UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BOULEVARD
ARLINGTON, TEXAS 76011-4511

August 20, 2018

Ken J. Peters, Senior Vice President
and Chief Nuclear Officer
Attention: Regulatory Affairs
Vistra Operations Company LLC
P.O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT – NRC DESIGN BASES ASSURANCE INSPECTION (TEAMS) REPORT 05000445/2018010 and 05000446/2018010

Dear Mr. Peters:

On July 12, 2018, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Nuclear Power Plant, Units 1 and 2, and discussed the results of this inspection with Mr. T. McCool, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented two findings of very low safety significance (Green) in this report. Both of these findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or significance of these NCVs, 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, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement; and the NRC resident inspector at the Comanche Peak Nuclear Power Plant.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Thomas R. Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-445 and 50-446
License Nos. NPF-87 and NPF-89

Enclosure:
Inspection Report 05000445/2018010 and 05000446/2018010
w/ Attachments:

1. Additional Request for Information
2. Supplemental Request for Information
3. Detailed Risk Evaluation

U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000445, 05000446
License Numbers:	NPF-87, NPF-89
Report Numbers:	05000445/2018010 and 05000446/2018010
Enterprise Identifier:	I-2018-010-0042
Licensee:	Vistra Operations Company, LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	Glen Rose, Texas
Inspection Dates:	June 25, 2018, to July 12, 2018
Inspectors:	J. Braisted, PhD, Reactor Inspector, Team Lead B. Correll, Reactor Inspector C. Speer, Resident Inspector D. Reinert, PhD, Resident Inspector M. Bloodgood, Emergency Response Specialist R. Deese, Senior Reactor Analyst
Accompanying Personnel:	C. Baron, Contractor, Beckman and Associates S. Gardner, Contractor, Beckman and Associates
Approved By:	T. Farnholtz, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting Inspection Procedure 71111.21M, "Design Bases Assurance (Teams)," at Comanche Peak Nuclear Power Plant, Units 1 and 2, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information. NRC-identified and self-revealed findings, violations, and additional items are summarized in the table below.

List of Findings and Violations

Failure to Establish Test Program to Verify Residual Heat Removal Suction Valve Capability			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000445/2018010-01; 05000446/2018010-01 Closed	None	71111.21M
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to establish a test program to ensure that residual heat removal suction isolation valves would perform adequately in service.			

Failure to Provide Procedural Guidance for the Failure of a Component Cooling Water Surge Tank Makeup Valve			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000445/2018010-02; 05000446/2018010-02 Closed	None	71111.21M
The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to provide procedural guidance for the failure of a component cooling water surge tank makeup valve.			

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

REACTOR SAFETY

71111.21M—Design Bases Assurance Inspection (Teams)

The inspectors evaluated the following components and listed applicable attributes, permanent modifications, and operating experience during the weeks of June 25 to June 29, 2018, and July 9 to July 12, 2018:

Component (5 Samples)

(1) 125 VDC Switchboard 1ED1

a) Component system health and history reports to verify the monitoring of potential degradation.

b) Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remain within minimum acceptable limits.

c) Protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

d) Procedures for circuit breaker preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

(2) Safety-Related Chiller 2-06

a) Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

b) Calculations for heat loading and thermal performance under accident conditions.

c) Operations procedures for system loading under accident conditions.

d) Preventative maintenance and testing program documents.

(3) Component Cooling Water (CCW) Pump 2-02

a) Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

b) Calculations for system flow, system flow balance, net positive suction head, surveillance test acceptance criteria minimum flow, and runout flow.

c) The impact of minimum and maximum allowable electrical power supply frequency on pump performance and net positive suction head.

d) Procedures for operation of the CCW system under accident conditions.

e) Design of the safety-related makeup flowpath to the CCW system.

f) Procedures related to cross-tying the CCW system between units.

(4) 6900 VAC Switchgear 1EA1

a) Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

b) Calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within minimum acceptable limits.

c) Protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

d) Procedures for circuit breaker preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

e) Corrective actions associated with a non-cited violation involving undervoltage relay settings documented in the 2013 Component Design Bases Inspection report (ML13214A346).

(5) 6900/480 VAC Transformer T1EB4

a) Component maintenance history and corrective action program reports to verify the monitoring of potential degradation.

b) Calculations for electrical distribution and electrical protection to verify that transformer capacity and voltages remained within minimum acceptable limits.

c) The protective device settings and circuit breaker ratings to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions.

d) Procedures for transformer preventive maintenance, inspection, and testing to compare maintenance practices against industry and vendor guidance.

Component Large Early Release Frequency (LERF) (1 Sample)

(1) Residual Heat Removal Valve 2-8701B

a) Procedures for valve operation during normal, shutdown, and post-accident conditions.

b) Calculations for valve pressure interlock setpoints and interlock surveillance test records.

c) Motor operated valve program calculations for required and available voltage during normal and alternate electrical lineups.

Permanent Modification (5 Samples)

(1) FDA-2010-000172-01-01, "Replace Manual Valve 1-8401A with a Motor Operated Valve"

(2) FDA-2010-000172-36-07, "Multiple Spurious Operations Cause Refueling Water Storage Tank Drain Down"

(3) FDA-2013-000185-01-00, "Lift Check Valve 2SI-8819A Requires Replacement with a Nozzle Check Valve due to Excessive Leakage Past the Seat"

(4) FDA-2014-000134-01-06, "Install 6 amp Fuses in 1E DC Battery Supply"

(5) FDA-2015-000089-01-00, "This FDA Validates That 67 CFR Pressure Regulators may be used in Locations where the Design Basis Event is Seismic or Environmentally Harsh"

Operating Experience (3 Samples)

(1) NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire"

(2) NRC Information Notice 2014-04, "Potential for Teflon® Material Degradation in Containment Penetrations, Mechanical Seals, and Other Components"

(3) NRC Information Notice 2015-05, "Inoperability of Auxiliary and Emergency Feedwater Auto-Start Circuits on Loss of Main Feedwater Pumps"

Evaluation of Inspection Sample Related Operator Procedures and Actions

(1) Control room operator actions resulting from a simulated steam generator tube rupture (SGTR) accident followed by a post reactor trip loss-of offsite power with a single failure of an intact steam generator atmospheric relief valve.

a) Control room crew was expected to enter procedures for standard post trip actions and SGTR.

b) Following the failure of an intact steam generator atmospheric relief valve, the crew was expected to cooldown using the two remaining atmospheric relief valves.

(2) In plant operator actions resulting from a loss of instrument air.

a) In plant operators were expected to manually fill the CCW surge tank.

b) Following the loss of instrument air to the CCW surge tank fill valves, the operators were expected to manually operate the fill valves.

INSPECTION RESULTS

Failure to Establish Test Program to Verify Residual Heat Removal Suction Valve Capability			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000445/2018010-01; 05000446/2018010-01 Closed	None	71111.21M

Introduction:

The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for the failure to establish a test program to ensure that residual heat removal suction isolation valves would perform adequately in service.

Description:

The inspectors reviewed the design and testing associated with the residual heat removal (RHR) suction isolation valves. Each RHR suction line is equipped with two redundant motor operated valves that isolate the higher pressure reactor coolant system from the lower pressure RHR system during normal plant operation. Following a design basis accident, licensed operators open the valves to initiate cooldown using the RHR system.

As discussed in final safety analysis report (FSAR) Appendix 5A, the RHR system is designed to bring the plant from hot shutdown to cold shutdown in a reasonable period of time, assuming the most limiting single failure. To address the limiting single failure of one emergency power train, the two valves in each RHR suction line are powered from different emergency power trains. This arrangement allows that, even with a single failure of an emergency electrical train, both RHR suction lines can maintain their isolation capability. However, the failure of either emergency power train will prevent the initiation of RHR cooling in the normal manner.

In the event of such a failure, the affected valve can be opened using proceduralized operator actions outside the control room. Normally, valve 8701B is supplied from the train A power supply and valve 8702A from the B power supply. If either of these valves cannot be opened using their normal power supplies, power and control cables for either valve can be swapped to its alternate, unaffected emergency power train. Several abnormal operating procedures include the use of this alternate power lineup for valves 8701B and 8702A.

The inspectors reviewed the periodic testing associated with these motor operated isolation valves and determined that not all valves were being tested in all potential post-accident configurations. Specifically, the licensee was not periodically testing to assure that valve 8701B could be opened using its alternate power supply. A latent failure within the alternate power lineup would result in RHR suction isolation valve 8701B failing to open and could cause a loss of RHR system function.

Corrective Actions: The licensee verified that individual active components within the alternate power supply lineup, including the motor control center breaker and valve operator, are routinely tested. The licensee also initiated an action to test the valves from their alternate power supplies during the next refueling outage.

Corrective Action Reference: CR-2018-004665.

Performance Assessment:

Performance Deficiency: The licensee's failure to establish a test program to assure that RHR suction isolation valve 8701B would perform satisfactorily in service, as required by 10 CFR Part 50, Appendix B, Criterion XI, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the lack of testing affected the objective because there was no method to determine the capability of the valve to perform its function in the event of a postulated single failure of an emergency electrical train during an accident which could affect the residual heat removal function.

Significance: The inspectors assessed the significance of the finding using NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings at Power," dated October 7, 2016. Using Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability.

Cross-cutting Aspect: No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure 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, since initial plant startup until July 11, 2018, the licensee failed to establish a test program to assure that testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures. Specifically, the licensee did not establish a test

program to assure that RHR suction isolation valve 8701B would perform satisfactorily when powered from its alternate power source. Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Failure to Provide Procedural Guidance for the Failure of a Component Cooling Water Surge Tank Makeup Valve			
Cornerstone	Significance	Cross-cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000445/2018010-02; 05000446/2018010-02 Closed	None	71111.21M
<u>Introduction</u>: The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure to provide procedural guidance for the failure of a CCW surge tank makeup valve.			
<u>Description</u>: The inspectors reviewed the design of the CCW system and source of makeup to the CCW system. Through a single flowpath, the reactor makeup water system provides the only safety-related makeup to the CCW surge tank in order to accommodate CCW system leakage, to ensure CCW pumps have sufficient net positive suction head, to allow for thermal expansion and contraction of the CCW system, and to provide a means of CCW system overpressure protection. Valve 4500-1 is a safety-related, fail-open, air-operated valve in this single flowpath and is considered part of the CCW system. This valve is normally closed. During a design basis accident, when level in the CCW surge tank reaches the “lo-lo” setpoint, the safeguards loops automatically isolate and an alarm response procedure directs the operators to ensure valve 4500-1 is open. If valve 4500-1 were to fail in the closed position, or if any other component in the single flowpath were to fail, there are currently no instructions or procedures to provide alternate makeup methods to the CCW surge tank. As discussed in CCW FSAR Section 9.2.2.2.1, the failure or malfunction of any single active or passive component does not prevent fulfillment of the CCW system safeguards functions. However, the only safety-related source of makeup to the CCW surge tank is a single flowpath from the reactor makeup water system. Because the CCW system would be required to operate in the long term following a design basis accident, a source of makeup water would be required to accommodate isolation valve leakage, among other purposes. A postulated single failure in this flowpath could prevent fulfillment of the CCW system safeguards functions. Additionally, as discussed in CCW design basis document DBD-ME-229, Section 5.4.2, and CCW FSAR Table 9.2-5, if the reactor makeup valve 4500-1 fails in the closed position as a result of an electrical or mechanical single failure within the valve, an operator action to open			

the valve by venting the diaphragm and/or forcing the valve open may be required. There were no instructions or procedures directing the operators to take these actions or to establish an alternate source of makeup water to the CCW surge tank to ensure functionality of the CCW system.

Corrective Actions: The licensee implemented a compensatory measure, failing open valve 4500-1 by removing air to it, until permanent corrective actions are accomplished.

Corrective Action Reference: IR-2018-004603 and IR-2018-004701.

Performance Assessment:

Performance Deficiency: The licensee’s failure to provide procedural guidance for the failure of a CCW surge tank makeup valve, as required by 10 CFR Part 50, Appendix B, Criterion V, was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, given a postulated single failure of valve 4500-1, or another component in the single makeup flowpath, the lack of procedural guidance for ensuring makeup to the CCW surge tank during an accident could affect the ability of the CCW system to perform its safeguards function.

Significance: The inspectors assessed the significance of the finding using NRC Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process for Findings at Power,” dated October 7, 2016. Using Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding required a detailed risk evaluation because the finding involved a deficiency affecting the design and qualification of a mitigating structure, system, or component that lost its operability or functionality and represented a loss of system function. A Region IV senior reactor analyst performed a detailed risk evaluation and determined that the bounding increase in core damage frequency for this issue was 7.9E-8/year for both units, and the finding was therefore of very low safety significance (Green). Additional information regarding the detailed risk evaluation is found in Attachment 3 of this report.

Cross-cutting Aspect: No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

Violation: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” required, 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.

Contrary to the above, since initial plant startup until July 12, 2018, the licensee failed to prescribe by documented instructions, procedures, or drawings, of a type appropriate to the circumstances activities affecting quality. Specifically, the licensee failed to provide

procedural guidance for the failure of CCW surge tank makeup valve 4500-1, or the failure of another component, in the single safety-related makeup flowpath.

Disposition: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

EXIT MEETINGS AND DEBRIEFS

On July 12, 2018, the inspectors presented the results of this design bases assurance inspection to Mr. T. McCool, Site Vice President, and other members of the licensee staff. The inspectors verified no proprietary information was retained or documented in this report.

DOCUMENTS REVIEWED

71111.21M—Design Bases Assurance Inspection (Teams)

Condition Reports (CRs) (Reviewed)

CR-2013-006252	CR-2010-004244	CR-2014-010113	CR-2017-001489
CR-2016-010346	CR-2011-001742	CR-2016-008215	CR-2018-004696
CR-2010-005563	CR-2015-008517	CR-2018-001530	CR-2015-007625
CR-2015-009942	CR-2015-009839	CR-2014-004995	CR-2013-008401
CR-2015-011497	CR-2015-010339	CR-2015-010000	CR-2015-009979
CR-2017-000269	CR-2016-007653	CR-2016-003348	CR-2015-011913
EV-CR-2014-003591	CR-2016-010346	TR-2014-009407	TR-2018-003301
CR-2008-000089	CR-2017-007437	CR-2018-001532	CR-2014-010279
CR-2015-004579	CR-2017-012024	CR-2018-000941	CR-2018-004367
CR-2018-001372	CR-2018-000940	CR-2017-011633	CR-2017-004995
CR-2018-004259	CR-2018-003136	CR-2018-002879	CR-2018-001671
CR-2017-002493	EV-CR-2012-007312	IR-2018-004369	CR-2017-002493
CR-2012-007312	OER-2017-004566		

Condition Reports (CRs) (Issued)

IR-2018-004602	IR-2018-004612	IR-2018-004637	CR-2018-004638
IR-2018-004367	CR-2018-004369	CR-2018-004448	IR-2018-004390
CR-2018-004403	IR-2018-004447	CR-2018-004597	CR-2018-004660
CR-2018-004665	IR-2018-004603	IR-2018-004624	IR-2018-004649
IR-2018-004660	IR-2018-004665	CR-2018-004447	IR-2018-004701

Work Orders

4747766	5180554	5438272	5464313
5588117	5063996	5212783	5582853
5538622	5609896	4240872	5261005
3604038	5186334	3659824	5273830
4610645	149566	5179229	5211820
4598867	4598838	5465353	149564
399132	4827245	4842555	4881066
4967904	5174262	4977059	5198308
5542215	5494586	5198308	4064912
5460952	5574642		

Procedures

Number	Title	Revision
ABN-301	Instrument Air System Malfunction	13
ABN-502A	Component Cooling Water System Malfunctions	9
ABN-602	Response to 6900/480V System Malfunction	8
ABN-803A	Response to a Fire in the Control Room or Cable Spreading Room	13

Procedures Number	Title	Revision
ABN-803B	Response to a Fire in the Control Room or Cable Spreading Room	10
ABN-804A	Response to Fire in the Safeguards Building	6
ABN-804B	Response to Fire in the Safeguards Building (Unit 2)	4
ABN-805A	Response to Fire in the Auxiliary Building or the Fuel Building (Unit 1)	8
ABN-805B	Response to Fire in the Auxiliary Building or the Fuel Building (Unit 2)	7
ABN-806A	Response to Fire in the Electrical and Control Building (Unit 1)	8
ABN-806B	Response to Fire in the Electrical and Control Building (Unit 2)	6
ABN-807A	Response to Fire in the Containment Building (Unit 1)	8
ABN-807B	Response to Fire in the Containment Building (Unit 2)	6
ABN-808A	Response to Fire in Service Water Intake Structure	6
ALM-0032A	Alarm Procedure 1-ALB-3B (Unit 1)	7
ALM-0032B	Alarm Procedure 1-ALB-3B (Unit 2)	3
ALM-0102A	Alarm Procedure 1-ALB-10B	12
ECA 3.1A	Steam Generator Tube Rupture with Loss of Reactor Coolant - Subcooled Recovery Desired (Unit 1)	9
ECA 3.1B	Steam Generator Tube Rupture with Loss of Reactor Coolant - Subcooled Recovery Desired (Unit 2)	9
EOP 0.0A	Reactor Trip or Safety Injection (Unit 1)	9
EOP 0.0B	Reactor Trip or Safety Injection (Unit 2)	9
EOP 3.0A	Steam Generator Tube Rupture (Unit 1)	9
EOP 3.0B	Steam Generator Tube Rupture (Unit 2)	9
EOP-0.0B	Reactor Trip or Safety Injection	9
INC-7756B	Channel Calibration Reactor Coolant System Wide Range Pressure and RHR Isolation Valve Interlock Test	4
IPO-002A	Startup from Hot Standby	21
IPO-003A	Power Operations	30
MSE-C0-6305	6.9KV 7.5 HK Circuit Breaker Enhanced Maintenance	3
MSE-GO-6300	Breaker Removal and Installation	3

Procedures Number	Title	Revision
MSE-P0-5304	GE DC Switchboards Inspection and Testing	2
MSE-P0-6000	6.9 KV Switchgear Clean and Inspection	7
MSE-P0-6305	Station Transformer Maintenance (Dry Type)	6
MSE-S0-6301	6.9KV Air Circuit Breaker Inspection and Cleaning	6
MSE-S0-6303	Molded Case Circuit Breaker Test and Inspection	8
MSE-S1-0602A	Unit 1 train A Electrical UV Relay Test, Response Time Test and Bus Transfer Test	2
MSE-S1-0603A	Unit 1 train A UV Relay Calibration and Response Time Surveillance Test	7
OPT-108A-2	RSP/STP Switch and Controller Lineup Verification Data Sheet	13
OPT-216A	Remote Shutdown Operability Test	14
OPT-430A	train A Integrated Test Sequence	7
OPT-512B	ECCS Operability	6
OPT-512B	Residual Heat Removal and SI Valve Subsystem Valve Test	11
OPT-612B	Reactor Coolant System Pressure Boundary Leakage Test For Loop 1 CL Injection Valves	3
PPT-S0-6000	Motor Operated Valve Risk-Informed IST	3
SOP-102B	Residual Heat Removal System	15
SOP-302A	Feedwater System	19
SOP-304A	Auxiliary Feedwater System	17
SOP-304B	Auxiliary Feedwater System	13
SOP-506	Spent Fuel Pool Cooling and Cleanup System	21
SOP-815B	Safety Chilled Water System	11
STA-716	Modification Process	26
STI-426.02	Processing important OE	0
TSP-509	Predictive Maintenance Thermographic Analysis Program	6

Calculations Number	Title	Revision or Date
2-EE-0011	Protection and Ampacity of Electrical Containment Penetration	11
2-ME-0071	Unit 2 Component Cooling Water Heat Loads and Temps for Various Operating Modes	1
2-ME-0121	Determine Available NPSH(A)	0

Calculations Number	Title	Revision or Date
2-ME-0177	Component Cooling Water Flow Distribution	0
EE01E-2EB3-2	Cable Sizing Report – Voltage	7
EE-1E-2EB4-2	Cable Sizing Report - Voltage	6
EE-1E-BT1ED1	125V DC Battery and Charger Sizing Calculation	7
EE-CA-0008-0871	Protective Relay Settings for Safeguard Buses OV/UV Relays and Associated Time Delay Relays	18
EE-CA-0008-157	Coordination Study of 6.9KV Power Distribution	4
EE-CA-0008-182	Coordination Study – 125V DC Class 1E Power Distribution System	3
EE-SC-U1-1E	Unit 1 and Unit 2 Class 1E System Short Circuit Study with Unit 1 Preferred Source Lineup	5
EE-VP-U1-1E	Unit 1 Class 1E System Voltage Profile	5
ER-ME-089	Resolution of NRC Information Notice IN-92-018 “Potential Loss of Remote Shutdown Capability Following Control Room Fire”	0
FSD/SS-TBX-340	Residual Heat Removal Initiation Window	April 29, 1982
IC(B)-064	Main Steam Valve Air Pressure	1
ME(3)-073	Component Cooling Water Surge Tank Volume	3
ME(B)-0267	Component Cooling Water Flow Distribution	1
ME(B)-071	Component Cooling Water Pump NPSH for MELB	3
ME(B)-093	Hydraulic Analysis of Component Cooling Water	1
ME-CA-0000-5478	Fire Safe Shutdown Analysis – MS) – Refueling Water Storage Tank Gravity Drain Down Time (to Containment Sumps)	0
ME-CA-0000-5483	Fire Safe Shutdown Analysis – MSO – HBC-0 Stop Nut Evaluation in SMB-000 Actuators under stall conditions	1
ME-CA-0206-5543	TDAFW Pump Crimped Exhaust Stack Evaluation	0
ME-CA-0206-5545	TDAFW Pump Crimped Flash Tank Vent Evaluation	0
ME-CA-0229-5127	The Concerns Raised by SMF-1999-001334 on Calculation ME(B)-255 Revision 1	0
ME-CA-0260-5471	RHR Temperature Limits	0
ME-CA-1100-3356	Component Cooling Water Flow Balance for LOCA with Flows Throttled	0
TE-93-56	Component Cooling Water Pump IST Basis	0
TNE-EE-CA-0008-265	Selection and Settings of Relays and CTs for Unit 1 and Unit 2	4

Drawings Number	Title	Revision
50020445	Penetration Assy Low Voltage Power	T
DDVEN-PL-7551-1000	Conax Penetration BOM	A
E1-0001	Plant One Line Diagram	CP-33
E1-0004	6.9 KV Auxiliaries One Line Diagram	CP-41
E1-0024, Sheet 4	Device Level One Line Diagram Fuse/Breaker Bill of Material	CP-89
E1-0031, Sheet 1	6.9 KV Switchgear Bus 1EA1	CP-10
E1-0031, Sheet 21	6.9 KV Switchgear Bus 1EA1 Diesel Breaker	CP-11
E1-0031, Sheet 3	6.9 KV Switchgear Bus 1EA1 Breaker 1EA1-2	CP-19
E1-0061, Sheet 22	Motor Operated Valve 1-8811A Sump to Number 1 Residual Heat Removal Pump	CP-9
E1-0061, Sheet 23	Motor Operated Valve 1-8811B Sump to Number 2 Residual Heat Removal Pump	CP-10
E1-0061, Sheet 4	Motor Operated Valve 1-8110 Charging Pump Miniflow Isolation	CP-10
E1-0061, Sheet 5	Motor Operated Valve 1-8111 Charging Pump Miniflow Isolation	CP-9
E1-0061, Sheet 66	Motor Operated Valve 1-8351A Seal Water Injection Isolation	CP-5
E1-0062, Sheet 24	Motor Operated Valve 1-8812A Refueling Water Storage Tank to RHR Pump 1 Isolation	CP-8
E1-0062, Sheet 25	Motor Operated Valve 1-8812B Refueling Water Storage Tank to RHR Pump 2 Isolation	CP-9
E1-0063, Sheet 2	Motor Operated Valve 1-8701B Residual Heat Removal Loop 2 Inlet Isolation Valve	CP-7
E1-0063, Sheet 4	Motor Operated Valve 1-8702B Residual Heat Removal Loop 2 Inlet Isolation Valve	CP-8
E1-2400, Sheet 134	Protective Device Settings – 6.9 kV Safeguard Buses	CP-1
E1-2400, Sheet 152	Protective Device Settings 6.9KV Safeguard Buses	CP-6
E1-2400, Sheet 153	Protective Device Settings 6.9KV Safeguard Buses	CP-8
E1-2400, Sheet 320	Protective Device Settings 480V Safeguard Buses	CP-6

Drawings Number	Title	Revision
E1-2400, Sheet 321	Protective Device Settings 480V Safeguard Buses	CP-6
E1-2400, Sheet 322	Protective Device Settings 480V Safeguard Buses	CP-5
E2-0024, Sheet 4	Device Level One Line Diagram Fuse/Breaker Bill of Material	CP-48
E2-0061, Sheet 4	Motor Operated Valve 2-8110 Charging Pump Miniflow Isolation	CP-6
E2-0061, Sheet 5	Motor Operated Valve 2-8111 Charging Pump Miniflow Isolation	CP-8
M1-0229	Flow Diagram Component Cooling Water System	CP-23
M1-0229, Sheet A	Flow Diagram Component Cooling Water System	CP-21
M1-0229, Sheet B	Flow Diagram Component Cooling Water System	CP-25
M1-0307, Sheet A	Flow Diagram Chilled Water System	CP-8
M1-0307, Sheet B	Flow Diagram Chilled Water System	CP-8
M1-0307, Sheet C	Flow Diagram Chilled Water System	CP-4
M2-0229	Flow Diagram Component Cooling Water System	CP-19
M2-0229, Sheet A	Flow Diagram Component Cooling Water System	CP-14
M2-0229, Sheet B	Flow Diagram Component Cooling Water System	CP-15
M2-0263	Flow Diagram Safety Injection System	CP-17
M2-0263, Sheet A	Flow Diagram Safety Injection System	CP-7
M2-0263, Sheet B	Flow Diagram Safety Injection System	CP-13
M2-0263, Sheet C	Flow Diagram Safety Injection System	CP-7
M2-0307, Sheet A	Flow Diagram Chilled Water System	CP-14
M2-0311	Flow Diagram Safety Chilled Water System	CP-9
M2-0311, Sheet A	Flow Diagram Safety Chilled Water System	CP-6
SK-0001-10-000172-01-00	Flow Diagram Chemical and Volume Control System Charging and Positive Displacement Pump Trains	00
SK-0003-10-000172-01-01	Chemical and Volume Control	01
SK-0009-10-000172-01-01	Vents and Drains System Flow Diagram Auxiliary Building Leak-offs	01

Miscellaneous Number	Title	Revision or Date
2323-ES-012A	Specification Electrical Penetration Assemblies	0
59EV-2010-000172-01-00	50.59 Evaluation	April 11, 2012
59SC-2010-000172-0-03	50.59 Screening	June 3, 2013
59SC-2013-000185-01-00	Replace 2SI-8819A	February 4, 2014
59SC-2015-000089-01-00	50.59 Screen for FDA-2015-000089-01	0
CP-0080B-002	Hermetic Turbopak Safety-Related Chillers	19
CP-0425-001	6.9kV Metal Clad Switchgear	28
CP-0430-002	Indoor Low Voltage Metal Enclosed Switchgear	29
D102601X012	Manual - 67C Series Instrument Supply Regulators	March 2017
DBD-EE-040	6.9kV Electrical Power System	18
DBD-EE-051	Protection Philosophy	44
DBD-EE-062	Containment Electrical Penetration Assemblies	17
FDA 2014-FDA 000130	Design Change UV Setpoints	1
FDA-2010-000172-36-07	Multiple Spurious Operations Drain Down of Refueling Water Storage Tank	7
FDA-2013-000185-01	Lift Check Valve 2SI-8819A Requires Replacement with a Nozzle Check Valve due to Excessive Leakage Past the Seat	February 4, 2014
FDA-2014-000134-01-06	Unfused DC Ammeter Circuits	6
FDA-2015-000089-01-00	This FDA validates that 67 CFR pressure regulators may be used in locations where the design basis event is seismic or environmentally harsh	October 2, 2015
Fire Watch Map	Fire Watch No. 18-0007	June 21, 2018
PQE ID:229	Qualification Evaluation: Elec Penetration	1
System Health Report	AC Distribution 480 MCCs	4th Qtr 2017
System Health Report	Switchyard Equipment (EPA, EPB, IPC, EP)	2nd Qtr 2018
TSN-468698	Pressure Regulator 0-60 psig	June 7, 2018
TSN-468699	Pressure Regulator 0-60 psig	June 7, 2018

Miscellaneous Number	Title	Revision or Date
VTMR-001-802-004	Testing and Maintenance of Molded Case Circuit Breakers	
VTMR-001-802-150	Installation and Maintenance Instructions AV-Line Switchboards	
WCAP-11736-A	Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners Group	0
White Paper	Evaluation of Timing Associated with Refueling Water Storage Tank Drain Down through a Spuriously Open Containment Sump Isolation Valve	
WPT-17834	Steam Generator Tube Rupture Margin to Overfill Addressing NSAL 07-11	0

Design Bases Documents Number	Title	Revision
DBD-EE-044	DC Power Systems	27
DBD-EE-051	Protection Philosophy	44
DBD-ME-229	Component Cooling Water System	41
DBD-ME-260	Residual Heat Removal System	29
DBD-ME-261	Safety Injection System	36
DBD-ME-311	Safety Chilled Water System	18

ADDITIONAL REQUEST FOR INFORMATION

Additional Request for Information
2018 Design Bases Assurance Inspection (Teams)
Comanche Peak Nuclear Power Plant

Please provide the following information by the start of the first on-site inspection week (June 25, 2018), separated by inspector and sample.

<u>Brian Correll</u>

1. NRC Information Notice 92-18
    a. Unit 1 and Unit 2 Control Room (and Cable Spread Room) Evacuation Fire Procedures
    b. Stone & Webster Calculation 02101.02 NM(B)-370 R/0
    c. Westinghouse letter WPT-15195
    d. Unit 1 and Unit 2 Fire Safe Shutdown Analysis for control room and cable spreading rooms (areas required for Alternative Shutdown/control room evacuation)
2. 125 VDC Switchgear 1ED1
    a. Battery vendor manual (for BT1ED1 and BT1ED3) showing the 4-hour and 8-hour discharge rate AH ratings
    b. Documentation/descriptions of battery maintenance activities that implement IEEE-450-1995 guidance
    c. FDA-2017-000002-01-00 design documentation

<u>Susan Gardner</u>

1. NRC Information Notice 2014-04 (OER-2017-004566)
    a. CR-2014-003591
    b. CR-2014-003762
    c. CR-2016-010346 (with root cause)
    d. Scope of Electrical and Instrument containment penetrations, that is, approximate number and size
    e. DBD-EE-062
    f. Any inspections or testing procedures for penetrations, including electrical penetration protection breakers
2. FDA-2014-000236-01-00
    a. Provide design documentation
3. 6.9 KV Switchgear 1EA1 and 6900/480 VAC Transformer (1EA2/1EB4) T1EB4
    a. EE-CA-0008-871 (calculation identified in DBD-EE-040)
    b. 1635-EE(B)-074 (calculation identified in DBD-EE-040)
    c. DBD-EE-51 "protection philosophy"
    d. Maintenance procedures for 6.9 KV breakers and cabinet 1EA1 including frequencies
    e. Any condition reports generated from last maintenance cycle on cabinet 1EA1
    f. Maintenance procedure for transformer T1EB4 and frequency
    g. Vendor manual for transformer T1EB4

Craig Baron

1. Component Cooling Water Pump 2-02
    a. Completed surveillance tests (3 most recent)

Miscellaneous

1. For IMS Item I.14, provide a list (with titles) of the procedures
2. For IMS Item I.20, provide the Technical Requirements Manual
3. For IMS Item II.04, provide a list (with titles) of the calculations

SUPPLEMENTAL REQUEST FOR INFORMATION

Supplemental Additional Request for Information
2018 Design Bases Assurance Inspection (Teams)
Comanche Peak Nuclear Power Plant

Susan Gardner

- 6.9 KV Switchgear 1EA1 and 6900/480 VAC Transformer (1EA2/1EB4) T1EB4
  - 6.9 KV system health report for last 2 quarters
  - Component health report, if any, for breakers
  - Vendor manual for breakers in 1EA1
  - System and/or Component health report that tracks transformer, T1EB4
  - Drawing and Bill of Material of electrical penetration for the RCP power cables.
  - Procedure or description of process of tracking breakers by serial number.
  - Schedule walkdown of T1EB4 and 1EA1 for Tuesday
  - If maintenance on 6.9KV breakers is performed on site, please request a walkdown of the breaker shop on Wednesday, if breakers are also sent out for maintenance, provide a copy of latest purchase order
- FDA-2014-000181-01-00
  - Provide design documentation

* Also, please delete the request for FDA-2014-000236-01-00 from the previous information request.

DETAILED RISK EVALUATION

Comanche Peak

Component Cooling Water Make-up Procedural Deficiency

Detailed Risk Evaluation

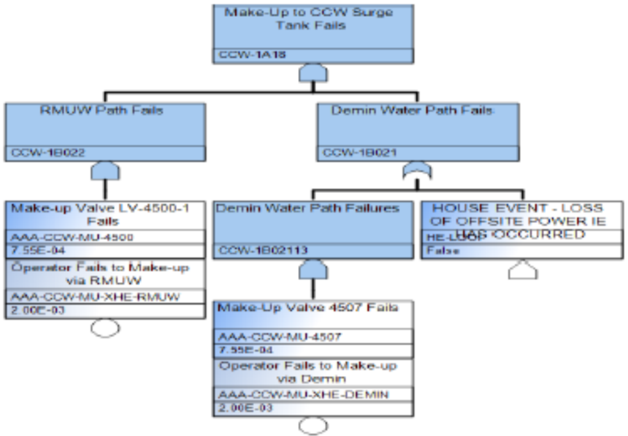
The Comanche Peak SPAR model did not model make-up to the component cooling water surge tank. Because the component cooling water system has high risk significance, the analyst modified the model to estimate the risk from the performance deficiency dealing with inadequate procedural guidance for the postulated failure of make-up capability to the surge tank.

The analyst modified Fault Trees CCW-1A, "Component Cooling Water Unit 1, Loop A," and CCW-1B, "Component Cooling Water Unit 1, Loop B," to add a failure mechanism for failure of make-up to the surge tank under "OR" gates CCW-1A-1, "Component Cooling Water Loop A is Unavailable," and CCW-1A-1, "Component Cooling Water Loop A is Unavailable." The failure mechanism was made in the form of an "AND" gate representing failure of component cooling water surge tank make-up capability from its two sources. Under this "AND" gate for make-up failure, a gate modeling failure of the Reactor Make-Up Water path and a gate modeling failure of the Demineralized Water path were made.

- Failure for the Demineralized Water path was modeled with an "OR" gate with House Event for a Loss of Offsite Power OR-ed with make-up valve 4507 failing AND-ed with a human error event for operators being unable to recover from the valve 4507 failure.

- Failure for the Reactor Make-Up Water path was modeled with an "AND" gate with make-up valve LV-4500-1 failing AND-ed with a human error event for operators being unable to recover from the valve LV-4500-1 failure.

The valve failure probabilities were assigned used Template Event ZT-AOV-FTOC, "Air Operated Valve Fails to Open," which had a failure probability of 7.55E-4. For the human error events, the analyst assigned extra time to the time available performance shaping factor for diagnosis while keeping all other performance shaping factors nominal to obtain a failure probability of 2.0E-3. The new gate which is typical of both component cooling water trains is pictorially depicted below:



To model the performance deficiency, the analyst changed the procedure performance shaping factor for action to "Not Available," making the probability 5.1E-2 that operators would fail to make-up to the component cooling water surge tank. The maximum exposure time of one year was used since the performance deficiency has existed for at least one year. This modeling of the performance deficiency yielded an estimate of 7.8E-8/year for the increase in core damage frequency from internal events.

Since the increase in core damage frequency was less than 1.0E-7/year, the increase in core damage frequency from external events and the increase in large early release frequency were not required to be estimated. Seismic events were reviewed however because of a known issue with leakage of the isolation valves for the non-safety portion of the component cooling water system. These isolation valves were assumed to leak at 42 gallons per minute in conjunction with seismically induced leakage in the non-safety portion of the system. The analyst estimated that time available to effect make-up to the surge tank would be shortened to change the time available performance shaping factor for action to barely adequate time; the experience performance shaping factor for action was low; and the stress performance shaping factor for action to extreme. This resulted in a human error probability under seismic conditions of 5.29E-1 and an increase in core damage frequency from seismic events for the performance deficiency of 2.6E-10/year. Adding this to the internal events estimate yielded a total increase in core damage frequency of 7.9E-8/year and the significance remained Green, or of very low safety significance, when seismic risk was also considered.

The analysis was performed using the Comanche Peak SPAR model, Version 8.55, ran on SAPHIRE, Version 8.1.8. Default truncation of 1E-12 was used. Dominant core damage sequences were all four types of losses of offsite power which were mitigated by the offsite power recovery and high pressure recirculation.

COMANCHE PEAK NUCLEAR POWER PLANT – NRC DESIGN BASES ASSURANCE INSPECTION (TEAMS) REPORT 05000445/2018010 and 05000446/2018010 – AUGUST 20, 2018

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☒ SUNSI Review: ADAMS: ☐ Non-Publicly Available ☒ Non-Sensitive Keyword: NRC-002
By: JDB ☒ Yes ☐ No ☒ Publicly Available ☐ Sensitive

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