UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON, TX 76011-4511

February 9, 2018

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

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000445/2017004 and 05000446/2017004

Dear Mr. Peters:

On December 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Nuclear Power Plant, Units 1 and 2. On January 3, 2018, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

NRC inspectors documented six findings of very low safety significance (Green) in this report. All 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.a 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.

If you disagree with a cross-cutting aspect assignment 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Comanche Peak Nuclear Power Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," a copy of this letter, its enclosure, and your response (if any) will be made available electronically for public inspection in the NRC's Public Document Room or the NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

To the extent possible, your response, if any, should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction.

Sincerely,

/RA/

Mark S. Haire, Chief
Project Branch A
Division of Reactor Projects

Docket Nos. 5000445 and 5000446
License Nos. NPF-87 and NPF-89

Enclosure:
Inspection Report 05000445/2017004 and 05000446/2017004
w/ Attachments:
1.) Supplemental Information
2.) Document Request

COMANCHE PEAK NUCLEAR POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000445/2017004 and 05000446/2017004 – Dated February 9, 2018

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000445, 05000446
License:	NPF-87, NPF-89
Report:	05000445/2017004 and 05000446/2017004
Licensee:	Vistra Operations Company, LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	6322 N. FM-56, Glen Rose, Texas
Dates:	October 1 through December 31, 2017
Inspectors:	J. Josey, Senior Resident Inspector R. Kumana, Resident Inspector J. Drake, Senior Reactor Inspector L. Carson, II, Senior Health Physicist S. Money, Health Physicist
Approved By:	Mark S. Haire Chief Project Branch A Division of Reactor Projects

Enclosure

SUMMARY

IR 05000445/2017004; 05000446/2017004; 10/01/2017 – 12/31/2017; Comanche Peak Nuclear Power Plant; Equipment Alignment; Inservice Inspection Activities; Maintenance Effectiveness; Maintenance Risk Assessments and Emergent Work Control; Operability Determinations and Functionality Assessments; Refueling and Other Outage Activities

The inspection activities described in this report were performed between October 1 and December 31, 2017, by the resident inspectors at Comanche Peak Nuclear Power Plant and inspectors from the NRC's Region IV office. Six findings of very low safety significance (Green) are documented in this report. All of these findings involved violations of NRC requirements. The significance of inspection findings is indicated by their color (i.e., Green, greater than Green, White, Yellow, or Red), determined using Inspection Manual Chapter 0609, "Significance Determination Process," dated April 29, 2015. Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas," dated December 4, 2014. Violations of NRC requirements are dispositioned in accordance with the NRC Enforcement Policy. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," dated July 2016.

Cornerstone: Initiating Events

- <u>Green</u>. The inspectors identified a non-cited violation of 10CFR50.55a(g)(4) involving the licensee's failure to perform pressure testing of all Class 1 pressure retaining components for Units 1 and 2, in accordance with the applicable edition of Section XI of the ASME Boiler and Pressure Vessel Code. Specifically, prior to November 1, 2017, the licensee failed to perform the required pressure test on those portions of the Class 1 pressure boundary between the first and second isolation valves in the injection and return path of safety-related systems for both units in accordance with ASME Code, Section XI, Article IWB-5000, "System Pressure Tests." This issue does not represent an immediate safety concern because the licensee performed an operability evaluation for Unit 2 which demonstrated a reasonable expectation of operability, and received an approved relief request prior to Unit 1 startup. This finding was entered into the licensee's corrective action program as Condition Reports CR-2017-010530 and CR-2017-011968.

  The inspectors determined that the licensee's failure to perform a system leakage test of all Class 1 pressure retaining components for each of the units was a performance deficiency. This finding was more than minor because it affected the initiating events cornerstone attribute of equipment reliability, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated June 19, 2012, Exhibit 1, "Initiating Events Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak-rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The finding has a problem identification and resolution cross-cutting aspect associated with evaluation. Although the licensee identified that they had failed to properly perform pressure testing on portions of various ASME Code Class 1 systems, they required prompting from the inspectors to thoroughly evaluate the issue to ensure that their resolution fully addressed ASME requirements and the need to obtain a relief request prior to restarting Unit 1 following the outage [P.2]. (Section 1R08.5)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," associated with the licensee's failure to take timely corrective actions for a condition adverse to quality. Specifically, since 2010 the licensee failed to take corrective actions for indications of degraded concrete in the service water intake structure. This issue does not represent an immediate safety concern because the licensee performed detailed engineering evaluations and tests to confirm the structural integrity of the intake structure, and will implement corrective actions to repair the affected concrete in the future. The licensee entered this issue into the corrective action program as Condition Report CR-2018-000088.

  The licensee's failure to take timely and adequate corrective actions to correct a condition adverse to quality was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to correct the degraded condition resulted in continued degradation with no evaluation to determine continued operability. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 4, "External Events Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because the finding is a deficiency affecting the design or qualification of a mitigating SSC, but the SSC maintained its operability. The finding has a human performance cross-cutting aspect associated with conservative bias, in that, the licensee failed to ensure that the individuals used decision making practices that emphasized prudent choices [H.14]. (Section 1R12)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, associated with the licensee's failure to accomplish activities affecting quality in accordance with documented procedures. Specifically, the licensee breached control room door E-40A, a tornado missile barrier, without following the requirements of standing order OSO-008. This issue does not represent an immediate safety concern because, when this was identified, the licensee took action to immediately close door E-40A. The licensee entered this issue into the corrective action program as Condition Report CR-2017-010661.

  The licensee's failure to follow the requirements of Standing Order OSO-008 when breaching door E-40A was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with protection against external factors attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because: (1) it was not a design deficiency; (2) it did not represent a loss of system and/or function; (3) it did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; (4) and it did not result in the loss of a high safety significant non-technical specification train. The finding has a problem identification and resolution cross-cutting aspect associated with resolution, in that the licensee failed to take adequate corrective actions from a previous issue with this door [P.3]. (Section 1R13)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to prescribe adequate procedures for closing the containment equipment hatch for both units. Specifically, the licensee's procedure failed to specify the correct bolting arrangement and torque for closure during emergency conditions. This issue does not represent an immediate safety concern because at the time of discovery rapid containment closure was not required, and the licensee implemented corrective actions to change the procedure to reflect the correct torque value and bolting pattern. The licensee entered this issue into the corrective action program as Condition Report CR-2017-011236.

  The licensee's failure to prescribe adequate procedures to perform quality related activities was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the inadequate procedure resulted in a potentially degraded containment barrier in the event of core damage during shutdown conditions. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," Exhibit 4, "Barrier Integrity Screening Questions," the inspectors determined the finding degraded the ability to close or isolate containment and required evaluation under Inspection Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," dated May 6, 2004. Using Large Early Release Frequency (LERF) type screening process, the inspectors determined the finding was a "Type B LERF" finding because the finding did not affect core damage frequency, and that a Phase 2 estimate was required because the containment equipment hatch was important to LERF in accordance with Table 6.1. The inspectors used Table 6.2 Phase 2 Risk Significance to determine the finding was of very low safety significance (Green) because it did not meet the threshold for a White finding for leakage from containment to the environment being greater than 100 percent containment volume per day through containment penetration seals, isolation valves or vent and purge systems. The finding was not assigned a cross-cutting aspect because the performance deficiency was not reflective of current performance. (Section 1R04)

- Green. Inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to follow the requirements of station procedure STI-606.01, Work Control Process, Revision 5. Specifically, the work order for emergent maintenance activities on Containment Access Corridor door S1-35G was not reviewed by operations for plant impacts prior to work commencing. As a result, S1-35G was blocked open with the Unit 1 containment personnel airlock and equipment hatch open, causing the Unit 2 primary plant ventilation system to be inoperable. This issue does not represent an immediate safety concern because the licensee took action to enter the required technical specification action statement and establish appropriate compensatory measures for the activity. The licensee entered this issue into the corrective action program as Condition Report CR-2017-011424.

The licensee's failure to follow procedural requirements and review work activities for plant impact when blocking open door S1-35G for emergent maintenance was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the SSC and barrier performance attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, with the Unit 1 containment personnel airlock and equipment hatch open and door S1-35G blocked open, the Unit 2 primary plant ventilation system could not perform its function. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 4, "External Events Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the auxiliary building secondary containment. The finding has a human performance cross-cutting aspect associated with team work, in that the licensee failed to ensure that work groups adequately communicate and coordinate their activities across organizational boundaries to ensure that nuclear safety is maintained [H.4]. (Section 1R15)

- <u>Green</u>. The inspectors identified a non-cited violation of 10 CFR 26.205 associated with the licensee's failure to calculate and control work hours of personnel subject to work hour controls. Specifically, the licensee failed to track work hours for personnel assigned to emergency containment equipment hatch closure duty, resulting in multiple violations of work hour requirements. This issue does not represent an immediate safety concern because at the time of discovery no personnel were in excess of work hour requirements, and the licensee took action to include the individuals in work hour controls during future outages. The licensee entered this issue into the corrective action program as Condition Report CR-2017-012922.

  The licensee's failure to control work hours for personnel subject to work hour controls was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to control work hours for personnel assigned to emergency containment hatch closure introduced a higher risk of human performance errors in maintaining containment integrity. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," Exhibit 4, "Barrier Integrity Screening Questions," the inspectors determined the finding was of very low safety significance because it did not degrade the ability to close or isolate containment and did not degrade the physical integrity of reactor containment. The finding was not assigned a cross-cutting aspect because the cause of the performance deficiency occurred more than three years prior and therefore is not reflective of current performance. (Section 1R20)

PLANT STATUS

Unit 1 began the inspection period at approximately 100 percent power. On October 8, 2017, Unit 1 was shut down for a planned refueling outage. Unit 1 returned to full power on November 11, 2017. Unit 1 operated at full power for the rest of the inspection period.

Unit 2 began the inspection period at approximately 100 percent power. On November 25, 2017, Unit 2 mas manually tripped due to a simultaneous loss of both main feed water pumps. Unit 2 returned to full power on December 3, 2017. Unit 2 operated at full power for the rest of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

On December 26, 2017, the inspectors completed an inspection of the station's readiness for seasonal extreme weather conditions. The inspectors reviewed the licensee's adverse weather procedures for cold weather and evaluated the licensee's implementation of these procedures. The inspectors verified that prior to the onset of cold weather, the licensee had corrected weather-related equipment deficiencies identified during the previous cold weather season.

The inspectors selected two risk-significant systems that were required to be protected from cold weather:

- emergency diesel generators
- fire protection

The inspectors reviewed the licensee's procedures and design information to ensure the systems or components would remain functional when challenged by adverse weather. The inspectors verified that operator actions described in the licensee's procedures were adequate to maintain readiness of these systems. The inspectors walked down portions of these systems to verify the physical condition of the adverse weather protection features.

These activities constituted one sample of readiness for seasonal adverse weather, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

On December 20, 2017, the inspectors completed an inspection of the station's readiness to cope with external flooding. After reviewing the licensee's flooding analysis, the inspectors chose two plant areas that were susceptible to flooding:

- safety chiller rooms
- uninterruptable power supply heating, ventilation, and air conditioning rooms

The inspectors reviewed plant design features and licensee procedures for coping with flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether credited operator actions could be successfully accomplished.

These activities constituted one sample of readiness to cope with external flooding, as defined in Inspection Procedure 71111.01.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walk-Down

a. Inspection Scope

The inspectors performed partial system walk-downs of the following risk-significant systems:

- October 15, 2017, Unit 1, containment hatches during refueling operations
- November 7, 2017, Unit 1, train A containment spray system following startup

The inspectors reviewed the licensee's procedures and system design information to determine the correct lineup for the systems. They visually verified that critical portions of the systems or trains were correctly aligned for the existing plant configuration.

These activities constituted two partial system walk-down samples as defined in Inspection Procedure 71111.04.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to prescribe adequate procedures for closing the containment equipment hatch for both units. Specifically, the licensee's procedure failed to specify the correct bolting arrangement and torque for closure during emergency conditions.

Description. During the Unit 1 refueling outage 1RF19, the inspectors observed the licensee's training on rapid containment equipment hatch closure in response to abnormal events. The inspectors also reviewed the licensee's procedures and supporting calculations. The inspectors noted that the engineering analysis used to demonstrate successful closure of containment in response to a refueling accident or loss of shutdown cooling made several assumptions to demonstrate adequate clamping force for the equipment hatch. The analysis assumed that 4 of the 16 bolts would be tightened, and that the 4 bolts selected would be 90 degrees apart. The analysis also assumed that the bolts would be tightened to a minimum preload value that the licensee defined as "snug tight." The inspectors noted that the procedure STI 600.01, "Inspecting Plant Equipment and Sensitive Equipment Controls," Revision 1, did not require the bolts to be tightened to "snug tight" and also did not require the four bolts to be spaced 90 degrees apart. Based on their observations of containment closure drills, the inspectors determined that the licensee was not selecting bolts 90 degrees apart. Instead, the bolts were approximately 67.5 degrees apart on the vertical axis and 112.5 degrees apart on the horizontal axis. The inspectors also determined that the personnel tightening the bolts were not verifying sufficient load on the bolts was being applied. The inspectors concluded that the licensee had not established adequate procedures for emergency closure of the containment equipment hatch.

The licensee entered this into the corrective action program as Condition Report CR-2017-011236. At the time the inspectors raised the concern, the time to boil on Unit 1 was sufficiently long that the licensee would have had time to properly close the hatch. The inspectors noted that earlier in the outage, the licensee would have improperly closed the hatch in the event of a loss of shutdown cooling. The licensee determined that the closure method used would be acceptable to ensure protection against a fuel handling accident.

The licensee performed an evaluation to determine whether the containment hatch would have prevented a release in the event of a loss of shutdown cooling. The licensee determined that there was sufficient margin in the bolt stress to ensure containment with the specific asymmetric spacing that was utilized during the outage.

However, the licensee was unable to demonstrate that the procedure would result in sufficient preload on the four bolts used for closure. The inspectors determined that the issue could have resulted in an inadequate seal in the event containment closure was required, but that the resulting leakage would be restricted to the gasket area. The licensee determined, assuming no preload on the bolts, that the worst case leakage would be bounded by 30 percent containment volume turnover per day.

Analysis. The licensee's failure to prescribe adequate procedures to perform quality related activities was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the inadequate procedure resulted in a potentially degraded containment barrier in the event of core damage during shutdown conditions. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of

Findings," Exhibit 4, "Barrier Integrity Screening Questions," the inspectors determined the finding degraded the ability to close or isolate containment and required evaluation under Inspection Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," dated May 6, 2004. Using the Large Early Release Frequency (LERF) type screening process, the inspectors determined that the issue was a "Type B LERF" finding because it did not affect core damage frequency, and that a Phase 2 estimate was required because the containment equipment hatch was important to LERF in accordance with Table 6.1. The inspectors used Table 6.2 Phase 2 Risk Significance to determine the finding was of very low safety significance (Green) because it did not meet the threshold for a White finding for leakage from containment to the environment because the estimated leakage was determined to be less than 100 percent containment volume turnover per day through containment penetration seals, isolation valves or vent and purge systems. The inspectors determined that the procedure used for containment hatch closure had not been updated in more than 3 years therefore the finding was not assigned a cross-cutting aspect because the performance deficiency was not reflective of current performance.

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances. Contrary to the above, from July 2015 through October 2017, the licensee failed to prescribe activities affecting quality by procedures of a type appropriate to the circumstances. This issue does not represent an immediate safety concern because, at the time of discovery, rapid containment closure was not required, and the licensee generated corrective actions to modify their procedures. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-011236, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-01; 05000446/2017004-01, Inadequate Procedure for Containment Closure)

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors evaluated the licensee's fire protection program for operational status and material condition. The inspectors focused their inspection on four plant areas important to safety:

- November 20, 2017, fire area EL62, Unit 1 stairwell
- November 20, 2017, fire area AF33, Unit 1 component cooling water pump 1-01
- November 22, 2017, fire area 2SH11, Unit 2 diesel generator day tank
- November 22, 2017, fire area SB2, Unit 1 containment spray pumps

For each area, the inspectors evaluated the fire plan against defined hazards and defense-in-depth features in the licensee's fire protection program. The inspectors evaluated control of transient combustibles and ignition sources, fire detection and suppression systems, manual firefighting equipment and capability, passive fire protection features, and compensatory measures for degraded conditions.

These activities constituted four quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On December 19, 2017, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, the inspectors chose two plant areas containing risk-significant structures, systems, and components that were susceptible to flooding:

- Unit 1, train A containment spray pump room
- Unit 1, train B containment spray pump room

The inspectors reviewed plant design features and licensee procedures for coping with internal flooding. The inspectors walked down the selected areas to inspect the design features, including the material condition of seals, drains, and flood barriers. The inspectors evaluated whether operator actions credited for flood mitigation could be successfully accomplished.

These activities constituted completion of one flood protection measures sample as defined in Inspection Procedure 71111.06.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

On November 17, 2017, the inspectors completed an inspection of the readiness and availability of risk-significant heat exchangers. The inspectors verified the licensee used the industry standard periodic maintenance method outlined in Electric Power Research Institute (EPRI) NP-7552 for the Unit 1 containment spray pump 1-01 lube oil cooler heat exchanger. Additionally, the inspectors walked down the Unit 1 containment spray pump 1-01 lube oil cooler heat exchanger to observe its performance and material condition and verified that the heat exchanger was correctly categorized under the Maintenance Rule and was receiving the required maintenance.

These activities constituted completion of one heat sink performance annual review sample, as defined in Inspection Procedure 71111.07.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

The activities described in subsections 1 through 4 below constitute completion of one inservice inspection sample, as defined in Inspection Procedure 71111.08.

.1 Non-destructive Examination Activities and Welding Activities

a. Inspection Scope

The inspectors directly observed the following nondestructive examinations:

SYSTEM	COMPONENT IDENTIFICATION	EXAMINATION TYPE
Safety Injection	TBX-1-4303-H3	Visual Test – 3
Safety Injection	TBX-1-4201-H3	Visual Test – 3
Safety Injection	TBX-1-4303-H3	Penetrant Test
Safety Injection	TBX-1-4201-H4	Penetrant Test
Steam Generator	TBX-1-3100-3-1	Ultrasonic Test
Pressurizer	TBX-1-2100-6	Ultrasonic Test
Pressurizer	TBX-1-2100-10	Magnetic Particle Test

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	COMPONENT IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant	TBX-1-4500-H1	Visual Test – 3
Safety Injection	TBX-1-4301-H1	Visual Test – 3
Safety Injection	TBX-1-4301-H2	Visual Test – 3
Residual Heat Removal	TBX-2-2530-H11-H1	Visual Test – 3
Component Cooling	CC-1-SB-006-H1	Visual Test – 3
Safety Injection	TBX-1-4306-H28	Visual Test – 3
Building and Structures	Equipment Hatch	Visual Test – 3
Reactor Coolant	TBX-1-1300A-Sup	Visual Test – 3
Reactor Coolant	TBX-1-4500-H1	Ultrasonic Test
Pressurizer	TBX-1-4504-1	Ultrasonic Test

During the review and observation of each examination, the inspectors observed that activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Code requirements and applicable procedures. The inspectors also reviewed the qualifications of nondestructive examination technicians performing inspections and determined they were current.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD PROCESS
Reactor Coolant	1CS-0016 FW-1-3A	Gas Tungsten Arc Welding
Reactor Coolant	1CS-0017 FW-1-3A	Gas Tungsten Arc Welding

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD PROCESS
Service Water	Weld 17-3	Gas Tungsten Arc Welding
Auxiliary Feedwater	Weld FW-44	Gas Tungsten Arc Welding
Auxiliary Feedwater	Weld FW-45	Gas Tungsten Arc Welding
Auxiliary Feedwater	Weld A	Gas Tungsten Arc Welding
Station Service Water	CP1-SWSRPL-03 Weld Repair of Sealing Surface	Gas Tungsten Arc Welding

The inspectors reviewed that the welding procedure specifications and the welders were properly qualified in accordance with ASME Code Section IX requirements. The inspectors also determined that essential variables were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities

The bare metal visual inspection of the reactor vessel upper head penetrations was not performed in this outage.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee’s implementation of its boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure STA-737, "Boric Acid Corrosion Detection and Evaluation," Revision 8. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components, and that engineering evaluations used corrosion rates applicable to the affected components and properly assessed the effects of corrosion induced wastage on structural or pressure boundary integrity. The inspectors observed that corrective actions taken were consistent with the ASME Code and 10 CFR 50, Appendix B requirements.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the steam generator tube eddy current examination scope and expansion criteria, and determined that these criteria met technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspectors also verified that the eddy current examination inspection scope included areas of degradations that were known to represent potential eddy current test challenges such as the top of tube sheet, tube support plates, and U-bends. The inspectors confirmed that no repairs were required at the time of the inspection.

Steam Generator Inspection

- The inspectors verified that the number and sizes of steam generator tube flaws/degradation identified were consistent with the licensee's previous outage operational assessment predictions.

- The inspectors verified that steam generator eddy current examination scope and expansion criteria met technical specification requirements.

- The inspectors verified that eddy current probes and equipment configurations used to acquire data from the steam generator tubes were qualified to detect the known/expected types of steam generator tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination of EPRI Document 1013706."

The inspectors reviewed the licensee's identification of the following tube degradation mechanisms:

- tube support plate wear
- foreign object wear

Secondary Side Inspections

- The inspectors reviewed the completed portions of secondary side inspection results and verified the licensee took or planned to take corrective actions in response to the observed degradation.

- The inspectors reviewed the licensee's planned actions in response to several foreign objects that had been identified on the secondary side of the steam generators.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed 25 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. From this review the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10CFR50.55a(g)(4) involving the licensee's failure to perform a system pressure test of all Class 1 pressure retaining components for Units 1 and 2, in accordance with the applicable edition of Section XI of the ASME Boiler and Pressure Vessel Code. Specifically, prior to November 1, 2017, the licensee failed to perform the required pressure test of the portions of the Class 1 pressure boundary between the first and second isolation valves in the injection and return path of safety-related systems for both units in accordance with ASME Code Section XI, Article IWB-5000 requirements.

Description. During an inservice inspection program review, the inspectors noted that the licensee had identified that they were not correctly performing system leakage tests of all Class 1 pressure retaining components for each of the units. The inspectors identified, through further review and discussion, that the licensee had not adequately addressed the implications of this for either unit. Unit 1 was in a refueling outage, and Unit 2 was operating at 100 percent reactor power. Specifically, the licensee was performing the system leakage tests for the portions of the Class 1 boundary between the first and second isolation valves in the injection and return path of interfacing safety-related systems either during shutdown activities or prior to reaching a pressure corresponding to 100 percent reactor power (2235 psig +/- 15 psig). Article IWB-5000, "System Pressure Tests," of Section XI of the ASME Code requires that all pressure retaining components be pressure tested via a system leakage test per IWB-5220, "System Leakage Test." The licensee is required by 10 CFR 50.55a(g)(4) to comply with the requirements imposed by Section XI of the ASME Code or request exemption from particular requirements via a relief request. The inspectors had additional concerns with the licensee's evaluation that a relief request was not needed and the extent of condition review that was assigned to the issue. In the condition report, the action initiated was, "Review 1RF19 VT-2 activities to ensure scheduling and plant conditions meet ASME code requirements."

The inspectors found that this issue had been previously identified by the licensee during a self-assessment evaluation in 2009. The evaluation stated in part, "Making preparations for end of interval Class 1 pressure test per IWB-5222(b) [Sub-article of IWB-5000]. The pressure test may meet the normal pressure testing requirements, but may not be sufficient to meet the requirements of the end of interval pressure test on Class 1 systems." The issue was entered into the licensee's corrective action program as Condition Report CR-2009-004987. The licensee incorrectly closed this issue with the determination that the system pressure tests met the ASME 10-year pressure test requirements.

The only corrective action the licensee had assigned to the current condition report was to perform a review of the 1RF19 VT2 pressure tests. Based on the nature of the performance deficiency, interviews with the program owner as to the cause of the performance deficiency, as well as previous instances where NRC inspectors or the licensee had found required ASME Code examinations and tests had been missed or incorrectly performed, the inspectors determined that this action was too narrowly focused. The licensee entered this issue into the corrective action program and submitted a relief request to allow the use of ASME Code Cases N-798 and N-800 to restore compliance with regulatory requirements. The NRC granted verbal Relief Request 1/2B3-2 for the Unit 1 and Unit 2 Third Ten Year Inservice Inspection Interval on November 1, 2017.

Analysis. The inspectors determined that the licensee's failure to perform a system leakage test of all Class 1 pressure retaining components for each of the units was a performance deficiency. This finding was more than minor because it affected the Initiating Events Cornerstone attribute of equipment reliability, and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, Exhibit 1, "Initiating Events Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident, and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The finding has a problem identification and resolution cross-cutting aspect associated with evaluation. Although the licensee identified that they had failed to properly perform pressure testing on portions of various ASME Code Class 1 systems, they failed to thoroughly evaluate the issue to ensure that their resolution fully addressed ASME requirements and the need to obtain a relief request prior to restarting Unit 1 following the outage [P.2].

Enforcement. Title 10 CFR 50.55a(g)(4) requires that components classified as ASME Code Class 1, Class 2, and Class 3 meet the requirements set forth in Section XI of the applicable editions of the ASME Boiler and Pressure Vessel Code, and Addenda. Title 10 CFR 50.55a(g)(4)(ii) requires that inservice examination of components be conducted during successive 120-month inspection intervals, and comply with the requirements of the latest edition and addenda of the Code applicable to the specific interval. The AMSE Code, Section XI, Article IWB-5000 requires, in part, that pressure retaining components shall be tested with a system leakage test conducted at a pressure not less than the pressure corresponding to 100 percent reactor power prior to startup at the end of each refueling outage, and at or near the end of each 10-year interval. Contrary to these requirements, prior to November 1, 2017, the licensee failed to perform the required system leakage test at a pressure not less than the pressure corresponding to 100 percent reactor power on all Class 1 pressure retaining components for each of the units prior to startup at the end of each refueling outage, and at or near the end of each 10-year interval. Each unit is currently in its third 10-year interval. Because this finding is of very low safety significance, a verbal relief request was granted, and this was entered into the corrective action program as Condition Reports CR-2017-010530 and CR-2017-011968, this violation is being treated as a NCV consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-02; 05000446/2017004-02, Failure to Perform Class 1 Piping Pressure Tests in Accordance with ASME Code)

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On November 21, 2017, the inspectors observed an evaluated simulator scenario performed by an operating crew. The inspectors assessed the performance of the operators and the evaluators' critique of their performance.

These activities constituted completion of one quarterly licensed operator requalification program sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Review of Licensed Operator Performance

a. Inspection Scope

Inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity and risk. The inspectors observed the operators' performance of the following activities:

- October 8, 2017, shutting down Unit 1 for refueling
- October 31, 2017, draining Unit 1 reactor vessel down to mid-loop conditions

In addition, the inspectors assessed the operators' adherence to plant procedures, including conduct of operations procedure and other operations department policies.

These activities constituted completion of one quarterly licensed operator performance sample, as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed four instances of degraded performance or condition of safety-significant structures, systems, and components (SSCs):

- November 17, 2017, Unit 1 containment spray system, full system review
- October 25, 2017, service water intake structure, delamination of concrete
- December 13, 2017, Units 1 and 2 Maintenance Rule (a)(3) assessment
- December 20, 2017, common instrument air system, compressor air leaks

The inspectors reviewed the extent of condition of possible common cause SSC failures and evaluated the adequacy of the licensee's corrective actions. The inspectors reviewed the licensee's work practices to evaluate whether these may have played a role in the degradation of the SSCs. The inspectors assessed the licensee's characterization of the degradation in accordance with 10 CFR 50.65 (the Maintenance Rule), and verified that the licensee was appropriately tracking degraded performance and conditions in accordance with the Maintenance Rule.

These activities constituted completion of four maintenance effectiveness samples, as defined in Inspection Procedure 71111.12.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," associated with the licensee's failure to take timely corrective actions for a condition adverse to quality. Specifically, the licensee failed to take corrective actions for indications of degraded concrete in the service water intake structure.

Description. On October 17, 2017, while touring the service water intake structure pump bay, the inspectors noted concrete spalling resulting in large pieces of concrete falling onto the walkway. The inspectors also observed exposed and corroded rebar, and additional rust blooms in the overhead concrete. The inspectors found that the licensee had been aware of indications of concrete degradation since December 2010. The inspectors reviewed the structural monitoring program for the service water intake structure and determined that the licensee had documented indications of increasingly degraded concrete over many inspections, but had not taken corrective actions. The licensee had generated work orders to address the spalling, but did not complete the planned work. Prior to the inspectors' tour, the inspections had last been completed in June 2017.

The inspectors determined that a long term degraded condition had existed since at least 2010 resulting in corrosion of rebar in the concrete and ultimately resulting in significant spalling. The licensee determined that with the existing spalling sufficient concrete remained to maintain structural integrity. This issue does not represent an immediate safety concern because the licensee performed detailed engineering evaluations and tests to confirm the structural integrity of the intake structure, and will implement corrective actions to repair the affected concrete in the future. The licensee entered this issue into the corrective action program as Condition Report CR-2018-000088.

Analysis. The licensee's failure to take timely and adequate corrective actions to correct a condition adverse to quality was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to correct the condition resulted in a degraded condition with no evaluation to determine continued operability. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 4, "External Events Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because the finding is a deficiency affecting the design or qualification of a mitigating SSC, but the SSC maintained its operability. The finding has a human performance cross-cutting aspect associated with conservative bias, in that, the licensee failed to ensure that the individuals used decision making practices that emphasized prudent choices [H.14].

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, from December 2010 through October 2017, for quality related structures to which 10 CFR Part 50, Appendix B applies, the licensee failed to assure that conditions adverse to quality were promptly identified and corrected. This issue does not represent an immediate safety concern because the licensee performed detailed engineering evaluations and tests and determined that the structural integrity of the intake structure remained adequate. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2018-000088, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-03; 05000446/2017004-03, Failure to Promptly Correct a Condition Adverse to Quality)

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed four risk assessments performed by the licensee prior to changes in plant configuration and the risk management actions taken by the licensee in response to elevated risk:

- October 5, 2017, Unit 1, risk mitigation plan during fuel movement

- October 18, 2017, Unit 1 and 2, opening control room door E-40A
- November 22, 2017, Unit 2, diesel generator 2-01 fiber optic cable replacement
- December 1, 2017, modification of XST1 with safety chiller 2-05 out of service

The inspectors verified that these risk assessments were performed timely and in accordance with the requirements of 10 CFR 50.65 (the Maintenance Rule) and plant procedures. The inspectors reviewed the accuracy and completeness of the licensee's risk assessments and verified that the licensee implemented appropriate risk management actions based on the result of the assessments.

Additionally, on November 8, 2017, the inspectors also observed portions of one emergent work activity that had the potential to affect the functional capability of mitigating systems. The inspectors observed emergent work on the Unit 1, train B safety related battery. The inspectors verified that the licensee appropriately developed and followed a work plan for these activities. The inspectors verified that the licensee took precautions to minimize the impact of the work activities on unaffected SSCs.

These activities constituted completion of five maintenance risk assessments and emergent work control inspection samples, as defined in Inspection Procedure 71111.13.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings, associated with the licensee's failure to accomplish activities affecting quality in accordance with documented procedures. Specifically, the licensee breached control room door E-40A, a tornado missile barrier, without following the requirements of standing order OSO-008.

Description. Comanche Peak Inspection Report 2017002 documented finding NCV 05000445/2017002-04; 05000446/2017002-04, Failure to Adequately Assess Risk and Implement Risk Management Actions for Proposed Maintenance. The cause of this finding was the licensee's failure to implement preplanned actions contained in station procedure ODA-308, "LCO Tracking Program," Revision 16, section 13.7.39, "Tornado Missile Shields," when disabling a hazard barrier, door E-40A. As a result of this finding the licensee developed and issued Standing Order OSO-008, "Control of Control Room Door for Material Loadout," to proceduralize required actions when opening door E-40A for other than routine passage. Specifically, operators were directed to implement the requirements of ODA-308 section 3.7.10, "Control Room Emergency Filtration/Pressurization System," Section 13.7.39, "Tornado Missile Shields," and STA-759, "Control Room Envelop Habitability Program," Attachments 8.B and 8.C. Among other requirements, these references directed operators to station a dedicated operator in continuous communication at door E-40A when opened for other than routine passage.

While touring the control room on September 28, 2017, inspectors noted that door E-40A was open for maintenance activities with no one stationed at the door as required by Standing Order OSO-008. Inspectors learned that the unit supervisor was unaware that the door had been opened, and that the clearance processing center senior reactor operator had authorized opening of the door. Furthermore inspectors determined that the clearance processing center senior reactor operator had failed to ensure that the requirements of Standing Order OSO-008 were implemented prior to opening door E-40A. Subsequently, activities were stopped and door E-40A was shut. Condition Report CR-2017-010661 was generated to capture this issue in the station's corrective action program.

Analyses. The licensee's failure to follow the requirements of Standing Order OSO-008 when breaching door E-40A was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with protection against external factors attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because: (1) it was not a design deficiency; (2) it did not represent a loss of system and/or function; (3) it did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; (4) and it did not result in the loss of a high safety significant non-technical specification train. The finding has a problem identification and resolution cross-cutting aspect associated with resolution, in that the licensee failed to take adequate corrective actions from a previous issue with this door [P.3].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Standing Order OSO-008, "Control of Control Room Door for Material Loadout," an Appendix B quality related procedure, provides instructions for breaching the control room boundary thru door E-40A. Contrary to the above, on September 28, 2017, the licensee failed to follow the requirements of Standing Order OSO-008 when breaching door E-40A. Specifically, the licensee failed to implement the required compensatory measure prior to breaching a tornado missile barrier/control room envelop barrier. This issue does not represent an immediate safety concern because when this was identified the licensee took action to immediately close door E-40A. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-010661, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-04; 05000446/2017004-04, Failure to Follow Procedure When Breaching Control Room Envelope)

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed four operability determinations that the licensee performed for degraded or nonconforming SSCs:

- October 12, 2017, Unit 1, environmental qualification

- October 16, 2017, Unit 1 and 2, door maintenance on primary plant ventilation boundary door S1-35G

- November 2, 2017, Unit 1, bent reach rod on safety related spring can for steam generator 1-04 blowdown piping

- November 8, 2017, Unit 1, B safety related battery cell 41 jumpered out

The inspectors reviewed the timeliness and technical adequacy of the licensee's evaluations. Where the licensee determined the degraded SSC to be operable, the inspectors verified that the licensee's compensatory measures were appropriate to provide reasonable assurance of operability. The inspectors verified that the licensee had considered the effect of other degraded conditions on the operability of the degraded SSC.

These activities constituted completion of four operability and functionality review samples as defined in Inspection Procedure 71111.15.

b. Findings

Introduction. Inspectors identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with the licensee's failure to follow the requirements of station procedure STI-606.01, "Work Control Process," Revision 5. Specifically, the work order for emergent maintenance activities on containment access corridor door S1-35G was not reviewed by operations for plant impacts prior to work commencing. As a result, S1-35G was blocked open with the Unit 1 containment personnel airlock and equipment hatch open, causing the Unit 2 primary plant ventilation system to be inoperable.

Description. On October 16, 2017, Unit 1 was shutdown in a refueling outage with the containment personnel airlock and equipment hatch open. With Unit 1 in this configuration, containment access corridor door S1-35G is required to be shut so that the Unit 2 primary plant ventilation system boundary is maintained.

While touring the Unit 1 safeguards building, inspectors noted that door S1-35G was blocked open for maintenance activities. Inspectors questioned this configuration because having previously been in the control room inspectors knew that Unit 2 had not entered Technical Specification 3.7.12, "Primary Plant Ventilation System," action statement A. Inspectors contacted the Unit 2 unit supervisor and informed the operations crew of the condition of door S1-35G. The unit supervisor subsequently declared the primary plant ventilation system inoperable and entered the limiting condition for operation.

During subsequent review the licensee determined that Work Order 5508594 had not been reviewed by operations for plant impact. Specifically, station procedure STI-606.01 Sections 5.16.1 and 6.8.7.1 requires, in part, that operations personnel review maintenance or work activities which impact plant equipment to determine effects on equipment operability. This resulted in the failure to identify the need to declare Unit 2 primary plant ventilation inoperable when the door was blocked open. Furthermore, the licensee also determined that inadequate communications between the work week manager and the Unit 2 unit supervisor regarding the scope of the work to be performed on door S1-35G was a causal factor for this issue.

Analysis. The licensee's failure to follow procedural requirements and review work activities for plant impact when blocking open door S1-35G for emergent maintenance was a performance deficiency. This performance deficiency is more than minor, and therefore a finding, because it is associated with the SSC and barrier performance attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, with the Unit 1 containment personnel airlock and equipment hatch open and door S1-35G blocked open, the Unit 2 primary plant ventilation system could not perform its function. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 4, "External Events Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it only represented a degradation of the radiological barrier function provided for the auxiliary building secondary containment. The finding has a human performance cross-cutting aspect associated with team work, in that the licensee failed to ensure that work groups adequately communicate and coordinate their activities across organizational boundaries to ensure that nuclear safety is maintained [H.4].

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Station procedure STI-606.01, "Work Control Process," Revision 5, an Appendix B quality related procedure, provides instructions for planning maintenance activities on safety-related equipment. Contrary to the above, on October 16, 2017, an activity affecting quality was not accomplished in accordance with a procedure that was appropriate to the circumstances. Specifically, procedure STI-606.01, Sections 5.16.1 and 6.8.7.1 require, in part, that operations personnel review maintenance or work activities which impact plant equipment to determine effects on equipment operability. This issue does not represent an immediate safety concern because the licensee took action to enter the required technical specification action statement and establish appropriate compensatory measures for the activity. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-011424, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-05; 05000446/2017004-05, Failure to Declare Equipment Inoperable)

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed two temporary plant modifications that affected risk-significant SSCs:

- October 16, 2017, Unit 1, control room cabinets temporary modification for integrated system testing

- November 8, 2017, Unit 1, B train safety related battery cell 41 jumper

The inspectors verified that the licensee had installed and removed these temporary modifications in accordance with technically adequate design documents. The inspectors verified that these modifications did not adversely impact the operability or availability of affected SSCs. The inspectors reviewed design documentation and plant procedures affected by the modifications to verify the licensee maintained configuration control.

These activities constituted completion of two samples of temporary modifications, as defined in Inspection Procedure 71111.18.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed seven post-maintenance testing activities that affected risk-significant SSCs:

- September 26, 2017, Unit 1, containment spray pump 1-03 following maintenance

- October 25, 2017, Unit 2, steam generator 3 atmospheric relief valve following maintenance

- October 26, 2017, Unit 2, diesel generator 2-01 following fiber optic cable replacement

- October 30, 2017, Unit 1, component cooling water discharge cross-connect valve 1-HV-4515 following maintenance

- November 2, 2017, Unit 1, containment personnel airlock testing following limit switch adjustment

- November 17, 2017, Unit 1, auxiliary feedwater pump 1-01 following maintenance

- November 29, 2017, Unit 2, solid state protection system testing following replacement of switch S810

The inspectors reviewed licensing- and design-basis documents for the SSCs and the maintenance and post-maintenance test procedures. The inspectors observed the performance of the post-maintenance tests to verify that the licensee performed the tests in accordance with approved procedures, satisfied the established acceptance criteria, and restored the operability of the affected SSCs.

These activities constituted completion of seven post-maintenance testing inspection samples, as defined in Inspection Procedure 71111.19.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

During the station's Unit 1 refueling outage that concluded on November 6, 2017, the inspectors evaluated the licensee's outage activities. The inspectors verified that the licensee considered risk in developing and implementing the outage plan, appropriately managed personnel fatigue, and developed mitigation strategies for losses of key safety functions. This verification included the following:

- review of the licensee's outage plan prior to the outage
- review and verification of the licensee's fatigue management activities
- monitoring of shut-down and cool-down activities
- verification that the licensee maintained defense-in-depth during outage activities
- observation and review of reduced-inventory and mid-loop activities
- observation and review of fuel handling activities
- monitoring of heat-up and startup activities

These activities constituted completion of one refueling outage sample as defined in Inspection Procedure 71111.20.

On November 25, 2017, Unit 2 was manually tripped due to a simultaneous loss of both main feed water pumps, and the unit was not returned to full power until December 3, 2017. However, the unit shutdown and repair activities did not require the unit to cool down or a containment entry for a shutdown tour. Therefore, the conditions were not met requiring the inspectors to formally complete the applicable sample elements defined in Inspection Procedure 71111.20 for the unplanned outage, though the inspectors did review the licensee's response to the unit trip in accordance with Inspection Procedure 71153 (see Section 4OA3 below).

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 26.205 associated with the licensee's failure to calculate and control work hours of personnel subject to work hour controls. Specifically, the licensee failed to track work hours for personnel assigned to emergency containment equipment hatch closure duty, resulting in multiple violations of work hour requirements.

Description. During the Unit 1 refueling outage 1RF19, the inspectors reviewed the licensee work hour control program. The inspectors reviewed the work hour controls for personnel designated to perform rapid containment equipment hatch closure in response to abnormal events. The licensee uses an electronic database to track personnel work hours such that their work schedules do not violate the NRC's fatigue rule. The inspectors found that personnel designated for containment hatch closure and containment coordination were not included in the licensee's database, and subsequently raised this apparent discrepancy to licensee personnel.

The licensee reviewed procedure STA-615, "Fatigue Management and Staff Work Hours," and determined that the personnel met the criteria for covered workers and should have been included in their program for controlling fatigue. The licensee concluded that 31 personnel had been incorrectly excluded from the program. The licensee determined that, during the Unit 1 refueling outage, 21 of these personnel exceeded fatigue rule work hour limits during the outage without an evaluation or waiver, resulting in 56 total violations of work hour limits. The inspectors reviewed logs and condition reports and determined that the personnel's fatigue did not result in any adverse consequences, and that emergency containment closure had not been required to be implemented during the outage.

The licensee entered this issue into the corrective action program as Condition Report CR-2017-012922. Through subsequent reviews the inspectors determined that the licensee relied on a manager's interpretation documented in a previous review of outage personnel subject to work hour limits conducted more than three years ago as their bases for not including these workers. Therefore inspectors determined that the performance deficiency was not reflective of current performance.

Analyses. The licensee's failure to control work hours for personnel subject to work hour controls was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the human performance attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to control work hours for personnel assigned to emergency containment hatch closure introduced a higher risk of human performance errors in maintaining containment integrity. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated October 7, 2016, and Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," Exhibit 4, "Barrier Integrity Screening Questions," the inspectors determined the finding was of very low safety significance because it did not degrade the ability to close or isolate containment and did not degrade the physical integrity of reactor containment. The finding was not assigned a cross-cutting aspect because the performance deficiency was not reflective of current performance.

Enforcement. 10 CFR 26.205 requires, in part, that licensees shall calculate and control the work hours of individuals who are subject to that section. Contrary to the above, from October 8, 2017 through November 1, 2017, for individuals performing duties to which 10 CFR 26 applies, the licensee failed to assure that work hours were calculated and controlled. In response to this issue, the licensee developed actions to include the individuals assigned to emergency containment equipment hatch closure duty in work hour controls during future outages. This issue does not represent an immediate safety concern because at the time of discovery no personnel were in excess of work hour requirements, and the licensee took action to include the individuals in work hour controls during future outages. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-012922, this violation is being treated as a non-cited violation consistent with

Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017004-06; 05000446/2017004-06, Failure to Control Work Hours for Covered Personnel)

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed two risk-significant surveillance tests and reviewed test results to verify that these tests adequately demonstrated that the SSCs were capable of performing their safety functions:

Other surveillance tests:

- October 30, 2017, Unit 1, component cooling water discharge cross-connect valve 1-HV-4515 stroke testing

- October 26, 2017, Unit 1 containment sump inspection

The inspectors verified that these tests met technical specification requirements, that the licensee performed the tests in accordance with their procedures, and that the results of the test satisfied appropriate acceptance criteria. The inspectors verified that the licensee restored the operability of the affected SSCs following testing.

These activities constituted completion of two surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety and Occupational Radiation Safety

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors performed this portion of the attachment as a post-outage review. During the inspection the inspectors interviewed licensee personnel, reviewed licensee documents, and evaluated licensee performance in the following areas:

- Radiological work planning, including work activities of exposure significance, and radiological work planning ALARA evaluations, initial and revised exposure estimates, and exposure mitigation requirements. The inspectors also verified that the licensee's planning identified appropriate dose reduction techniques, reviewed any inconsistencies between intended and actual work activity doses, and determined if post-job (work activity) reviews were conducted to identify lessons learned.

- Verification of dose estimates and exposure tracking systems, including the basis for exposure estimates, and measures to track, trend, and, if necessary, reduce occupational doses for ongoing work activities. The inspectors evaluated the licensee's method for adjusting exposure estimates and reviewed the licensee's evaluations of inconsistent or incongruent results from the licensee's intended radiological outcomes.

- Problem identification and resolution for ALARA planning. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constitute completion of three of the five required samples of occupational ALARA planning and controls program, as defined in Inspection Procedure 71124.02, and completes the inspection.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

a. Inspection Scope

The inspectors evaluated the accuracy and operability of the licensee's personnel monitoring equipment, verified the accuracy and effectiveness of the licensee's methods for determining total effective dose equivalent, and verified that the licensee was appropriately monitoring occupational dose. The inspectors interviewed licensee personnel, walked down various portions of the plant, and reviewed licensee performance in the following areas:

- source term characterization, including characterization of radiation types and energies, hard to detect isotopes, and scaling factors

- external dosimetry including National Voluntary Laboratory Accreditation Program (NVLAP) accreditation, storage, issue, use, and processing of active and passive dosimeters

- internal dosimetry, including the licensee's use of whole body counting, use of in vitro bioassay methods, dose assessments based on airborne monitoring, and the adequacy of internal dose assessments

- special dosimetric situations, including declared pregnant workers, dosimeter placement and assessment of effective dose equivalent for external exposures (EDEX), shallow dose equivalent, and neutron dose assessment

- problem identification and resolution for occupational dose assessment, including review of audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution

These activities constitute completion of the five required samples of occupational dose assessment program, as defined in Inspection Procedure 71124.04.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index: Residual Heat Removal Systems (MS09)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of October 1, 2016, through September 30, 2017, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for residual heat removal systems for Units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index: Cooling Water Support Systems (MS10)

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of October 1, 2016, through September 30, 2017, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for cooling water support systems for Units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review

a. Inspection Scope

Throughout the inspection period, the inspectors performed daily reviews of items entered into the licensee's corrective action program and periodically attended the licensee's condition report screening meetings. The inspectors verified that licensee personnel were identifying problems at an appropriate threshold and entering these problems into the corrective action program for resolution. The inspectors verified that the licensee developed and implemented corrective actions commensurate with the significance of the problems identified. The inspectors also reviewed the licensee's problem identification and resolution activities during the performance of the other inspection activities documented in this report.

b. Findings

No findings were identified.

.2 Annual Follow-up of Selected Issues

a. Inspection Scope

The inspectors selected one issue for an in-depth follow-up:

- The inspectors reviewed the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused on an issue associated with incorrect shaft sizing for the turbine driven auxiliary feedwater pumps documented in Condition Report CR-2017-004821. The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and compensatory actions. The inspectors verified that the licensee appropriately prioritized the planned corrective actions and that these actions were adequate to correct the condition.

These activities constituted completion of one annual follow-up sample as defined in Inspection Procedure 71152.

b. Findings

No findings were identified.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

.1 Manual Reactor Trip Following A Loss of Main Feedwater

a. Inspection Scope

On November 25, 2017, during normal operations at 100 percent reactor power, the on shift operators of Comanche Peak Unit 2 inserted a manual reactor trip following a loss of all main feedwater due to a simultaneous loss of both main feedwater pumps. The plant responded with no complications. The operators placed the unit in Mode 3 with steam dumps and the auxiliary feedwater system in operation.

The inspectors evaluated operator performance, and reviewed operator logs, and plant computer data. Inspectors also reviewed the licensee's evaluation of the system response and cause of the trip, as well as the licensee's immediate corrective actions. Inspectors did not identify a performance deficiency during the initial plant response to the event.

b. Findings

No findings were identified.

These activities constituted completion of one event follow-up sample, as defined in Inspection Procedure 71153.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 25, 2017, the inspectors presented the inservice inspection results to Mr. T. McCool, Site Vice President, and other members of the licensee staff. The inspectors re-exited these inspection results to Mr. Jack Hicks, Regulatory Affairs, and other members of the licensee staff on December 11, 2017. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On December 7, 2017, the inspectors presented the radiation safety inspection results to Mr. K. Peters, Senior Vice President and Chief Nuclear Officer, T. McCool, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

On January 3, 2018, the inspectors presented the overall resident inspection results to Mr. Ken Peters, Senior Vice President and Chief Nuclear Officer, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspectors had been returned or destroyed.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

K. Peters, Senior Vice President and Chief Nuclear Officer
T. McCool, Site Vice President
S. Dixon, Consulting License Analyst, Regulatory Affairs
J. Dreyfuss, Manager, Plant
R. Garcia, Supervisor, Radiation Protection.
J. Goodrich, Supervisor, Radiation Protection.
J. Gumnick, Manager, Radiation Protection
R. Knapp, Supervisor, Radiation Protection
T. Hope, Manager, Regulatory Affairs
K. Mandrell, Supervisor, Radiation Protection.
E. McGurk, Supervisor, Radiation Protection
G. Merka, License Analyst, Regulatory Affairs
L. Windham, Manager, Corrective Action Program
D. Davis, Manager, Organizational Development
R. Deppi, Director, Project Engineering and Support
D. Goodwin, Director, Work Management
T. Hope, Manager, Regulatory Affairs
J. Hull, Manager, Nuclear Emergency Preparedness
A. Marzloft, Director, Nuclear Oversight
K. Peters, Senior Vice President and Chief Nuclear Officer
D. Volkening, Audit Manager, Nuclear Oversight
S. Sewell, Director, Engineering and Regulatory Affairs
J. Taylor, Director, Site Engineering
C. Tran, Manager, Engineering Programs

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000445/2017004-01 05000446/2017004-01	NCV	Inadequate Procedure for Containment Closure
05000445/2017004-02 05000446/2017004-02	NCV	Failure to Perform Class 1 Piping Pressure Tests in Accordance with ASME Code
05000445/2017004-03 05000446/2017004-03	NCV	Failure to Promptly Correct a Condition Adverse to Quality
05000445/2017004-04 05000446/2017004-04	NCV	Failure to Follow Procedure When Breaching Control Room Envelope
05000445/2017004-05 05000446/2017004-05	NCV	Failure to Declare Equipment Inoperable
05000445/2017004-06 05000446/2017004-06	NCV	Failure to Control Work Hours for Covered Personnel

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
MSM-G0-9827	Equipment Hatch Emergency Closure	1
STI-600.01	Protecting Plant Equipment and Sensitive Equipment Controls	1

Condition Reports

TR-2017-011749 CR-2017-011445 CR-2017-011444 CR-2017-011955 CR-2017-011441 CR-2017-011236

Calculations

Number	Title	Revision
MM-90-2671	Technical Evaluation for Equipment Hatch	11/28/90

Section 1R08: Inservice Inspection Activities

Procedures

Number	Title	Revision
STA-737	Boric Acid Corrosion Detection and Evaluation	8
STA-690	Erecting and Control of Scaffolding	4
STI-737.01	Boric Acid Corrosion Detection and Evaluation	0
STI-211.04	Fall Protection	0
CP-201	Comanche Peak Nuclear Power Plant Welding Procedure Specification	10
CP-202	Comanche Peak Nuclear Power Plant Welding Procedure Specification	11
CP-301	Comanche Peak Nuclear Power Plant Welding Procedure Specification	11
EPG-703	Inservice Inspection Program	

Procedures

Number	Title	Revision
EPG-725	Pressure Testing	
NDE 2.01	Liquid Penetrant Examination	8
NDE 2.02	ASME Section XI Liquid Penetrant Examination	8
NDE 3.01	Magnetic Particle Examination	6
NDE 3.02	ASME Section XI Magnetic Particle Examination	6
NQA 3.09-8.61	Requirements for ASME and ANSI B31.1 Visual Weld Inspection	9
STA-421	Control of Issue Reports	21
STA-422	Corrective Action Program	34
STA-661	Non-Plant Equipment Storage and Use Inside Seismic Category I Structures	5
STA-728	Storage and Handling of Flammable/Combustible Material and Compressed/Cryogenic Gases	5
STI-422.01	Initiation of Issue Reports	0
STI-422.02	Issue Report Reviews	0
STI-422.01	Operability Determination and Functionality Assessment Program	4
TX-ISI-8	VT-1 and VT-3 Visual Examination Procedure	10
TX-ISI-11	Liquid Penetrant Examination for Comanche Peak Nuclear Power Plant	17
TX-ISI-70	Magnetic Particle Examination for Comanche Peak Nuclear Power Plant	14
TX-ISI-210	Ultrasonic Examination Procedure for Welds in Ferritic Steel Vessels	9
TX-ISI-IWE	Metal Containment Visual Examination	5

Miscellaneous

Number	Title	Date
	Comanche Peak Nuclear Power Plant 2016 Targeted Self-Assessment Plan for the In-service Inspection (ISI) Program	September 27, 2016
DEH	Rockwell Edward Forged Steel Univalve Globe Stop Valve	June 14, 1983
SA-2009-025	Inservice Inspection and Assessment Program	September 17, 2009
	Comanche Peak Nuclear Power Plant Self-Assessment Report	June 18, 2013
EVAL-2016-003	Equipment Reliability	March 2016

Condition Reports (CRs)

CR-2009-004987	CR-2013-005386	CR-2017-003610	CR-2017-004173	CR-2017-004286
CR-2017-004864	CR-2017-004903	CR-2017-004950	CR-2017-005055	CR-2017-005144
CR-2017-005148	CR-2017-005296	CR-2017-005321	CR-2017-005343	CR-2017-005418
CR-2017-005695	CR-2017-005952	CR-2017-006029	CR-2017-006046	CR-2017-006113
CR-2017-007940	CR-2017-008135	CR-2017-008491	CR-2017-008542	CR-2017-008715
CR-2017-010530	CR-2017-011530	CR-2017-011527	CR-2017-011536	CR-2017-011429
CR-2017-011526				

Section 1R12: Maintenance Effectiveness

Procedures

Number	Title	Revision
STI-744.09	Structural Monitoring Inspection Guide	0

Condition Reports

TR-2017-001543	CR-2010-011535	CR-2012-010932	CR-2017-011931	CR-2017-011560

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

Number	Title	Revision
STI-606.01	Work Control Process	5
STA-694	Station Verification Activities	8

Procedures

Number	Title	Revision
STA-602	Temporary Modifications and Transient Equipment Placement	18

Condition Reports

IR-2017-012531

Work Orders

5527486

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

Number	Title	Revision
STI-422.01	Operability Determinations and Functionality Assessment Program	2

Condition Reports

IR-2017-012531 CR-2017-012317 CR-2017-011424 CR-2017-10567

Section 1R20: Refueling and Other Outage Activities

Procedures

Number	Title	Revision
STI 604.04	Outage Safety Function Guide	2

Condition Reports

CR-2017-012922 CR-2017-012441 CR-2017-012439

Section 2RS2: Occupational ALARA Planning and Controls

Procedures

Number	Title	Revision
RPI-606	Radiation Work and General Access Permits	37
STA-656	Radiation Work Control	22
STA-657	ALARA Job Planning & Debriefing	19
STA-421	Control of Issue Reports	21

Procedures

Number	Title	Revision

Audits and Self-Assessments

Number	Title	Date
EVAL-2017-008	Work Management/Radiation Protection	July 13, 2017

Condition Reports

CR-2016-003856	CR-2016-004354	CR-2016-004423	CR-2016-004869
CR-2016-004680	CR-2016-006435	CR-2017-004991	

Radiological Work Permit ALARA Packages

Number	Title
20171405	1RF19 RCP Motor 1-04 Swap
20171600	1RF19 Westinghouse Refueling Activities
2017600	2RF15 Westinghouse Refueling Activities
20172300	2RF16 Steam Generator Secondary Side Activities
20172400	Steam Generator Work Primary Side

Miscellaneous Documents

Title	Date
1RF18 Outage ALARA Report	2017
2RF16 Outage ALARA Report	2017
Comanche Peak Annual ALARA Report	January 2017

Section 2RS4: Occupational Dose Assessment

Procedures

Number	Title	Revision
RPI-402	Personnel Decontamination	31
RPI-500	Bioassay Program	14
RPI-501	In Vitro Sampling	9
RPI-506	Calculation and Tracking of Personnel Exposures to Airborne Radioactive Material	10
RPI-507	Internal Dose Calculation	6
RPI-509	Personnel Dosimetry Program	16

Procedures

Number	Title	Revision
RPI-515	Neutron Dose Measurement and Recording	19
RPI-516	Dose Determination	33
RPI-528	Multiple Dosimetry Badging	14
RPI-614	Skin Dose Calculation	7
RPI-626	Alpha Monitoring Program	8
STA-650	General Health Physics Plan	8
STA-655	Exposure Monitoring Plan	22

Audits and Self-Assessments

Number	Title	Date
EVAL-2017-008	Work Management/Radiation Protection Audit	June 19, 2017

Condition Reports

CR-2016-004691	CR-2016-005139	CR-2017-005127	CR-2017-005352
CR-2017-005634	TR-2017-012115	CR-2017-013309	CR-2017-005634

Miscellaneous Documents

Number	Title	Date
100555-0	NVLAP Accreditation Certificate (Mirion)	July 1, 2017
2016	Dry Active Waste Sample Data	June 1, 2016
2016	Unit 1 RCS Filter Sample Data	January 12, 2016
2017	Unit 2 RCS Filter Sample Data	February 13, 2017

PAPERWORK REDUCTION ACT STATEMENT

This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, Control Number 31500011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid Office of Management and Budget control number.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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 http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Information Request
September 5, 2017
Notification of Inspection and Request for Information
Comanche Peak Unit 1
NRC Inspection Report 05000445/2017004

INSERVICE INSPECTION DOCUMENT REQUEST

Inspection Dates: October 16 – 27, 2017

Inspector: Jim Drake, Senior Reactor Inspector

A. Information Requested for the In-Office Preparation Week

The following information should be sent to the Region IV office in hard copy or electronic format (ims.certrec.com preferred), in care of Jim Drake, by October 2, 2017, to facilitate the selection of specific items that will be reviewed during the onsite inspection week. The inspector will select specific items from the information requested below and then request from your staff additional documents needed during the onsite inspection week (Section B of this enclosure). We ask that the specific items selected from the lists be available and ready for review on the first day of inspection. Please provide requested documentation electronically if possible. If requested documents are large and only hard copy formats are available, please inform the inspector(s), and provide subject documentation during the first day of the onsite inspection.

If you have any questions regarding this information request, please call the inspector as soon as possible.

On October 16, 2017, a reactor inspector from the Nuclear Regulatory Commission's (NRC) Region IV office will perform the baseline inservice inspection at Comanche Peak Unit 1, using NRC Inspection Procedure 71111.08, "Inservice Inspection Activities." Experience has shown that this inspection is a resource intensive inspection both for the NRC inspectors and your staff. The date of this inspection may change dependent on the outage schedule you provide. In order to minimize the impact to your onsite resources and to ensure a productive inspection, we have enclosed a request for documents needed for this inspection. These documents have been divided into two groups. The first group (Section A of the enclosure) identified information to be provided prior to the inspection to ensure that the inspectors are adequately prepared. The second group (Section B of the enclosure) identifies the information the inspectors will need upon arrival at the site. It is important that all of these documents are up to date and complete in order to minimize the number of additional documents requested during the preparation and/or the onsite portions of the inspection.

We have discussed the schedule for these inspection activities with your staff and understand that our regulatory contact for this inspection will be Mr. James Barnette of your licensing organization. The tentative inspection schedule is as follows:

Preparation week: October 9 - 13, 2017
Onsite weeks: October 16 – 27, 2017

Our inspection dates are subject to change based on your updated schedule of outage activities. If there are any questions about this inspection or the material requested, please contact the lead inspector Jim Drake at (817) 200-1558. (mail to: James.Drake@nrc.gov).

A.1 ISI/Welding Programs and Schedule Information

1. A detailed schedule (including preliminary dates) of:

   1.1. Nondestructive examinations planned for ASME Code Class Components performed as part of your ASME Section XI, risk informed (if applicable), and augmented inservice inspection programs during the upcoming outage.

   1.2. Examinations planned for Alloy 82/182/600 components that are not included in the Section XI scope (If applicable)

   1.3. Examinations planned as part of your boric acid corrosion control program (Mode 3 walkdowns, bolted connection walkdowns, etc.)

   1.4. Welding activities that are scheduled to be completed during the upcoming outage (ASME Class 1, 2, or 3 structures, systems, or components)

2. A copy of ASME Section XI Code Relief Requests and associated NRC safety evaluations applicable to the examinations identified above.

   2.1. A list of ASME Code Cases currently being used to include the system and/or component the Code Case is being applied to.

3. A list of nondestructive examination reports which have identified recordable or rejectable indications on any ASME Code Class components since the beginning of the last refueling outage. This should include the previous Section XI pressure test(s) conducted during start up and any evaluations associated with the results of the pressure tests.

4. A list including a brief description (e.g., system, code class, weld category, nondestructive examination performed) associated with the repair/replacement

activities of any ASME Code Class component since the beginning of the last outage and/or planned this refueling outage.

5. If reactor vessel weld examinations required by the ASME Code are scheduled to occur during the upcoming outage, provide a detailed description of the welds to be examined and the extent of the planned examination. Please also provide reference numbers for applicable procedures that will be used to conduct these examinations.

6. Copy of any 10 CFR Part 21 reports applicable to structures, systems, or components within the scope of Section XI of the ASME Code that have been identified since the beginning of the last refueling outage.

7. A list of any temporary noncode repairs in service (e.g., pinhole leaks).

8. Please provide copies of the most recent self-assessments for the inservice inspection, welding, and Alloy 600 programs.

9. Copy of the procedures for welding techniques, and NDE that will be used during the outage.

A.2 Boric Acid Corrosion Control Program

1. Copy of the procedures that govern the scope, equipment and implementation of the inspections required to identify boric acid leakage and the procedures for boric acid leakage/corrosion evaluation.

2. Please provide a list of leaks (including code class of the components) that have been identified since the last refueling outage and associated corrective action documentation. If during the last cycle, the unit was shut down, please provide documentation of containment walkdown inspections performed as part of the boric acid corrosion control program.

A.3 Steam Generator Tube Inspections (If applicable)

1. A detailed schedule of:

   - Steam generator tube inspection, data analyses, and repair activities for the upcoming outage (if occurring).

   - Steam generator secondary side inspection activities for the upcoming outage (if occurring).

2. Copy of SG history documentation given to vendors performing eddy current (ET) testing of the SGs during the upcoming outage.

3. Copy of procedure containing screening criteria used for selecting tubes for in-situ pressure testing and the procedure to be used for in-situ pressure testing.

4. Copy of previous outage SG tube operational assessment. Also include a copy of the following documents as they become available:

- Degradation assessment
- Condition monitoring assessment

5. Copy of the document defining the planned SG ET scope (e.g., 100 percent of unrepaired tubes with bobbin probe and 20 percent sample of hot leg expansion transition regions with rotating probe) and identify the scope expansion criteria, which will be applied. Also identify and describe any deviations in this scope or expansion criteria from the EPRI Guidelines.

6. Copy of the document describing the ET acquisition equipment to be applied including ET probe types. Also identify the extent of planned tube examination coverage with each probe type (e.g. rotating probe -0.080 inches, 0.115 inches pancake coils and mid-range +point coil applied at the top-of-tube-sheet plus 3 inches to minus 12 inches).

7. Identify and quantify any SG tube leakage experienced during the previous operating cycle. Also provide documentation identifying which SG was leaking and corrective actions completed and planned for this condition.

8. Copy of steam generator eddy current data analyst guidelines and site validated eddy current technique specification sheets. Additionally, please provide a copy of EPRI Appendix H, "Examination Technique Specification Sheets," qualification records.

9. Provide past history of the condition and issues pertaining to the secondary side of the steam generators (including items such as loose parts, fouling, top of tube sheet condition, crud removal amounts, etc.).

   Indicate where the primary, secondary, and resolution analyses are scheduled to take place.

A.4 Additional Information Related to All Inservice Inspection Activities

1. A list with a brief description of inservice inspection, and boric acid corrosion control program related issues (e.g., PVAR) entered into your corrective action program since the beginning of the last refueling outage. For example, a list based upon data base searches using key words related to piping such as: inservice inspection, ASME Code, Section XI, NDE, cracks, wear, thinning, leakage, rust, corrosion, boric acid, or errors in piping examinations.

2. Provide training (e.g. Scaffolding, Fall Protection, FME, Confined Space) if they are required for the activities described in A.1 through A.3.

3. Please provide names and phone numbers for the following program leads:

   Inservice inspection (examination, planning)
   Containment exams
   Reactor pressure vessel head exams
   Snubbers and supports
   Repair and replacement program
   Licensing

Site welding engineer
Boric acid corrosion control program
Steam generator inspection activities (site lead and vendor contact)

B. Information to be Provided Onsite to the Inspector at the Entrance Meeting (October 16, 2017):

B.1 Inservice Inspection / Welding Programs and Schedule Information

1. Updated schedules for inservice inspection/nondestructive examination activities, including planned welding activities, and schedule showing contingency repair plans, if available.

2. For ASME Code Class welds selected by the inspector from the lists provided from section A of this enclosure, please provide copies of the following documentation for each subject weld:

   - Weld data sheet (traveler).
   - Weld configuration and system location.
   - Applicable Code Edition and Addenda for weldment.
   - Applicable Code Edition and Addenda for welding procedures.
   - Applicable welding procedures used to fabricate the welds.
   - Copies of procedure qualification records (PQRs) supporting the weld procedures from B.1.b.v.
   - Copies of welder's performance qualification records (WPQ).
   - Copies of the nonconformance reports for the selected welds (If applicable).
   - Radiographs of the selected welds and access to equipment to allow viewing radiographs (if radiographic testing was performed).
   - Copies of the preservice examination records for the selected welds.
   - Readily accessible copies of nondestructive examination personnel qualifications records for reviewing.

3. For the inservice inspection related corrective action issues selected by the inspectors from section A of this enclosure, provide a copy of the corrective actions and supporting documentation.

4. For the nondestructive examination reports with relevant conditions on ASME Code Class components selected by the inspectors from Section A above, provide a copy of the examination records, examiner qualification records, and associated corrective action documents.

5. A copy of (or ready access to) most current revision of the inservice inspection program manual and plan for the current interval.

6. For the nondestructive examinations selected by the inspectors from section A of this enclosure, provide a copy of the nondestructive examination procedures used to perform the examinations (including calibration and flaw characterization/sizing procedures). For ultrasonic examination procedures qualified in accordance with ASME Code, Section XI, Appendix VIII, provide documentation supporting the procedure qualification (e.g. the EPRI performance demonstration qualification summary sheets). Also, include qualification documentation of the specific equipment to be used (e.g., ultrasonic unit, cables, and transducers including serial numbers) and nondestructive examination personnel qualification records.

B.2 Boric Acid Corrosion Control Program

1. Please provide boric acid walk down inspection results, an updated list of boric acid leaks identified so far this outage, associated corrective action documentation, and overall status of planned boric acid inspections.

2. Please provide any engineering evaluations completed for boric acid leaks identified since the end of the last refueling outage. Please include a status of corrective actions to repair and/or clean these boric acid leaks. Please identify specifically which known leaks, if any, have remained in service or will remain in service as active leaks.

B.3 Steam Generator Tube Inspections (If Being Inspected, otherwise N/A)

1. Copies of the Examination Technique Specification Sheets and associated justification for any revisions.

2. Please provide a copy of the eddy current testing procedures used to perform the steam generator tube inspections (specifically calibration and flaw characterization/sizing procedures, etc.).

3. Copy of the guidance to be followed if a loose part or foreign material is identified in the steam generators.

4. Identify the types of SG tube repair processes which will be implemented for defective SG tubes (including any NRC reviews/evaluations/approvals of this repair process). Provide the flaw depth sizing criteria to be applied for ET indications identified in the SG tubes.

5. Copy of documents describing actions to be taken if a new SG tube degradation mechanism is identified.

6. Provide procedures with guidance/instructions for identifying (e.g. physically locating the tubes that require plugging) and plugging SG tubes.

7. List of corrective action documents generated by the vendor and/or site with respect to steam generator inspection activities.

B.4 Codes and Standards

1. Ready access to (i.e., copies provided to the inspector(s) for use during the inspection at the onsite inspection location, or room number and location where available):

    - Applicable Editions of the ASME Code (Sections V, IX, and XI) for the inservice inspection program and the repair/replacement program.

2. Copy of the performance demonstration initiative (PDI) generic procedures with the latest applicable revisions that support site qualified ultrasonic examinations of piping welds and components (e.g., PDI-UT-1, PDI-UT-2, PDI-UT-3, PDI-UT-10, etc.).

3. Boric Acid Corrosion Guidebook Revision 1 – EPRI Technical Report 1000975.