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

August 10, 2017

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 05000445/2017002 and 05000446/2017002

Dear Mr. Peters:

On June 30, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Comanche Peak Nuclear Power Plant, Units 1 and 2. On July 11, 2017, the NRC inspectors discussed the results of this inspection with you and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented five findings of very low safety significance (Green) in this report. All of these findings involved violations of NRC requirements. Further, inspectors documented a licensee-identified violation which was determined to be of very low safety significance in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2.a of the NRC 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, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; 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 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 05000445/2017002 and 050446/2017002
w/ Attachments:
1.) Supplemental Information
2.) Document Request
3.) Licensee Event Report
Detailed Risk Evaluation

COMANCHE PEAK NUCLEAR POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000445/2017002 and 05000446/2017002 – DATED AUGUST 10, 2017

DISTRIBUTION
KKennedy, RA
SMorris, DRA
TPruett, DRP
AVegel, DRS
JClark, DRS
RLantz, DRP
JJosey, DRP
RKumana, DRP
MHaire, DRP
RAlexander, DRP
MKirk, DRP
TSullivan, DRP
SJanicki, DRP
DLackey, DRP
JBowen, RIV/OEDO
KFuller, RC
VDricks, ORA
JWeil, OCA
MWatford, NRR
AMoreno, RIV/OCA
BMaier, RSLO
THipschman, IPAT
EUribe, IPAT
MHerrera, DRMA
RIV ACES
ROP Reports

Electronic Distribution for Comanche Peak Nuclear Power Plant

ADAMS ACCESSION NUMBER: ML17223A116

■ SUNSI Review By: MHaire/dll	ADAMS ■ Yes ☐ No	■ Non-Sensitive ☐ Sensitive	■ Publicly Available ☐ Non-Publicly Available			
OFFICE	SRI:DRP/A	RI:DRP/A	SPE:DRP/A	BC:EB1	BC:EB2	BC:OB
NAME	JJosey	RKumana	RAlexander	TFarnholtz	GWerner	VGaddy
SIGNATURE	**/RA/**	**/RA/**	**/RA/**	**/RA/**	**/RA/** JMateychick for	**/RA/**
DATE	08/07/2017	08/04/2017	08/03/17	8/7/2017	08/08/2017	8/4/17
OFFICE	BC:PSB2	TL-IPAT	BC:DRP/A			
NAME	HGepford	THipschman	MHaire			
SIGNATURE	**/RA/**	**/RA/**	**/RA/**			
DATE	08/04/2017	08/03/2017	8/9/17			

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000445, 05000446
License:	NPF-87, NPF-89
Report:	05000445/2017002 and 05000446/2017002
Licensee:	Vistra Operations Company, LLC
Facility:	Comanche Peak Nuclear Power Plant, Units 1 and 2
Location:	6322 N. FM-56, Glen Rose, Texas
Dates:	April 1 through June 30, 2017
Inspectors:	J. Josey, Senior Resident Inspector R. Kumana, Resident Inspector S. Janicki, Project Engineer M. Chambers, Physical Security Inspector L. Carson II, Sr. Health Physicist J. O'Donnell, CHP, Health Physicist K. Clayton, Senior Operations Engineer I. Anchondo, Reactor Inspector
Approved By:	Mark S. Haire Chief, Project Branch A Division of Reactor Projects

SUMMARY

IR 05000445/2017002; 05000446/2017002; 04/01/2017 – 06/30/2017; Comanche Peak Nuclear Power Plant; Fire Protection; Fire Protection; Inservice Inspection Activities; Maintenance Effectiveness; Maintenance Risk Assessments and Emergent Work Control; Follow-up of Events and Notices of Enforcement Discretion

The inspection activities described in this report were performed between April 1 and June 30, 2017, by the resident inspectors on site and inspectors from the NRC's Region IV office. Five 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 (Green, White, Yellow, or Red), which is determined using Inspection Manual Chapter 0609, "Significance Determination Process." Their cross-cutting aspects are determined using Inspection Manual Chapter 0310, "Aspects within the Cross-Cutting Areas." 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."

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of Operating Licenses NPF-87 and NPF-89, License Condition 2.G, "Fire Protection Program," for the failure to control transient combustibles in accordance with the station's fire protection report. Specifically, Fire Protection Report, Revision 29, Section 5.3.8, "Fire Area EO – Control Room," includes Deviation 3c-1, "Control Room Missile Door," which requires, in part, that since the control room missile door in the west wall is not a 3-hour rated fire door, the area of the turbine deck within 100 feet of the door is to be void of combustibles. Contrary to this, the licensee allowed storage of combustible materials in this area without required compensatory measures. This issue does not represent an immediate safety concern because the licensee removed the combustible materials upon identification. The licensee entered this issue into corrective action program as Condition Report CR-2017-5564.

  The failure to control transient combustible material in accordance with the approved fire protection report is a performance deficiency. The performance deficiency was more than minor and therefore a finding because it was associated with the protection against external factors attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the introduction of transient combustible materials decreased the external event mitigation for fire prevention. Using NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," June 19, 2012, the inspectors determined that the finding pertained to a failure to adequately implement fire prevention and administrative controls for transient combustible materials. As a result, the inspectors were directed to Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," September 20, 2013. The inspectors evaluated the finding through Appendix F, Attachment 1, "Fire Protection Significance Determination Process Worksheet," September 20, 2013, and determined that the finding was of very low safety consequence (Green) because the Fire Prevention and Administrative Controls finding would not prevent the reactor from reaching and maintaining a safe shutdown condition. The finding has a problem identification and resolution cross-cutting aspect associated with resolution, in that, the licensee failed to take effective corrective actions to address issues in a timely manner.

Specifically, the licensee had previously identified this issue in Condition Report CR-2014010224 but had failed to take corrective actions to address it [P.3]. (Section 1R05)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” that occurred when the licensee failed on two occasions to perform an adequate operability determination associated with multiple safety-related pipe supports. Specifically, the operability determination of multiple carbon steel pipe support clamps exposed to boric acid and a bent sway strut pipe restraint lacked the engineering rigor necessary to provide a high degree of confidence to support the operability of the components. Subsequently, the inspectors concluded that the licensee established reasonable expectation for operability once engineering provided the control room with further analysis on the degraded conditions, and the new information was reviewed and accepted. This issue was entered into the licensee’s corrective action program as Condition Report CR-2017-05418.

  The licensee's failure to perform adequate operability determinations per plant procedures was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating System cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee: (1) failed to perform the required corrosion evaluation for a comparison of material wastage against design dimensions of the pipe support clamps; (2) failed to perform a visual inspection of the material condition of the pipe support clamps as required by the work order; (3) used non-seismic design tolerances for the qualification of a seismically qualified strut in the immediate operability determination; and (4) failed to consider that the bent condition of the strut occurred after the previously accepted visual examinations on the same pipe support. All these issues could have resulted in safety-related components failing to perform their specified safety function during accident conditions. 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 the finding: (1) it was not a design deficiency; (2) did not represent a loss of system and/or function; (3) 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 did not result in the loss of a high safety-significant non-technical specification train. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to adequately assess the degraded condition of the pipe supports in a complete and accurate manner to support a reasonable expectation of operability [P.1]. (Section 1R08)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” associated with the licensee’s failure to assure that design changes were subject to design control measures commensurate with those applied to the original design. Specifically, the licensee changed internal components for safety-related, steam generator atmospheric relief valve booster relays but failed to verify that these new components could withstand the environment created during a high energy line break. This issue does not represent an immediate safety concern because the licensee performed an operability determination which established a reasonable expectation for operability, and implemented corrective actions to replace the relays with qualified relays. The licensee entered this issue into the corrective action program for resolution as Condition Report CR-2017-006236.

The failure to ensure that changes to the facility were subject to design control measures commensurate with those applied to the original design was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated 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 the finding: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality, (2) did not represent a loss of system and/or function, (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their technical specification allowed outage time, and (4) does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant for greater than 24 hours in accordance with the licensee's maintenance rule program. The inspectors did not assign a cross-cutting aspect because the performance deficiency was not reflective of present performance. (Section 1R12)

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those structure, systems and components to which this appendix applies, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, from initial construction through March 2017, the licensee failed to fully incorporate applicable moderate energy line break design requirements for fire protection piping located in the vicinity of the station service water pumps, the latter which are needed to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition following a moderate energy line break. This issue does not represent an immediate safety concern because when the lines were identified the licensee took prompt action to isolate and depressurize them, and the licensee has implemented plant modifications. The licensee entered this issue into the corrective action program as Condition Report CR-2016-008147.

  The failure to incorporate applicable design requirements into specifications for moderate energy line break protection was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, from initial construction through March 2017, the licensee failed to fully incorporate applicable design requirements for components needed to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition following a moderate energy line break. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated July 1, 2012, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated

October 7, 2016, the inspectors determined the finding required a detailed risk evaluation because the finding involved a deficiency affecting the design and qualification of a mitigating structure, system, or component, and resulted in a loss of operability, and represented an actual loss of function of at least a single train for longer than its allowed outage time. A senior reactor analysts from Region IV performed a detailed risk evaluation and determined that the bounding increase in core damage frequency for this issue was 5.1E-8/year for Unit 1 and 2.9E-10/year for Unit 2, and was therefore of very low safety significance (Green). The inspectors did not assign a cross-cutting aspect because the performance deficiency was not reflective of present performance. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to adequately assess risk and implement required risk management actions for a planned maintenance activity. Specifically, the licensee failed to evaluate the risk and implement required risk management actions associated with disabling a hazard barrier and breeching the control room envelope when blocking open door E-40A. This issue did not represent an immediate safety concern because, at the time of identification, the licensee stopped the activity and secured the door. The licensee entered this issue into the corrective action program for resolution as Condition Report CR-2017-006019.

  The failure to adequately assess the risk and implement required risk management actions for proposed maintenance activities was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the associated objective to ensure physical design barriers protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," dated May 19, 2005, Flowchart 2, "Assessment of Risk Management Actions," the inspectors determined the need to calculate the risk deficit to determine the significance of this issue. A senior reactor analyst determined the finding to have very low safety significance (Green) based on combining the effects of the degradation of the radiological barrier and tornado missile barrier functions. The analyst performed a qualitative review of the screening criteria in Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," for the degradation of the radiological barrier function for the control room and considered the short exposure time (2.9E-5 years) and the Comanche Peak specific high winds frequency (3.0E-4/year) for the tornado missile barrier function of the control room to determine that the incremental core damage probability deficit and the incremental large early release probability deficit were less than 1E-6 and 1E-7, respectively. The finding has a human performance cross-cutting aspect associated with procedure adherence, in that operations personnel failed to follow procedures when allowing door E-40A to be opened. (Section 1R13)

PLANT STATUS

Unit 1 began the inspection period at approximately 100 percent power. On May 20, 2017, unit 1 reduced power to 68 percent for main turbine testing and returned to full power the same day. Unit 1 operated at full power for the rest of the inspection period.

Unit 2 began the inspection period at approximately 98 percent power. On April 2, 2017, Unit 2 was shut down for a planned refueling outage. Unit 2 returned to full power on May 10, 2017. On June 2, 2017, unit 2 lowered power to 73 percent due to high turbine generator temperatures. On June 5, unit 2 was shut down to repair the turbine generator and remained shut down 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 Summer Readiness for Offsite and Alternate AC Power Systems

a. Inspection Scope

On June 26, 2017, the inspectors completed an inspection of the station's off-site and alternate-ac power systems. The inspectors inspected the material condition of these systems, including transformers and other switchyard equipment to verify that plant features and procedures were appropriate for operation and continued availability of off-site and alternate-ac power systems. The inspectors reviewed outstanding work orders and open condition reports for these systems. The inspectors walked down the switchyard to observe the material condition of equipment providing off-site power sources.

These activities constituted one sample of summer readiness of off-site and alternate-ac power systems, 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:

- April 25, 2017, Unit 2, component cooling water heat exchanger 2-01
- May 19, 2017, Unit 2, containment spray train B

- June 14, 2017, Unit 1, turbine driven auxiliary feedwater pump 1-01

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 three partial system walk-down samples as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

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 five plant areas important to safety:

- April 5, 2017, Fire Area SG10a, Unit 1, emergency diesel generator (EDG) room train A
- April 24, 2017, Fire area EO65, Unit 1 and 2, control room
- April 25, 2017, Fire area TB105g, Unit 1, turbine deck
- April 25, 2017, Fire area TB205g, Unit 2, turbine deck
- May 1, 2017, Fire areas AA26, AA27, AE32, AF33, Unit 1 and 2, component cooling water pump rooms

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 five quarterly inspection samples, as defined in Inspection Procedure 71111.05.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of Operating Licenses NPF-87 and NPF-89, License Condition 2.G, "Fire Protection Program," for the failure to control transient combustibles in accordance with the station's fire protection report.

Description. On April 24, 2017, while touring the turbine deck, the inspectors noted a large amount of combustible material stored within the 100 foot combustible material exclusion area of door E-40A, control room to turbine deck. The inspectors determined that the licensee had performed and evaluation of this condition under EV-TR-2017-003925-1 and determined it to be acceptable. Inspectors questioned this evaluation because the station's Fire Protection Report, Revision 29, Section 5.3.8, "Fire Area EO – Control Room," contains Deviation 3c-1, "Control Room Missile Door," which requires, in part, that since the control room missile door in the west wall is not a 3-hour rated fire door, the area of the turbine deck within 100 feet of the door is to be void of combustibles.

The inspectors reviewed EV-TR-2017-003925-1 and noted that this evaluation documented that the combustible material exclusion area was created because a fire rating for door E-40A could not be found (Condition Report CR-2014-010224) and referenced Calculation 0210-063-0043, Maximum Permissible Fire Loading/Non-Rated Features Analysis, Revision 14, as a basis for a 3-hour fire rating for the door. Inspectors reviewed Condition Report CR-2014-010224 and noted that it had been generated because combustible materials had been stored within 100 feet of door E-40A without proper controls. Furthermore, this condition report identified that Deviation 3c-1 requires the area of the turbine deck within 100 feet of the door is to be void of combustible material since door E-40A is not a 3-hour rated fire door. Inspectors also reviewed Calculation 0210-063-0043 and determined that this calculation was not a design analyses and used judgement as a basis for a 3-hour fire rating on door E-40A, which was non-conservative.

Inspectors determined that the licensee had failed to implement the requirements of the station's approved Fire Protection Report when storing combustible materials within the combustible material exclusion zone without proper controls. Inspectors informed the licensee of their concern and the licensee initiated Condition Report CR-2017-005564 to capture this issue in the station's corrective action program. The licensee also removed all combustible material from the exclusion area.

During subsequent reviews inspectors noted that the licensee had also initiated Condition Report CR-2016-004166 because Nuclear Oversight had determined that the corrective actions for Condition Report CR-2014-010224 were not fully effective. Condition Report CR-2016-004166 was subsequently closed with no actions taken.

Analyses. The failure to control transient combustible material in accordance with the approved fire protection report is a performance deficiency. The performance deficiency was more than minor and therefore a finding because it was associated with the protection against external factors attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the introduction of transient combustible materials decreased the external event mitigation for fire prevention. Using NRC Inspection Manual Chapter 0609, Attachment 4, "Initial Characterization of Findings," June 19, 2012, the inspectors determined that the finding pertained to a failure to adequately implement fire prevention and administrative controls for transient combustible materials. As a result, the inspectors were directed to Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," September 20, 2013. The inspectors evaluated the finding through Appendix F, Attachment 1, “Fire Protection Significance Determination Process Worksheet,” September 20, 2013, and determined that the finding was of very low safety consequence (Green) because the Fire Prevention and Administrative Controls finding would not prevent the reactor from reaching and maintaining a safe shutdown condition. The finding has a problem identification and resolution cross-cutting aspect associated with resolution, in that, the licensee failed to take effective corrective actions to address issues in a timely manner. Specifically, the licensee had previously identified this issue in Condition Report CR-2014-010224 but had failed to take corrective actions to address it [P.3].

Enforcement. Comanche Peak Unit 1, Operating License NPF-87, Condition 2.G, “Fire Protection,” requires, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment 78 and as approved in the Safety Evaluation Report and its supplements through Supplement 24.

Comanche Peak Unit 2, Operating License NPF-89, Condition 2.G, “Fire Protection,” requires, in part, that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report through Amendment 87 and as approved in the Safety Evaluation Report and its supplements through Supplement 27.

The station’s approved fire protection program includes Fire Protection Report, Revision 29, Section 5.3.8, “Fire Area EO – Control Room,” which contains Deviation 3c-1, “Control Room Missile Door,” which requires, in part, that since the control room missile door in the west wall is not a 3-hour rated fire door, the area of the turbine deck within 100 feet of the door is to be void of combustibles. Contrary to the above, on April 24, 2017, the licensee failed to maintain the area around the control room missile door void of combustibles. Specifically, the licensee allowed storage of combustible materials within 100 feet of the control room missile door in the west wall without required compensatory measures for the deviation from the Fire Protection Report. This issue does not represent an immediate safety concern because upon identification, the licensee removed the combustible materials from the 100 foot exclusion area. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-005564, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017002-01; 05000446/2017002-01, Failure to Control Transient Combustible Material in Accordance with a Fire Protection Procedure)

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 inspector directly observed the following nondestructive examinations:

SYSTEM	COMPONENT IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant	Pressurizer Upper Heat-to-Shell (TCX-1-2100-5)	Ultrasonic
Reactor Coolant	Pressurizer Spay Nozzle to Vessel (TCX-1-2100-12)	Ultrasonic
Reactor Coolant	Pressurizer Safety Nozzle to Vessel (TCX-1-2100-13)	Ultrasonic
Reactor Vessel Head	Control Rod Drive Mechanism Penetrations 75, 76, 77, 78	Ultrasonic

The inspector reviewed records for the following nondestructive examinations:

SYSTEM	COMPONENT IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant	Pressurizer Longitude Weld (TCX-1-2100-9)	Ultrasonic
Reactor Coolant	Pressurizer Safety Nozzle to Vessel (TCX-1-2100-14, TCX-1-2100-15, TCX-1-2100-16)	Ultrasonic
Service Water	SW-1-132-046-A43R (Strut) (SW-1-AB-001-H1 during refueling outage RF5, RF12, and RF14)	Visual (VT-3)

During the review and observation of each examination, the inspector observed whether activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspector reviewed one indication that was previously examined, and observed that the licensee evaluated and accepted the indication in accordance with the ASME Code and/or an NRC approved alternative. The inspector also reviewed the qualifications of all nondestructive examination technicians performing the inspections to determine whether they were current.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspector reviewed the results of the licensee’s volumetric inspection of the reactor vessel head to determine whether the inspection met ASME Code Case N-729-1. The inspector also reviewed whether the required inspection coverage was achieved and whether limitations were properly recorded. The inspector reviewed whether the personnel performing the inspection were certified examiners to their respective nondestructive examination method.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

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

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspector reviewed the steam generator tube eddy current examination scope and expansion criteria to determine whether these criteria met technical specification requirements, EPRI guidelines, and commitments made to the NRC. The inspector also reviewed whether 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 inspector confirmed that no repairs were required at the time of the inspection.

Steam Generator Inspection

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

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

- circumferential primary water stress corrosion cracking (PWSCC) at bulge/over-expansion locations with the hot leg (HL) tubesheet
- circumferential PWSCC at the HL tubesheet expansion transition
- tube wear at anti-vibration bars, preheater baffle plates, and quatrefoil tube support plates
- tube wear due to loose parts

The inspector verified that the licensee's eddy current examination scope included the new degradation mechanism, fully enveloped the problem, and has taken appropriate corrective actions before plant start up. The licensee will now include circumferential primary water stress corrosion cracking as a new degradation mechanism at the multiple locations specified above.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed 14 condition reports concerning inservice inspection activities to evaluate whether the licensee implemented appropriate corrective actions for inservice inspection issues. From this review the inspector 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

Green. The inspector identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," that occurred when the licensee failed on two occasions to perform an adequate operability determination associated with safety-related pipe supports. Specifically, the operability determination of multiple carbon steel pipe support clamps exposed to boric acid and a bent sway strut pipe restraint, lacked the engineering rigor to provide a high degree of confidence to support the component's operability.

Description. Procedure STI-422.01, "Operability Determination and Functionality Assessment Program," institutes a definition for reasonable expectation of operability in Section 4.18. In particular, this definition establishes that "the supporting basis for the reasonable expectation of Technical Specification System, Structure, and Component (SSC) operability should provide a high degree of confidence that the SSC remains operable." This requirement is applicable during both immediate and prompt operability determinations per procedure Section 6.2.1, and 6.2.2, respectively.

On April 27, 2016, the licensee performed a visual inspection on an ASME Code, Class 3, sway strut pipe restraint in the service water system. The examination documented a slight bend along the strut resulting in a failed visual examination. The immediate operability determination in Condition Report CR-2016-03811, documented that "a slightly bent or bowed" strut was acceptable per Site Specification CPES-P-1079, "Specification Field Fabrication and Erection of Pipe Supports." An evaluation under the prompt operability determination documented that the bent condition was within the design calculation tolerances and that the strut restraint remained operable. Further discussions at the time of the inspection established that the licensee believed that the bent strut was part of the original construction of the pipe support.

The immediate operability determination referenced Step 4.1.2.9 of Site Specification CPES-P-1079, and established that a slight bend on the strut restraint was an acceptable condition for operation without further evaluation. Step 4.1.2.9 states:

> "Seismic Category None supports shall be installed within +/- 5 degrees from the angle indicated on the Design Drawing. Support rods shall be installed such that they do not exhibit slack. Slightly bent or bowed rods are acceptable provided they support the dead load of the pipe."

The inspector determined that "Seismic Category None" is a designation for nonsafety-related components that are not seismically qualified and are not required to have a quality assurance inspection. These requirements are specified in Table 5.1.1.1, "Pipe Support Design Document Classification Matrix," of Site Specification CPES-P-1079. However, the inspector verified that the strut was classified as safety-related and was seismically qualified per design documents. Furthermore, the inspector determined that the prior two visual examinations on the same strut that were performed as part of the ASME Section XI program had not identified an unacceptable condition such as a slight bend on the component. Therefore, the inspector determined that the bent condition did not exist prior to the failed visual examination and that the licensee had failed to consider these facts in their operability determination. Consequently, the licensee failed to establish a reasonable expectation of operability.

On May 4, 2016, the licensee identified rust particles under the insulation of the discharge line to the reactor coolant system in the Chemical and Volume Control System. Upon removal of the insulation, the affected components exhibited excessive discoloration in the form of corrosion products and dry boric acid. These components included three ASME Code, Class 3, carbon steel pipe support clamps. The licensee proceeded to clean the affected components under Work Order 5268838. A step included in the work order directed the licensee to perform a material condition inspection to look for obvious degradation such as pitting or corrosion. As a conclusion, the licensee determined that the pipe support clamps remained operable because the cross-sectional properties of the clamps with respect to membrane or bending strength remain unaffected.

The inspector questioned the level of technical details and assumptions provided in the operability determination evaluations. Specifically, the inspector noted that statements such as, "the inspected surfaces exhibited minor boric acid staining and material loss," and “with minimal material lost given the amount of corrosion product, the corrosion was of an intermittent nature,” were provided without quantifying the condition of the clamps.

The inspector determined that the boric acid evaluation performed per the boric acid corrosion control program had failed to take into account corrosion rates as required by Procedure STA-737.01, “Boric Acid Corrosion Detection and Evaluation,” Rev 0. Furthermore, the inspector concluded that the visual inspection per Work Order 5268838 had not been performed, but rather signed off by engineering per teleconference referencing the evaluation provided as part of the operability evaluation. The inspector concluded that the licensee had not provided the technical rigor required to demonstrate a reasonable expectation of operability as required by Section 6.2.1 and 6.2.2 of Procedure STI-422.01.

Analysis. The licensee's failure to perform adequate operability determinations per plant procedures was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating System cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee: (1) failed to perform the required corrosion evaluation for a comparison of material wastage against design dimensions of the pipe support clamps; (2) failed to perform a visual inspection of the material condition of the pipe support clamps as required by the work order; (3) used non-seismic design tolerances for the qualification of a seismically qualified strut in the immediate operability determination; and (4) failed to consider that the bent condition of the strut occurred after the previously accepted visual examinations on the same pipe support. All these issues could have resulted in safety-related components failing to perform their specified safety function during accident conditions. 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 the finding: (1) it was not a design deficiency; (2) did not represent a loss of system and/or function; (3) did not represent an actual loss of function of at least a single train for longer than its technical specification allowed outage time; and (4) did not result in the loss of a high safety-significant non-technical specification train. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with resolution because the licensee failed to adequately assess the degraded condition of both pipe supports in a complete and accurate manner to support a reasonable expectation of operability [P.1].

Enforcement. Title 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure STI-422.01, “Operability Determination and Functionality Assessment Program,” Section 4.18 institutes a definition for reasonable expectation of operability. In particular, this definition establishes that, “the supporting basis for the reasonable expectation of Technical Specification System, Structure, and Component (SSC) operability should provide a high degree of confidence that the SSC remains operable.” Contrary to the above, on April 27 and May 4, 2017, the licensee failed to accomplish activities affecting quality in accordance with the applicable procedure. Specifically, the licensee discovered multiple degraded conditions of safety-related pipe supports but failed to implement adequate actions that provided a reasonable expectation of operability as required by Procedure STI-422.01. Since the affected components were located in the operating Unit, the inspector concluded that the licensee had established reasonable expectation for operability once engineering had provided the control room with further analysis on the degraded conditions and the new information was reviewed and accepted. Because the violation was of very low safety significance and it was entered into the corrective action program as Condition Report CR-2017-005418, this violation is being treated as a non-cited violation consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000446/2017002-02, Inadequate Operability Evaluation for Safety Related Pipe Supports)

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

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On May 31, 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

On April 2, 2017, the inspectors observed the performance of on-shift licensed operators in the Unit 2 main control room. At the time of the observations, the unit was in a period of heightened activity due to performing a planned shutdown for refueling. The inspectors observed the operators' performance of the unit shutdown, and transition to shutdown cooling.

In addition, the inspectors assessed the operators' adherence to plant procedures, including the 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.

.3 Biennial Review

The licensed operator requalification program involves two training cycles that are conducted over a 2-year period. In the first cycle, the annual cycle, the operators are administered an operating test consisting of job performance measures and simulator scenarios. In the second part of the training cycle, the biennial cycle, operators are administered an operating test and a comprehensive written examination. For this cycle, the licensee is completing the second, or biennial cycle, which ends March 31, 2017. The licensee completed the operating tests in December 2016 and the inspectors documented the results of the operating test performance and content reviews in inspection report(s) 05000445/2016004 and 05000446/2016004.

On April 17, 2017, the licensee informed the inspectors of the completed biennial cycle results for Units 1 and 2 for both the written examinations and the operating tests:

- 11 of 13 crews passed the simulator portion of the operating test

- The 2 crews that were evaluated as unsatisfactory consisted of twelve operators and one individual failed the simulator scenarios as an individual (not tied to crew performance)

- 74 of 74 licensed operators passed the job performance measure portion of the operating test

- Out of 74 operators, 2 retired from the company in December 2016, and 71 of the remaining 72 licensed operators passed the written examination

The final failure count on any portion of the biennial exams was 14 operators. Using 74 operators as the total number of operators that took any portion of the exam, this resulted in an 18.9 percent failure rate. This is below the threshold for a finding (greater than 20 percent failure is a green finding) as described in Inspection Manual Chapter 0609, "Significance Determination Process," Appendix I, "Licensed Operator Requalification Significance Determination Process."

The inspectors also reviewed the written examinations for content quality, overlap, and remediation packages. The individuals that failed any portions of their operating tests and/or written examinations were remediated, retested, and passed their retake operating tests and/or written examinations prior to returning to shift.

The inspectors completed one inspection sample of the biennial licensed operator requalification program.

a. Inspection Scope

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness

a. Inspection Scope

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

- June 8, 2017, Unit 2 component cooling water system

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 one maintenance effectiveness sample, as defined in Inspection Procedure 71111.12.

b. Findings

No findings were identified.

.2 Quality Control

a. Inspection Scope

On April 25, 2017, the inspectors reviewed the licensee's quality control activities through a review of parts installed in the steam generator atmospheric relief valves that were purchased as commercial-grade parts but were dedicated prior to installation in a quality-grade application.

These activities constituted completion of one quality control sample, as defined in Inspection Procedure 71111.12.

b. Findings

Introduction. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to assure that design changes were subject to design control measures commensurate with those applied to the original design. Specifically, the licensee changed internal components for safety-related booster relays but failed to verify that these new components could withstand the environment created during a high energy line break.

Description. While reviewing a commercial grade dedication package for steam generator atmospheric relief valve booster relays inspectors noted that Condition Report CR-2017-004594 was written because of identified discrepancies with the currently installed relay. Specifically, the licensee had determined that in 2005 a design change had been performed which allowed new relay models to be installed in the plant, and these new model contained elastomers that were not qualified for the environmental conditions the relays could be exposed to under accident conditions. Inspectors determined that these changes to the facility were design changes that should have been subject to design control measures commensurate with those applied to the original design, but were not.

The inspectors reviewed the licensee's evaluation documented in Condition Report CR-2017-004594. While this evaluation identified that the relays were not qualified for the environment it focused only on procuring replacement relays. Based on this inspectors determined that the licensee's evaluation was inadequate. Specifically, the inspectors noted that (1) the operability evaluation performed by the licensee failed to establish a reasonable expectation of operability for all of the relays, and the licensee had not initiated a past operability to address the inoperable relay, (2) the licensee had no actions to correct the identified condition adverse to quality of a design change implemented without appropriate controls, and (3) the licensee had no actions in place to ensure that the inadequate relays were controlled as blocked stock to ensure they were not subsequently re-installed in the facility.

The inspectors informed the licensee of their concerns, and the licensee subsequently added actions to Condition Report CR-2017-004594 to address these issues. The additional actions included a revised operability evaluation completed by the licensee which adequately established a reasonable expectation for operability while the licensee procured and installed replacement booster relays which fully met the required environmental qualifications.

Analysis. The failure to ensure that changes to the facility were subject to design control measures commensurate with those applied to the original design was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and affected the associated 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 the finding: (1) was not a deficiency affecting the design and qualification of a mitigating structure, system, or component, and did not result in a loss of operability or functionality, (2) did not represent a loss of system and/or function, (3) did not represent an actual loss of function of at least a single train for longer than its allowed outage time, or two separate safety systems out-of-service for longer than their technical specification allowed outage time, and (4) does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant for greater than 24 hours in accordance with the licensee's maintenance rule program. The inspectors did not assign a cross-cutting aspect because the performance deficiency was not reflective of present performance (i.e., the design change was completed in 2005).

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that, design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, the volume booster relays in the Unit 1 and Unit 2 atmospheric relief valves, items that are safety-related and to which Appendix B requirements apply, did not have design changes subject to the same design control measures commensurate with those applied to the original design. Specifically, between August 2005 and April 25, 2017, the licensee implemented changes to the relays and failed to control critical materials inside of the relays. This issue does not represent an immediate safety concern because the licensee performed an operability determination which established a reasonable expectation for operability, and implemented corrective actions to replace the relays with qualified relays. Because this finding is of very low safety significance, and has been documented in the corrective action program as CR-2017-006236, this violation is being treated as an NCV consistent with Section 2.3.2.a of the NRC Enforcement Policy. (05000445/2017002-03; 05000446/2017002-03, Relays not Environmentally Qualified)

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

a. Inspection Scope

The inspectors reviewed five 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:

- March 27, 2017, Unit 2, refueling outage 2RF16 defense in depth plan
- April 17, 2017, Unit 2, equipment train A controls during orange risk window due to train B outage
- May 4, 2017, Unit 1 and Unit 2, controls in place when opening hazard barrier door E-40A
- May 31, 2017, Unit 2, sequencer maintenance during main turbine automatic voltage regulator testing
- June 26, 2017, Unit 1, risk management actions during containment spray maintenance window

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.

The inspectors also observed portions of the unit 2 polar crane troubleshooting during a reactor vessel head lift on April 8 and 9, 2017, an emergent work activity that had the potential to cause an initiating event.

The inspectors verified that the licensee appropriately developed and followed a work plan for this activity. The inspectors verified that the licensee took precautions to minimize the impact of the work activity on SSCs.

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

b. Findings

Introduction. The inspectors identified a non-cited violation of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the licensee's failure to adequately assess risk and implement required risk management actions for a planned maintenance activity. Specifically, the licensee failed to evaluate the risk and implement required risk management actions associated with disabling a hazard barrier and breeching the control room envelope when blocking open door E-40A.

Description. While touring the turbine deck on May 4, 2017, inspectors noted that door E-40A was blocked open by a pallet jack with no workers in the immediate area of the door. Inspectors questioned this because this door is a tornado missile boundary and part of the control room pressure boundary. Inspectors noted that site procedure ODA-308, LCO Tracking Program, Revision 16, section 13.7.39, Tornado Missile Shields, contains a preplanned risk assessment and required risk management actions associated with disabling this barrier. Specifically, for routine entry and exit, the person opening/closing the door is in administrative control of the door and for all other activities a dedicated individual stationed at the door in continuous communication with the control room who could rapidly shut the door is required.

Inspectors went to the control room and engaged the shift manager with their concerns about the current configuration of door E-40A. During their discussion they determined that the shift manager had authorized the opening of the door, but had not reviewed and implemented the required risk management actions specified in the risk assessment documented in procedure ODA-308. The shift manager subsequently directed that the activity be stopped, and door E-40A be shut. Condition Report CR-2017-006019 was generated to capture this issue in the station's corrective action program.

Analysis. The failure to adequately assess the risk and implement required risk management actions for proposed maintenance activities was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the associated objective to ensure physical design barriers protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," dated May 19, 2005, Flowchart 2, "Assessment of Risk Management Actions," the inspectors determined the need to calculate the risk deficit to determine the significance of this issue. A senior reactor analyst determined the finding to have very low safety significance (Green) based on combining the effects of the degradation of the radiological barrier and tornado missile barrier functions. The analyst performed a qualitative review of the screening criteria in Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," for the degradation of the radiological barrier function for the control room and considered the short exposure time (2.9E-5 years) and the Comanche Peak specific high winds frequency (3.0E-4/year) for the tornado missile barrier function of the control room to determine that the incremental core damage probability deficit and the incremental large early release probability deficit were less than 1E-6 and 1E-7, respectively. The finding has a human performance cross-cutting aspect associated with procedure adherence, in that operations personnel failed to follow procedures when allowing door E-40A to be opened [H.8].

Enforcement. Title 10 CFR 50.65(a)(4) states, in part, “Before performing maintenance activities (including, but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from proposed maintenance activities.” Contrary to the above, prior to performing maintenance activities, the licensee failed to manage the associated increase in risk from the proposed maintenance activity. Specifically, on May 4, 2017, the licensee failed to implement required risk management actions associated with disabling a hazard barrier and breeching the control room envelope when blocking open door E-40A. This issue did not represent an immediate safety concern because, at the time of identification, the licensee stopped the activity and secured the door. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2017-006019, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the NRC Enforcement Policy. (NCV 05000445/2017002-04; 05000446/2017002-04, Failure to Adequately Assess Risk and Implement Risk Management Actions for Proposed Maintenance.)

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

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

- April 7, 2017, Unit 2, CR-2017-004391 negative pressure boundary door degraded window
- April 13, 2017, Unit 2, CR-2017-004737 2-02 reactor coolant pump bolting issue
- April 20, 2017, Unit 1 and Unit 2, CR-2017-004594 steam generator atmospheric relief valve booster relays contain unqualified material for environmental qualification
- April 28, 2017, Unit 2, CR 2017-005384, residual heat removal pump comprehensive operability test flow oscillations
- May 4, 2017, Unit 2, CR-2017-005101, EDG turbocharger wall thickness below design minimum
- May 5, 2017, Unit 2, IR-2017-006042 turbine driven auxiliary feedwater pump exceeded response time acceptance criteria in surveillance procedure
- May 18, 2017, Unit 2, IR-2017-006454 oil leak on chiller X-05

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

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- May 8, 2017, Unit 2, train A main steam line isolation and response time testing following valve actuator adjustment
- May 9, 2017, Unit 2, train A slave relay testing
- May 11, 2017, Unit 2, turbine driven auxiliary feedwater pump run following maintenance
- May 19, 2017, Unit 2, diesel generator 2-01 post maintenance re-torques
- June 15, 2017, Unit 1, turbine driven auxiliary feedwater pump following replacement of sentinel valve

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 five 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 refueling outage that concluded on May 8, 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.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed seven 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:

In-service tests:

- April 17, 2017, Unit 2, residual heat removal pump 2-02 testing

Containment isolation valve surveillance tests:

- May 8, 2017, Unit 2, train A main steam line isolation and response time testing

Other surveillance tests:

- April 12, 2017, Unit 2, train B integrated test sequence testing
- April 18, 2017, Unit 2, reactor coolant pump flow transmitter testing
- April 19, 2017, Unit 2, turbine driven auxiliary feedwater pump testing
- May 19, 2017, Unit 2, train A containment spray pump testing
- May 22, 2017, Unit 2, train B reactor trip breaker testing

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 seven surveillance testing inspection samples, as defined in Inspection Procedure 71111.22.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

The inspectors evaluated the licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities. The inspectors assessed the licensee's implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures. During the inspection, the inspectors interviewed licensee personnel, walked down various areas in the plant, performed independent radiation dose rate measurements, and observed postings and physical controls. The inspectors reviewed licensee performance in the following areas:

- Radiological hazard assessment, including a review of the plant's radiological source terms and associated radiological hazards. The inspectors also reviewed the licensee's radiological survey program to determine whether radiological hazards were properly identified for routine and non-routine activities and assessed for changes in plant operations.

- Instructions to workers including radiation work permit requirements and restrictions, actions for electronic dosimeter alarms, changing radiological conditions, and radioactive material container labeling.

- Contamination and radioactive material control, including release of potentially contaminated material from the radiologically controlled area, radiological survey performance, radiation instrument sensitivities, material control and release criteria, and control and accountability of sealed radioactive sources.

- Radiological hazards control and work coverage. During walk-downs of the facility and job performance observations, the inspectors evaluated ambient radiological conditions, radiological postings, adequacy of radiological controls, radiation protection job coverage, and contamination controls. The inspectors also evaluated dosimetry selection and placement as well as the use of dosimetry in areas with significant dose rate gradients. The inspectors examined the licensee's controls for items stored in the spent fuel pool and evaluated airborne radioactivity controls and monitoring.

- High radiation area and very high radiation area controls. During plant walk-downs, the inspectors verified the adequacy of posting and physical controls, including areas of the plant with the potential to become risk-significant high radiation areas.

- Radiation worker performance and radiation protection technician proficiency with respect to radiation protection work requirements. The inspectors determined if workers were aware of significant radiological conditions in their workplace, radiation work permit controls/limits in place, and electronic dosimeter dose and dose rate set points. The inspectors observed radiation protection technician job performance, including the performance of radiation surveys.

- Problem identification and resolution for radiological hazard assessment and exposure controls. The inspectors reviewed audits, self-assessments, and corrective action program documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of the seven required samples of radiological hazard assessment and exposure control program, as defined in Inspection Procedure 71124.01.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope

The inspectors evaluated whether the licensee controlled in-plant airborne radioactivity concentrations consistent with as low as reasonably achievable principles and that the use of respiratory protection devices did not pose an undue risk to the wearer. During the inspection, the inspectors interviewed licensee personnel, walked down various areas in the plant, and reviewed licensee performance in the following areas:

- Engineering controls, including the use of permanent and temporary ventilation systems to control airborne radioactivity. The inspectors evaluated installed ventilation systems, including review of procedural guidance, verification of systems used during high-risk activities, and verification of airflow capacity, flow path, and filter/charcoal unit efficiencies. The inspectors also reviewed the use of temporary ventilation systems used to support work in contaminated areas, such as high-efficiency particulate air/charcoal negative pressure units. Additionally, the inspectors evaluated the licensee's airborne monitoring protocols, including verification that alarms and set points were appropriate.

- Use of respiratory protection devices, including an evaluation of the licensee's respiratory protection program for use, storage, maintenance, and quality assurance of National Institute for Occupational Safety and Health-certified equipment, air quality and quantity for supplied-air devices and self-contained breathing apparatus (SCBA) bottles, qualification and training of personnel, and user performance.

- Self-contained breathing apparatus for emergency use, including the licensee's capability for refilling and transporting SCBA bottles to and from the control room and operations support center during emergency conditions, hydrostatic testing of SCBA bottles, status of SCBA staged and ready for use in the plant including vision correction, mask sizes, etc., SCBA surveillance and maintenance records, and personnel qualification, training, and readiness.

- Problem identification and resolution for airborne radioactivity control and mitigation. The inspectors reviewed audits, self-assessments, and corrective action documents to verify problems were being identified and properly addressed for resolution.

These activities constituted completion of the four required samples of in-plant airborne radioactivity control and mitigation program, as defined in Inspection Procedure 71124.03.

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 Safety System Functional Failures (MS05)

a. Inspection Scope

For the period of April 1, 2016, through March 31, 2017, the inspectors reviewed licensee event reports (LERs), maintenance rule evaluations, and other records that could indicate whether safety system functional failures had occurred. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," Revision 3, to determine the accuracy of the data reported.

These activities constituted verification of the safety system functional failures performance indicator for units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

The inspectors reviewed the licensee's reactor coolant system chemistry sample analyses for the period of April 1, 2016, through March 31, 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 reactor coolant system specific activity performance indicator for units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Reactor Coolant System Total Leakage (BI02)

a. Inspection Scope

The inspectors reviewed the licensee's records of reactor coolant system total leakage for the period of April 1, 2016, through March 31, 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 reactor coolant system leakage performance indicator for units 1 and 2, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors verified that there were no unplanned exposures or losses of radiological control over locked high radiation areas and very high radiation areas during the period of June 1, 2016, to March 31, 2017. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 mrem. 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 occupational exposure control effectiveness performance indicator as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.5 Radiological Effluent Technical Specifications (RETS)/Offsite Dose Calculation Manual (ODCM) Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed corrective action program records for liquid or gaseous effluent releases that occurred between June 1, 2016, and March 31, 2017, and were reported to the NRC to verify the performance indicator 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 RETS/ODCM radiological effluent occurrences performance indicator 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 ambient air temperature limits for starting the station's risk significant alternate power diesel generators documented in Condition Reports CR-2016-001817 and CR-2017-002072. 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 (Closed) Licensee Event Report 05000445/2016-001-01, Unanalyzed Condition Involving Potential Moderate Energy Line Break

a. Inspection Scope

On September 13, 2016, based on initial observations by NRC inspectors, the licensee determined that pressurized fire protection piping in the service water intake structure was not properly shielded for moderate energy line break protection of service water components which resulted in inoperability of one train of service water for both Unit 1 and Unit 2.

During extent of condition walk downs conducted on October 6, 2016, October 10, 2016, November 17, 2016, December 5, 2016, and December 22, 2016, additional piping in the Unit 1 and Unit 2 safeguards and auxiliary buildings was found to not be shielded correctly as well, resulting in inoperability of one train of various safety related equipment for both units. The licensee determined the most likely cause of this event was that the methodology used to conduct the initial moderate energy line break walk downs was flawed and allowed some threats to be missed. The licensee's corrective actions include shielding the affected piping, performing a 100 percent walk down of rooms containing moderate energy line break piping identified for shielding, and revising the systems interaction program maintenance procedure.

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

b. Findings

Introduction. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," associated with the licensee's failure to assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those structure, systems and components to which this appendix applies, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, from initial construction through March 2017, the licensee failed to fully incorporate applicable design requirements for components needed to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition following a moderate energy line break.

Description. On September 13, 2016, inspectors performed walkdowns in the service water intake structure and identified a vertical run of unshielded, pressurized fire protection piping that appeared to pose a moderate energy line break threat to the service water pumps. Inspectors determined that in the event of a moderate energy line break crack along any portion of the unshielded piping, the resultant spray had the potential to impact the function of any one of the four service water pumps. However, only one train would have been affected during the event due to the physical configuration/separation relative to the source line and target pumps and/or associated motor control centers that support pump operation. Inspectors informed the licensee of their concern.

Engineering personnel performed a subsequent walkdown of the intake structure and determined that the identified piping was not correctly shielded and operability of the service water pumps was in question. The licensee took immediate action to isolate and depressurize the fire protection line in question which addressed the operability concern. The licensee entered this issue into the station corrective action program as Condition Report CR-2016-008147 for resolution.

Part of the licensee's actions was to perform extent of condition walkdowns for unshielded moderate energy piping in the safeguards building for Unit 1 and 2. During the extent of condition walk downs conducted on October 6, 2016, October 10, 2016, November 17, 2016, December 5, 2016, and December 22, 2016, additional piping in the Unit 1 and Unit 2 safeguards and auxiliary buildings was found to not be appropriately shielded against a moderate energy line break, resulting in the inoperability of various safety related equipment for both units.

- Unit 2 Train B 480 VAC motor control center 2EB2-1 (Unit 2 Train B emergency core cooling, battery charger, containment spray, and containment isolation valve equipment)

- Unit 1 Train B 480V MCC 1EB4-2, and Unit 1 Train B Distribution Panel 1ED2-2 (Unit 1 Train B safety-related pumps, panels, sequencer, and transformers)

- Unit 1 Train B 480V MCC 1 EB4-1 (Unit 1 Train B safety-related pumps, valves, fans, battery chargers, and transformers)

- Unit 2 Train B 480V MCC 2E134-1 (Unit 2 Train B safety-related pumps, valves, fans, battery chargers, and transformer)

- Unit 1, Train B 480V MCC 1E84-1 (Unit 1 Train B safety-related pumps, valves, fans, battery chargers, and transformers)

In each of these instances the licensee took prompt action to isolate and depressurize the identified moderate energy piping pending modification. The licensee subsequently determined that the most probable cause of the issue was the use of a flawed methodology during the initial moderate energy piping walkdowns conducted in 1989. The licensee reported this issue to NRC in Event Report 52239, and Licensee Event Report 16-002-00.

Analyses. The failure to incorporate applicable design requirements into specifications for moderate energy line break protection was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, from initial construction through March 2017, the licensee failed to fully incorporate applicable design requirements for components needed to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition following a moderate energy line break. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," dated July 1, 2012, and Inspection Manual Chapter 0609, Appendix A, "Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," dated October 7, 2016, the inspectors determined the finding required a detailed risk evaluation because the finding involved a deficiency affecting the design and qualification of a mitigating structure, system, or component, and resulted in a loss of operability, and represented an actual loss of function of at least a single train for longer than its allowed outage time. A senior reactor analysts from Region IV performed a detailed risk evaluation and determined that the bounding increase in core damage frequency for this issue was 5.1E-8/year for Unit 1 and 2.9E-10/year for Unit 2, and was therefore of very low safety significance (Green). Additional information is included in the detailed risk evaluation in Attachment 3 of this report. The inspectors did not assign a cross-cutting aspect because the performance deficiency was not reflective of present performance.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that, "measures shall be established to assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies, are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, measures established by the licensee did not assure that applicable regulatory requirements and the design bases, as defined in 10 CFR 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies, were correctly translated into specifications, drawings, procedures, and instructions. Specifically, from initial construction through March 2017, the licensee failed to fully incorporate applicable design requirements for components needed to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition following a moderate energy line break. This issue does not represent an immediate safety concern because when the lines were identified the licensee took prompt action to isolate and depressurize them, and the licensee has implemented plant modifications. Since this violation was of very low safety significance (Green) and has been entered into the corrective action program as Condition Report CR-2016-008147, this violation is being treated as a non-cited violation consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000445/2017002-05; 05000446/2017002-05, Failure to Translate Design Requirements Into the As Built Facility)

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 14, 2017, the inspectors presented the radiation safety inspection results to Mr. 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 April 21, 2017, the inspector presented the inservice inspection activities inspection results to Mr. S. Sewell, Director of Engineering and Regulatory Affairs, and other members of the licensee staff. The licensee acknowledged the issues presented. The licensee confirmed that any proprietary information reviewed by the inspector had been returned or destroyed.

On May 4, 2017, the inspectors presented the licensed operator requalification program inspection results to Mr. J. Ruby, Exam Lead, Licensed Operator Requalification Training, and other members of the licensee staff. The licensee representatives acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On March 27, 2017, the inspectors presented the resident inspector quarterly inspection results to Mr. K. 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

J. Barnette, Consultant, Licensing Technologist
A. Birdett, Engineer, Steam Generator
S. Dixon, Consulting License Analyst, Regulatory Affairs
J. Goodrich, Supervisor, Radiation Protection.
J. Gumnick, Manager, Radiation Protection
R. Knapp, Supervisor, Radiation Protection
T. Hope, Manager, Regulatory Affairs
J. Howard, Engineering, Inservice Inspection
T. McCool, Site Vice President
E. McGurk, Supervisor, Radiation Protection
K. Peters, Senior Vice President and Chief Nuclear Officer
J. Ruby, Exam Lead, Licensed Operator Requalification Training
S. Sewell, Director, Engineering and Regulatory Affairs
J. Taylor, Director, Site Engineering
C. Tran, Manager, Engineering Programs
G. Woods, Supervisor, Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000445/2017002-01 05000446/2017002-01	NCV	Failure to Control Transient Combustible Material in Accordance with a Fire Protection Procedure (Section 1R05)
05000446/2017002-02	NCV	Inadequate Operability Evaluation for Safety-related Pipe Supports (Section 1R08)
05000445/2017002-03 05000446/2017002-03	NCV	Relays not Environmentally Qualified (Section 1R12)
05000445/2017002-04 05000446/2017002-04	NCV	Failure to Adequately Assess Risk and Implement Risk Management Actions for Proposed Maintenance (Section 1R13)
05000445/2017002-05 05000446/2017002-05	NCV	Failure to Translate Design Requirements Into the As Built Facility (Section 4OA3)

Closed

05000445-2016-001-01	LER	Unanalyzed Condition Involving Potential Moderate Energy Line Break (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
STA-634	Extreme Temperature Equipment Protection Program	6

Section 1R04: Equipment Alignment

Miscellaneous Documents

Number	Title
	Guarded Equipment Management Sign Posting Log

Section 1R05: Fire Protection

Calculations

Number	Title	Revision
0210-063-0043	Maximum Permissible Fire Loading/Non-Rated Features Analysis	14

Condition Reports

CR-2017-003925	CR-2014-010224	CR-2016-004166	CR-2016-004167

Procedures

Number	Title	Revision
FPI-403	Fire Preplan Instruction Manual	5
STA-729	Control of Transient Combustibles, Ignition Sources and Fire Watches	11

Miscellaneous Documents

Number	Title
DBD-ME-002	

Section 1R08: Inservice Inspection Activities

Procedures

Number	Title	Revision/ Date
SG-SGMP-17-8	Comanche Peak 2RF16 (April 2017) Steam Generator Degradation Assessment	0

Procedures

Number	Title	Revision/ Date
STI-422.01	Operability Determination and Functionality Assessment Program	4
SG-CDME-08-28	Steam Generator Condition Monitoring and Operational Assessment for Comanche Peak Unit 2, April 2008 (2RF10)	July 17, 2008
MRS-SSP-3393	Eddy Current Data Analysis Guidelines for Comanche Peak Unit 2 D5 steam generators	0
STA-737	Boric Acid Corrosion Detection and Evaluation	8
MRS-TRC-2317	Use of Appendix H and I Qualified Techniques at Comanche Peak 2RF16 Steam Generator Inspection	0
SG-SGMP-14-9	Steam Generator Condition Monitoring and Operational Assessment for Comanche Peak Unit 2, April 2014 Outage (2RF14)	0
WDI-PJF-1316984-FSR-001	Reactor Vessel Head Examination Final Report April 2017	0
DBD-CS-018	Design Criteria for Pipe Stress and Pipe Supports	11
WDI-SSP-1326	Reactor Vessel Head Penetration Inspection Service Procedure for Comanche Peak Unit 2	0
WDI-STD-1040	Procedure for Ultrasonic Examination of Reactor Vessel Head Penetrations	14
TX-ISI-302	Ultrasonic Examination of Austenitic Piping Welds	5
TX-ISI-210	Ultrasonic Examination Procedure of Welds in Ferritic Steel Vessel	9
TX-ISI-8	VT-1 and VT-3 Visual Examination Procedure	9
CPES-P-1079	Specification Field Fabrication and Erection of Pipe Supports	11
	Reactor Vessel Closure Head Visual Examination	5

Drawings

Number	Title	Revision
MS-2-001-402-C72S	Large Bore Piping Support	CP-4
BRHL-SW-1-AB-001	Station Service Water	CP-5
BRP-SW-1-AB-001	Station Service Water	CP-1

Condition Reports

CR-2017-005066	CR-2017-005109	CR-2017-005222	CR-2016-004050	CR-2016-003811
CR-2016-005566	CR-2016-005600	CR-2016-007429	CR-2015-008795	CR-2015-009112
CR-2015-009115	CR-2015-009180	CR-2015-009181	CR-2015-009272	

Work Orders

5264718	5171029	5268838

Section 1R11: Licensed Operator Requalification Program and Licensed Operator Performance

Miscellaneous Documents

Number/Type	Title	Date
Written Exams	2017 Exam -Weeks 1-6	April 2017
Sample Plans	2017 Exam -Weeks 1-6	April 2017
ES-601-1	NRC Exam Security Agreement Form	March 2017

Condition Reports

IR-2017-003933	TR-2017-003859	TR-2017-002923	TR-2017-001656
TR-2017-000548	TR-2016-010799	TR-2016-000851	TR-2017-001568

Section 1R12: Maintenance Effectiveness

Miscellaneous Documents

Number	Title	Revision/ Date
	Technical Evaluation 94-00367-00-00	March 17, 1994
CJ7239-1	Dedication Plan for Fisher Volume Booster Fischer (CW) P/N CW2625-12-HT	1
Q1717.0	Nuclear Environmental Test Procedure for Fischer Volume Booster	1

Condition Reports

CR-2017-004594	CR-2005-000085	CR-2005-003263

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Condition Reports

CR-2017-006354 CR-2014-004903 CR-2011-010090 CR-2009-008296

Drawings

Number	Title	Revision
M2-0249	Flow Diagram Generator Primary Water	18

Miscellaneous Documents

Number	Title
EV-IR-2017-006354-1	Unit 2 ODMI for Primary Water Pump 5” line to vent line weld crack
EV-IR-2017-006354-2	Evaluation to install temporary tieback support for Primary Water
EV-IR-2017-006354-3	Maximum crack size before shutdown

Work Orders

5444458

Section 1R15: Operability Determinations and Functionality Assessments

Drawings

Number	
E2-0064	S2-0910-2-8010A

Procedures

Number	Title	Revision
OPT-214A	Diesel Generator operability test	22
STI-422.01	Operability determinations and functionality assessment program	2
STA-707	10CFR50.59 and 10CFR72.48 reviews	21
MSM-C0-3346	Emergency Diesel Engine Turbocharger Maintenance	6
INC-214	Installation of electrical conductor seal assemblies	0

Miscellaneous Documents

Number	Title	Revision/ Date
59SC-2017-000057-01-00	50.59 Screening – EDG G90 Elliot Turbocharger – Thin Wall Turbine Casing	April 17, 2017
VDRT-5427168	Diesel Generator Turbocharger Casing Minimal Wall Thickness	April 19, 2017
EVAL-2005-001433-01-00	Evaluation of EDG 2-01 Turbocharger wall thinning	April 7, 2005
VL-04-000971	Vendor Letter – G90 Elliot Turbocharger Thin Wall Turbine Case	April 6, 2004
FDA-2017-000057-01	U1/U2 Generic use-as-is disposition to address EDG Turbocharger wall thinning	0

Work Orders

5174039	5429766

Section 1R19: Post-Maintenance Testing

Calculations

Number	Title	Revision
ME-CA-00005476	Residual Heat Removal design performance limit for Inservice Testing	0

Condition Reports

CR-2017-005384

Procedures

Number	Title	Revision
OPT-203B	Residual Heat Removal System	14
OPT-206B	AFW System	22

Miscellaneous Documents

Number	Title	Date
EV-CR-2017-005384-1	RHR pump comprehensive pump test flow band evaluation	April 21, 2017

Work Orders

5171407

Section 1R20: Refueling and Other Outage Activities

Condition Reports

CR-2017-004725 CR-2017-004789 CR-2017-004811

Procedures

Number	Title	Revision
OPT-203B	Residual Heat Removal System	14
IPO-002B	Plant Startup from Hot Standby	10
NUC-301	Low power physics testing	21

Work Orders

5174286

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
STA-601	Authority for Equipment Operation	17

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures

Number	Title	Revision
RPI-115	Alarm Response	9
RPI-212	Radioactive Source Control	13
RPI-213	Survey and Release of Material and Personnel	26
RPI-400	Decontamination Program	21
RPI-509	Personnel Dosimetry Program	16
RPI-602	Radiological Surveillance and Posting	58
RPI-606	Radiation Work and General Access Permits	37
RPI-623	Radiological Briefings	10
RPI-626	Alpha Monitoring Program	8
RPI-700	Sealed Source Leak Testing	13
RPI-802	Performance of Source Checks	23
STA-650	General Health Physics Plan	8
STA-655	Exposure Monitoring Program	22

Procedures

Number	Title	Revision
STA-656	Radiation Work Control	22
STA-660	Control of High Radiation Areas	17

Audits and Self-Assessments

Number	Title	Date
CR-2016-005928	CPNPP Strategic Self-Assessment Report	October 16, 2016
CR-2016-005929	CPNPP Targeted Self-Assessment Report	September 15, 2016

Condition Reports

CR-2016-003921	CR-2016-004059	CR-2016-004387	CR-2016-004390	CR-2016-004806
CR-2016-004879	CR-2016-005048	CR-2016-006813	CR-2016-006813	CR-2017-000761
CR-2017-004224	CR-2017-004307	CR-2017-004787		

Radiation Work Permits

Number	Title	Revision
20172100	2RF16 RP Support in Containment	0
20172214	2RF16 Reactor Vessel Annulus BMI, Seal Table Activities, Eddy Current Testing and Containment Close-Out	0
20172300	2RF16 Secondary Side Steam Generator Activities	0
20172400	2RF16 Primary Side Steam Generator Activities	1
20172600	2RF16 Westinghouse (WEC) Refueling Activities	2

Radiation Surveys

Number	Title	Date
	U-2 RB 825’ S/G Platform #s 1, 2, & 3	April 13, 2017
	U-2 RB 825’ S/G Channel Head Survey Points (EPRI) Generator #s 1, 2, 3, & 4.	April 13, 2017
M-20161101-10	U-2 RB 808’ All Rooms 2-154 Quarterly Comprehensive	November 1, 2016
M-20170125-11	U-2 RB 808’ All Rooms 2-154 Quarterly Comprehensive	January 25, 2017
M-20170215-17	Aux 842’ Valve & Pipe Gallery X-230 Trending Routine	February 15, 2017
M-20170225-1	Aux 832’ Piping Area X-213 Bi-Weekly	February 25, 2017

Procedures

Number	Title	Revision
M-20170329-13	Aux 832' Piping Area X-213 Post Resin Transfer Flush Survey	March 29, 2017
M-20170330-19	Aux 842' Valve & Pipe Gallery X-230 Post 24 Hour Resin Transfer	March 30, 2017

Air Sampling Results

Number	Title	Date
12-Apr-2017-0005	U2 Equipment Hatch 832	April 12, 2017
12-Apr-2017-0037	Unit 2 LTDN HX A/S	April 12, 2017
13-Apr-2017-0006	Platform 1 Iodine	April 13, 2017
13-Apr-2017-0007	Platform 1 Particulate	April 13, 2017

Section 2RS3: In-plant Airborne Radioactivity Control and Mitigation

Procedures

Number	Title	Revision
ODA-102	Conduct of Operations	27
RPI-888	Calibration of Air Sampling Equipment	4
RPI-902	Issue and Control of Respiratory Protection	17
RPI-903	Cleaning, Decontamination, & Disinfecting of Respiratory Protection Equipment	14
RPI-904	Accountability & Inspection of Respiratory Protection Equipment (Maintenance & Repair)	13
SOP-817A	Safeguards Ventilation Systems	10
SOP-816	Primary Plant Ventilation Systems	13
SOP-801A	Containment Ventilation	14
SOP-802	Control Room Ventilation	13
STA-659	Respiratory Protection Program	17
STI-659.1	Use of Respiratory Protection	0
STI-704.1	Processing Respiratory Health Screens	1
STA-704	Respiratory Health Screen Program	17

Procedures

Number	Title	Revision
TRA-103	Respiratory Protection Training	11

Audits and Self-Assessments

Number	Title	Date
CR-2016-005928	CPNPP Strategic Self-Assessment Report	October 16, 2016
CR-2016-005929	CPNPP Targeted Self-Assessment Report	September 15, 2016
AI-TR-2017-000435	1 CRE habitability targeted self-assessment	January 27, 2017

Condition Reports

CR-2016-007561 CR-2016-010369 CR-2016-008843 CR-2016-002439 CR-2017-001125

CR-2016-006015

Miscellaneous Documents

Title	Date
SCBA Qualification Records	March 31, 2017
SCBA Qualification Records Ops & Fire Brigade	March 13, 2017
SCBA Inspection Records	December 16, 2016
SCBA Inspection Records	October 15, 2016
CPNPP Respirator Model Types	March 13, 2017

Section 4OA2: Problem Identification and Resolution

Condition Reports

CR-2016-001817 CR-2017-002072

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Condition Reports

CR-2016-008147

The following items are requested for the
Occupational Radiation Safety Inspection

Comanche Peak Nuclear Power Plant

Inspection Dates April 10 - 17, 2017
Integrated Report 2017002

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **April 3, 2017.**

Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for Inspection Procedure 71124.0**1** should be in a file/folder titled "**1**- A," applicable organization charts in file/folder "**1**- B," etc.

If information is placed on *ims.certrec.com*, please ensure the inspection exit date entered is at least 30 days later than the onsite inspection dates, so the inspectors will have access to the information while writing the report.

In addition to the corrective action document lists provided for each inspection procedure listed below, please provide updated lists of corrective action documents at the entrance meeting. The dates for these lists should range from the end dates of the original lists to the day of the entrance meeting.

If more than one inspection procedure is to be conducted and the information requests appear to be redundant, there is no need to provide duplicate copies. Enter a note explaining in which file the information can be found.

If you have any questions or comments, please contact **Louis C. Carson II at (817) 200-1221 or Louis.Carson@nrc.gov.**

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 3150-0011.

1. Radiological Hazard Assessment and Exposure Controls (71124.01) and Performance Indicator Verification (71151)
Date of Last Inspection: **May 13, 2016**

A. List of contacts and telephone numbers for the Radiation Protection Organization Staff and Technicians

B. Applicable organization charts

C. ALL radiation protection related licensee assessments and audits, all independent or third party radiation protection related assessments and audits, all radiation protection related self-assessments, and all radiation safety related LERs, including but not limited to radiation monitoring instrumentation and radioactive effluents, releases and / or spills, written since **May 2016**.

D. Procedure indexes for the radiation protection procedures

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. Radiation Protection Program Description
2. Radiation Protection Conduct of Operations
3. Personnel Dosimetry Program
4. Posting of Radiological Areas
5. High Radiation Area Controls
6. RCA Access Controls and Radiation Worker Instructions
7. Conduct of Radiological Surveys
8. Radioactive Source Inventory and Control
9. Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and sub-tiered systems) since **May 2016**.

a. Initiated by the radiation protection organization
b. Assigned to the radiation protection organization

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

If not covered above, a summary of corrective action documents since **May 2016** involving unmonitored releases, unplanned releases, or releases in which any dose limit or administrative dose limit was exceeded (for Public Radiation Safety Performance Indicator verification in accordance with IP 71151)

Additionally, a copy of **ALL** radiation protection AND chemistry department root cause evaluations, apparent cause evaluation, and condition evaluations performed **since May 2016.**

G. List of radiologically significant work activities scheduled to be conducted during the inspection period (If the inspection is scheduled during an outage, please also include a list of work activities greater than 1 rem, scheduled during the outage with the dose estimate for the work activity.)

H. List of active radiation work permits

I. Radioactive source inventory list

a. All radioactive sources that are required to be leak tested

b. All radioactive sources that meet the 10 CFR Part 20, Appendix E, Category 2 and above threshold. Please indicate the radioisotope, initial and current activity (w/assay date), and storage location for each applicable source.

J. The last two leak test results for the radioactive sources inventoried and required to be leak tested. If applicable, specifically provide a list of all radioactive source(s) that have failed its leak test within the last two years

K. A current listing of any non-fuel items stored within your pools, and if available, their appropriate dose rates (Contact / @ 30cm)

L. Computer printout of radiological controlled area entries greater than 100 millirem since the previous inspection to the current inspection entrance date. The printout should include the date of entry, some form of worker identification, the radiation work permit used by the worker, dose accrued by the worker, and the electronic dosimeter dose alarm set-point used during the entry (for Occupational Radiation Safety Performance Indicator verification in accordance with IP 71151).

3. In-Plant Airborne Radioactivity Control and Mitigation (71124.03)
Date of Last Inspection: **October 2015**

A. List of contacts and telephone numbers for the following areas:

1. Respiratory Protection Program
2. Self-contained breathing apparatus

B. Applicable organization charts

C. Copies of audits, self-assessments, vendor or NUPIC audits for contractor support (SCBA), and LERs, written since date of last inspection related to:

1. Installed air filtration systems
2. Self-contained breathing apparatuses

D. Procedure index for:

1. Use and operation of continuous air monitors
2. Use and operation of temporary air filtration units
3. Respiratory protection

E. Please provide specific procedures related to the following areas noted below. Additional Specific Procedures may be requested by number after the inspector reviews the procedure indexes.

1. Respiratory protection program
2. Use of self-contained breathing apparatuses
3. Air quality testing for SCBAs
4. Use of installed plant systems, such as containment purge, spent fuel pool ventilation, and auxiliary building ventilation

F. A summary list of corrective action documents (including corporate and sub-tiered systems) written since date of last inspection, related to the Airborne Monitoring program including since **October 2015**

1. Continuous air monitors
2. Self-contained breathing apparatuses
3. Respiratory protection program

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide in document formats which are "searchable" so that the inspector can perform word searches.

G. List of SCBA qualified personnel - reactor operators and emergency response personnel

H. Inspection records for self-contained breathing apparatuses (SCBAs) staged in the plant for use since date of last inspection: **October 2015**

I. SCBA training and qualification records for control room operators, shift supervisors, STAs, and OSC personnel for the last year.

A selection of personnel may be asked to demonstrate proficiency in donning, doffing, and performance of functionality check for respiratory devices

J. List of respirators (available for use) by type (APR, SCBA, PAPR, etc.), manufacturer, and model.

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
February 24, 2017
Notification of Inspection and Request for Information
Comanche Peak Unit 2
NRC Inspection Report 05000446/2017002

INSERVICE INSPECTION DOCUMENT REQUEST

Inspection Dates: April 10 - 21, 2017

Inspection Procedures: IP 71111.08 "Inservice Inspection (ISI) Activities"

Inspector: Isaac Anchondo

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 Isaac Anchondo, by March 17, 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.

Based on the current schedule, on April 10, 2017, reactor inspector from the Nuclear Regulatory Commission's (NRC) Region IV office will perform the baseline inservice inspection at Comanche Peak, Unit 2, using NRC Inspection Procedure 71111.08, "Inservice Inspection

Activities." Experience has shown that this inspection is a resource intensive inspection both for the NRC inspector 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 inspector are adequately prepared. The second group (Section B of the enclosure) identifies the information the inspector 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: April 3, 2017
Onsite weeks: April 10 - 21, 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 Isaac Anchondo at (817) 200-1152 (isaac.anchondo@nrc.gov).

A.1 ISI/Welding Programs and Schedule Information

a) A detailed schedule (including preliminary dates) of:

i. 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.

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

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

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

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

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

c) 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.

d) 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.

e) 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.

f) 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.

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

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

A.2 <u>Reactor Pressure Vessel Head</u>

Provide a detailed scope of the planned bare metal visual examinations (e.g., volume coverage, limitations, etc.) of the vessel upper head penetrations and/or any nonvisual nondestructive examination of the reactor vessel head including the examination procedures to be used.

i. Provide the records recording the extent of inspection for each penetration nozzle including documents which resolved interference or masking issues that confirm that the extent of examination meets 10 CFR 50.55a(g)(6)(ii)(D).

ii. Provide records that demonstrate that a volumetric or surface leakage path examination assessment was performed.

Copy of current calculations for EDY, and RIY as defined in Code Case N-729-1 that establish the volumetric and visual inspection frequency for the reactor vessel head and J-groove welds.

A.3 <u>Boric Acid Corrosion Control Program</u>

a) 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.

b) 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.4 Steam Generator Tube Inspections

a) A detailed schedule of:

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

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

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

c) 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.

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

    i. Degradation assessment

    ii. Condition monitoring assessment

e) 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.

f) 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).

g) 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.

h) 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.

i) 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.5 Additional Information Related to all Inservice Inspection Activities

a) A list with a brief description of inservice inspection, and boric acid corrosion control program related issues (e.g., Condition Reports) 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.

b) Provide training (e.g. Scaffolding, Fall Protection, FME, Confined Space) if they are required for the activities described in A.1 through A.4.

c) 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(s) at the Entrance Meeting (April 6, 2017):

B.1 Inservice Inspection / Welding Programs and Schedule Information

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

b) 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.

c) For the inservice inspection related corrective action issues selected by the inspector from Section A of this enclosure, provide a copy of the corrective actions and supporting documentation.

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

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

f) For the nondestructive examinations selected by the inspector 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 Reactor Pressure Vessel Head (RPVH)

a) Provide drawings showing the following (if performing any RPVH inspection activities):

   i. RPVH and control rod drive mechanism nozzle configurations.

   ii. RPVH insulation configuration.

   Note: The drawings listed above should include fabrication drawings for the nozzle attachment welds as applicable.

b) Copy of the documents which demonstrate that the procedures to be used for volumetric examination of the reactor vessel head penetration J-groove welds were qualified by a blind demonstration test in accordance with 10 CFR 50.55a(g)(6)(ii)(D).

c) Copy of volumetric, surface and visual examination records for the prior inspection of the reactor vessel head and head penetration J-groove welds.

B.3 Boric Acid Corrosion Control Program

a) 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.

b) 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.4 Steam Generator Tube Inspections

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

b) 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.).

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

d) 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.

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

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

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

B.5 Codes and Standards

a) 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):

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

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

Comanche Peak Medium Energy Line Break Licensee Event Report

Detailed Risk Evaluation

Comanche Peak Nuclear Power Plant Licensee Event Report 16-002-01, "Unanalyzed Condition Involving Potential Moderate Energy Line Break," described six vulnerabilities in the licensee's equipment configurations for medium energy line breaks (MELB). Each of these conditions is contained in the tables in this evaluation.

The length of piping which could cause the loss of the affected components by having a MELB was estimated and the values are contained in the tables.

All MELBs were assumed to fail the affected components every time the piping would leak or break. The pipe leak and/or break frequency was estimated by the use of system piping data from the 2010 Component Reliability update for data from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." The analyst assumed non-service water system piping and added the 2.53E-10 per hour per foot for small leakage and 2.53E-11 per hour per foot for large leakage to estimate the MELB hazard.

The MELB deficiencies were assumed to have existed since initial plant power operations began. Exposure time was limited to one year per Section 2.0, "Exposure Time Modeling," in the Risk Assessment of Operational Events (RASP) Handbook.

The analyst assumed for condition number 1 in the tables below that each unit was vulnerable to having a service water pump affected and therefore considered a loss of service water pump for each unit in the estimate of increase in core damage frequency for each unit. The Comanche Peak Plant Risk Information e-Book was used to determine that service water pump 1-1 was the most risk significant service pump for Unit 1; and service water pump 2-1 was the most significant service water pump for Unit 2.

The analyst assumed for condition number 3 in the tables below that only one component could be affected, therefore in the tables, only the most risk significant service water pump was considered in the final estimated increase in core damage frequency.

The analyst first estimated the increase in core damage frequency due to the additional failure probability of the components from a MELB event. The results are contained in the following table:

<table>
<tr><th colspan="2" rowspan="2">Condition Number and Affected Component(s)</th><th rowspan="2">Feet of piping</th><th rowspan="2">Piping failure probability</th><th rowspan="2">Nominal failure probability</th><th rowspan="2">Revised failure probability</th><th colspan="2">Increase in core damage frequency</th></tr>
<tr><th>Unit 1</th><th>Unit 2</th></tr>
<tr><td rowspan="2">1</td><td>Service Water Pump 1-1</td><td>50</td><td>3.34E-7</td><td>1.57E-4</td><td>1.58E-4</td><td>2.44E-10</td><td>N/A</td></tr>
<tr><td>Service Water Pump 2-1</td><td>50</td><td>3.34E-7</td><td>1.57E-4</td><td>1.58E-4</td><td>N/A</td><td>2.44E-10</td></tr>
<tr><td>2</td><td>Motor Control Center 2EB2-1</td><td>5</td><td>3.34E-8</td><td>3.33E-5</td><td>3.34E-5</td><td>N/A</td><td>Negligible</td></tr>
<tr><td rowspan="3">3</td><td>Switchgear 1EA2</td><td>20</td><td>1.34E-7</td><td>3.33E-5</td><td>3.34E-5</td><td>1.32E-10</td><td>N/A</td></tr>
<tr><td>Motor Control Center 1EB4-2</td><td>20</td><td>1.34E-7</td><td>3.33E-5</td><td>3.34E-5</td><td>Negligible</td><td>N/A</td></tr>
<tr><td>Distribution Panel 1ED2-2</td><td>20</td><td>1.34E-7</td><td>6.50E-5</td><td>6.51E-5</td><td>Negligible</td><td>N/A</td></tr>
</table>

Condition Number and Affected Component(s)		Feet of piping	Piping failure probability	Nominal failure probability	Revised failure probability	Increase in core damage frequency	
						Unit 1	Unit 2
4	Motor Control Center 1EB4-1	5	3.34E-8	3.33E-5	3.333E-5	1.0E-11	N/A
5	Motor Control Center 2EB4-1	5	3.34E-8	3.33E-5	3.34E-5	N/A	1.0E-11
6	Motor Control Center 1EB4-1	10	6.68E-8	3.33E-5	3.34E-5	1.0E-11	N/A
Total Increase In Core Damage Frequency (per year) for each Unit						3.96E-10	2.54E-10

The analyst then estimated the increase in core damage frequency from each of the MELB events by increasing the initiating event frequency for cases where there was a clear initiator (e.g., loss of bus initiator, loss of service water initiator). If no initiator was modelled, the analyst assumed the MELB caused a transient and applied the frequency of the MELB event to the conditional core damage probability (CCDP) for a transient. The following table contains the results:

Condition Number and Affected Component(s)		Feet of piping	Piping failure frequency (per year)	Nominal initiating event frequency	Revised initiating event frequency	Transient CCDP	Increase in core damage frequency	
							Unit 1	Unit 2
1	Service Water Pump 1-1	50	1.22E-4	5.74E-2	5.76E-2	N/A	Negligible	N/A
	Service Water Pump 2-1	50	1.22E-4	5.74E-2	5.76E-2	N/A	N/A	Negligible
2	Motor Control Center 2EB2-1	5	1.22E-5	6.90E-1	6.90E-1	9.33E-6	N/A	1.2E-10
3	Switchgear 1EA2*	20	4.88E-5	False	4.88E-5	N/A	4.55E-8	N/A
	Motor Control Center 1EB4-2	20	4.88E-5	6.90E-1	6.90E-1	5.35E-6	2.61E-10 (not used)	N/A
	Distribution Panel 1ED2-2**	20	4.88E-5	7.37E-4	7.86E-4	N/A	3.52E-8 (not used)	N/A
4	Motor Control Center 1EB4-1	5	1.22E-5	6.90E-1	6.90E-1	2.22E-4	2.7E-9	N/A
5	Motor Control Center 2EB4-1	5	1.22E-5	6.90E-1	6.90E-1	2.22E-4	N/A	2.7E-9
6	Motor Control Center 1EB4-1	10	2.44E-5	6.90E-1	6.90E-1	2.22E-4	2.7E-9	N/A
Total Increase In Core Damage Frequency (per year) for each Unit							5.1E-8	2.8E-9

*The analyst assumed that an initiator for failure of bus 1EA2 could be modeled by increasing the failure probability of bus 1EA1 to obtain representative results because the SPAR model did not contain an initiator of loss of bus 1EA2.

**A MELB affecting distribution panel 1ED2-2 was assumed to initiate a complete loss of DC bus 1ED2.

Adding the results of the two effects resulted in an estimated increase in core damage frequency of 5.1E-8/year for Unit 1 and 2.9E-10/year for Unit 2. Based on these increases, the finding was determined to be of very low safety significance (Green). The estimates were obtained by use of Version 8.28 of the Comanche Peak SPAR model ran on SAPHIRE, Version 8.1.5. The dominant core damage sequences were losses of switchgear and losses of service water. Offsite power and feed and bleed availability remained to mitigate the significance of dominant initiators.