2010-006071	2008-003180	2009-003891	2007-000550	2007-000968
2003-003275	2005-004223	2010-006028	2009-003817	2004-001249
2010-005878	2005-004220	2010-006031	2010-005942	2004-000307
2010-005983	2005-004187	2010-006047	2004-003683	2010-005941
2008-001911	2007-000967	2003-002504	2004-001166	

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
16345-ME-071	Operation of Emergency Diesel Generators without Full Flow to Jacket Water Heat Exchanger	5
1-EB-302-4	Auxiliary Feedwater Pump Rooms	5
1-EB-302-5	Residual Heat Removal Pump Rooms- Unit 1	4
1-EB-302-7	Containment Spray Pump Rooms	5
1-EB-303-2	Component Cooling Water Pump Rooms Unit 1	6
1-EB-305-P3	Pressure Loss and Fan Evaluation – Unit 1	0
1-EB-311-1	ESF Local Cooler Areas Summary	6
1-EB-311-11	Adequacy of Safety Chilled Water system and Fan Coolers	2
1-EB-311-3	ESF Local Cooler Areas Summary – Space Heater Gains and Maximum Temperatures	6
1-EB-311-4	Cooling Load on Safety Chillers – Normal Operation	3
1-EB-311-5	Water Volume in Safety Chilled Water System	0

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
1-EB-311-8	Safety Chilled Water Pump Evaluation – Unit 1	2
2-NU-0094	Hydrogen Distribution in Unit 2 Battery Rooms X-116 and X-123	0
2-NU-0107	Effect of Unit 22 Operation on Hydrogen Levels in Unit 1 and Unit 2 Battery Rooms	0
3-A-8-002	Hydrogen Distribution in Battery Rooms	January 18, 1989
EE-1E-1EB2	480 Volt AC Switchgear CP1-EPSWEB-02 (1EB2) Bus Based Calculation	1
EE-1E-1EB3-3	480 Volt AC Motor Control Center CP1-EPMCEB-07 (1EB3-3) Bus Based Calculation	1
EE-1E-1EB4-3	480 Volt AC Motor Control Center CP1-EPMCEB-08 (1EB4-3) Bus Based Calculation	1
EE-1E-1ED1	125 Volt DC Switchboard CP1-EPSWED-01 (1ED1) Bus Based Calculation	2
EE-1E-1ED2	125 Volt DC Switchboard CP1-EPSWED-02 (1ED2) Bus Based Calculation	2
EE-1E-1ED3	125 Volt DC Switchboard CP1-EPSWED-03 (1ED3) Bus Based Calculation	1
EE-1E-1ED4	125 Volt DC Switchboard CP1-EPSWED-04 (1ED4) Bus Based Calculation	1
EE-1E-2ED2	125 Volt DC Switchboard CP2-EPSWED-02 (2ED2) Bus Based Calculation	4
EE-1E-BT1ED1	125 Volt DC Battery and Charger Sizing Calculation, CP1-EPBTED-01, CP1-EPBCED-01, CP1-EPBCED-03	1
EE-1E-BT1ED2	125 Volt DC Battery and Charger Sizing Calculation, CP1-EPBTED-02, CP1-EPBCED-02, CP1-EPBCED-04	1
EE-1E-BT1ED3	125 Volt DC Battery and Charger Sizing Calculation, CP1-EPBTED-03, CP1-EPBCED-05, CP1-EPBCED-07	0

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
EE-1E-BT1ED4	125 Volt DC Battery and Charger Sizing Calculation, CP1-EPBTED-04, CP1-EPBCED-06, CP1-EPBCED-08	0
EE-B-018	Verification of 6.9 kV Switchgear Bus Sizing	1
EE-BAT/CHG-METHODOLOGY	125 Volt DC Battery and Battery Charger Sizing Methodology	1
EE-CA-0008-0265	Protective Relay Settings for 6.9 kV Safeguards Buses	4
EE-CA-0008-0871	Protective Relay Settings for Safeguard Buses Overvoltage / Undervoltage Relays and Associated Time Delay Relays	12
EE-MCC-METHODOLOGY	480 Volt AC MCC, Distribution Panel, and Switchgear Methodology	11
EE-VP-U1-1E	Unit 1 Class 1E Voltage Profile	1
EM(B)-069	Thru Thickness Liner Stresses	0
IC(B)-002	Air Accumulator Sizing	March 1, 1991
IC(S)-010	Alarm Setpoint for Service Water Temperature Out of Component Cooling Water Heat Exchanger	0
IC(S)-011	Station Service Water Component Cooling Water Heat Exchanger Outlet Flow Loop Accuracy calculation	4
ME(B)-053	Auxiliary Feedwater System Performance	3
ME(B)-088	Station Service Water System Steady State Hydraulic Calculation	5
ME(B)-152	Determining Pressure and Flow Rate Decrease of SWS-AFW Line Interface Caused By High and Low Leakoff Connections	1
ME(B)-181	Component Cooling Water Heat Loads and Temperatures for Various Heat Loads	7
ME(B)-240	Condensate Storage Tank Technical Specifications Limit	4
ME(B)-391	Flow to Diesel Generators	5

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
ME-0098	Nitrogen Metering Orifices for the Condensate Storage Tank and the Reactor Makeup Water Storage Tank	0
MEB-054	Auxiliary Feedwater Pumps Net Positive Suction Head	3
MEB-172	Condensate Storage Tank – Determination of Critical Depth for Vortex flow and Usable tank Volume	2
MEB-391	Minimum Allowed Service Water Flow to the Diesel Generators	5
ME-CA-0000-3264	Safe Shutdown Impoundment Hydrothermal Analysis	3
ME-CA-0000-3339	Flow of Service Water into Auxiliary Feedwater System, with Backflow to Idle Service Water Train	0
ME-CA-0000-3342	Air Accumulator Check Valve Leakage	2
ME-CA-0000-4070	Equipment Qualification Total Integrated Dose to ABB Relays in Switchgear Located in Rooms 1-083, 2-083, 1-103, and 2-103	0
ME-CA-0000-4089	Component Cooling Water Heat Exchanger Fouling Factor Analysis	2
ME-CA-0000-5295	Comanche Peak Unit 1 Minimum Condensate Storage Tank Volume for Replacement Steam Generators/Uprate	1
ME-CA-0000-5335	Service Water and Auxiliary Feedwater Design Temperatures for Stress Analysis	1
ME-CA-0011-3075	Diesel Jacket Water Heat Exchanger Fouling Factor Analysis	2
ME-CA-0206-3147	Auxiliary Feedwater Flow Distribution Evaluation on Turbine Driven Overspeed	0
ME-CA-0206-5085	Evaluation of the Effects of Increasing the Motor Driven Auxiliary Feedwater Minimum Flow to 200gpm	0
ME-CA-0215-4054	Diesel generator Fuel oil Storage Requirements and Tank Level Setpoints	2
ME-CA-0303-4060	The Effects on the Safety Chiller Caused by the Addition of Two Rows of Coils to the Spent Fuel Pool Pump Room Cooler	0

CALCULATIONS

NUMBER	TITLE	REVISION/DATE
ME-CA-0305-3027	Minimum Temperature Transient in Battery Rooms Following Station Blackout	0
ME-VS-0000-3342	Air Accumulator Check Valve Leakage-Decay Rate, Pressure and Time	2
SI-CA-0000-4005	System Interaction Evaluation for UPS Fan Coil Units	1
SI-CA-0803-3381	Uninterruptable Power Supply Room Flooding Analysis	1
SMI-103 C-2	Stress Analysis of Diaphragm Connection Ring for Condensate Storage tank	April 20, 1978
X-EB-HV-15	Hydrogen Level in Battery Rooms, Units 1 and 2	2
X-ES-303-01-04	Spent Fuel Pool Heat Exchanger & Pump Rooms	4

DESIGN BASIS DOCUMENTS

NUMBER	TITLE	REVISION
DBD-EE-004	Accident Monitoring Instrumentation	38
DBD-EE-030	Environmental Qualification of Safety Related Mechanical Equipment	4
DBD-EE-033	Detailed Control Room Design	18
DBD-EE-040	6.9 kV Electrical Power System	14
DBD-EE-044	DC Power Systems	24
DBD-EE-051	Protection Philosophy	35
DBD-EE-057	Separation Criteria	29
DBD-ME-009	Tornado Venting Analysis	11
DBD-ME-011	Diesel Generator Sets	30
DBD-ME-031	Environmental Qualification for Safety-Related Electrical Equipment	5
DBD-ME-037	Balance of Plant Safety-Related Setpoints	8
DBD-ME-206	Auxiliary Feedwater System	24
DBD-ME-233	Station Service water System	20

DESIGN BASIS DOCUMENTS

NUMBER	TITLE	REVISION
DBD-ME-239	Component Cooling water System	36
DBD-ME-250	Reactor Coolant System	41
DBD-ME-260	Residual Heat Removal System	23
DBD-ME-261	Safety Injection System	25
DBD-ME-305	Uncontrolled Access Area Ventilation System	11

DRAWINGS

NUMBER	TITLE	REVISION/DATE
2323-SI-0316	Condensate Storage Tank	3
23323-MI-0751	Ventilation Auxiliary Building Plan EL. 790’-6” & 792’-0”	8
BRP-AF-1-YD-002	Auxiliary Feedwater	CP-2
BRP-AF-1-YD-003	Auxiliary Feedwater	CP-2
BRP-RH-1-RB-001	Residual Heat Removal	CP-3
BRP-RH-1-RB-002	Residual Heat Removal	CP-3
BRP-RH-2-RB-001	Residual Heat Removal	CP-7
BRP-RH-2-RB-002	Residual Heat Removal	CP-8
E1-0001	Plant One Line Diagram, Units 1 and 2	CP-30
E1-0003 Sh. A	Plant One Line Diagram, Unit 1 and Common Distribution Panels	CP-18
E1-0004	6.9 kV Auxiliaries One Line Diagram, Safeguard Buses	CP-37
E1-0004 Sh. A	6.9 kV Auxiliaries One Line Diagram, Safeguard Buses	CP-27
E1-0020	125 Volt DC One Line Diagram	CP-20
E1-0020 Sh. A	125 Volt DC One Line Diagram	CP-14
E1-0022 Sh. 3	Under / Overvoltage Protection Logic Diagram for Class 1E 6.9 kV / 480 V Buses	CP-1
E1-0022 Sh. 4	Under / Overvoltage Protection Logic Diagram for Class 1E 6.9 kV / 480 V Buses	CP-1

DRAWINGS

NUMBER	TITLE	REVISION/DATE
E1-0022 Sh. 5	Solid State Safeguard Sequencer Logic Diagram	CP-2
E1-0027 Sh. 1A	6.9 kV Three Line Diagram, Safeguard Buses	CP-9
E1-0030 Sh. 21	6.9 kV Switchgear Safeguard Bus 1EA1 Undervoltage Auxiliary Relays	CP-11
E1-0031 Sh. 41	6.9 kV Switchgear Bus 1EA1 Station Service Water PP 11 Tag CP1-SWAPSW-01 BKR 1APSW1 Schematic Diagram	CP-9
E1-0031 Sh. 41A	6.9 kV Switchgear Bus 1EA1 Station Service Water PP 11 Tag CP1-SWAPSW-01 BKR 1APSW1 Schematic Diagram	CP-4
E1-0031 Sh. 42	6.9 kV Switchgear Bus 1EA1 Station Service Water PP 11 Tag CP1-SSWAPSW-01 BKR 1APSW SW Development and Connection Diagram	CP-6
E1-0031 Sh. 43	6.9 kV Switchgear Bus 1EA2 Station Service Water PP 12 Tag CP1-SWAPSW-02 BKR 1APSW2 Schematic Diagram	CP-10
E1-0031 Sh. 43A	6.9 kV Switchgear Bus 1EA2 Station Service Water PP 12 Tag CP1-SWAPSW-02 BKR 1APSW2 Schematic Diagram	CP-3
E1-0031 Sh. 44	6.9 kV Switchgear Bus 1EA2 Station Service Water PP 12 Tag CP1-SWAPSW-02 BKR 1APSW2 SW Development and Connection Diagram	CP-7
E1-0043	Service Water System Typical Internal Wiring Diagrams and Developments	CP-3
E1-0043 Sh. 5	Motor Operated Valve 1-HV-4286 Station Service Water PP 01 Discharge to Strainer Isolation Valve	CP-6
E1-0043 Sh. 6	Motor Operated Valve 1-HV-4287 Station Service Water. PP 12 Discharge	CP-4
E1-0071 Sh. 51	1-SSII-1 125 Volt DC Status Indicating Lights Schematic Diagram	CP-3
E1-2400 Sh. 136	Protective Device Settings, 6.9 kV Safeguards Buses	CP-1
E1-2400 Sh. 137	Protective Device Settings, 6.9 kV Safeguards Buses	CP-3
E1-2400 Sh. 151	Protective Device Settings, 6.9 kV Safeguards Buses	CP-2

DRAWINGS

NUMBER	TITLE	REVISION/DATE
Gibbs & Hill Dwg. 2323-E1-1007-01	Yard Electrical Duct bank Manhole & Handhole Plan Sheet 2	8
Gibbs & Hill Dwg. 2323-E1-1008	Yard Electrical Duct bank & Manhole Sections & Details Sheet 1	11
Gibbs & Hill Dwg. 2323-E1-1009	Yard Electrical Duct bank, Manhole, and Handhole Sections & Details Sheet 2	17
M1-0200	Mechanical Symbols & Notes	CP-26
M1-0206, Sh. 1	Auxiliary Feedwater System Pump Trains	CP-15
M1-0206, Sh. 2	Auxiliary Feedwater System Yard Layout	CP-19
M1-0206, Sh. 6	Instrumentation and Control Diagram Auxiliary Feedwater System	CP-8
M1-0215-D	Flow Diagram Starting Air Piping CP1-MEDGEE-01	CP-24
M1-0215-E	Flow Diagram Starting Air Piping CP1-MEDGEE-02	CP-26
M1-0229	Flow Diagram Component Cooling Water System	CP-22
M1-0229, Sh. A	Flow Diagram Component Cooling Water System	CP-21
M1-0229, Sh. B	Flow Diagram Component Cooling Water System	CP-25
M1-0233	Flow Diagram Station Service Water	CP-39
M1-0233, Sh. A	Flow Diagram Station Service Water	CP-18
M1-0234	Flow Diagram Station Service Water	CP-24
M1-0260	Flow Diagram Residual Heat Removal System	CP-35
M1-0305	Flow Diagram Ventilation Uncontrolled and Battery Rooms	CP-19
M1-0305 Sh. A	Flow Diagram Ventilation Uncontrolled and Battery Rooms	CP-10
M1-0313	Flow Diagram-Ventilation Control Building Uninterruptable Power Supply Area Air Conditioning Systems	CP-21
M1-2229, Sh. 4	Instrumentation and Control Diagram, Component Cooling Water System	CP-15

DRAWINGS

NUMBER	TITLE	REVISION/DATE
M1-2229, Sh. 4C	Instrumentation and Control Diagram, Component Cooling Water System	CP-2
M1-2233, Sh. 1	Instrumentation and Control Diagram, Station Service Water System	CP-9
M1-2233, Sh. 1A	Instrumentation and Control Diagram, Station Service Water System	CP-6
M1-2233, Sh. 3	Instrumentation and Control Diagram, Station Service Water System	CP-3
M1-2233, Sh. 5	Instrumentation and Control Diagram, Station Service Water System	CP-4
M1-2233, Sh. 8	Instrumentation and Control Diagram, Station Service Water System	CP-6
M1-2233, Sh. 9	Instrumentation and Control Diagram, Station Service Water System	CP-3
M2-0215-D	Flow Diagram Starting Air Piping CP2-MEDGEE-01	CP-16
M2-0215-E	Flow Diagram Starting Air Piping CP2-MEDGEE-02	CP-18
MI-2200 Sh. 10D	Instrumentation & Control Diagram, Safety System Inoperable Indicator Logic	CP-5
Okonite Dwg. CS-4510	1/c Okoguard Shielded Okolon 8 kV Cable	October 2, 1986
Promatec Dwg.A-572	Silicone Foam Internal Conduit Seal, Comanche Peak Steam Electric Station	2
TUS Drawing 182C79225 Sh. 11	Elementary Diagram Nuclear Safety Related Class 1E, 1ED1, 1ED2	CP-1
TUS Drawing 182C79225 Sh. 11A	Elementary Diagram Nuclear Safety Related Class 1E, 1ED1, 1ED2	4
VL-02-004105	Corrosion Control service Diaphragm	July 23, 2002

ENGINEERING REPORTS

NUMBER	TITLE	REVISION/DATE
ER-ME-102	Resolution of NRC Generic Letter 95-07, "Pressure Locking and Thermal Binging of Safety Related Power Operated Gate Valves"	1
ER-ME-109	Evaluation of safety Related Pump Degradation Issues	1
RXE-TA-CPX/O-029	RCS Wide Range Pressure Uncertainty – Rosemount Part 21 Evaluation	June 24, 1993
TDI-EDG-001-A	Basis for Modification to Inspection Requirements for Transamerica Delaval, Inc., Emergency Diesel Generator	March 17, 1994
TE-98-641	Technical Evaluation	November 13, 1998
WCAP-11736-A	Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners Group	0
WCAP-16871-P	Comanche Peak Nuclear Power Plant Stretch Power Uprate Engineering Report	0
WPT-11968	Comanche Peak Unit One Loop Uncertainty Analysis	1

MAINTENANCE WORK ORDERS

394247	394778	395614	402050	402864
402924	404449	407606	412506	3440477
3453227	3485939	3486199	3488272	3489447
3507694	3511936	3513380	3555824	3560573
3609385	3611518	3612656	3612674	3612688
3619638	3639449	3690197	3713847	3719801
3719842	3724599	3727026	3731665	3735061
3735638	3739607	3744503	3749750	3756843
3764685	3773217	3774647	3779591	3782126
3782128	3788137	3793893	3796302	3799143
3802231	3809896	3823790	3835500	3835687
3839979	3839995	3843734	3843761	3843782
3851441	3857453	3860295	3863761	3865850
3867225	3867732	3868946	3872207	3874472
3880428	3884330	3889558	3889580	3889583
3892664	3897575	3906451	3922392	3932833
3935380	3940102	3952909	3960511	3960513
3962178	5053850	1-05-164257-00	1-92-28105-00	3-03-319213-01
3-04-308883-01	3-04-319175-01	3-04-319177-01	3-05-345078-01	3-06-319173-01
3-98-338537-01	3-98-338601-01	4-05-161517-00	4-96-104945-00	5-02-505264-AA

5-04-502507-AA	5-04-504417-AA	5-04-504419-AA	5-04-505262-AA	5-05-501062-AA
5-05-503515-AA	5-05-504410-AA	5-05-504534-AA	5-05-505261-AA	5-06-505034-AA
5-07-502-507	5-07-506001-AB	5-07-506003-AA	5-07-506004-AA	

VENDOR MANUALS

NUMBER	TITLE	REVISION/DATE
2323-ES-13A Sh. 6-11	Gibbs & Hill Cable Specification Technical Data Sheets	November 7, 1975
CP-0001-009	Valve Vendor Manual	27
CP-0034-001A	Delaval Instruction Manual Volume I	59
CP-0034-001B	Book 1, Parts Manual, Volume II “As Built” Configuration	31
CP-0034-001B	Book 2, Parts Manual, Volume II Aftermarket Supplement	1
CP-0034-001C	Book 1, Associated Publications Volume III	62
CP-0034-001C	Book 2, Associated Publications Volume III	62
CP-0034-001C	Book 3, Associated Publications Volume III	62
CP-0034-001E	TDI Diesel Generator Owners Group Sub Vendor Maintenance	July 15, 1988
CP-0034-001F	Delta Switchboard Company	6
CP-0034-001H	Book 1, NEI Peebles Electric Company (PORTEC)	14
CP-0034-001H	Book 2, NEI Peebles Electric Company (PORTEC)	14
S02994066S6-001	Iris Power Engineering Inc. Installation Guide-Bus Coupler Epoxy and RFCT Type Sensors	May 18, 2005

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION/DATE
2323-MS-34	Comanche Peak Steam Electric Station Units 1 and 2 Specification Diesel Generator Sets	5
CPSES-200300946	Letter: Westinghouse to TXU, 1045DEP Diaphragm Material for Reactor Heat-up Tank and Reactor Makeup Water Storage Tanks	March 26, 2003

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION/DATE
CPSES-200701065	Comanche Peak Steam Electric Station, Response to NRC Generic Letter 2007-01, “Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients”	June 20, 2007
EB-T-3507	Minutes of Comanche Peak Steam Electric Station/Stone & Webster Engineering Corporation/ Ebasco Services Incorporated Heating Ventilation and Air Conditioning Systems Interface Meeting of September 10, 1987	September 16, 1987
FX90-1245	Diesel Generator Roll Failure	March 16, 1990
QCG-10291	Cooper Energy Services Part 21 Letter	July 9, 1996
S0150391	Purchasing document for Condensate Storage Tank Diaphragm	February 1, 1995
TBX-283/2	NSSS Process Control Setpoints	March 11, 1980
VL 05-000294	Letter: Corrosion Control Service Inc. to Texas Utilities, Condensate Storage Tank and Reactor Makeup Water Storage Tank (unexpected material failure of flotation media at Callaway)	October 28, 2003
VL-06-001694	Cooper-Enterprise Clearinghouse R4/RV4 Preventive Maintenance Program For Nuclear Standby Applications	0
VP-2001-0025	Verification Plan Bus Coupler Package, 6.9 kV (Receipt Inspection) 1E Dedication Documentation For Partial Discharge Monitor Equipment	March 18, 2002
VP-2001-0026	Verification Plan Bus Coupler Package, 6.9 kV (Source Inspection) 1E Dedication Documentation For Partial Discharge Monitor Equipment Factory Tests	March 21, 2001
WPT-15517	Letter: Westinghouse to TU Electric Company, Condensate Storage Tank Diaphragm	February 17, 1995
WPT-17445	Letter; Westinghouse to Luminant, Impact of Auxiliary pump Heat on Westinghouse and Combustion Engineering Analyses/Methodologies	April 13, 2010

Comanche Peak Unit 1
CST Bladder
SDP Phase 3

Performance Deficiency. The licensee failed to maintain the condensate storage tank (CST) bladder in a configuration that would prevent it from tearing, sinking, and causing a loss of suction to the auxiliary feedwater pumps.

Assumptions.

1. The nitrogen cover gas was secured for a period of 90 days. At the end of this period, the CST bladder was observed to be adhering to the walls of the tank. It is assumed that this adherence was a direct and immediate consequence of securing the gas supply. It is also possible that bladder was adhered to the wall prior to the 90-day period because the nitrogen was being injected under the bladder and not on the edges, where doing so would have prevented the adherence. However, for this evaluation, the exposure period is assumed to be 90 days.

2. There are two potential damage states resulting from the CST bladder configuration. In both cases, the bladder rips apart as the water level is lowering during an Auxiliary Feedwater (AFW) actuation, with the heavier-than-water fabric sinking into the tank to the region of the two AFW suction pipes (one associated with the turbine-driven AFW pump and other feeding both of the motor-driven AFW pumps). The suction pipes are 10 inches in diameter and two feet apart, protruding from the side of the tank and turning 90 degrees downward.

   In Damage State #1, the bladder fabric falls to the bottom of the tank, blocks the suctions to both pipes, and effectively isolates the CST from the AFW pumps. In this case, operators can re-direct the AFW suction to the service water system as long as they secure the AFW pumps before they are damaged from overheating from loss of suction.

   In Damage State #2, all of the AFW pumps either are lost because of overheating from a loss of suction or because fabric pieces from the bladder are drawn into the suction pipes and migrate to the pumps causing irrecoverable damage. This damage state precludes a recovery from the service water system and eliminates the AFW function for the duration of the recovery.

   Based on qualitative judgment of the observed condition and reference to similar events at other nuclear plants, the following assumptions are made:

   - There is a 1 percent probability that the CST bladder fabric will rip, sink, and be drawn to the suctions of the AFW pumps, in response to an AFW actuation, and that this sequence of events will occur fast enough to result in a risk impact.

   - In the event of a loss of suction, all three AFW pumps (2 motor-driven, one turbine-driven) could become damaged if they are not secured within a few minutes before overheating (there is no low suction trip). However, following an AFW actuation, operators are instructed by procedure to secure the turbine-driven AFW pump as long as both motor-driven AFW pumps are running. The analyst calculated that this situation (that is, a running turbine-driven pump would be secured early in the accident) would happen approximately 95 percent of the time, reflecting a 5 percent cumulative probability that one of the motor-driven pumps would be either out of service or would fail to start or run. Additionally, there is about a 5 percent chance that the turbine-driven pump will be out of service at the time of the accident or will otherwise fail in some fashion. Given that operators secure the turbine-driven pump before suction is lost, there is an additional possibility that they will start it as soon as the two-motor driven pumps overheat from a loss of suction. In this case, the turbine-driven pump could overheat or be lost by ingestion of bladder fabric pieces. The analyst assumed that this would add an additional 15 percent probability that the turbine-driven pump would be lost in spite of the procedural guidelines. Given the above, it assumed that a loss of suction from the CST will result in a 25 percent chance that all three AFW pumps will become unavailable for additional mitigation. In this situation, only feed and bleed operations could ultimately circumvent core damage. This is defined as Damage State #2 above, and is assigned a probability of occurrence as 0.01(0.25) = 2.5E-3, reflecting that there is an assumed 1 percent probability that a loss of suction will occur and a 25 percent chance that this condition will ultimately result in a loss of all of the AFW pumps.

- Conversely, it is assumed that there is a 75 percent probability that at least one AFW pump (presumably the turbine-driven pump) will survive the loss of suction event and remain available for mitigation by taking suction from the service water system (after operators align the alternate flowpath). This is defined as Damage State #1 and is assigned a probability of occurrence of 0.01(0.75) = 7.5E-3.

- Although it is understood that there is a possibility that bladder fabric could be sucked into the pump casings and cause direct damage, this scenario was considered to be unlikely based on a qualitative analysis of the bladder material properties, but this possibility can still be considered as contributing to the overall 25 percent assumption that all AFW pumps will be lost in the scenario.

3. The Comanche Peak SPAR model, Revision 3.51 was used to determine the risk of the finding. Several changes were made to the model, as follows:

- The original model assigned a nonrecovery probability for operators to recover feedwater following a plant trip of 0.25. This was considered to be too high, and was changed to a value of 0.04 based on consultation with the Idaho National Laboratory (INL).

- The original model assigned a nonrecovery probability for switching AFW suction to the service water system of 1.0E-2. Based on a review of the complexity of the evolution and using the SPAR-H method, the nonrecovery probability was increased to 7.7E-2. For diagnosis, high stress was assumed, and all other PSFs were considered nominal. For action, time was assumed to be approximately equal to the time required, stress was high, and experience/training (the procedure had never been performed) was considered to be low, with all other PSFs nominal.

- The original model assumed that loss of a single emergency dc bus would preclude recovery of AFW suction from service water. This was incorrect and was changed to show that both buses must be lost for this to occur.

- The SPAR model assumes that two PORVs are needed for feed and bleed operations. Some sequence cutsets include situations where an offsite power event occurs, an EDG is lost, and, after a 4-hour discharge, the battery and dc bus are de-energized. The analyst was uncertain if a two-PORV requirement would still exist at 4 hours post-shutdown, but did not change this assumption in the model. The licensee stated that only one PORV would be needed if both CCPs were in operation, but the loss of power to one emergency ac bus would cause the loss of one CCP. Therefore, the SPAR assumption appears valid at least for the short-term sequences.

4. The SPAR model was run to determine the delta-CDF of Damage States #1 and #2. To model Damage State #1, the basic event AFW-TNK-FC-CST, CST or Pump Suction Path is Unavailable, was set to 1.0. The result was a delta-CDF of 1.332E-3/yr. The following sequences were dominant:

Sequence	Delta-CDF	Percent of Total	Cumulative
LOMFW 22	2.653E-4	19.9	19.9
LOOPGR 17	2.334E-4	17.5	37.4
LOCHS 13	2.122E-4	15.9	53.3
LOOPSC 17	1.305E-4	9.8	63.1
LODC1ED2	9.248E-5	6.9	70.0
TRANS 22	8.598E-5	6.5	76.5
LOSWS 16	7.910E-5	6.0	82.5

To model Damage State #2, basic event AFW-MDP-CF-RUN, Common Cause Failure of Motor-Driven AFW Pumps to Run, was set to 1.0, and AFW-TDP-FR-P01, Turbine Driven Feed Pump P01 Fails to Run, was set to 1.0. The combination of these events removes the AFW mitigation function and precludes a recovery by realigning the suction to the service water system. The only remaining mitigation is the application of a feed and bleed line-up. The result was a delta-CDF of 1.663E-2/yr. The following sequences were dominant:

Sequence	Delta-CDF	Percent of Total	Cumulative
LOMFW 22	3.456E-3	21.1	21.1
LOCHS 13	2.765E-3	16.9	38.0
LOOPGR 17	2.736E-3	16.8	54.8
LOOPSC 17	1.530E-3	9.4	64.2
LODC1ED2 22	1.200E-3	7.3	71.5

The combined delta-CDF for internal initiators is calculated as follows:

Damage State	Probability of Failure	Delta-CDF	Adjusted Delta-CDF	Exposure Period Adjustment (90/365)
1 – loss of CST	0.0075	1.332E-3	9.99E-6	2.46E-6
2 – loss of AFW	0.0025	1.663E-2	4.16E-5	1.03E-5
Total Internal Delta- CDF				**1.28E-5/yr.**

5. External Initiators.

Fire. Fires that result in transients or other events are accounted for in the initiating event frequencies incorporated in the SPAR model. Therefore, only fires that target sequence-relevant mitigating equipment and result in AFW actuations would contribute risk additional to the internal result. The analyst concluded that the frequency of fires that would meet this qualification is very low compared to the frequency of events in the internal SPAR model, particularly those that contribute the most risk to this finding. Therefore, the risk contribution from fires was screened qualitatively.

Seismic. The analyst assumed that a seismic event that caused a loss of offsite power would preclude recovery of offsite power for at least 24 hours. Using information in the RASP manual, a calculation was performed to determine the frequency of LOOPS caused by seismic events. This frequency was reduced by the LOOPs that would cause a loss of the CST, since this would result in a baseline result. The SPAR model was used to determine the difference in CCDP between a nonrecoverable LOOP and a nonrecoverable LOOP that also includes a loss of all AFW. The result of the calculation was that seismic events would contribute a delta-CDF of 1.36E-6/yr. if AFW is always lost upon an AFW actuation. This result was modified to account for a 1 percent chance of losing AFW and a 90-day exposure period, resulting in an inconsequential seismic contribution of 3.4E-9/yr.

Other. The risk contribution from high winds, flooding, and other external events was screened qualitatively.

Large Early Release. In accordance with IMC 0609, Appendix H, because there are no significant core damage sequences that involve steam generator tube ruptures or inter-system LOCAs, the contribution of large early release is negligible.

APPENDIX M - TABLE 4.1
Qualitative Decision-Making Attributes for NRC Management Review

1. The SDP is the preferred path for determining the significance of findings in the Reactor Oversight Process.

2. Inspection Manual Chapter 0609 Appendix M is provided for use when the existing SDP guidance is not adequate to provide a reasonable estimate of the significance.

3. Inspection Manual Chapter 0609 Appendix M could be used for this case. Appendix M utilizes a qualitative significance determination process focused on the below table where six of eight attributes would have some level of applicability:

Decision Attribute	**Applicable to Decision?**	**Basis for Input to Decision – Provide qualitative and/or quantitative information for management review and decision-making.**
Finding can be bounded using qualitative and/or quantitative information.	Yes	Risk evaluation tools are available to address the significance of a loss of suction to the auxiliary feedwater pumps. However, the probability of losing suction as a consequence of the condensate storage tank bladder fabric adhering to the tank walls cannot be estimated with any confidence or reference to industry data. A significance determination Phase 3 analysis was performed to determine bounding risk significance. The results of this analysis are as follows: Assuming a 100 percent probability that suction to the auxiliary feedwater pumps would be lost, with a 25 percent probability that this event would result in pump damage from either ingestion of fabric material into the pump casing or from overheating, results in a delta-CDF of 1.3E-3/yr. using the Comanche Peak SPAR model, revised to correct errors and modify basic events, and assuming a 90-day exposure period. A 10 percent probability results in a delta-CDF or 1.3E-4/yr. A 1 percent probability results in a delta-CDF of 1.3E-5/yr. A 0.1 percent probability results in a delta-CDF of 1.3E-6/yr. The analysts and inspectors considered that the best estimate for failure of the bladder and loss of suction to the pumps is 1 percent. However, the uncertainty of this estimate is very high and could only be refined with

Decision Attribute	Applicable to Decision?	Basis for Input to Decision – Provide qualitative and/or quantitative information for management review and decision-making.
		extensive analysis or testing. Additionally, there is a large amount of uncertainty concerning the probability that the pumps would be damaged from a loss of suction event, thereby preventing a recovery by lining up suction to the service water system. Given this situation, it is appropriate to reduce the risk estimate to account for the large amount of uncertainty. In light of this consideration and the attached risk analysis, the agency considers that the violation is best characterized as having low to moderate risk significance (White).
Defense-in-Depth affected.	Yes	Loss of secondary cooling by means of the steam generators results in defeating the preferred means of cooling the reactor core post-shutdown. The only remaining mitigation system that can be used to cool down the reactor coolant system enough to initiate shutdown cooling is feed and bleed cooling.
Performance Deficiency effect on the Safety Margin maintained.	N/A	
The extent the performance deficiency affects other equipment.	N/A	
Degree of degradation of failed or unavailable component(s)	Yes	Although the bladder fabric was not observed to be degraded based on a limited inspection of the visible surfaces, the fact that the fabric was adhered to the condensate storage tank wall is considered significant degradation in light of the expected forces that could be applied during an actuation of auxiliary feedwater system. With the wall surface and bladder fabric both having hydrophilic properties, there is a reasonable possibility that the fabric could rip and tear even if it was initially in a pristine condition. This is because it was not designed to withstand the shear forces that could exist in this scenario.

Decision Attribute	Applicable to Decision?	Basis for Input to Decision – Provide qualitative and/or quantitative information for management review and decision-making.
Period of time (exposure time) affect on the performance deficiency.	Yes	It is highly likely that the bladder fabric became adhered to the tank wall immediately after nitrogen injection was secured to the tank. This condition persisted for 90 days until steps were taken by the licensee to separate the bladder fabric from the wall. Therefore, the auxiliary feedwater system was potentially nonfunctional for this time period.
The likelihood that the licensee's recovery actions would successfully mitigate the performance deficiency.	Yes	Control room operators would have a short period of time to diagnose the loss of suction to the auxiliary feedwater pumps. Two actions would be critical in mitigating the consequences of this event: • Operators would need to secure the pumps to prevent overheating and potential damage. This would require action within minutes of the onset of the event. • Assuming that operators succeeded in stopping the pumps in time to prevent damage, they could follow procedures to initiate an alternate suction path from the service water system.
Additional qualitative circumstances associated with the finding that regional management should consider in the evaluation process.	Yes	• An event at the Farley Nuclear Plant is strongly analogous to the Comanche Peak situation. In the Farley Nuclear Plant event, the bladder fabric tore and sank to the bottom of the tank resulting in the blocking of pump suction piping. Materials were similar in both cases. The primary difference was that in the Farley Nuclear Plant case there was no nitrogen between the water surface and the bladder, while at Comanche Peak, there was a nitrogen bubble present. Although it is possible that the nitrogen bubble would migrate to the sides of tank walls and separate the fabric, the amount of pressure available to perform this action was calculated to be less than 0.5 inches of water. This was probably not sufficient to separate the fabric from the wall and thereby prevent the large shear forces that would ensue from a rapid drawdown of the tank. • The tank wall surface and the bladder fabric are both hydrophilic, meaning that capillary forces would be very strong. These forces could be overcome fairly easily with application of a normal force, as occurred when nitrogen was injected

Decision Attribute	Applicable to Decision?	Basis for Input to Decision – Provide qualitative and/or quantitative information for management review and decision-making.
		through the topside connection, but the amount of shear force needed to separate the fabric from the wall could be severe. It is expected that the primary component of the force that would occur during an actuation of the auxiliary feedwater system would be in the shear orientation. • There is some uncertainty with the timing of the potential ripping and sinking of the bladder fabric. With a specific gravity of 1.1, the fabric might sink very slowly and not affect the pump suctions until the entire contents of the condensate storage tank was used by the auxiliary feedwater system. In this case, the performance deficiency would have little significance. However, there is also a possibility that currents in the tank would quickly draw the torn fabric sections to the suction openings.

Result of management review (COLOR): White

Corrosion
Control
Service
Inc.

P.O. Box 3708
Davenport, Iowa 52808
Phone (563) 324-9076
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Texas Utilities
PO Box 1002
6322 North FM 56
Glen Rose, TX 76043

July 29, 2002
Resent October 15, 2004 - Dave Weyandt

RE: NEED FOR DIAPHRAGM INSPECTION DURING OPERATION
REACTOR MAKEUP WATER STORAGE TANKS
CONDENSATION STORAGE TANKS
DEMINERALIZED WATER STORAGE TANKS
RECYCLE HOLDUP TANKS

Attn Purchasing:

We strive to circumvent any problems from arising during diaphragm operations. We feel that we have achieved the optimum in design and material composition, this insures the diaphragm will not fail from material degradation or seam failure. These are the most commonly accepted modes of failure.

Our diaphragms provide a passive means of keeping the water chemistry in the tanks as close as possible to specification while in storage. Air will not enter or exit through the diaphragm material. All airborne contaminates, dust, leaves, birds, bugs, tree frogs, etc. are also excluded.

The diaphragm's design concept is to permit the diaphragm to have a leisurely, unrestricted life of floating stress-free up and down as the liquid level changes. During the time the tank is filling the diaphragm floats on the water's surface. This is due to buoyancy. The water level raises the diaphragm's side wall material. The diaphragm side wall folds in layers similar to an accordion. Gravity will not let this occur in any other manner. The diaphragms are the same size as the tanks they are installed in. This allows for the diaphragm to displace the same volume as leaves the tank.

CCSI is committed to insuring trouble-free operation of our diaphragms. Based on our recent observations, DIAPHRAGMS could be subject to conditions during plant operation that warrant your personnel considering visually inspecting them.

Site specific conditions may exist that require the diaphragm to be visually inspected to insure it is still enjoying the leisurely life we assumed it would have.

Our areas of concern are the accumulation of too much air on the water side of the diaphragm and the complete absence on the water side of air between the tank wall and the diaphragm material. Some systems collect gases under the diaphragm due to upsets or purging. Others lose gases due to absorption by the water. Some systems seem to self-balance. The diaphragm, to function freely without undo stress, must have some air remaining between the tank wall and the diaphragm material on the waterside.

Our concerns are greater for the absence of gases since we have observed the diaphragm material sticking tighter than wallpaper to the tank wall. We believe this condition was caused by the absorption of all the gases into the water. The time frame for complete absorption to happen is unknown. This is a major factor of concern. In order for the diaphragm to function without stress, it needs only a fraction of an inch of air between it and the tank wall. Gravity will unfold the side wall material of the diaphragm as the tank level goes down.

A complete absence of air when lowering the water level could cause the side wall material of the diaphragm to be held firmly against the tank wall. This creates a condition that could cause the diaphragm to unfold from the bottom dragging the material down the wall instead of gravity unfolding it from the top down. If enough friction is generated with atmospheric pressure being the clamping force, it could result in the diaphragm being ruptured. The diaphragm's sidewalls freedom of movement is our specific concern. Our concern is that the necessary air left around the circumference of a tank wall could be completely absorbed. We base this on our recent observation. We are not able to calculate the rate of absorption because the area of exposure of the air to the water is very small. The diaphragm material completely covers the surface of the water. The area where the absorption occurs is in the minor areas around the circumference.

An extreme excess of air allowed to accumulate under the diaphragm could create positive pressure in the tank. This can occur if the tank is operating at a relativity low level during accumulation and then raised to maximum level.

Our concern is that if there is not an inspection after startup, we won't know if there is a problem that could be been avoided. The solution to our concerns is as simple as removing the excess accumulated gases or by simply adding a two hundred twenty cubic foot cylinder of nitrogen every six months to insure the diaphragm's freedom of movement on the sidewalls of the tank.

Visual verification of the diaphragm condition during operation is easily accomplished using a mirror or two, depending upon the placement of the manway on the top of the tank. Excessive accumulation of gas will be apparent if the diaphragm is inflated close to the tank roof. An absence of gas will be evident by the diaphragm being pasted to the tank wall instead of hanging vertical from the mounting angle. The ideal condition to find would be if there was a donut-shaped ring of gas trapped around the circumference of the mounting ring extending down to where the diaphragm side wall meets the water.

These conditions are easily corrected but there is no way to insure they are not present without visual verification. Our policy to find the cause before fixing the problem helps insure trouble-free operation of this system. Operations personnel should be informed that there is a probability of these conditions during operation of the plant.

CCSI's discoveries of these concerns came from a history of observations. We continually strive to share this information to insure reliability of our diaphragm system.

Please contact us if you need any further explanation or have other questions. We will provide any assistance or support needed to insure that your personnel understanding of the above concerns is complete.

Sincerely,
CCSI DIAPHRAGM DIVISION

DONALD L. FRANTZ
PRESIDENT