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

February 13, 2017

EA-16-069

Adam C. Heflin, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION – NRC INTEGRATED INSPECTION REPORT 05000482/2016004

Dear Mr. Heflin:

On December 31, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Wolf Creek Generating Station. On February 8, 2017, the NRC inspectors discussed the results of this inspection with Mr. S. Smith, Plant Manager, and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented one finding of very low safety significance (Green) in this report. This finding involved a violation of NRC requirements. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation or significance of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement; and the NRC resident inspector at the Wolf Creek Generating Station.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; and the NRC resident inspector at the Wolf Creek Generating Station.

This letter also provides you the final significance determination of the preliminary Greater than Green finding identified in NRC Inspection Report 05000482/2016008, issued on August 19, 2016 (Agencywide Documents Access and Management System (ADAMS) ML16235A132). The finding was associated with the failure to adequately establish and adjust preventive maintenance activities for emergency diesel generator excitation system diodes. At your request, a Regulatory Conference was held on September 21, 2016, to discuss your position on the preliminary Greater than Green finding and to present new information based on testing conducted by your staff. A copy of your presentation provided at the Regulatory Conference is attached to the summary of the Regulatory Conference dated September 27, 2016

(ML16271A482). In your presentation, you discussed information important to characterize the safety significance of the finding associated with the emergency diesel generator excitation diodes. Specifically, you presented additional test data and evaluation that demonstrated that the diodes had failed due to manufacturing defects, and that the failure to establish preventive maintenance schedules for diode replacement was not the cause of the failure of emergency diesel generator B on October 6, 2014.

After considering the information reviewed during our inspections and the information you provided at the Regulatory Conference, the NRC has concluded that the failure to adequately develop and adjust preventive maintenance activities was not the root cause of the failure of the diode on June 9, 2014. As a result, this performance deficiency alone did not cause the failure of the excitation system diode. Based on this new understanding of the facts, the NRC has concluded that the finding is appropriately characterized as Green, a finding of very low safety significance. Section 4OA3 of the enclosure provides additional information.

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 in 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/

Nicholas H. Taylor, Branch Chief
Project Branch B
Division of Reactor Projects

Docket No.: 50-482
License No.: NPF-42

Enclosure:
Inspection Report 05000482/2016004
w/ Attachments:
1. Supplemental Information
2. Request for Information for the Occupational Radiation Safety Inspection

WOLF CREEK GENERATING STATION – NRC INTEGRATED INSPECTION REPORT 05000482/2016004 DATED FEBRUARY 13, 2017

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ADAMS ACCESSION NUMBER: ML17045A201

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

OFFICE	ASRI/DRP/B	SRI/DRP/B	RI/DRP/B	C:DRS/PSB2	C:DRS/EB1	C:DRS/EB2
NAME	MLangelier	DDodson	FThomas	HGepford	TFarnholtz	GWerner
SIGNATURE	/RA/	/RA/	/RA/	/RA/	/RA/	/RA/
DATE	2/10/17	2/9/17	02/09/17	02/07/17	02/06/2017	2/3/2017
OFFICE	TL/DRS/IPAT	C:DRS/OB	C: ACES	C:DRP/B		
NAME	THipschman	VGaddy	MHay	NTaylor		
SIGNATURE	/RA/HAF for	/RA/	/RA/JGK for	/RA/		
DATE	2/7/2017	2/3/17	2/8/17	2/13/17		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	05000482
License:	NPF-42
Report:	05000482/2016004
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane NE Burlington, KS 66839
Dates:	October 1 through December 31, 2016
Inspectors:	M. Langelier, P.E., Acting Senior Resident Inspector D. Dodson, Senior Resident Inspector F. Thomas, Resident Inspector R. Kopriva, Senior Reactor Inspector J. O'Donnell, CHP, Health Physicist M. Phalen, Senior Health Physicist G. Pick, Senior Reactor Inspector
Approved By:	Nicholas H. Taylor Chief, Project Branch B Division of Reactor Projects

SUMMARY

IR 05000482/2016004; 10/01/2016 – 12/31/2016; Wolf Creek Generating Station; Maintenance Effectiveness and Follow-up of Events and Notices of Enforcement Discretion

The inspection activities described in this report were performed between October 1 and December 31, 2016, by the resident inspectors at Wolf Creek Generating Station and inspectors from the NRC's Region IV office. One finding of very low safety significance (Green) is documented in this report. This finding involved a violation 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 Technical Specification 5.4.1.a, for the licensee's failure to adequately develop and adjust preventive maintenance activities in accordance with Procedure AP 16B-003, "Planning and Scheduling Preventive Maintenance," Revision 5. Specifically, the licensee did not create a preventive maintenance task for emergency diesel generator excitation system diodes, which resulted in degradation of the excitation system diodes in emergency diesel generator B. The licensee restored compliance by establishing preventive maintenance tasks 49286, 49287, 49288, and 49289, which refurbish the power rectifier assemblies and replace the diodes on a 12-year replacement frequency. The licensee entered this issue into the corrective action program as Condition Report 88665.

  The failure to adequately develop and adjust emergency diesel generator excitation system diode preventive maintenance activities in accordance with Procedure AP 16B-003, "Planning and Scheduling Preventive Maintenance," was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems 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. The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," this finding was not a deficiency affecting the design or qualification of a mitigating structure, system, or component that maintained its operability or functionality; the finding did not represent a loss of system and/or function; the finding did not represent an actual loss of function of at least a single train for greater than its Technical Specification allowed outage time; and the finding did not represent an actual loss of function of one or more non-Technical Specification trains of equipment designated as high safety-significant. Therefore, the finding was of very low safety significance (Green). This finding has a cross-cutting aspect in the area of problem identification and resolution, operating experience, because the organization did not systematically and effectively evaluate relevant internal and external operating experience in a timely manner. This issue is indicative of current performance because the station did not take any formal corrective actions to address the station's failure to adequately consider operating experience [P.5]. (Section 4OA3)

PLANT STATUS

Wolf Creek Generating Station began the inspection period shutdown for refueling outage 21. The reactor was restarted on November 20, 2016, and commenced power ascension. The plant reached full power on November 26, 2016, and remained at full power for the remainder of the inspection period.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

On December 16, 2016, the inspectors completed an inspection of the station's readiness for seasonal extreme weather conditions. The inspectors reviewed the licensee's adverse weather procedures for extreme cold weather and evaluated the licensee's implementation of these procedures. The inspectors verified that prior to the onset of extreme cold weather, the licensee had corrected weather-related equipment deficiencies identified during the previous winter season.

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

- Essential service water system
- Condensate transfer and storage system

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

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

b. Findings

No findings were identified.

.2 Readiness to Cope with External Flooding

a. Inspection Scope

On December 15, 2016, the inspectors completed an inspection of the station's readiness to cope with external flooding. After reviewing the licensee's flooding analysis, the inspectors chose the cooling lake spillways as a focus area of the inspection as the maximum probable flood level (which results in no susceptible areas to flooding) was dependent on the lake spillway's ability to pass water at the design rate.

The inspectors reviewed plant design features and design calculations for external flooding. The inspectors walked down the cooling lake spillways to inspect the design features, including the material condition of the spillways and to ensure no debris would affect the spillways' function. The inspectors evaluated the periodic inspections in place for the spillways.

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

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walk-Down

a. Inspection Scope

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

- October 15, 2016, spent fuel pool cooling pump B
- October 26, 2016, essential service water pump A

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 trains were correctly aligned for the existing plant configuration.

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

b. Findings

No findings were identified.

.2 Complete Walk-Down

a. Inspection Scope

On December 15, 2016, the inspectors performed a complete system walk-down inspection of the train B emergency diesel generator. The inspectors reviewed the licensee's procedures and system design information to determine the correct system lineup for the existing plant configuration. The inspectors also reviewed outstanding work orders, open condition reports, and other open items tracked by the licensee's operations and engineering departments. The inspectors then visually verified that the system was correctly aligned for the existing plant configuration.

These activities constituted one complete system walk-down sample, as defined in Inspection Procedure 71111.04.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

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

- October 13, 2016, fire area RB-2, reactor building - general area, elevation 2000 feet

- October 17, 2016, fire areas F-2 and F-3, fuel pool heat exchanger rooms - trains A and B, elevation 2000 feet

- October 25, 2016, fire area A-16, component cooling water heat exchangers, pumps, pump room coolers, piping and valves - trains A and B, elevation 2026 feet

- December 20, 2016, fire area C-21, lower cable spreading room 3501, elevation 2032 feet

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

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

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

On December 21, 2016, the inspectors completed an inspection of the station's ability to mitigate flooding due to internal causes. After reviewing the licensee's flooding analysis, the inspectors chose the control building 2016 foot elevation area including rooms 3403, 3404, 3405, 3407, 3408, 3409, 3410, 3411, 3413, 3414, 3415, and 3416, which contain risk-significant structures, systems, and components (SSCs) that were susceptible to flooding.

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

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

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

.1 Nondestructive Examination Activities and Welding Activities

a. Inspection Scope

The inspector directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-C. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 202°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F303. Reactor Outlet Safe End to Pipe Weld at 202°.	Eddy Current

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-D. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 338°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F403. Reactor Outlet Safe End to Pipe Weld at 338°.	Eddy Current
Safety Injection System	Penetrant Report Number: PT-4119, Work Order (WO) # 15-402944-009, Identification: EJ-04-R008, 2000 foot elevation outside Reactor Coolant System B.	Penetrant
Reactor Coolant System	Report Number: PT-4147, Pressurizer Skirt Weld, WO # 15-402944003.	Penetrant
High Pressure Coolant Injection System	Penetrant Report Number: PT-4158, WO # 15-402944-010, Identification: EM-01-A003, Saddle Weld located in 88 Pipe Chase.	Penetrant
Reactor Coolant System	Penetrant Report Number: PT-4162, WO # 16-417262-003, Core Exit Thermocouple Nuclear Assembly 77 Seal Weld.	Penetrant
Reactor Coolant System	Radiograph Report Number: RT-4389, Weld # PW-2, WO # 13-364166-007, Valve BBV0130. RCS Hot Leg Drain.	Radiograph
Reactor Coolant System	Radiograph Report Number: RT-4388, Weld # PW-1A, WO # 13-364166-007, Valve BBV0130. RCS Hot Leg Drain.	Radiograph
Reactor Coolant System	Radiograph Report Number: RT-4391, Weld # W-2, WO # 13-371092-000, Valve BBV0266. RCS Hot Leg Drain.	Radiograph

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Radiograph Report Number: RT-4390, Weld # W-1B, WO # 13-371092-000, Valve BBV0266. RCS Hot Leg Drain.	Radiograph
High Pressure Coolant Injection	Ultrasonic Report Number: JEW-003, Pipe to Elbow Weld Joint EM-05-S005B, Loop D – SI injection to RHR 6 inch line. Exam angle: 45°.	Ultrasonic
High Pressure Coolant Injection	Ultrasonic Report Number: JEW-003, Elbow to Pipe Weld Joint EM-05-S005C, Loop D – Safety Injection to Residual Heat Removal 6 inch line. Exam angle: 45°.	Ultrasonic
Reactor Coolant System	Ultrasonic Report Number: TMC-008, Pipe to Elbow Weld Joint BB-04-F010, Pressurizer Spray Line. Exam angle: 45°.	Ultrasonic
Reactor Coolant System	Ultrasonic Report Number: TMC-008, Pipe to Elbow Weld Joint BB-04-S007-D, Pressurizer Spray Line. Exam angle: 45°.	Ultrasonic
Chemical and Volume Control System	Ultrasonic Report Number: TMC-007, Pipe to Pipe Weld Joint BG-21-F014A. Exam angle: 45°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-C. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 202°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F303. Reactor Outlet Safe End to Pipe Weld at 202°.	Ultrasonic

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-D. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 338°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F403. Reactor Outlet Safe End to Pipe Weld at 338°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Head Inspection Final Examination Report Fall 2016: WDI-PJF-1318261-FSR-001, Revision 0.	Ultrasonic
High Pressure Coolant Injection System	WO # 15-402944-010, Component A003, Pipe Support. Location 88 Foot Pipe Chase.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-01-C016, Pipe Support. Location 1974 Foot Aux Room 1108.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-01-H015, Pipe Support. Location 1974 Foot Aux Room 1108.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-02-R005, Pipe Support. Location 2000 Foot Aux Room 1323.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-R018, Pipe Support. Location 2010 Foot Containment Loop B.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-A005, Pipe Support. Location 2000 Foot Containment.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-05-R001, Pipe Support. Location 2000 Foot Containment Loop D.	Visual

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-C013, Pipe Support. Location 1974 Foot Aux Room 1122.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-C033, Pipe Support. Location 2000 Foot Containment.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-R015, Pipe Support. Location 2000 Foot Containment.	Visual
High Pressure Coolant Injection System	WO # 15-402944-010, Component EM-03-C012, Pipe Support. Location 2000 Foot Aux Room 1323.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop A Hot Leg Top Center NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop A Hot Leg Lower Inside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop A Hot Leg Lower Outside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop A Cold Leg - NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop B Hot Leg Top Center NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop B Hot Leg Lower Inside NR Thermowell.	Visual

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop B Hot Leg Lower Outside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop B Cold Leg - NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop C Hot Leg Top Center NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop C Hot Leg Lower Inside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop C Hot Leg Lower Outside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop C Cold Leg - NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop D Hot Leg Top Center NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop D Hot Leg Lower Inside NR Thermowell.	Visual
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop D Hot Leg Lower Outside NR Thermowell.	Visual

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	STS PE-040I, Reactor Pressure Vessel Loop Nozzle Visual Examination, Revision 2. Loop D Cold Leg - NR Thermowell.	Visual

The inspector reviewed records for the following nondestructive examinations:

SYSTEM	IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-A. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 22°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F103. Reactor Outlet Safe End to Pipe Weld at 22°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-B. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 158°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F203. Reactor Outlet Safe End to Pipe Weld at 158°.	Eddy Current
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-A. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 22°.	Ultrasonic

SYSTEM	IDENTIFICATION	EXAMINATION TYPE
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F103. Reactor Outlet Safe End to Pipe Weld at 22°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # RV-301-121-B. Reactor Outlet Nozzle to Safe End Dissimilar Metal Weld at 158°.	Ultrasonic
Reactor Coolant System	Reactor Vessel Nozzle Safe End Weld Examinations Final Report: WDI-PJF-1316944-FRS-001, Revision 0, Weld # BB-01-F203. Reactor Outlet Safe End to Pipe Weld at 158°.	Ultrasonic
Safety Injection System	WO # 15-402944-009, Component EJ-03-R015, Pipe Support. Safety Injection Room 1108.	Visual
Safety Injection System	WO # 15-402944-009, Component EJ-03-C007, Pipe Support. Safety Injection Room 1108.	Visual

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

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

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant System	Valve BBV0090, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-402713-000, ANSI B31.1, Field Weld W-1A.	Manual Gas Tungsten Arc Welding

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant System	Valve BBV0090, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15 402713 000, ANSI B31.1, Field Weld W-2.	Manual Gas Tungsten Arc Welding
Reactor Coolant System	Line BB099BCD-3/4, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-402713-000, ANSI B31.1, Field Weld W-3A.	Manual Gas Tungsten Arc Welding

The inspector reviewed records of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Reactor Coolant System	Line BB099BCD-3/4, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-401589-000, ANSI B31.1, Field Weld W-1A Pipe to Pipe.	Manual Gas Tungsten Arc Welding
Reactor Coolant System	Valve BBV0089, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-401589-000, ANSI B31.1, Field Weld W-2 Valve to Pipe.	Manual Gas Tungsten Arc Welding
Reactor Coolant System	Valve BBV0089, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-401589-000, ASME Section III, Class 2, Field Weld W-3A Valve to Pipe.	Manual Gas Tungsten Arc Welding
Reactor Coolant System	Line BB32BCA-6, Reactor Coolant System Pressurizer Relief Header Drain Valve. WO # 15-401589-000, ASME Section III, Class 2, Field Weld W-4A Pipe to Pipe.	Manual Gas Tungsten Arc Welding

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

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The inspector reviewed the results of the licensee's bare metal visual inspection of the Reactor Vessel Upper Head Penetrations to determine whether the licensee identified any evidence of boric acid challenging the structural integrity of the reactor head components and attachments. The inspector also verified that the required inspection coverage was achieved and limitations were properly recorded. The inspector reviewed whether the personnel performing the inspection were certified examiners to their respective nondestructive examination method.

b. Findings

On September 3, 2016, during the performance of Procedure STN PE-040G, "Transient Event Walkdown," quality control personnel identified an active leak (approximately one half to one gallon per minute) coming from Core Exit Thermocouple (CETNA) #77, near the canopy seal area. The core exit thermocouple housing is internally threaded and torqued down onto a seating surface at the interface between the housing and the top of the reactor head adapter. This connection is a mechanical joint and leakage via this pathway is not pressure boundary leakage as defined by Technical Specifications. The seal weld is not a structural part of the pressure boundary and is not required to meet the structural requirements of ASME Boiler and Pressure Vessel Code, Section III, NB-3000. The threads are the load carrying part of the joint design.

Boron accumulation was noted on the reactor head, control rod drive mechanism penetration nozzles, canopy seal welds, and housings from approximately 50° to 180° of the reactor head. Significant boron accumulation was identified on the housing, nozzle, and at the penetration to head interface. Boron at the head interface had a red rust color, indicative of corrosion. Rust was noted on the head adjacent to the boron accumulation. The licensee's boric acid control program manager, design engineering, and certified quality control nondestructive examination personnel, evaluated the amount of metal wastage on the reactor head caused by the boric acid. The results of their evaluations were that the metal wastage on the reactor head was minimal, having no detrimental effects on the design criteria of the reactor head.

The licensee constructed a plan for addressing inspection concerns and follow-up cleaning of the reactor vessel head due to the leak on the core exit thermocouple nozzle assembly. This included removing 14 control rod drive mechanism coil stacks and 13 dummy cans, to remove boron accumulation and install canopy seal weld clamps. The coil stack assemblies are part of the control rod drive mechanism and include a set of coils that generate magnetic flux. Dummy cans are used to cap spare penetrations on the reactor vessel head not required for operation. The licensee also conducted ultrasonic volumetric testing under the reactor head on all of the reactor head penetrations. This provided positive confirmation that there was no leakage coming from underneath the reactor vessel head. Following removal of the coil stacks and the under-head inspection, clamps were installed on the location of the leaking canopy seal weld, penetration #77, and four additional spare locations. The four spare penetrations being clamped were identified as the most susceptible to future leakage based on internal and external operating experience and vendor recommendations.

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspector reviewed the licensee's implementation of its boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspector reviewed the documentation associated with the licensee's boric acid corrosion control walk-down as specified in Procedure AP 16F-001, "Boric Acid Corrosion Control Program," Revision 8. The inspector reviewed whether the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components, and whether engineering evaluation used corrosion rates applicable to the affected components and 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, and 10 CFR 50, Appendix B requirements.

The inspector reviewed 13 licensee boric acid evaluations where boric acid deposits were found on reactor coolant system piping components and other components:

COMPONENT NUMBER	DESCRIPTION	CONDITION REPORT
ENV002	Containment Spray Pump A, WO # 15-407867-000.	CR-99749
EJFE0618	Flow Element for Residual Heat Removal Heat Exchanger 1A to Accumulator Injection, Work Request 16-115489.	CR-101820
EJFCV0618	Residual Heat Removal Heat Exchanger A, Work Request 15-113396.	CR-97652
EJHV8804A	Residual Heat Removal A to Chemical Volume Control System Centrifugal Charging Pumps Isolation, Work Requests 15-110732 and 15-113766.	CR-92506 CR-98261
BBPV8702A	Reactor Coolant System Hot Leg 1 to Residual Heat Removal Pump A Suction, Work Request 15-110750.	CR-92519

PEJ01A	Residual Heat Removal Pump, WO # 13-365489-009.	CR-92501
HFV0161	Secondary Liquid Waste Plant Discharge Upstream Manual Isolation Valve, WO # 15-398783-010.	CR 95103
EP8818D	Residual Heat Removal to Accumulator Injection Line Check Valve, Work Request 15-110614.	CR-92365
BNV0002	Fuel Pool Cleanup Pumps to Reactor Water Storage Tank Isolation Valve, Work Request 15-115352.	CR-101520
EMV0213	Safety Injection Pump B Discharge Header Upstream Drain, Work Request 16-117153.	CR-105232
SJV0772	Reactor Coolant System/Pressurizer Liquid Pressurized Sample Flow Control Valve, Work Request 14-106695.	CR-83239
EMV0159	Safety Injection Pump B to Accumulator Injection Header Drain Valve, Work Request 16-117152.	CR-105231
BGFT0157	Reactor Coolant Pump Seal 1 Water Leak-off, Work Request 15-114684.	CR-99986

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 repairs were required at the time of the inspection.

Steam Generator Inspection

The initial scope of inspection for the steam generators during Refueling Outage 21 were as follows:

Primary Side:

- Eddy current bobbin probe examinations of all four steam generators (50 percent of all inservice tubes, full length tube-end to tube-end) were performed.

- Eddy current rotating pancake probe – top of tube sheet, Hot Leg and Cold Leg:

  - Steam Generator A, 100 percent – as a result of the inspection findings during the last Refueling Outage 20

  - Steam Generator B, 50 percent

  - Steam Generator C, 50 percent

  - Steam Generator D, 50 percent

- 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.

Secondary Side:

The inspector observed and reviewed secondary side inspection results and verified the licensee took corrective actions in response to the observed degradation. Inspections performed were:

- Full bundle advanced scale conditioning agents cleaning in all four steam generators

- Sludge lancing in all four steam generators

- Top of tube sheet visual inspections

- Post-top of tube sheet sludge lancing cleanliness inspections and foreign object search and retrieval

- Foreign object search and retrieval inspections of the annulus and tube lane in all four steam generators

- Post-top of tube sheet in-bundle sludge lancing inspections in Steam Generators A and D

- Upper bundle in-bundle inspection in Steam Generator D

- Steam drum inspections in Steam Generators A and D, including feed ring and J-nozzle inspections

- Ultrasonic thickness measurements at select locations

- Sludge/deposit chemistry analysis for all four steam generators

- Scale profiling in Steam Generator D

- Top of tube sheet sludge height mapping

<u>Visual Examinations</u>:

The inspector observed and reviewed the visual examination inspection results. Inspections performed were:

- Inspections of selected steam generator tube plugs, including factory installed plugs, or their replacements from the primary side

- Inspection of primary channel head cladding, divider plate, stub runner, and associated welds

b. <u>Findings</u>

On October 9, 2016, a single circumferential indication, categorized as inside diameter primary water stress corrosion cracking was identified in Steam Generator D, Row 19, Column 83. The indication was located approximately 0.88 inches below the top of tube sheet (in the hot leg). Per EPRI guidelines and the Degradation Assessment, the scope of eddy current inspections was increased to complete 100 percent of the tubes in the Steam Generator D top of tube sheet hot leg region. Previously, there had been two indications of inside diameter primary water stress corrosion cracking in steam generators at Wolf Creek: One indication was found in Steam Generator B during Refueling Outage 19 and one in Steam Generator A during Refueling Outage 20. Due to the identification of these indications, the inspection scope was expanded, and no additional indications were identified.

Also on October 9, 2016, a single circumferential indication was identified at the top of the tube sheet hot leg region in Steam Generator C, Row 41, Column 70. This indication is located 0.20 inches below the top of the tube sheet, hot leg side. This indication has been categorized as outside diameter stress corrosion cracking. Per EPRI guidelines, this indication requires an increase in the scope of eddy current inspections to 100 percent of the top of the tube sheet in the hot leg regions in Steam Generator C. This represents a degradation mechanism that had not been observed previously in the steam generators at Wolf Creek.

Due to the two new indications identified, and since the licensee was now required to do 100 percent inspection of the top of tube sheet in the hot leg regions in Steam Generators A, C, and D, the licensee conservatively increased their current inspections of Steam Generator B from 50 percent of tubes to 100 percent. Therefore, during Refueling Outage 21, the licensee completed 100 percent eddy current inspection of the top of tube sheet in the hot leg regions in all four steam generators. Also, due to the flaw identified in one of the tubes in Steam Generator C, the significance of the tube degradation had to be determined using EPRI Report 1025132, "Steam Generator Management Program: Steam Generator In-Situ Pressure Test Guidelines," Revision 4, October 2012. In-situ testing is the process of proof (burst) and leakage testing of steam generator tubing in the steam generators. In-situ testing is used to verify structural and leakage integrity of the steam generators, when analysis is not adequate to confirm structural integrity of the steam generator tube. Per the EPRI testing guidelines, the licensee was required to perform an in-situ pressure test on the degraded steam generator tube. The inspector witnessed the entire pressure test. The test was successful and the tube maintained its integrity and did not leak. The degraded tube was then plugged.

Tube Repair:

The inspector verified that the licensee implemented repair methods that were consistent with the repair processes allowed in the plant technical specification requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service. The licensee repaired a total of seven tubes. The following repairs were made:

- Steam Generator A – 1 tubes plugged
- Steam Generator B – 2 tubes plugged
- Steam Generator C – 1 tubes plugged
- Steam Generator D – 3 tubes plugged

.5 Water Jet Peening

a. Inspection Scope

During construction of light water reactors, pressurized water reactors and boiling water reactors, Alloy 600, a nickel-chromium-iron alloy was used in many locations of the reactor coolant system. Operating experience has shown that Alloy 600 and related weld filler materials, UNS N06082 and UNS W86182 (also referred to as Alloys 82 and 182 or collectively as Alloy 82/182), are susceptible to stress corrosion cracking in the reactor coolant environment. For the primary systems of pressurized water reactors, this degradation mechanism is called primary water stress corrosion cracking. Industry operating experience to date has identified cracks at Alloy 82/182 welds, including some through-wall leaks, at several locations in the reactor coolant system piping and/or components:

- Reactor pressure vessel outlet nozzle butt welds
- Control rod drive mechanism nozzles and welds
- Bottom mounted instrument nozzles and welds

- Pressurizer nozzle butt welds (cracks detected at Wolf Creek by inspection prior to mitigation by weld overlay)

- Pressurizer heater sleeves

- Steam generator primary side channel head drain attachment welds (Wolf Creek has experienced leakage at these locations)

- Steam generator primary side nozzle butt welds

Alloy 82/182 dissimilar metal welds are used on all inlet and outlet nozzles to attach a stainless steel safe-end to the low alloy steel nozzle of the vessel. The intent of the water jet peening project was to mitigate the susceptibility of these locations to primary water stress corrosion cracking by changing the stresses on the wetted surfaces using a water jet peening technique. The water jet peening process is designated as a surface stress improvement mitigation technique per ASME Code Case N-770-4, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI." Water jet peening treatment of the entire wetted surface of the susceptible material (Alloy 82/182) was performed using a qualified procedure that is documented to meet the performance criteria of ASME Code Case N-770-4, Appendix I.

Pre-peening examinations associated with water jet peening mitigation were performed in accordance with the licensee's Inservice Inspection Program, which included ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1," that described augmented examination requirements. Code Case N-770-1 was the basis for the augmented examinations of reactor vessel nozzle dissimilar metal welds currently required by the NRC. The nozzle tubes for the bottom mounted nozzles are wrought Alloy 600 and the nozzles are attached to the vessel with Alloy 82/182 J-groove welds. Water jet peening covered the highly stressed portions of the nozzle tube inside and outside surfaces. "Highly stressed" is defined as a calculated stress greater than 10 kilopound per square inch (ksi).

Peening was performed over longer lengths of the inside and outside diameters to assure that at risk material was fully covered by the mitigation process. The entire wetted surfaces of the Alloy 82/182 J-groove welds were treated by water jet peening. The water jet peening technique employed gets its mitigating effect by leaving the surface in a residual stress state of biaxial compression. Cracks do not initiate in compression so performing peening prior to crack initiation helps prevent primary water stress corrosion cracking. The water jet peening process was applied in accordance with 10 CFR 50, Appendix B, Criterion IX. Because the effectiveness of peening cannot be confirmed by measurements on the treated surfaces in the field, the process had been qualified by laboratory testing and analysis.

Qualification testing confirmed that appropriate levels and depths of residual stress were achieved. Qualification testing and analysis confirm that residual compressive stress levels were sufficient to maintain compression on the wetted surfaces under normal reactor operating conditions for the remainder of plant life. For the reactor vessel inlet and outlet nozzle dissimilar metal welds, the residual compressive stress extends to a depth of at least 1 millimeter (0.040 inches) below the treated surfaces. For the bottom mounted nozzles, the depth of the compressive residual stress extends to a minimum of 0.5 millimeter (0.020 inches).

The purpose of the activities covered by the licensee's Change Package 12655 were to reduce the susceptibility of the reactor vessel nozzle dissimilar metal welds and bottom mounted nozzle dissimilar metal welds to primary water stress corrosion cracking by creating a favorable compressive residual stress on the wetted surfaces of the susceptible materials in these components, which eliminates future initiation of primary water stress corrosion cracking. The licensee completed water jet peening of all of the hot and cold leg vessel nozzles and all of the bottom mounted instrumentation nozzles. The licensee's Change Package 12655 did not address relaxation of the NRC augmented examinations due to current uncertainty of the future NRC position on ASME Code Case N-770-4.

b. Findings

No findings were identified.

.6 Identification and Resolution of Problems

a. Inspection scope

The inspector reviewed 88 condition reports that dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. 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

No findings were identified.

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

.1 Review of Licensed Operator Requalification

a. Inspection Scope

On December 19, 2016, the inspectors observed simulator training for 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 November 19 and 20, 2016, the inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity and risk due to reactor startup from refueling outage 21.

In addition, the inspectors assessed the operators' adherence to plant procedures, including AP 21-001, "Conduct of Operations," Revision 78, and other operations department policies.

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed two instances of degraded performance or condition of safety-related SSCs:

- August 7, 2016, train A control room air conditioner (SGK04A) air handling unit condensate drain line plugged

- August 15, 2016, train B Class 1E air conditioner (SGK05B) air handling unit condensate drain line plugged

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.

On November 22, 2016, the inspectors reviewed the licensee's periodic 10 CFR 50.65(a)(3) assessment of the Maintenance Rule Program.

The inspectors verified that a periodic evaluation was completed within the time constraints of 10 CFR 50.65(a)(3). The inspectors also verified that the licensee reviewed its (a)(1) goals, (a)(2) performance criteria, monitoring, preventive maintenance activities, effectiveness of corrective actions, and that industry operating experience was taken into account where practicable.

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

b. Findings

No findings were identified.

.2 Quality Control

a. Inspection Scope

On November 22, 2016, the inspectors reviewed the licensee's quality control activities through a review of whether quality control verifications were properly specified in accordance with the licensee's Quality Assurance Program, and were implemented as specified, during work associated with the cleaning and inspection of the reactor vessel head.

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

b. Findings

No findings were identified.

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

a. Inspection Scope

On November 17, 2016, the inspectors observed portions of the troubleshooting and repairs of number six and number seven switchyard transformers after loss of the switchyard east side bus due to an electrical fault. This emergent work activity had the potential to cause an initiating event.

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

These activities constituted completion of one emergent work control inspection sample, as defined in Inspection Procedure 71111.13.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

a. Inspection Scope

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

- October 25, 2016, operability determination of train A emergency diesel generator governor drive casing leak

- November 8, 2016, operability determination for the reactor head due to indications of wastage in three locations

- November 22, 2016, operability determination of combined safety injection/residual heat removal to accumulator D outlet line vent valve

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

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

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

Permanent Modifications

a. Inspection Scope

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

- December 13, 2016, condensate drains for the turbine driven auxiliary feedwater pump

- December 28, 2016, compressor for the Class 1E electrical switchgear air conditioning unit

The inspectors reviewed the design and implementation of the modifications. The inspectors verified that work activities involved in implementing the modifications did not adversely impact operator actions that may be required in response to an emergency or other unplanned event. The inspectors verified that post-modification testing was adequate to establish the operability of the SSCs as modified.

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- October 27, 2016, post maintenance testing of train B emergency diesel generator after extent of condition inspection of governor drive assembly

- November 19, 2016, inservice test of turbine driven auxiliary feedwater pump after installation of modification to the pump's condensate drains

- December 14, 2016, inservice test of train B safety injection pump after planned maintenance work on pump lubricating oil cooling system

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 three 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 November 21, 2016, 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

Prior to the refueling outage, on August 31, 2016, the licensee noted an upward trend in unidentified reactor coolant system leakage. On September 2, 2016, the licensee's daily calculated reactor coolant system unidentified leakage spiked to approximately 1.35 gallons per minute (gpm), exceeding the technical specification limit of 1.0 gpm. As a result, the licensee initiated a technical specification required shutdown on September 2, 2016, in accordance with technical specification 3.4.13. Subsequent to the shutdown, on September 3, 2016, the licensee determined the unidentified reactor coolant system leakage to actually be approximately 0.65 gpm. The licensee's analysis demonstrated that the over-estimation of the leak rate was caused by an abnormal letdown system lineup that caused their mass-balance calculation to be inaccurate.

Following shutdown and containment entry, the source of the leak was identified as the canopy seal weld on penetration 77 on the reactor vessel head, which serves one of the core exit thermocouples. The canopy seal weld is not considered to be part of the pressure boundary of the reactor coolant system. This function is served by the threaded mechanical joint on the nozzle assembly. In accordance with ASME Code Section XI, leakage from the threaded mechanical joint serving the core exit thermocouple nozzle assembly is not considered pressure boundary leakage. Once the leak was determined not to be pressure boundary leakage, the licensee exited technical specification 3.4.13 and maintained the plant in Mode 3 rather than continue with the previously required action to reach Mode 5 within 36 hours (i.e. cooldown the plant). The plant remained in Mode 3 until September 7, 2016, at which time the plant was cooled down and depressurized to Mode 5.

Following the technical specification required shutdown, the licensee remained shut down to commence their refueling outage, during which the licensee repaired the leak using a mechanical clamping assembly.

On November 6, 2016, resident inspectors conducted an inspection of the reactor vessel head during final head cleaning activities performed by the licensee. Resident inspectors performed another head inspection on November 7, 2016, after the licensee completed their final head cleaning activities. During the final head inspection for cleanliness, the licensee discovered three small locations on the head having minor wastage. This wastage indicated a loss of metal from the reactor vessel head as a result of boric acid corrosion from the canopy seal weld leakage. The deepest wastage impression recorded was 0.0625 inches. The actual construction of the reactor head contains between 0.662 and 2.779 inches of margin above the minimum design thickness in the areas of the wastage. The deepest impression recorded resulted in a margin reduction of approximately 9.4 percent.

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 four 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:

- November 8, 2016, STS EJ-100B, “[Residual Heat Removal] System Inservice Pump B Test,” Revision 45

Other surveillance tests:

- October 1, 2016, STS PE-155, “[Local Leak Rate Test] Valve Lineup for Penetration 55,” Revision 6
- October 12, 2016, STS IC-806B, “4KV Undervoltage – Loss of Voltage – Channel Calibration of 1 Second Time Delay Circuit NB02,” Revision 5
- November 21, 2016, STS KJ-100A, “Integrated [Diesel Generator] & Safeguards Actuation Test - Train A,” Revision 61

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 four 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 condition, 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 constitute 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.

2RS2 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

The inspectors assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors performed this portion of the attachment during the refueling outage, in order to directly observe the licensee's ALARA process activities including planning, implementation of radiological work controls, execution of work activities, and ALARA review of work-in-progress. During the inspection the inspectors interviewed licensee personnel, reviewed licensee documents, and evaluated licensee performance in the following areas:

- Implementation of ALARA and radiological work controls. The inspectors observed pre-job briefings, reviewed planned radiological administrative, operational, and engineering controls, and compared the planned controls to field activities.

- Radiation worker and radiation protection technician performance during work activities performed in radiation areas, airborne radioactivity areas, or high radiation areas.

- Problem identification and resolution for ALARA and radiological work 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 constitute completion of two of the five required samples of occupational ALARA planning and controls program, as defined in Inspection Procedure 71124.02.

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 October 1, 2015, through September 30, 2016, 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, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index: Emergency Alternating Current (AC) Power Systems (MS06)

a. Inspection Scope

The inspectors reviewed the licensee’s mitigating system performance index data for the period of August 26, 2015, through September 30, 2016, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, to determine the accuracy of the reported data.

These activities constituted verification of the mitigating system performance index for emergency ac power systems, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index: Heat Removal Systems (MS08)

a. Inspection Scope

The inspectors reviewed the licensee’s mitigating system performance index data for the period of October 1, 2015, through September 30, 2016, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, to determine the accuracy of the reported data.

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

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's mitigating system performance index data for the period of October 1, 2015, through September 30, 2016, to verify the accuracy and completeness of the reported data. The inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, to determine the accuracy of the reported data.

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

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed corrective action program records documenting losses of radiological control over locked high radiation areas and very high radiation areas during the period of January 1, 2015, to September 30, 2016. The inspectors reviewed a sample of radiologically controlled area exit transactions showing exposures greater than 100 millirem. 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 constitute verification of the occupational exposure control effectiveness performance indicator, as defined in Inspection Procedure 71151.

b. Findings

No findings were identified.

.6 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 January 1, 2015, and September 30, 2016, 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 constitute verification of the radiological effluent technical specifications (RETS)/offsite dose calculation manual (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 Semiannual Trend Review

a. Inspection Scope

The inspectors reviewed the licensee's corrective action program, performance indicators, system health reports, and other documentation to identify trends that might indicate the existence of a more significant safety issue. The inspectors verified that the licensee was taking corrective actions to address the following identified adverse trends:

- Licensee identified trend in events related to inadequate monitoring by operators
- Inspector identified trend in seismic instrumentation faults

These activities constituted completion of one semiannual trend review sample, as defined in Inspection Procedure 71152.

b. Observations and Assessments

The inspectors' review of the trends identified above produced the following observations and assessments:

- Inspectors reviewed the licensee's response to a trend evaluation for operator fundamentals involving monitoring. During the period from May 2016 to November 2016, the licensee identified five events that occurred during the time period that were attributed to an aspect of inadequate monitoring. Two of the events included a failure to use available diverse indications. One event involved the failure to verify valve position using a standing order required hands on approach versus an eyes on approach. The remaining two events involved inadequate attention to detail during monitoring of critical equipment.

From this evaluation, the licensee identified two areas for improvement for operator fundamentals including teamwork and monitoring of critical parameters. The licensee has initiated evaluations to determine the causes of the degradation in these two areas and to plan corrective actions.

The inspectors evaluated the licensee's response to this trend and determined the actions to date are appropriate.

- Inspectors identified an adverse trend related to faults on seismic instrumentation. During the refueling outage and specifically from September 14, 2016, to November 13, 2016, various warning and fault lights illuminated on the control room seismic panel for five different seismic instruments on seven separate occasions. The fault lights indicate that either the instrument has lost communication to the seismic panel or that the battery on the seismic instrument has failed. None of these incidents were found to be caused by battery issues. Troubleshooting of the issues by the licensee has not determined a cause and no corrective actions beyond resetting the instrumentation have been taken. In response to inspector concerns, the licensee initiated Condition Report 110503 to evaluate the trend and the need for additional corrective actions.

c. <u>Findings</u>

No findings were identified.

.3 <u>Annual Follow-up of Selected Issues</u>

a. <u>Inspection Scope</u>

From September 26 through October 18, 2016, during an in-office inspection, the inspectors reviewed the NRC-identified and licensee-identified issues documented in Inspection Report 05000482/2015405 (ML15316A250) for an in-depth follow-up. The inspectors reviewed procedures, digital asset listings, and corrective action documents. The inspectors interviewed personnel involved in implementing the corrective actions.

The inspectors assessed the licensee's 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 appropriate.

Additionally, the inspectors selected one issue for an in-depth follow-up:

Between October 22 and October 28, 2016, the actuators for atmospheric relief valves ABV0001, ABV0002, ABV0003, and ABV0004 failed their as-found diagnostic air pressure drop tests.

The inspectors assessed the licensee's problem identification threshold, cause analyses, extent of condition reviews and compensatory actions. The inspectors assessed the licensee's prioritization and adequacy of the planned corrective actions.

These activities constitute completion of two annual follow-up samples 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) LER 05000482/2016-001-01, Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability

On October 6, 2014, during a scheduled 24-hour run, the train B emergency diesel generator unexpectedly tripped and a fire was observed in the electrical cabinet NE106 associated with the exciter circuitry. Inspectors previously reviewed LER 05000482/2016-001-00 associated with this event. Documentation of this review and a preliminary White finding was provided in NRC Inspection Report 05000482/2016008, dated August 19, 2016 (ML16235A132). This revision to the LER revised the root cause and added reporting pursuant to 10 CFR 21. A final significance determination for the preliminary White finding is documented in Section 4OA3.2.b of this report. This LER is closed.

.2 (Closed) Apparent Violation 05000482/2016008-01, Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes

a. Inspection Scope

Apparent Violation (AV) 05000482/2016008-01, "Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System," was opened (ML16235A132) to allow determination of the risk significance of the issue given the potential exposure period of the performance deficiency. The inspectors performed an in-depth review of the licensee's root cause evaluations associated with Condition Report 88665, operating experience related to the event, other related condition reports, and documentation listed in Attachment 1. In addition, the inspectors performed on-site tours, interviewed site personnel, and reviewed corrective actions associated with the condition. Following the regulatory conference of September 21, 2016, the inspectors evaluated additional information provided by the licensee to inform the NRC decision. Apparent Violation 05000482/2016008-01 is closed to the enforcement action discussed below.

b. Findings

Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes

Introduction. The inspectors identified a Green non-cited violation of Technical Specification 5.4.1.a, for the licensee's failure to adequately develop and adjust preventive maintenance activities in accordance with Procedure AP 16B-003, "Planning and Scheduling Preventive Maintenance," Revision 5. Specifically, the licensee did not create a preventive maintenance task for emergency diesel generator excitation system diodes, which resulted in degradation of the excitation system diodes in emergency diesel generator B.

Description. Apparent Violation 05000482/2016008-01, "Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes," outlines details concerning the emergency diesel generator B fire on October 6, 2014. Specifically, the licensee's treatment of 2012 operating experience; the events of June and October 2014; root cause analyses (Condition Report 88665) details; inspectors' conclusions and concerns; risk insights; and corrective actions are detailed in the apparent violation.

Following the issuance of Apparent Violation 05000482/2016008-01, the licensee requested a regulatory conference that was completed on September 21, 2016. During and following the regulatory conference, the inspectors evaluated new information provided by the licensee to inform the NRC's final significance determination decision. The detailed presentation provided by the licensee at the regulatory conference can be reviewed in the meeting summary (ML16271A482).

One event that was discussed extensively during and following the regulatory conference included an independent failure of the engine speed governor on June 9, 2014. This event, which was determined by the licensee to be caused by a wiring defect from the manufacturer, caused the emergency diesel generator to experience overload transients that were more severe than the generator was designed to withstand. During the regulatory conference on September 21, 2016, the licensee presented new testing results that showed that over half of the excitation system diodes that were originally installed in the emergency diesel generators had flaws. The licensee's analysis concluded that these flaws were manufacturing defects that reduced their ability to withstand load transients such as the transients experienced on June 9, 2014. During the regulatory conference, the license stated that the largest expected step increase during design basis events is approximately 1.4 megawatts. Additionally, step increase transients exceeding 1.4 megawatts, such as the approximately 3 megawatt transients experienced during the governor malfunction on June 9, 2014, would be expected to cause damage to excitation system diodes with pre-exisitng flaws.

The inspectors noted a number of industry sources that demonstrated the age-related degradation of the power diodes, including the following:

- Operating experience described in licensee Condition Report 55103 in July 2012 described a service life concern with emergency diesel generator power diodes and described the industry average life span for these diodes as approximately 12 years.

The document recommended utilities consider adjusting preventative maintenance scope and schedule accordingly.

- Electric Power Research Institute Technical Report 1011232, “Emergency Diesel Generator Voltage Regulator Maintenance Issues,” dated December 2004, described, in part, that across the industry components such as diodes appear to be failing because of age. On the subject of obsolescence, the document also recommends that utilities consider replacement of power diodes.

- On September 8, 2016, KCI Engineering Consultants provided, “Diode Failure Analysis for emergency diesel generator Exciter Bridge Rectifier Circuit,” to the licensee, which described a number of stresses that negatively affect the service life of semiconductor devices, including: electrical load (overload), operating temperature, humidity, mechanical stress, static electricity, and effect of repeated stress. The report also described that semiconductors experience a “wearout failure period” near end of life and that failures can occur prematurely due to exposure to the stresses discussed above.

Following the regulatory conference on September 21, 2016, NRC inspectors discovered additional overpower transients that occurred on June 29, 1999, and December 20, 2007, which were in excess of the 1.4 megawatt threshold for damage as described by the licensee. These previous overpower transients would have been expected to cause damage to excitation system diodes with pre-existing flaws, and were not evaluated by the licensee for at least fifteen years. Because the failed subject diodes likely contained flaws and experienced multiple overpower transients in excess of those experienced during design basis conditions, the inspectors determined that a transient-related aging mechanism caused degradation of the diodes and contributed to the failure of a diode on June 9, 2014, as predicted by the operating experience.

As described in Apparent Violation 05000482/2016008-01, the inspectors determined that the July 2012 operating experience was not adequately evaluated, in that the licensee’s power diodes were susceptible to age-related failure mechanisms as described in the operating experience. The inspectors also determined that the licensee should have utilized the operating experience and revised maintenance procedures to prevent aging mechanisms from impacting emergency diesel generator B reliability and availability. The inspectors noted that Procedure AP 16B-003, “Planning and Scheduling Preventive Maintenance,” Revision 5, provides direction for implementing the preventive maintenance program. Section 6.2, “Establishing [preventive maintenance] Activities,” states, “Develop [preventive maintenance] activities by considering the following…Operating Experience (OE) (Industry and Station).” Section 6.2.2, states, “[Preventive maintenance] frequencies are established and adjusted in accordance with…the following considerations…The age of the installed equipment.”

However, the inspectors determined that overload transients on June 9, 2014, introduced a series of voltage transients in the static exciter system that exceeded the capacity of the degraded diodes, which caused the initial diode failure and the fire on October 6, 2014. The inspectors concluded that the transient-related diode aging mechanism was not the root cause of the diode failure but did contribute to the failure. As a result, the inspectors reassessed the risk significance of the failure to adequately develop and adjust preventive maintenance activities and determined that this performance deficiency alone did not cause the failure of the excitation system diodes in emergency diesel generator B on June 9, 2014.

The licensee's corrective actions following the fire included replacing the power potential transformer and selecting the alternate rectifier bank to restore the availability of emergency diesel generator B. In addition to immediate actions taken, the licensee replaced all power diodes within all four rectifier bridges (two rectifier bridges for each emergency diesel generator). On October 27, 2015, the licensee implemented a corrective action to generate new preventive maintenance tasks 49286, 49287, 49288, and 49289 to periodically replace the diodes within the power rectifier and other excitation system components as recommended by the operating experience.

Analysis. The failure to adequately develop and adjust emergency diesel generator excitation system diode preventive maintenance activities in accordance with Procedure AP 16B-003, "Planning and Scheduling Preventive Maintenance," was a performance deficiency. This finding is more than minor because it is associated with the equipment performance 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. The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," this finding was not a deficiency affecting the design or qualification of a mitigating system, structure, or component that maintained its operability or functionality; the finding did not represent a loss of system and/or function; the finding did not represent an actual loss of function of at least a single train for greater than its technical specification allowed outage time; and the finding did not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant. Therefore, the finding was of very low safety significance (Green). This finding has a cross-cutting aspect in the area of problem identification and resolution, operating experience, because the organization did not systematically and effectively evaluate relevant internal and external operating experience. This issue is indicative of current performance because the station had not taken any corrective actions to address the station's failure to adequately consider operating experience [P.5].

Enforcement. Technical Specification 5.4.1.a, requires, in part, that procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2. Section 9.b of Appendix A to Regulatory Guide 1.33, Revision 2, requires that "preventive maintenance schedules be developed to specify…inspection or replacement of parts that have a specific lifetime." The licensee established Procedure AP 16B-003, "Planning and Scheduling Preventive Maintenance," Revision 5, to meet the Regulatory Guide 1.33 requirement. Section 6.2 of Procedure AP 16B-003 requires that preventive maintenance activities be developed by considering operating experience and preventive maintenance frequencies are established and adjusted in accordance with the age of installed equipment. Contrary to the above, until October 27, 2015, the licensee did not ensure that preventive maintenance activities were developed by considering operating experience and preventive maintenance frequencies were not established and adjusted in accordance with the age of installed equipment. Specifically, the licensee did not ensure that adequate preventive maintenance activities were developed for emergency diesel generator excitation system diodes by considering operating experience documented in Condition Report 55103, and preventive maintenance frequencies were not established or adjusted for emergency diesel generator excitation system diodes that were original plant equipment. The licensee restored compliance by establishing preventive maintenance tasks 49286, 49287, 49288, and 49289, which refurbish the power rectifier assemblies and replace the diodes on a 12-year replacement frequency. The licensee entered this finding into the corrective action program as Condition Report 88665. This violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000482/2016004-01, “Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes.”

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

4OA6 Meetings, Including Exit

Exit Meeting Summary

On October 18, 2016, the inspectors presented the inspection results for the annual follow up of issues documented in inspection report 05000482/2015405 inspection to Mr. T. Bailey, Cyber Security Program Manager, and other members of the licensee staff. The inspectors did not review any proprietary information.

On October 21, 2016, the inspectors presented the radiation safety inspection results to Mr. J. McCoy, Vice President, Engineering, 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 October 21, 2016, the inspector presented the inservice inspection results to Mr. J. McCoy, Vice President, Engineering and other members of your staff. The licensee acknowledged the issues presented. The inspector acknowledged review of proprietary material during the inspection, which had been or would be returned to the licensee.

On December 8, 2016, the inspectors presented the final significance determination for the train B emergency diesel generator issue, documented in section 4OA3, to Mr. C. Reasoner, 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 February 8, 2017, the inspectors presented the resident inspection results to Mr. S. Smith, Plant Manager, 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

R. Audano, Superintendent, Maintenance
R. Ayers, Supervisor, Radiation Protection
T. Baban, Manager, Engineering Programs
T. Bailey, Cyber Security Program Owner
R. Barraclough, Boric Acid Program Owner, Engineering
J. Cuffe, Supervisor, Radiation Protection
R. French, Supervisor, Radiation Protection
N. Good, Licensing Engineer
C. Hafenstine, Manager, Regulatory Affairs
A. Heflin, President and Chief Executive Officer
G. Hicks, Supervisor, Quality Control
R. Hobby, Licensing Engineer
J. Knust, Licensing Engineer
J. McCoy, Vice President, Engineering
W. Muilenburg, Supervisor, Licensing
L. Ratzlaff, Manager, Maintenance
C. Reasoner, Site Vice President
M. Skiles, Manager, Security
T. Slenker, Supervisor, Operations Support
S. Smith, Plant Manager
L. Stone, Licensing Engineer
D. Tougaw, Inservice Inspection Program Owner, Engineering
L. Upson, Manager, Strategic Initiatives
P. Wagner, Steam Generator Program Owner, Engineering
J. Yunk, Manager, Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000482/2016004-01	NCV	Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes (4OA3)

Closed

05000482/2016-001-01	LER	Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability (4OA3)
05000482/2016008-01	AV	Failure to Adequately Establish and Adjust Preventive Maintenance for Emergency Diesel Generator Excitation System Diodes (4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

Number	Title	Revision
SYS EF-205	ESW/Circ Water Cold Weather Operations	39
SYS OPS-008	Cold Weather Operations	4

Drawings

Number	Title	Revision
C-1C0227	SBO Diesel Generator Foundation & Missile Barrier Concrete Neat Line & Reinforcing Foundation Plan at El. Varies	2
M-12AN01	Piping & Instrumentation Diagram, Demineralized Water Storage and Transfer System	36
M-12AP01	Piping & Instrumentation Diagram, Condensate Storage and Transfer System	17
M-12EF01	Piping & Instrumentation Diagram, Essential SVC [Service] Water System	29
M-12EF02	Piping & Instrumentation Diagram, Essential Service Water System	42
S-0573	SBO Diesel Generator Grading & Drainage Plan	1

Condition Reports

90048

Miscellaneous

Number	Title	Revision/Date
69461-C-001	Probable Maximum Precipitation Calculation	0
69461-C-002	Peak Discharge Calculation	0

Miscellaneous

Number	Title	Revision/Date
69461-C-002-000-CN001	Peak Discharge Calculation Change Notice 1	June 9, 2016
69461-C-003	Water Surface Elevation Calculation	2
69461-C-003-002-CN001	Water Surface Elevation Calculation Change Notice 1	January 19, 2015
69461-C-003-002-CN002	Water Surface Elevation Calculation Change Notice 2	February 16, 2015
69461-C-003-002-CN003	Water Surface Elevation Calculation Change Notice 3	June 9, 2016
WR-WC-DI-1	Wind Wave & High Water Level at ESW Screenhouse	1
WR-WC-PF-2	Precipitation and Infiltration Analysis	0
WR-WC-PF-3	Unit Hydrograph – Post Project Condition	0
WR-WC-PF-4	Flood Hydrograph	0
WR-WC-PF-5	Reservoir Routing – SPRAT Computer Runs	1
WR-WC-PF-6	Backwater Analysis – Water Elev. At Plant Site – PMF	1
WR-WC-PF-7	Wind Wave Runup and Plant Flooding	1

Section 1R04: Equipment Alignment

Procedures

Number	Title	Revision
AP 21G-001	Control of Locked Component Status	72
CKL EC-120	Fuel Pool Cooling and Cleanup System Normal Valve Lineup/Breaker Checklist	14A
CKL EF-120	Essential Service Water Valve, Breaker and Switch Lineup	54
CKL KJ-121	Diesel Generator NE01 and NE02 Valve Checklist	39
STS EC-100B	Spent Fuel Pool Cooling Pump “B” Inservice Pump Test	28A
STS KJ-005B	Manual/Auto Start, Sync & Loading of EDG [Emergency Diesel Generator] NE02	62

Drawings

Number	Title	Revision
M-12EC01	Piping & Instrumentation Diagram Fuel Pool Cooling and Clean-Up System	22
M-12EC02	Piping & Instrumentation Diagram Fuel Pool Cooling and Clean-Up System	07
M-12EF01	Piping & Instrumentation Diagram Essential SVC Water System	29
M-12EF02	Piping & Instrumentation Diagram Essential Service Water System	42
M-12JE01	Piping & Instrumentation Diagram Emergency Fuel Oil System	19
M-12KJ04	Piping & Instrumentation Diagram Standby Diesel Generator "B" Cooling Water System	18
M-12KJ05	Piping & Instrumentation Diagram Standby Diesel Generator "B" Intake Exhaust, F.O. & Start Air Sys.	17
M-12KJ06	Piping & Instrumentation Diagram Standby Diesel Generator "B" Lube Oil System	21

Condition Reports

108258	108285	108783	108802

Miscellaneous

Number	Title	Date
D-KJ-B-004	Clearance Order C-22, KJ01C Starting Air Compressor, 511 Diesel Generator Building/Elev 2000-0 / Area 1	December 14, 2016

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
AP 10-106	Fire Preplans	17
E-1F9905	Fire Hazard Analysis	8

Section 1R06: Flood Protection Measures

Drawings

Number	Title	Revision
M-12KC01	Piping & Instrumentation Diagram Fire Protection Turbine Building	20
M-12KC02	Piping & Instrumentation Diagram Fire Protection System	22
M-12KC04	Piping & Instrumentation Diagram Fire Protection Halon System	03
M-12KC05	Piping & Instrumentation Diagram Fire Protection System	03

Miscellaneous

Number	Title	Revision
FL-08	Control Building Flooding	2

Section 1R08: Inservice Inspection Activities

Procedures

Number	Title	Revision
AI 16F-001	Evaluation Of Boric Acid Leakage	10
AI 16F-002	Boric Acid Leakage Management	10
AP 16F-001	Boric Acid Corrosion Control Program	8
AP 25B-200	Radiography Guidelines	15
CWD-1	Control of Welding Documentation	14
FMC-1	Filler Material Control	10
GWS-ASME	ASME General Welding Standard	7
GWS-AWS	AWS Structural Steel	3
GWS-SM	Sheet Metal General Welding Standard	3
GWS-SW	Stud Welding General Welding Standard	2
LMT-08-PDI-UT-1	Curtiss-Wright Nuclear – LMT. Ultrasonic Examination of Ferritic Piping Welds	0
LMT-08-PDI-UT-2	Curtiss-Wright Nuclear – LMT. Ultrasonic Examination of Austenitic Piping Welds	0
LMT-08-PDI-UT-3	Curtiss-Wright Nuclear – LMT. Ultrasonic Through Wall Sizing in Piping Welds	0
LMT-08-UT-002	Curtiss-Wright Nuclear – LMT. Ultrasonic Thickness Measurements and Surface Contouring	0

Procedures

Number	Title	Revision
LMT-08-UT-004	Curtiss-Wright Nuclear – LMT. Ultrasonic Examination of Vessel Welds and Adjacent Base Metal > 2.0" in Thickness	0
LMT-08-UT-119	Curtiss-Wright Nuclear – LMT. Conventional Ultrasonic Instrument Linearity	1
PQ-1	Welding/Brazing Procedure Qualification	8
QCP-20-501	Visible Dye Penetrant Examination	10
QCP-20-502	Magnetic Particle Examination AC/DC Yoke and AC Coil Techniques	9
QCP-20-508	Radiographic Examination of Welds	4B
QCP-20-519	IWE/IWL Visual Examination	9
QCP-20-541	VT-3 Visual Examination	3
VE-1	Visual Examination of Welds	3
WDI-SSP-1320	Manual Examination of Wolf Creek (SAP) Reactor Vessel Head Penetrations	0
WDI-STD-1040	Procedure for Ultrasonic Examination of Reactor Vessel Head Penetrations	13
WDI-STD-1041	Reactor Vessel Head Penetration Ultrasonic Examination Analysis	11
WPD-1	Welding Program Description	6
WPS1-0808T01	Welding Procedure Specifications, Gas Tungsten Arc Welding of P8 materials, as-welded	2
WQ-1	Qualification of Welders and Brazers	15

Drawings

Number	Title	Revision
M-06EJ-03	Residual Heat Removal System Aux. Building "A & B" Train	2
M-06EM01 Sht. 2	High Pressure Coolant Injection System Aux Building. Hanger # 0-EM01-H015/111(Q)	1
M-15EJ-03	Hanger location Dwg. Residual Heat Removal System Aux Building "A" and "B" Train	10
M-15EJ04	Hanger Location Dwg. Residual Heat Removal System Reactor Building	22
M-16EM01 Sht. 1	High Pressure Coolant Injection System Aux Building. Pipe Support Dwg. A003/121	2

Drawings

Number	Title	Revision
M-16EM01 Sht. 2	High Pressure Coolant Injection System Aux Building. Pipe Support Dwg. C016/111	5
M-16EM03 Sht. 1	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. A005/231	6
M-16EM03 Sht. 1	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. C012/232	4
M-16EM03 Sht. 2	High Pressure Coolant Injection System Reactor Building. Hanger # 1-EM03-C033/232(Q)	4
M-16EM03 Sht. 2	High Pressure Coolant Injection System Reactor Building. Hanger # 1-EM03-R015232 (Q)	2
M-16EM03 Sht. 2	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. C012/232	4
M-16EM03 Sht. 2	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. C013/232	6
M-16EM03 Sht. 2	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. R018/232	5
M-16EM05 Sht. 2	High Pressure Coolant Injection System Reactor Building. Pipe Support Dwg. R001/231	3
M-189-50EJ-02-02	Residual Heat Removal "B" Train RHR Pump Discharge	1
M-189-50EJ-02-04 Sht 3	Residual Heat Removal "B" Train RHR Pump Discharge and Safety Injection Loop No. 2 & 3	0
M-189-50EM-03-05 Sht. 1	High Pressure Coolant Injection Safety Injection Pumps to RHR System	1
M-240-00009 W19	Valve Assembly ¾ Diaphragm, "Y" Type Globe	0
M-703-00054 W11 Sht. 1	Westinghouse Electric Corp., Closure Head (SAP) General Assembly (Drw. 1455E85)	1

Relief Request

Numbers	Title	Date
ET 12-0010	Docket 50-482: 10 CFR 50.55a Request No. 13R-07, Relief from ASME Code Case N729-1 Requirements for Examination of Reactor Vessel Head Penetration Welds	July 2, 2012

Relief Request Numbers	Title	Date
ET 12-0024	Docket 50-482: Response to Request for Additional Information Regarding 10 CFR 50.55a Request No. 13E-07, “Relief from ASME Code Case N-729-1 Requirements for Examination of Reactor Vessel Head Penetration Welds”	October 15, 2012
ET 16-0005	Docket No. 50-482: Wolf Creek Generating Station Inservice Inspection Plan and 10 CFR 50.55a Request Nos. 14R-01 and 14R-02 for the Fourth Inservice Inspection Program Interval	February 23, 2016
ET 16-0028: Letter: McCoy to NRC	Docket No. 50-482: Response to Verbal Request for Additional Information Related to Relief Request No. 14R-03, Request for Relief from Paragraph -3200(b) of ASME Code Case N-729-1 for Reactor Vessel Head Penetration Nozzle Welds	October 20, 2016
ET 16-0030: Letter: McCoy to NRC	Docket No. 50-482: Relief Request No. 14R-03, Request for Relief from Paragraph-3200(b) of ASME Code Case N-729-1 for Reactor Vessel Head Penetration Nozzle Welds and Relief Request 14R-04, Request for Relief from the requirements of ASME Code Case N-729-1	November 1, 2016
Letter: Markley to Heflin	Wolf Creek Generating Station- Request for Relief Nos. 4VR-01 and 4GR-01 for the Fourth 10-Year Inservice Testing Program Interval (TAC NO. MR4991 and MF5005)	August 10, 2015
Letter: Oesterle to Heflin.	Wolf Creek Generating Station- Request for Relief Nos. 13R-08 and 13R-09 for the Third 10-Year Inservice Inspection Program Interval (TAC NOS. MR3321 and MR 3322)	December 10, 2014
WO # 14-0002	Docket 50-482: 10 CFR 50.55a Request Nos. 13R-08 and 13R-09 for the Third Inservice Inspection Program Interval	January 8, 2014
WO # 16-0052	Docket No. 50-482: Relief Request No. 14R-03, Request for Relief from Paragaph-32000(b) of ASME Code Case N-729-1 for Reactor Vessel Head Penetration Nozzle Welds and Relief Request No. 14R04, Request for Relief from the Requirements of ASME Code Case N-729-01	October 11, 2016

Condition Reports

92498	92499	92747	92749	92750
92754	92756	92759	92760	92763
92928	93748	93749	94190	94193
94365	94366	94621	94642	94969
95209	95476	95481	95986	95987
95988	95989	95990	96262	96387
97477	97518	98260	98414	98492
98495	98707	98809	99182	99323
99356	99591	99592	99593	99594
99778	99781	100478	100519	101285
101521	101522	101756	101759	101845
102130	102143	102627	102647	102737
102960	103078	103278	103407	103415
103416	103447	103448	103459	103460
103461	103463	103464	103465	103487
103674	103705	104691	104692	105231
105232	105233	105309	105516	105651
105718	105726	107099		

Condition Reports (Generated during the outage)

108021	108047	108076	108149	108151
108152	108154	108251	108256	108277
108282	108293	108294	108327	108328
108362	108368	108369	108370	108371
108378	108384	108386	108387	108388
108389	108390	108391	108392	108393
108410	108414	108415	108424	108445
108446	108455	108456	108469	108473
108479	108495	108523	108526	108527
108528				

Work Orders

13-368504-002	13-368504-003	13-368504-004	13-368504-005	15-401539-000
15-401757-000	15-401758-000	15-401758-001	15-401758-003	15-401758-004
15-401758-007	15-401758-009	15-401980-005	15-401980-008	15-402484-000
15-402486-000	15-402898-000	15-404248-000	15-406147-000	15-406147-002
15-407543-000	15-407544-000	15-407545-000	15-407865-000	15-409675-000
16-410214-001	16-410636-000	16-410796-000	16-410796-001	16-410796-002
16-411319-000	16-411319-005	16-411319-006	16-413009-000	16-414471-000
16-414658-000	16-414659-000	16-416291-000	16-416292-000	16-417220-000
16-417262-005	16-418573-000			

Miscellaneous

Number	Title	Revision/Date
2008 Addenda – Section V	2007 ASME Boiler and Pressure Vessel Code – Nondestructive Examination	July 1, 2008
2008 Addenda – Section XI	2007 ASME Boiler and Pressure Vessel Code – Section XI	July 1, 2008
Boric Acid Quarterly Report: 1st Quarter, 2015	Boric Acid Quarterly Report (Inside Containment). Inspection Performed March 25, 2015. Report Generated: April 27, 2015. Report Date Range: January 1, 2015 - March 31, 2015	April 29, 2015
Boric Acid Quarterly Report: 3rd Quarter, 2015	Boric Acid Quarterly Report (All Leaks). Inspection Performed September 16, 2015. Report Generated: October 20, 2015. Report Date Range: July 1, 2015 – November 30, 2016	October 29, 2016
FCN-012655	Water Jet Peening of Reactor Vessel Nozzles and Bottom Mounted Nozzles to Mitigate PWSCC	2
M-706B-0001	Design Document Change Notice – Implementation of Water Jet Peening of Reactor Vessel	3
QH-2014-0871	Quick Hit Self-Assessment of the Non-Accredited WCNOC Welding Program Training	September 18, 2014
Report # 000998.401	Mitigation of Reactor Vessel Hot and Cold Leg Nozzle DMWs and Reactor Vessel Bottom Mounted Nozzles and Associated DVWs by Water Jet Peening	1
SA-2012-0023	ISI Program Self-Assessment	March 8, 2012
WCRE-30	Inservice Inspection Program Plan – Wolf Creek Generating Station Interval 4	1

Section 1R11: Licensed Operator Requalification Program

Procedures

Number	Title	Revision
GEN 00-003	Hot Standby To Minimum Load	99
RXE 01-002	Reload Low Power Physics Testing	25A

Drawings

Number	Title	Revision
M-12BG01	Piping & Instrumentation Diagram Chemical And Volume Control System	19

Miscellaneous

Number	Title	Revision/Date
	Post Refuel 21 Team Training Scenario	0
APF 21-001-02	Mode 3, Day Shift	November 19, 2016
APF 21-001-02	Mode 3, Night Shift	November 19, 2016
APF 21-001-02	Mode 4, Day Shift	November 18, 2016

Section 1R12: Maintenance Effectiveness

Procedures

Number	Title	Revision
AI 28A-023	Evaluation of Maintenance Rule Functional Failure CRs	4
ALR 00-098B	SSE [Significant Seismic Event]	14
ALR 00-098D	OBE [Operating Basis Earthquake]	15A
ALR 00-098E	Seismic Recorder On	15
AP 23M-001	WCGS Maintenance Rule Program	12
AP 26C-004	Operability Determination And Functionality Assessment	33
APF 06-002-01	Emergency Action Levels	17A
EDI 23M-050	Engineering Desktop Instruction – Monitoring Performance to Criteria and Goals	9
OFN SG-003	Natural Events	30
STS PE-040E	RPV [Reactor Pressure Vessel] Head Visual Inspection	6

Procedures

Number	Title	Revision
STS SG-001	Seismic Instrumentation Channel Test	17A
SYS GK-200	Non-Functional Class 1E A/C Unit	36

Drawings

Number	Title	Revision
E-12573-179-031-01	Westinghouse Electric Corporation, 173” ID PWR, Stud, Nut, and Washer	1
M-703-00054	Westinghouse Electric Corporation, Closure Head (SAP) General Assembly	W11
M-703-00055	Westinghouse Electric Corporation, Closure Head (SAP) General Assembly	W02
M-709-00099	Control Rod Drive Mechanism Housing Details	W01
M-797-00146	Wolf Creek Simplified Head Assembly, Reactor Head Assembly	W05

Condition Reports

51639	70745	91473	92075	93798
105971	105973	106262	106416	106417
106438	106904	107185	109166	109175
109242	109249	109283	109284	109312
109351	109352	107686		

Work Orders

16-416293	16-416523	16-416867	16-419401	15-402918

Miscellaneous

Number	Title	Revision/Date
AIF 28A-100-12	Wolf Creek Generating Station Basic Cause Evaluation	4
APF-05D-001-02	Calculation Change Notice - ASME Code Design Stress Report For Wolf Creek Power Plant Reactor Vessel (RBB01) CCN No. BB-S-018-000-CN005	November 14, 2016
CENC-1313	Analytical Report For Kansas Gas & Electric Company Wolf Creek Nuclear Power Plant Reactor Vessel	February 1978

Miscellaneous

Number	Title	Revision/Date
QH-2016-1324	2016 Maintenance Rule Program (a)(3) Assessment Report at Wolf Creek Generating Station	June 6 – July 14, 2016
SG-01	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-01	
SG-02	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-02	
SG-03	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-03	
SG-04	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-04	
SG-05	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-05	
SG-06	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-06	
SG-07	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-07	
SG-08	Maintenance Rule Final Scope Evaluation, Seismic Instrumentation System – SG-08	

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

Procedures

Number	Title	Revision
AP 21C-001	Wolf Creek Substation	18
AP 22B-001	Outage Risk Management	18A

Drawings

Number	Title	Revision
KD-7496	Switchyard One Line Diagram	61

Condition Reports

109467	109469

Work Orders

16-419750

Miscellaneous

Title	Date
RF21 OCC Shift Update (Day)	November 18, 2016
RF21 OCC Shift Update (Night)	November 17, 2016

Section 1R15: Operability Evaluations

Procedures

Number	Title	Revision
AP 21-001	Conduct of Operations	77
AP 26C-004	Operability Determination and Functionality Assessment	33
AP 28-001	Operability Evaluations	24
STS PE-040E	RPV [Reactor Pressure Vessel] Head Visual Inspection	6

Drawings

Number	Title	Revision
E-12573-179-031-01	Westinghouse Electric Corporation, 173” ID PWR, Stud, Nut, and Washer	1
M-12EP01	Piping & Instrumentation Diagram Accumulator Safety Injection	15
M-703-00054	Westinghouse Electric Corporation, Closure Head (SAP) General Assembly	W11
M-703-00055	Westinghouse Electric Corporation, Closure Head (SAP) General Assembly	W02

Condition Reports

93798	108699	108996	109175	109283

Work Orders

16-418830

Miscellaneous

Number	Title	Revision/Date
APF-05D-001-02	Calculation Change Notice - ASME Code Design Stress Report For Wolf Creek Power Plant Reactor Vessel (RBB01) CCN No. BB-S-018-000-CN005	November 14, 2016

Miscellaneous

Number	Title	Revision/Date
Colt-Pielstick-PCS.5V	Instructions – Engine Control System	August 2010
M-706-00050	Analytical Report for Kansas Gas & Electric Company, Wolf Creek Nuclear Power Plant, Reactor Vessel	December 5, 1978
OE EP-16-007	Operability Evaluation associated with CR 108996	0
OE KJ-16-005	Operability Evaluation associated with CR 108699	0

Section 1R18: Plant Modifications

Procedures

Number	Title	Revision
AI 05-025	Post Modification Testing Plan	2
CNT-MM-401	Installation/Modification and Inspection of Mechanical Equipment	2

Drawings

Number	Title	Revision
E-13GK13A	Schematic Diagram Class 1E Electrical Equipment A/C Unit	07
E-13GK13B	Schematic Diagram Class 1E Electrical Equipment A/C Unit Panels	2
M-12GK03	Piping and Instrumentation Diagram Control Building H.V.A.C	21
M-622.1-00003	1E Electrical Equipment Air Handling Unit Layout SGK05A(3-4-1) & SGK05B(3-4-1)	6
M-622.1A-00001	SGK05A & SGK05B Air Conditioner Refrigeration Schematic	W12
M-622.1A-00002	SGK05A & SGK05B Air Conditioner Electrical Schematic	W13
M-622.1A-00050	SGK05A & B Water Cooled Condenser Baffle Detail and Location	W03
M-622.1A-00315	SGK05A & SGK05B Air Conditioner Main Enclosure Panel Layout	01
SK-M-13GK01	Small Piping Isometric Room Coolers and Compressors Vents and Drains – Aux. Bldg. [Auxiliary Building]	2

Condition Reports

100802	107185	107789	108049	108416

Condition Reports

108425	108493	108553	108554	108583
108604	108724	108737	108751	108753
108871	108917	108995	109018	109076

Work Orders

12-357275

Miscellaneous

Number	Title	Revision/Date
AIF 05-025-01	Post Modification Testing Plan – SGK05A&B PMT – Bitzer Compressor Replacement	April 25, 2016
APF 05A-001-01	Design Input Data Sheet (DIDS) for DCP [Design Change Package] 0414566	July 7, 2015
APF 05-005-05	Return to Operations – DCP [Design Change Package] 014566	November 4, 2016
APF 26A-003-02	50.59 Screen for DCP [Design Change Package] 0414566 - SGK05A&B PMT – Bitzer Compressor Replacement (Revision 4)	November 3, 2016
M-622.1A-VDS-1.07	SGK05A/B Condenser Vendor Data Sheet	April 11, 2005
M-622.1A-VDS-1.08	SGK05A/B Cooling Coil and Compressor Vendor Data Sheet	W02
M-622.1A-VDS-1.09	SGK05A/B Water Regulating Valve Vendor Data Sheet	W01
M-622.1A-VDS-1.10	SGK05A/B Electric Motor Vendor Data Sheet	W02
013404	Design Change Package for Drain System Modification to Prevent Steam in AFW Pump Rooms	0 to 9
013404	Post Modification Testing Plan for Design Change Package 013404	3
014566	Field Change Notice for Change Package 014566 – SGK05A&B PMT – Bitzer Compressor Replacement	4
015167	Minor Change Package for Protective Cover over Aux Boiler Room Drain Hub	0 & 1

Section 1R19: Post-Maintenance Testing

Procedures

Number	Title	Revision
AI 16-003	Temporary Grounding	6
AP 16E-002	Post Maintenance Testing Development	18A
AP 21-001	Conduct of Operations	77
AP 26C-004	Operability Determination And Functionality Assessment	32
MPE NE-004	Alternator Inspection	13A
MPM M018Q-01	Standby Diesel Generator Inspection	23A
STS AL-103	TDAFW Pump Inservice Test	68
STS EM-100B	Safety Injection Pump “B” Inservice Pump Test	34
STS KJ-005B	Manual/Auto Start, Sync & Loading Of EDG NE02	62
STS KJ-011B	EDG NE02 24 Hour Run	37
SYS KJ-123	Post Maintenance Run Of Emergency Diesel Generator A	64
SYS KJ-124	Post Maintenance Run Of Emergency Diesel Generator B	64
SYS KJ-124	Post Maintenance Run Of Emergency Diesel Generator B	65

Drawings

Number	Title	Revision/Date
6998D62	Colt Industries Type “WNR”-Volt Reg. & Excitation System	March 16, 1978
E-13KJ03A	Schematic Diagram Diesel Gen. KKJ01B Engine Control (Start/Stop Circuit)	18
E-13KJ03B	Schematic Diagram Diesel Generator KKJ01B Engine Control (D/G Trips)	6
E-13KJ04	Schematic Diagram Diesel Generator KKJ01B Annunciator And Miscellaneous Circuits	8
E-13KJ07	Diesel Generator KKJ01B Governor Control	6
E-13NE02	Standby Generation System Three Line Meter And Relay Diagram	16
E-13NE11	Schematic Diagram 4.16KV DG NE02 Feeder Breaker 152NB0211	19
E-13NE13	Schematic Diagram Diesel Generator KKJ01B Exciter/Voltage Control	14

Drawings

Number	Title	Revision/Date
M-12EM01	Piping & Instrumentation Diagram High Pressure Coolant Injection System	43
M-12EM02	Piping & Instrumentation Diagram High Pressure Coolant Injection System	22
M-12EG02	Piping & Instrumentation Diagram Component Cooling Water System	27

Condition Reports

108258	108783	108802

Work Orders

12-357275	15-399836	16-410087	16-413640	16-413666
16-413676	16-413677	16-419074		

Miscellaneous

Number	Title	Revision/Date
	B EDG Trip CR 108783 Timeline	October 14, 2016
AIF 28-001-01	Event Review Team Summary associated with CR 108783	October 25, 2016
APF 15C-002-01	Procedure Cover Sheet - Post Maintenance Run Of Emergency Diesel Generator B	October 25, 2016
APF 15C-004-04	On The Spot Change – Alternator Inspection	October 19, 2016
APF 15C-004-04	On The Spot Change – Standby Diesel Generator Inspection	September 28, 2016
APF 21-001-02	On-Coming CRS/WC SRO/RO/BOP Review	October 14, 2016
APF 21E-001-14	Manual Clearance Order/Work Order Holders List – Troubleshoot NE002 Diesel	3
APF 23E-001-01	EDG Start Log Form	October 27, 2016
APF 24E-001-03	Work Order Bill Of Materials – NE002	3
APF 29B-003-01	Surveillance Test Routing Sheet – Manual/Auto Start, Synchronization & Loading Of Emergency D/G NE02	October 27, 2016

Miscellaneous

Number	Title	Revision/Date
Colt-Pielstick-PC-2.5V	Renewal Parts List	
M-018-00309	Instruction Manual For Emergency Diesel Generator System	W136
MGEF EOOP-05-02	Transformer Insulation Resistance Testing Procedure Sign-Off Sheet	2
MGEF-TL-001-01	Termination & Connector Sign-Off Sheet	10

Section 1R20: Refueling and Other Outage Activities

Procedures

Number	Title	Revision
AP 10-104	Breach Authorization	35
AP 12-003	Foreign Material Exclusion	15
AP 14A-003	Scaffold Construction and Use	24
AP 22B-001	Outage Risk Management	18A
GEN 00-008	RCS [Reactor Coolant System] Level Less Than Reactor Vessel Flange Operations	29
GEN 00-009	Refueling	38
STS IC-727B	Solid State Protection System Train B Response Time	3
SYS GN-120	Containment Cooling System Operation	40
SYS PN-200	Energizing and Deenergizing Inverters PN09 or PN10	19

Drawings

Number	Title	Revision
D-283137	Model D-100-160 3" Class 1500 Valve Assembly ASME Code Class-2	3
M-12BB01	Piping & Instrumentation Diagram Reactor Coolant System, Sheet 1	32
M-12BG03	Piping and Instrumentation Diagram Chemical & Volume Control System	48
M-703-00054	Westinghouse Electric Corporation Water Reactor Division, PGH. PA. U.S.A. Closure Head (SAP) General Assembly	W11

Condition Reports

99097	101020	101867	102519	102596
103481	106111	106763	107102	107340
107341	107686	107693	107719	107720
107721	107722	107838	107842	107847
107903	107904	107958	107976	108061
108135	108146	108151	108152	108154

Work Orders

16-418198

Miscellaneous

Number	Title	Revision/Date
ALARA Review Package 20161102	Incore Tunnel Inspections & Maintenance. This RWP to be used for all entries into Incore Tunnel	September 16, 2016
ALARA Review Package 20162601	Routine Outage Access (No Locked High Radiation Areas Access)	July 29, 2016
ALARA Review Package 20162602	Routine Outage Access (No Very High Radiation Areas Access)	July 29, 2016
APF 22B-001-04	Shutdown Risk Assessment Mode 4	8
APF 22B-001-07	Shutdown Risk Assessment Mode 5 or 6, RCS Lowered Inventory (indicated Level At Or Below 100.1 In. And Higher than 3 Ft. Below the Reactor Vessel Flange) OR RCS Reduced Inventory – Below 3 Ft. Below Vessel Flange (Indicated Level of <64.1 In.)	
APF 22B-001-09	No Mode – Defueled	9
APF 22B-001-10	Shutdown Safety Function Status & Assessment Summary	9
CO1232103	Work Order Holder Training	8A
CO: R-OP-S-010		
LER 2015-004-00	Licensee Event Report: Inadequate Procedure Error Results in Two Containment Isolation Valves being in a Condition Prohibited by Technical Specifications	0

Miscellaneous

Number	Title	Revision/Date
LER 2015-004-01	Licensee Event Report: Incorrect Decision Results in Two Containment Isolation Valves being in a Condition Prohibited by Technical Specifications	1
LER 2016-001-00	Licensee Event Report: Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability	0
LER 2016-001-01	Licensee Event Report: Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability	1
Radiation Work Permit 20161102	Incore Tunnel Inspections & Maintenance. This RWP to be used for all entries into Incore Tunnel	3
Radiation Work Permit 20162601	Routine Outage Access (No Locked High Radiation Areas Access)	0
Radiation Work Permit 20162602	Routine Outage Access (No Very High Radiation Areas Access)	0
SCA-97-0089	PN010	1
SCA-97-0090	PN010	0

Section 1R22: Surveillance Testing

Procedures

Number	Title	Revision
I-ENG-026	Local Leak Rate Test	7
STN IC-252B	Calibration Of RHR Pump B Mini Flow Valve Control Switch	9
STS CV-210B	ECCS SI And RHR Inservice Check Valve Test	20
STS EJ-100B	RHR System Inservice Pump B Test	45
STS GP-006	CTMT Closure Verification	23A
STS IC-806B	4KV Undervoltage – Loss Of Voltage – Channel Calibration Of 1 Second Time Delay Circuit NB02	5
STS KJ-001A	Integrated D/G And Safeguards Actuation Test – Train A	61
STS PE-155	LLRT Valve Lineup For Penetration 55	6
SYS EC-120	Fuel Pool Cooling And Cleanup System Startup	54

Drawings

Number	Title	Revision
M-12EJ01	Piping and Instrumentation Diagram Residual Heat Removal System	53
M-12EP01	Piping & Instrumentation Diagram Accumulator Safety Injection	14
M-12HE01	Piping & Instrumentation Diagram Boron Recycle System	04
M-12HE02	Piping & Instrumentation Diagram Boron Recycle System	12

Condition Reports

108965	108973

Work Orders

15-403314	16-419150

Miscellaneous

Number	Title	Date
APF 15C-004-04	On The Spot Change – Integrated D/G And Safeguards Actuation Test – Train A	November 14, 2016
APF 29B-003-01	Surveillance Test Routing Sheet – CTMT Closure Verification	September 22, 2016
APF 29B-003-01	Surveillance Test Routing Sheet – ECCS SI & RHR Inservice Check Valve Test	November 8, 2016
APF 29B-003-01	Surveillance Test Routing Sheet – Integrated D/G & Safeguards Actuation Test Train A	November 15, 2016
APF 29B-003-01	Surveillance Test Routing Sheet – LLRT Valve Lineup For Penetration 55	October 01, 2016
APF 29B-003-01	Surveillance Test Routing Sheet – RHR System Inservice Pump B Test	November 2, 2016
M-721-00099	IM for Residual Heat Removal Pump – Manual Breakdown Complete: Test Performance Curve No. 37768A	September 27, 1977

Section 2RS1: Radiological Hazard Assessment and Exposure Controls

Procedures

Number	Title	Revision
AP 19C-002	Special Nuclear Material Safeguards and Accountability	15

Procedures

Number	Title	Revision
AP 19D-100	Radioactive Source Program	5
AP 25A-001	Radiation Protection Manual	16A
AP 25A-200	Access to Locked High or Very High Radiation Areas	28
AP 25B-100	Radiation Work Guidelines	49
AP 25B-200	Radiography Guidelines	14
FHP 02-019	Spent Fuel Pool Exclusion Area	5
RPP 02-105	RWP [Radiation Work Permit]	43A
RPP 02-215	Posting of Radiological Controlled Areas	32
RPP 02-605	Control & Inventory of Radiation Sources	17
RPP 02-610	Receipt of Radioactive Material	10

Condition Reports

92608	93288	93882	94955	95099
95253	95426	96206	96651	100071
100433	102344	102581	102941	103884
103887	105081	105276	106072	106570
108510	108513	108517		

Radiological Work Permits

Number	Title	Revision
20163049	Under RV Head Full Body Entry to Support RV Head Inspections	1
20163050	Remote Under RV Head Thermal Sleeve Inspections	0
20164482	Expanded Scope Activities Remove/Install Canopy Seal Weld Clamps	1
20165048	RP / Decon Group Activities for Containment High Contamination Decon of the Reactor Head	0
20165059	RP / Decon Group Activities for Aux Building High Contamination Decon and Containment High Contamination Decon	4

Air Sample Surveys

Number	Title	Date
16-0842	Aux Building 2000' Charging Pump Work	October 15, 2016
16-0845	CTMT 2047' Reactor Vessel Head Work	October 15, 2016
16-0866	CTMT Inside Bio-Shield – Steam Generators	October 15, 2016
16-0988	CTMT 2047' Water Jet Peening Project	October 19, 2016
16-1930	Aux Building 1974' Valve 8405A	October 17, 2016

Audits, Self-Assessments, and Surveillances

Number	Title	Date
K-15-002 Audit 16-01-RP/PC	Radiological Protection and Process Control	February 24, 2016
QA-Audit 16-01-RP-PC	Radiological Protection and Process Control	
QS-2016-1728	QA Observation of Decon of Stud Hole Plugs	May 24, 2016
QS-2016-1790	QA Observation of RP Department Activities	June 28, 2016

Miscellaneous

Number	Title	Date
	National Source Tracking System - Confirmation Form 2015 Annual Inventory Reconciliation	January 5, 2015
	National Source Tracking System - Confirmation Form 2016 Annual Inventory Reconciliation	January 7, 2016
STS HP-001	Sealed Source Surveillance Tests	January 14, 2016
STS HP-001	Sealed Source Surveillance Tests	July 6, 2016
WC011005	Small Article Monitor (SAM) 11 Calibration	January 27, 2016
WC011006	Small Article Monitor (SAM) 11 Calibration	January 28, 2016
WC093361	Small Article Monitor (SAM) 12 Calibration	January 21, 2016

Section 2RS2: Occupational ALARA Planning and Controls

Procedures

Number	Title	Revision
AP 25A-401	ALARA Program	24
AP 25A-410	ALARA Committees	22
RPP 02-105	Radiation Work Permits (RWP)	44

Audits and Self-Assessments

Number	Title	Date
16-01-RP/PC	Quality Assurance Audit Report: Radiological Protection/Process Control	February 24, 2016
QS-2016-1699	Quality Surveillance: Senior ALARA Committee Meeting	April 12, 2016
QS-2016-1784	Quality Surveillance: ALARA Brief for Demin Resin Sluice	June 28, 2016

Condition Reports

96810	97400	97558	97635	99421
100459	102809	102813	102823	107129
108151	108314	108435		

Radiation Work Permits

Number	Title	Revision
163049	Under RV Head Full Body Entry to Support RV head Inspections	1
163050	Remote Under RV Head Thermal Sleeve Inspections	0
163055	Water Jet Peening (WJP) Project	1
163220	Primary Side Steam Generator Eddy Current Testing	1
164010	Excess Letdown Heat Exchanger (EBG02) Tube Bundle.	2
164200	Secondary Side S/G Sludge Lance	0
164420	Scaffolding	2
164482	Expanded Scope Activities Remove/Install Canopy Seal Weld Clamps	1
165048	RP / Decon Group Activities for Containment High Contamination Decon of the Reactor Head	0

ALARA Review Packages and In-Progress Reviews

Number	Title	Date
163049	ALARA Review Package	October 15, 2016
163050	ALARA Review Package	October 12, 2016
163055	ALARA Review Package	September 22, 2016
163220	ALARA Review Package	September 8, 2016
163220	In-Progress Review (50%)	October 5, 2016
164010	ALARA Review Package	August 16, 2016
164010	In-Progress Review (Exposure Estimate Revision)	September 27, 2016
164200	ALARA Review Package	September 6, 2016
164200	In-Progress Review (50%)	October 11, 2016
164420	ALARA Review Package	August 25, 2016
164420	In-Progress Review (80%)	October 12, 2016
164482	ALARA Review Package	October 7, 2016
164482	In-Progress Review (80%)	October 18, 2016
165048	ALARA Review Package	September 29, 2016

Miscellaneous Documents

Title	Date
ALARA Sub-Committee Meeting Minutes	October 12, 2016
ALARA Sub-Committee Meeting Minutes	October 13, 2016
RF 20 Summary for Radiation Protection	May 13, 2015
Wolf Creek Dose Excellence Plan: 2015 – 2019	August 28, 2015

Section 4OA1: Performance Indicator Verification

Procedures

Number	Title	Revision
AI 26A-007	NRC Performance Indicators	11
AI 26A-008	NRC / INPO / WANO Performance Indicator and MOR Reporting	1

Procedures

Number	Title	Revision
Desktop Guide	NRC / INPO / WANO Performance Indicator Guidelines	1

Condition Reports

102344	103887	108475

Miscellaneous

Number	Title	Revision/Date
	MSPI Indicator Margin Remaining in Green	September 2016
	PI Summary (4th quarter 2014 through 3rd quarter 2016)	October 10, 2016
	Unavailability For Residual Heat Removal (EJ-02)	October 19, 2016
102201	Functional Failure Determination Checklist	February 11, 2016
102680	Functional Failure Determination Checklist	March 31, 2016
103123	Functional Failure Determination Checklist	March 23, 2016
103274	Functional Failure Determination Checklist	March 30, 2016
AI, AP, FC-1	System Health Report	July 1, 2015, through September 30, 2015
AI, AP, FC-1	System Health Report	October 1, 2015, through December 31, 2015
AI, AP, FC-1	System Health Report	January 1, 2016, through March 31, 2016
AI, AP, FC-1	System Health Report	April 1, 2016, through June 30, 2016
AI, AP, FC-1	System Health Report	July 1, 2016, through December 31, 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Residual Heat Removal System - Unavailability Index	September 2016

Miscellaneous

Number	Title	Revision/Date
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Residual Heat Removal System - Unreliability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Emergency AC Power System - Unavailability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Emergency AC Power System - Unreliability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Heat Removal System - Unavailability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Heat Removal System - Unreliability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI High Pressure Injection System - Unavailability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI High Pressure Injection System - Unreliability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Cooling Water System - Unavailability Index	September 2016
Consolidated Data Entry 4.0	MSPI Derivation Report – MSPI Cooling Water System - Unreliability Index	September 2016
EJ	System Health Report	July 1, 2015, through September 30, 2015
EJ	System Health Report	October 1, 2015, through December 31, 2015
EJ	System Health Report	January 1, 2016, through March 31, 2016
EJ	System Health Report	April 1, 2016, through June 30, 2016
KJ, NE, JE	System Health Report	July 1, 2015, through September 30, 2015

Miscellaneous

Number	Title	Revision/Date
KJ, NE, JE	System Health Report	October 1, 2015, through December 31, 2015
KJ, NE, JE	System Health Report	January 1, 2016, through March 31, 2016
KJ, NE, JE	System Health Report	April 1, 2016, through June 30, 2016
KJ, NE, JE	System Health Report	July 1, 2016, through December 31, 2016
LER 2015-004-00	Licensee Event Report: Inadequate Procedure Error Results in Two Containment Isolation Valves being in a Condition Prohibited by Technical Specifications	0
LER 2015-004-01	Licensee Event Report: Incorrect Decision Results in Two Containment Isolation Valves being in a Condition Prohibited by Technical Specifications	1
LER 2016-001-00	Licensee Event Report: Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability	0
LER 2016-001-01	Licensee Event Report: Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability	1

Section 4OA2: Identification and Resolution of Problems

Procedures

Number	Title	Revision
AI 15D-018	System Administration for Milestone CDAs	0
AI 15D-021	Control of Portable Media and Mobile Devices	7
AI 28A-010	Screening Condition Reports	25B
STS AB-201D	Atmospheric Relief Valve Inservice Valve Test	27A
STS KA-010	N2 Accumulator Inservice Check Valve Test	17

Drawings

Number	Title	Revision
M-12KA05	Piping & Instrumentation Diagram Compressed Air System	7

Condition Reports

24479	25515	25946	97791	97792
99565	99567	99577	99578	99579
99634	99954	100064	100130	100144
100163	100173	100174	100194	100195
100196	100197	106021	106062	106390
107082	107102	107103	107660	108319
108498	108702	108858	108859	108860
109024	109134	109166	109284	109312
109331	109351	109352	109911	109912
110503				

Work Orders

15-404929	15-406200	15-407702	16-419401

Miscellaneous

Number	Title	Revision/Date
	Basic Engineering Disposition – Evaluation of Atmospheric Relief Valve Leakage Identified in RF21	November 22, 2016
	Defensive Strategy Diagram	
12-1120-TR-001	Altran Solutions Laboratory Evaluation of Five AOV Diaphragms	0
13-094326	Wolf Creek Receiving Inspection Report for PO#761624	February 17, 2013
110-P0001	Commercial Grade Dedication for Diaphragms for Masoneilan Spring Diaphragm Pneumatic Actuators	0
KA-03-W	KA System Back-up Nitrogen Accumulators Capacity Calculation	3
MCP 14984	Remove Emergency Operations Facility Workstation	0

Section 4OA3: Event Follow-Up

Condition Reports

20476	62093	85015	88665	95726
2007-4719				

Miscellaneous

Number	Title	Revision/Date
	DG Diode RCA 88665 Q&A White Paper	March 15, 2016
	Initiating Events Facts and Cause White Paper	
	Post Regulatory Conference White Paper Addressing Outstanding NRC Regulatory Conference Questions	
	Presentation Slides for Regulatory Conference Apparent Violation of Technical Specification 5.4.1a, October 6, 2014 B-EDG Trip and Fire	September 21, 2016
991297	Performance Improvement Request	
992347	Performance Improvement Request	
LER 2016-001-01	Power Potential Transformer Overloading Results in Emergency Diesel Generator Inoperability	1

The following items are requested for the
Occupational Radiation Safety Inspection
at Wolf Creek
October 17-21, 2016
Integrated Report 2016004

Inspection areas are listed in the attachments below.

Please provide the requested information on or before **September 26, 2016.**

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 Martin J. Phalen at (817) 200-1158 or martin.phalen@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: **March 9 to 13, 2015**

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

B. Applicable organization charts

C. Audits, self-assessments, and LERs written since date of last inspection, related to this inspection area

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.

- Radiation Protection Program Description
- Radiation Protection Conduct of Operations
- Personnel Dosimetry Program
- Posting of Radiological Areas
- High Radiation Area Controls
- RCA Access Controls and Radiation Worker Instructions
- Conduct of Radiological Surveys
- Radioactive Source Inventory and Control
- Declared Pregnant Worker Program

F. List of corrective action documents (including corporate and sub-tiered systems) since date of last inspection

- Initiated by the radiation protection organization
- 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 date of last inspection 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)

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).

2. Occupational ALARA Planning and Controls (71124.02)
Date of Last Inspection: **June 22-26, 2015**

A. List of contacts and telephone numbers for ALARA program personnel

B. Applicable organization charts

C. Copies of audits, self-assessments, and LERs, written since date of last inspection, focusing on ALARA

D. Procedure index for ALARA Program

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.

- ALARA Program
- ALARA Committee
- Radiation Work Permit Preparation

F. A summary list of corrective action documents (including corporate and sub-tiered systems) written since date of last inspection, related to the ALARA program. In addition to ALARA, the summary should also address Radiation Work Permit violations, Electronic Dosimeter Alarms, and RWP Dose Estimates

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 work activities greater than 1 rem, since date of last inspection, Include original dose estimate and actual dose

H. Site dose totals and 3-year rolling averages for the past 3 years (based on dose of record)

I. Outline of source term reduction strategy

J. If available, provide a copy of the ALARA outage report for the most recently completed outages for each unit

K. Please provide your most recent Annual ALARA Report.