UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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July 25, 2016

Mr. Brian D. Boles
Site Vice President
FirstEnergy Nuclear Operating Co.
Davis-Besse Nuclear Power Station
5501 N. State Rte. 2, Mail Stop A–DB–3080
Oak Harbor, OH 43449–9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000346/2016002

Dear Mr. Boles:

On June 30, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Davis-Besse Nuclear Power Station. The enclosed report documents the results of this inspection, which were discussed on July 14, 2016, with you and other members of your staff.

Based on the results of this inspection, the NRC has identified three issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. Additionally, a licensee-identified violation is listed in Section 4OA7 of this report. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the NRC Enforcement Policy. These NCVs are described in the subject inspection report.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to: (1) the Regional Administrator, Region III; (2) the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and (3) the NRC Resident Inspectors' Office at the Davis-Besse Nuclear Power Station.

In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspectors' Office at the Davis-Besse Nuclear Power Station.

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

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 4
Division of Reactor Projects

Docket No. 50–346
License No. NPF–3

Enclosure:
IR 05000346/2016002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–346
License No:	NPF–3
Report No:	05000346/2016002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	April 1 through June 30, 2016
Inspectors:	D. Kimble, Senior Resident Inspector T. Briley, Resident Inspector T. Bilik, Senior Reactor Inspector J. Cassidy, Senior Health Physicist J. Rutkowski, Project Engineer N. Valos, Senior Reactor Analyst
Approved by:	J. Cameron, Chief Branch 4 Division of Reactor Projects

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SUMMARY

Inspection Report (IR) 05000346/2016002; 4/1/16 – 6/30/16; Davis-Besse Nuclear Power Station; Identification and Resolution of Problems; Follow-Up of Events and Notices of Enforcement Discretion.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified. Each of the findings were considered non-cited violations (NCVs) of NRC regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using IMC 0609, "Significance Determination Process" dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas" effective date December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," Revision 5, dated February 2014.

NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A self-revealed finding of very low safety significance and an associated NCV of Title 10, *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Action," were identified for the licensee's failure to have adequately addressed an identified adverse trend involving 10 and 15 ampere Gould – Shawmut A25X series fuses. Specifically, the licensee had identified adverse trends related to failures of A25X series fuses in 2005, and again in 2015, and had entered these adverse trends into their corrective action program (CAP) as Condition Reports (CRs) 2005–05314 and 2015–03516. However, the evaluation performed under CR 2015–03516 did not recognize that the fuse failures were occurring much more frequently than originally anticipated and that the previously created Preventative Maintenance (PMs) were not adequate to prevent failures. Additionally, the evaluation did not adequately incorporate industry experience that also identified a trend of failures with the A25X series fuses. Corrective actions by the licensee included replacement of the existing stock of uninstalled A25X series fuses with equivalent fuses of a different style and from a different manufacturer and identification of a plan to replace in-plant installed fuses.

  This finding was of more than minor safety significance because it affected the attribute of equipment performance of the Initiating Events cornerstone of reactor safety, and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it did not represent a deficiency that caused a reactor trip as well as the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feedwater, etc.) The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross cutting aspect of "Resolution" to the finding because the licensee failed to take action to resolve the identified adverse trend associated with premature failures of A25X series fuses in a timely manner. (P.3) (Section 4OA3.1)

- Green. A self-revealed finding of very low safety significance and an associated NCV of Technical Specification (TS) 3.4.13, "Reactor Coolant System (RCS) Operational Leakage," were identified for the failure of a licensee vendor to have ensured that a replacement reactor coolant pump (RCP) seal cavity vent line flexible hose was subjected to adequate quality control testing following manufacture. Specifically, a manufacturing weld defect on the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line was not detected by post-manufacture testing, such that the hose developed a very minor leak during power operations for the reactor operating cycle occurring before the licensee's spring 2016 refueling outage. Corrective actions by the licensee included replacement of the failed flexible hose assembly and revising the procurement requirements for subsequently ordered flexible hose assemblies to include enhanced helium tracer probe leak testing.

  This finding was of more than minor safety significance because it affected the attribute of equipment performance of the Initiating Events cornerstone of reactor safety, and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it was determined that the finding could not have resulted in exceeding the RCS leak rate for a small loss of coolant accident or affected other systems used to mitigate a loss of coolant accident. The inspectors determined that the finding had a cross-cutting aspect in the area of human performance. The inspectors assigned the cross cutting aspect of "Field Presence" to the finding because the licensee failed to ensure that their vendor performed adequate quality control testing following manufacture of a safety-related flexible hose assembly. (H.2) (Section 4OA3.4)

Cornerstone: Mitigating Systems

- Green. An NRC-identified finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified for the licensee's failure to have entered a degraded condition associated with Auxiliary Feedwater (AFW) Train No. 2 into their CAP until challenged by the inspectors. Specifically, a flow transient that occurred on May 7, 2016, and that caused damage to components in the AFW recirculation line during AFW Train No. 2 testing was not entered into the licensee's CAP until May 8, 2016, following challenges from the inspectors. This omission on the part of the licensee's staff had the effect of bypassing certain features of the licensee's CAP associated with evaluating and documenting the operability of safety-related equipment. The physical event and equipment issues were entered into the licensee's CAP as CR 2016–06515 on May 8, 2016, following prompting by the inspectors. Corrective actions taken by the licensee included repairs to all damaged equipment, detailed inspections of AFW Train No. 2, and an engineering analysis into why the event occurred. The matter of the licensee's failure to enter the event into their CAP in a timely manner was documented as CR 2016–06516, with corrective actions including the coaching and counseling of personnel involved regarding the proper use of the CAP.

  This finding was of more than minor safety significance because it affected the equipment performance attribute of the Mitigating Systems cornerstone of reactor safety and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of the unit's AFW system. The finding was determined to be of very low safety significance because it did not represent a deficiency affecting the design or qualification of a mitigating system, structure, or component (SSC); it did not, in and of itself, represent a loss of system and/or function; it did not represent an actual loss of function of at least a single train for greater than its TS allowed outage time, or two separate safety systems being out-of-service for greater than their TS allowed outage times; and it did not represent an actual loss of function of one or more non-TS trains of equipment designated as high safety significant in accordance with the licensee's maintenance rule program. The inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross-cutting aspect of "Identification" to the finding because the licensee's staff failed to identify the issue with AFW Train No. 2 within their CAP completely, accurately, and in a timely manner in accordance with program requirements. (P.1) (Section 4OA2.4)

Licensee-Identified Violation

Cornerstone: Mitigating Systems

- A violation of very low safety significance that was identified by the licensee has been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's CAP. This violation and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

As discussed in NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), the unit began the inspection period shut down to facilitate the plant's scheduled 19th refueling outage (RFO). On May 8, 2016, the reactor was taken critical to begin its 20th operating cycle. Main electrical generator synchronization to the electrical power grid occurred on May 9, 2016, and the unit reached full power on May 12, 2016. On May 31, 2016, the unit experienced a feedwater transient (see Section 1R11.2). Plant power was initially stabilized at approximately 89 percent, and ultimately reduced to approximately 85 percent in response to the event. On June 1, 2016, plant power was raised to approximately 98 percent, and the unit continued operating at or near full power through the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate Alternating Current Power Systems

a. Inspection Scope

During the period from June 20, 2016, through June 30, 2016, the inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee’s procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors’ review included, but were not limited to:

- Coordination between the TSO and the plant during off-normal or emergency events;
- Explanations for the events;
- Estimates of when the offsite power system would be returned to a normal state; and
- Notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- Actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- Compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;

- Re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- Communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

The inspectors also reviewed CAP items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

These reviews by the inspectors constituted a single summer readiness of offsite and alternate AC power systems inspection sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Alignment Verifications

a. Inspection Scope

The inspectors performed partial system physical alignment verifications of the following risk-significant systems:

- The Station Blackout Diesel Generator (SBODG) while Emergency Diesel Generator (EDG) No. 1 was out of service and unavailable for shutdown risk management during the week ending April 16, 2016;
- Control Room Emergency Ventilation System (CREVS) / Control Room Emergency Air Temperature Control System (CREATCS) Train No. 1 while CREVS/CREATCS Train No. 2 was out of service for preventative maintenance (PM) (lubrication/inspection of CREVS fan/motor and cleaning/inspection of compressor) during the week ending June 11, 2016; and
- Auxiliary Feedwater (AFW) Train No. 1 while AFW Train No. 2 was out of service for PM (pressure instrument calibrations and motor-operated valve inspections) during the week ending June 11, 2016.

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Safety Analysis Report (USAR), TS requirements, outstanding work orders (WOs), CRs, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization.

These activities by the inspectors constituted three partial system alignment verification inspection samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Alignment Verification

a. Inspection Scope

During the week ending May 7, 2016, the inspectors performed a complete system alignment inspection of the station's containment spray (CS) system to verify the functional capabilities of the system. This system was selected because CS is considered both important to safety and risk-significant in the licensee's probabilistic risk assessment. The inspectors physically inspected accessible system components and piping to verify mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the licensee's CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved.

These activities constituted a single annual complete system alignment verification inspection sample as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Protection Zone Inspections

a. Inspection Scope

The inspectors conducted fire protection zone inspection tours which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Steam Generator (SG) West D-Ring Area (Containment, Room 216 – Fire Area D) during the week ending April 16, 2016;
- SG East D-Ring Area (Containment, Room 218 – Fire Area D) during the weeks ending April 16, 2016, through April 23, 2016;

- 565’ and 653’ Elevations (Containment, Rooms 214, 215, 217, 220, 700 and 701 – Fire Area D) during the weeks ending April 16, 2016, through April 23, 2016;
- No. 1 Emergency Core Cooling System Pump Room; Elevation 545’ (Auxiliary Building, Room 105 – Fire Area AB) and Pipe Tunnel and Equipment Pipe Chase Area; Elevation 545’ (Auxiliary Building, Rooms 100 and 101 – Fire Area B) during the weeks ending April 16, 2016, through April 23, 2016;
- 585’ and 603’ Elevations (Containment, Rooms 315, 316, 317, 407, 410, 410A – Fire Area D) during the week ending April 23, 2016; and
- No. 1 Mechanical Penetration Room and Pipeway Area; Elevation 565’ (Auxiliary Building, Rooms 202 and 208 – Fire Area AB) during the weeks ending April 16, 2016, through April 30, 2016.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event. The inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s CAP.

These activities constituted six quarterly fire protection zone inspection tour samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

.1 Annual Heat Sink Performance

a. Inspection Scope

During the course of the licensee’s 19th RFO, the inspectors reviewed the licensee’s inspection, cleaning, and testing of the No. 1 Component Cooling Water (CCW) Heat Exchanger (HX) to verify the following:

- That potential deficiencies did not mask the licensee’s ability to detect HX degraded performance;
- That any common cause issues that had the potential to increase risk were being identified; and

- That the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk.

The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions.

This review by the inspectors constituted a single annual heat sink performance inspection sample as defined in IP 71111.07–05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From March 28, 2016, through April 8, 2016, the inspectors conducted a review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the RCS, SG tubes, emergency feedwater (EFW) systems, and risk-significant piping, components, and containment systems.

The inspectors' reviews documented in Sections 1R08.1 through 1R08.5 below constituted a single ISI inspection sample as defined in IP 71111.08.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed the following non-destructive examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic (UT) examination of reactor coolant pump (RCP) cover to case studs, RC–RCP–1–1–1–STUDS;
- UT of elbow-to-pipe weld, 3" reactor coolant weld, RC–30–CCA–8–1–SWB;
- UT of pipe-to-elbow weld, 3" RC weld, RC–30–CCA–8–1–SWC;
- UT of elbow-to-pipe weld, 3" RC weld, RC–30–CCA–8–1–SWD;
- UT of pipe-to-elbow weld, 3" RC weld, RC–30–CCA–8–1–SWE;
- Dye Penetrant examination of RCS pipe-to-elbow weld, RC–CCA–23–S–CV030;
- Dye Penetrant examination of RCS pipe-to-elbow weld, RC–MK–A–93–SW51;
- Visual examination (VT–1) of Diesel Generator Jacket Water HX E10–1 Attachment Weld, DG–JKT WTR HTXCHR–1–1–AW; and
- VT–3 examination of Diesel Generator Jacket Water HX 10–1 Supports; DG–JKT WTR HTXCHR–1–1–Supports.

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC approved alternative:

- UT, feedwater 8” valve-to-elbow weld, (WO 200534112);
- UT, feedwater weld, (WO 200611890); and
- VT–3, ROTSG 1–1 lower primary manway stud, (WO 200560403).

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems since the beginning of the last RFO to determine if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the construction code and ASME Code, Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedures were qualified in accordance with the requirements of construction code and the ASME Code Section IX:

- 26” main-steam pipe weld, MS–3A–EBB–1–33–FW34, per WO 200532106;
- AFW header welds, AF–SG–1–1–B6–SW256, per PO 45249949;
- Main feedwater (MFW) piping welds, FW–7–EBB–3–118–FW51, per WO 200511889;
- AFW piping welds, AF–ISIM2–206N–FW61, per WO 200511892; and
- RCS head vent piping welds, RC–CCA–23–F–CV028, per WO 200532090.

b. <u>Findings</u>

No findings were identified.

.2 <u>Reactor Pressure Vessel Upper Head Penetration Inspection Activities</u>

a. <u>Inspection Scope</u>

No exams were required this outage. Therefore, no NRC reviews were completed for this IP attribute.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage. Therefore, no NRC reviews were completed for this IP attribute.

b. <u>Findings</u>

No findings were identified.

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

a. <u>Inspection Scope</u>

The inspectors performed an independent walkdown of the RCS and related lines in the containment, which had received a recent licensee boric acid walkdown and verified whether the licensee’s boric acid corrosion control VTs emphasized locations where boric acid leaks can cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of RCS components with boric acid deposits to determine if degraded components were documented in the CAP.

The inspectors also evaluated corrective actions for any degraded RCS components to determine if they met the ASME Section XI Code:

- DH84, body-to-bonnet leak;
- DB–CF30, flange leak;
- DB–WC121, boric acid on valve; and
- DB–WC393, body-to-bonnet leak.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI:

- CR 2014–01755: Packing Leak Was Found on MU281;
- CR 2014–03060: Packing Leak Was Found on DH50A;
- CR 2014–03331: Decay Heat Pump No. 2 Has an Inboard Pump Seal Leak; and
- CR 2015–14239; Boric Acid Leakage from HP39.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors interviewed the SG program owner and reviewed documentation related to the SG ISI program to determine if:

- In-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) 1025132, “Steam Generator Management Program: Steam Generator In-Situ Pressure Test Guidelines,” Revision 4, and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- The numbers and sizes of SG tube flaws/degradation identified was bound by the licensee’s previous outage operational assessment predictions;
- The SG tube eddy current testing scope and expansion criteria were sufficient to meet TS requirements, and the EPRI 1013706, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 7;
- The SG tube eddy current testing examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- The licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- The licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;

- The licensee implemented an inappropriate “plug on detection” tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- The licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons-per-day, or the detection threshold during the previous operating cycle;
- The eddy current testing probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, “Performance Demonstration for Eddy Current Examination,” of EPRI 1013706, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Revision 7; and
- The licensee performed secondary side SG inspections for location and removal of foreign materials.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no reviews by the inspectors were completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG-related problems entered into the licensee’s CAP and conducted interviews with licensee staff to determine if:

- The licensee had established an appropriate threshold for identifying ISI/SG-related problems;
- The licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- The licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Simulator Training

a. Inspection Scope

During the week ending on April 30, 2016, the inspectors observed a crew of licensed operators in the plant’s simulator during training scenarios associated with the upcoming RCS heat up and plant startup operations from the licensee’s 19th RFO. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that training was being conducted in accordance with licensee procedures. In addition, the inspectors verified that the licensee's personnel were observing NRC examination security protocols, as applicable. The inspectors evaluated the following areas:

- Licensed operator performance;
- The clarity and formality of communications;
- The ability of the crew to take timely and conservative actions;
- The crew's prioritization, interpretation, and verification of annunciator alarms;
- The correct use and implementation of abnormal and emergency procedures by the crew;
- Control board manipulations;
- The oversight and direction provided by licensed Senior Reactor Operators (SROs); and
- The ability of the crew to identify and implement appropriate TS actions and Emergency Plan actions and notifications, as applicable.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

These observations and activities by the inspectors constituted a single quarterly licensed operator requalification program simulator training inspection sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation of Control Room Activities

a. Inspection Scope

During the course of the inspection period, the inspectors performed several observations of licensed operator performance in the plant's control room to verify that operator performance was adequate and that plant evolutions were being conducted in accordance with approved plant procedures. Specific activities observed that involved a heightened tempo of activities or periods of elevated risk included, but were not limited to:

- High Pressure Injection (HPI) Train No. 1 and HPI Train No. 2 nozzle flushing following identification of a radiological hot spot during the week ending April 2, 2016;
- Draining of the reactor refueling canal and installation of the reactor vessel closure head during the week ending April 30, 2016;
- Selected portions of the heat up of the RCS to normal operating pressure and temperature during the week ending May 7, 2016;
- Control rod testing and insertion timing during the week ending May 7, 2016;
- Initial core physics testing following reactor restart during the week ending May 14, 2016;
- Main electrical generator synchronization to the power grid and plant power escalation following reactor RFO operations during the week ending May 14, 2016;

- Periodic at-power control rod exercise testing, main turbine valve testing, and associated plant power maneuvers on Sunday, May 29, 2016; and
- Control room operator response to a plant feedwater transient on May 31, 2016.

The inspectors evaluated the following areas during the course of the control room observations:

- Licensed operator performance;
- The clarity and formality of communications;
- The ability of the crew to take timely and conservative actions;
- The crew's prioritization, interpretation, and verification of annunciator alarms;
- The correct use and implementation of normal operating, annunciator alarm response, and abnormal operating procedures by the crew;
- Control board manipulations;
- The oversight and direction provided by on-watch SROs and plant management personnel; and
- The ability of the crew to identify and implement appropriate TS actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

These observation activities by the inspectors of operator performance in the station's control room constituted a single quarterly inspection sample as defined in IP 71111.11–05.

b. Findings

(1) Mispositioned Instrument Air Valves Result in Plant Transient

On May 31, 2016, at approximately 10:21 a.m., planned testing of Steam and Feedwater Rupture Control System (SFRCS) Actuation Channel No. 1 was in progress. This was the first performance of this test since the unit returned to operation following RFO 19. Unexpectedly, operators in the control room received several overhead annunciator alarms coincident with a rapid swing in plant power and indications that the SG 1–2 MFW Regulating Valve (SP6A) had gone closed and then reopened. In accordance with established procedures for responding to such an event, control room operators took manual control of integrated control system (ICS) stations for reactor demand, SG/reactor demand, both MFW regulating valves, both MFW startup valves, and both MFW loop demands. The control room crew was then able to arrest the transient and stabilize plant power at approximately 89 percent.

Initial evaluation of the transient by the licensee revealed that two instrument air (IA) valves associated with control air for SP6A (IA1008D, SVSP6A1 Bypass; and IA1008A SVSP6A1 Maintenance Isolation) were out of their normal positions. The mispositioned valves had the effect of placing P6A in a "half trip" condition, such that when SFRCS Actuation Channel No. 1 was being tested SP6A unintentionally responded to the test signal.

The licensee entered this issue into their CAP as CRs 2016–07282, 2016–07286, 2016–07337, and 2016–08363. Because the licensee had yet to complete their investigation and analysis of the event and the IA valve mispositioning by the end of this inspection period, the issue is being treated as an unresolved item (URI) pending the inspectors' review of the licensee's completed cause evaluation and proposed corrective actions. (URI 05000346/2016002–01)

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems and components:

- RCS temperature instrumentation.

The inspectors reviewed events such as where ineffective equipment maintenance had or could have resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for system, structure, and component (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization.

The inspectors also performed a quality control review for the recent maintenance activities associated with RCS temperature instrumentation, as discussed in IP 71111.12, Section 02.02.

The maintenance effectiveness review activities conducted by the inspectors constituted a single inspection sample as defined in IP 71111.12–05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Unplanned maintenance and repair activities associated with a small RCS leak on the RCP No. 1–1 first stage seal package vent line during the licensee's 19th RFO;
- Internal inspection activities of the borated water storage tank (BWST) by divers during the week ending April 16, 2016;
- Installation of the reactor vessel closure head following reactor refueling activities during the week ending April 30, 2016;
- Repair activities associated with solenoid valve issues for several extraction steam non-return valves during the week ending May 14, 2016;
- Replacement activities for two failed feedwater flow modules within the plant's ICS during the week ending May 21, 2016; and
- Work activities associated with MFW valve issues and response to a plant feedwater transient on May 31, 2016.

These activities were selected based on their potential risk significance during at power as well as shutdown operations. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

The inspectors' review of these maintenance risk assessments and emergent work control activities constituted six inspection samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

.1 Operability Evaluations and Functionality Assessments

a. Inspection Scope

The inspectors reviewed the following issues:

- The operability and functionality of RCS hot leg temperature instrumentation following identification of the installation of improper insulation, as documented in CR 2016–04587; and
- The operability and functionality of Control Rod 7–3 absolute position indication following the identification of an erratic position indication signal, as documented in CR 2016–07289.

The inspectors selected these potential operability issues based on the risk significance of the associated SSCs. The inspectors examined the technical adequacy of the evaluations to ensure that TS operability was properly justified, and also to ensure that the applicable SSCs remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and USAR to the licensee’s evaluations to determine whether the applicable SSCs were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were appropriately controlled. The inspectors verified, where applicable, that the bounding limitations of the evaluations were valid. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with the operability evaluations and functionality assessments.

The review of these operability evaluations and functionality assessments by the inspectors constituted two inspection samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

.2 Periodic Review of Operator Workarounds

a. Inspection Scope

Operator workarounds (OWAs) are operator actions taken to compensate for degraded or non-conforming conditions. OWAs that cannot be implemented effectively can contribute to an increase in overall plant risk. As a result, the inspectors verified that the licensee is identifying OWAs at an appropriate threshold, entering them into the CAP, and addressing them in a manner that effectively manages the related adverse effects. As part of the review, the inspectors considered all existing plant conditions and the cumulative impact of the entire population of OWAs put in place by the licensee.

During the week ending June 30, 2016, the inspectors evaluated the licensee's OWAs with respect to mitigating systems to determine if the functions of the mitigating systems were adversely impacted. Additionally, the inspectors assessed whether or not the OWAs had adversely impacted any operator’s ability to implement abnormal or emergency operating procedures. The inspectors placed particular emphasis on any OWAs that had not been effectively evaluated by the licensee; that had been formalized or proceduralized as the long-term corrective actions for a degraded or nonconforming condition (and therefore may not have been properly tracked by the licensee); and that may have increased the potential for human error, such as OWAs that:

- Required operations that were not consistent with current training and system knowledge;
- Required a change from longstanding operational practices;
- Required operation of a system or component in a manner that was inconsistent with similar systems or components;
- Created the potential for the compensatory action to be performed on equipment or under conditions for which it was not intended;
- Impaired access to required indications, increased dependence on oral communications, or impacted the timeliness of time–critical event mitigating actions under adverse environmental conditions;
- Required the use of equipment and interfaces that had not been designed with consideration of the task being performed;
- Required the licensee to assess and manage an increase in risk; or
- Required a license amendment in accordance with 10 CFR 50.59.

These activities by the inspectors constituted a single OWAs review inspection sample as required by IP 71111.15, Section 02.01(a).

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following permanent changes to the facility:

- Engineering Change Package (ECP) No. 12–0272, *Replacement of the Control Rod Drive System with a Digital Control Rod Drive System*;
- ECP No. 14–0396, *Update to SG Orifice Place Settings*; and
- ECP No. 15–0187, *Replacement of Train 2 HPI Pump Motor (MP58–2)*.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation documents against the design basis, the USAR, and the TS, as applicable, to verify that these permanent changes to the facility did not affect the operability or availability of any safety-related systems, or systems important to safety. The inspectors observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with applicable design control documents; that the modifications operated as expected; and that operation of the modifications did not adversely impact the operability of any interfacing systems. The inspectors verified that relevant procedure, design, and licensing documents were properly updated. Finally, the inspectors discussed the plant modifications with operations, engineering, and training department personnel to ensure that the individuals were aware of how plant operation with these modifications in place could impact overall plant performance.

The inspectors' reviews of these permanent plant modifications constituted three inspection samples as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Quarterly Resident Inspector Observation and Review of Post-Maintenance Testing Activities

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMT) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Radiography for EFW to AFW system piping connections during the weeks ending April 9, 2016, through May 14, 2016;
- Operational and functional testing of the Train No. 1 Station Batteries (1P/1N) following planned replacement during the weeks ending April 9, 2016, through April 30, 2016;
- Operational acceptance testing of the FLEX RCS charging pumps following system installation during the weeks ending April 16, 2016, through April 30, 2016;
- Baseline acceptance testing of No. 2 HPI Pump following motor replacement during the week ending April 30, 2016;
- Verification of fuel assembly locations following core reload activities during the weeks ending April 30, 2016, through May 7, 2016;
- Operational and functional testing of AFW Train No. 1 following overhaul of the pump and turbine during the week ending May 7, 2016;
- Cycle 20 core physics testing following initial reactor criticality after refueling during the week ending May 14, 2016;
- Control rod insertion time testing following installation of a new digital control rod drive system and other miscellaneous maintenance activities on the system during the week ending May 7, 2014;
- RC system leakage testing following startup from refueling activities during the week ending May 7, 2016; and
- At-power testing during initial power ascension following SG orifice plate adjustments during the week ending May 14, 2016.

These activities were selected based upon the SSC's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the USAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with the PMTs to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety.

The inspectors' reviews of these activities constituted ten PMT inspection samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors continued their review of the licensee's comprehensive outage plan, shutdown defense-in-depth plan, and contingencies for the plant's 19th RFO, which was in progress at the beginning of the inspection period, and ended on May 9, 2016, when the unit's main generator was synchronized to the electrical power grid. These reviews were performed to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the portion of the RFO that took place during the inspection period, the inspectors observed elements of the RCS heatup and pressurization from cold shutdown, reactor startup and zero power core physics testing, main turbine roll up, synchronization of the main generator to the electrical power grid, escalation to full plant power, and monitored licensee controls over the outage activities listed below:

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the shutdown defense-in-depth plan for key safety functions and compliance with the applicable TS when taking equipment out of service;
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- Installation and configuration of RCS pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and shutdown defense-in-depth plan requirements were met, and controls over switchyard activities;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- Controls over activities that could affect reactivity;
- Maintenance of containment and associated ventilation systems, as required by TS;
- Licensee fatigue management, as required by 10 CFR Part 26, Subpart I;
- Refueling activities, including fuel handling, spent fuel assembly inspections, and fuel assembly reconstitution; and
- Licensee identification and resolution of problems related to RFO activities.

In combination with the activities described in Section 1R20 of NRC Integrated IR 05000346/2016001 (ADAMS Accession No. ML16118A435), these RFO review activities completed a single RFO inspection sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the results for the following testing activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- Extended 8-hour endurance test run of the No. 2 EDG during the week ending April 4, 2016 (Routine);
- Periodic at-power turbine valve testing during the week ending June 4, 2016 (Routine);
- No. 2 HPI Pump baseline inservice testing (IST) during the week ending April 4, 2016; and
- Local leak rate testing for Containment Penetration No. 68A, the pressurizer quench tank sample line, during the weeks ending April 4, 2016, through May 21, 2016 (Containment Isolation Valve).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- Did preconditioning occur;
- The effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- Acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- Plant equipment calibration was correct, accurate, and properly documented;
- As-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- That measuring and test equipment calibration was current;

- That test equipment was used within the required range and accuracy;
- That applicable prerequisites described in the test procedures were satisfied;
- That test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- That test data and results were accurate, complete, within limits, and valid;
- That test equipment was removed after testing;
- Where applicable for IST activities, that testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis;
- Where applicable, that test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- Where applicable, that actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- That prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- That equipment was returned to a position or status required to support the performance of its safety functions; and
- That all problems identified during the testing were appropriately documented and dispositioned in the CAP.

These activities conducted by the inspectors constituted two routine surveillance testing inspection samples, a single IST inspection sample, and a single containment isolation valve inspection sample as defined in IP 71111.22, Sections 02 and 05.

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)

.1 Radiological Hazard Assessment

a. Inspection Scope

The inspectors assessed the licensee’s current and historic isotopic mix, including alpha emitters and other hard-to-detect radionuclides. The inspectors evaluated whether survey protocols were reasonable to identify the magnitude and extent of the radiological hazards.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements as needed to verify conditions were consistent with documented radiation surveys.

The inspectors assessed the adequacy of pre-work surveys for select radiologically risk-significant work activities.

The inspectors evaluated the radiological survey program to determine if hazards were properly identified. The inspectors discussed procedures, equipment, and performance of surveys with radiation protection staff and assessed whether technicians were knowledgeable about when and how to survey areas for various types of radiological hazards (i.e., hot particles, alpha emitters, airborne radioactivity).

The inspectors reviewed work in potential airborne areas to assess whether air samples were being taken appropriately for their intended purpose and reviewed various survey records to assess whether the samples were collected and analyzed appropriately. The inspectors also reviewed the licensee's program for monitoring contamination which has the potential to become airborne.

These inspection activities constituted a partial sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.2 Instructions to Workers

a. Inspection Scope

The inspectors reviewed select radiation work permits (RWPs) used to access high-radiation areas and evaluated the specified work control instructions or control barriers. The inspectors also assessed whether workers were made aware of the work instructions and area dose rates.

The inspectors reviewed electronic alarming dosimeter dose and dose rate alarm setpoint methodology. For selected electronic alarming dosimeter occurrences, the inspectors assessed the worker's response to the alarm, the licensee's evaluation of the alarm, and any follow-up investigations.

The inspectors reviewed the licensee's methods for informing workers of changes in plant operations or radiological conditions that could significantly impact their occupational dose.

The inspectors reviewed the labeling of select containers of licensed radioactive material that could cause unplanned or inadvertent exposure to workers.

These inspection activities constituted one sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control

a. Inspection Scope

The inspectors observed locations where the licensee monitors material leaving the radiologically controlled area and assessed the methods used for control, survey, and release of material from these areas. As available, the inspectors observed health physics personnel surveying and releasing material for unrestricted use.

The inspectors observed workers leaving the radiologically controlled area and assessed their use of tool and personal contamination monitors and reviewed the licensee's criteria for use of the monitors.

These inspection activities constituted a partial sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage

a. Inspection Scope

The inspectors evaluated ambient radiological conditions during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination controls. The inspectors evaluated the licensee's use of electronic alarming dosimeters in high-noise areas as high-radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

For select airborne area RWPs, the inspectors reviewed airborne radioactivity controls and monitoring, the potential for significant airborne levels, containment barrier integrity, and temporary filtered ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials stored within pools and assessed whether appropriate controls were in place to preclude inadvertent removal of these materials from the pool.

These inspection activities constituted one sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

.5 High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors observed posting and physical controls for high radiation areas and very high radiation areas to assess adequacy.

The inspectors conducted a selective inspection of posting and physical controls for high radiation areas and very high radiation areas to assess conformance with performance indicators (PIs).

The inspectors reviewed procedural changes to assess the adequacy of access controls for high and very high radiation areas, to determine whether procedural changes substantially reduced the effectiveness and level of worker protection.

The inspectors assessed the controls in the high radiation areas greater than 1 rem/hour and areas with the potential to become high radiation areas greater than 1 rem/hour for compliance with TS and procedures.

The inspectors assessed the controls for very high radiation areas and areas with the potential to become very high radiation areas. The inspectors also assessed whether individuals were unable to gain unauthorized access to these areas.

These inspection activities constituted one sample as defined in IP 71124.01–05

b. Findings

No findings were identified.

.6 Radiation Worker Performance and Radiation Protection Technician Proficiency

a. Inspection Scope

The inspectors observed radiation worker performance and assessed their performance with respect to radiation protection work requirements, the level of radiological hazards present, and RWP controls.

The inspectors assessed worker awareness of electronic alarming dosimeter set points, stay times, or permissible dose for radiologically significant work as well as expected response to alarms.

The inspectors observed radiation protection technician performance and assessed whether the technicians were aware of the radiological conditions and RWP controls and whether their performance was consistent with training and qualifications for the given radiological hazards.

The inspectors observed radiation protection technician performance of radiation surveys and assessed the appropriateness of the instruments being used, including calibration and source checks.

These inspection activities constituted one sample as defined in IP 71124.01–05.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Reasonably-Achievable Planning and Controls (71124.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors selected three to five work activities of the highest exposure significance or involved work in high-dose rate areas.

The inspectors reviewed the radiological work planning as-low-as-reasonably-achievable (ALARA) evaluations, initial and revised exposure estimates, and exposure mitigation requirements. The inspectors determined if the licensee reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose reduction techniques; considered, commensurate with the risk of the work activity, alternate reduction features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined if the licensee's work planning considered the use of remote technologies as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors reviewed if these ALARA requirements were integrated into work procedure and/or RWP documents.

These inspection activities constituted a partial sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.2 Implementation of As-Low-As-Reasonably-Achievable and Radiological Work Controls

a. Inspection Scope

The inspectors reviewed the radiological administrative, operational, and engineering controls planned for selected radiologically significant work activities and evaluated the integration of radiological work controls and ALARA requirements into work packages or work procedures.

The inspectors conducted observations of in-plant work activities and assessed whether the licensee had effectively integrated the planned administrative, operational, and engineering controls into the actual field work to maintain occupational exposure ALARA. The inspectors observed pre-job briefings, and determined if the planned controls were discussed with workers. The inspectors evaluated the in-plant placement and use of shielding, contamination controls, airborne controls, RWP controls, and other engineering work controls against the licensee's ALARA plans.

The inspectors assessed licensee activities associated with work in progress to ensure the licensee was tracking doses, performed timely in-progress reviews, and, when jobs did not trend as expected, the licensee appropriately communicated to workers, supervisors, and radiation protection technicians additional methods to be used to reduce dose. The inspectors verified health physics and ALARA staff were involved with the management of radiological work control if/when in-field activities deviated from the planned controls. The inspectors assessed whether the Outage Control Center and station management provided sufficient support for ALARA re-planning as needed.

The inspectors assessed the involvement of ALARA staff with emergent work activities during outage or on-line maintenance and when possible, attended in-progress review discussions, outage status meetings, and/or ALARA committee meetings.

These inspection activities constituted a partial sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

.3 Radiation Worker Performance

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas to assess whether workers demonstrated the ALARA philosophy in practice and followed procedures. The inspectors observed radiation worker performance to evaluate whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

The inspectors interviewed individuals from selected work to assess their knowledge and awareness of planned and/or implemented radiological and ALARA work controls.

These inspection activities constituted one sample as defined in IP 71124.02–05.

b. Findings

No findings were identified.

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

.1 Engineering Controls

a. Inspection Scope

The inspectors reviewed select portable or installed airborne monitoring protocols used to monitor and warn of changing airborne concentrations in the plant to assess whether the alarms and set points were sufficient to prompt licensee/worker action to ensure that doses were maintained within the limits of 10 CFR Part 20 and ALARA. The inspectors determined whether the licensee established trigger points for evaluating levels of airborne betA–emitting and alphA–emitting radionuclides.

These inspection activities constituted a partial sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.2 Use of Respiratory Protection Devices

a. Inspection Scope

The inspectors assessed whether the respiratory protection devices used to limit the intake of radioactive materials were certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration or have been approved by the NRC per 10 CFR 20.1703(b). The inspectors evaluated whether the devices were used consistent with their National Institute for Occupational Safety and Health/Mine Safety and Health Administration certification or any conditions of their NRC approval.

The inspectors evaluated whether selected individuals qualified to use respiratory protection devices had been deemed fit to use the devices by a physician.

The inspectors observed selected individuals donning, doffing, and functionally checking the device as appropriate and assessed whether these individuals know how to safely use the device and how to properly respond to any device malfunction or unusual occurrence (loss of power, loss of air, etc.).

These inspection activities constituted a partial sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

2RS7 Radiological Environmental Monitoring Program (71124.07)

.1 Site Inspection

a. Inspection Scope

The inspectors walked down select air sampling stations and dosimeter monitoring stations to determine whether they were located as described in the offsite dose calculation manual (ODCM) and to determine the equipment material condition.

The inspectors reviewed calibration and maintenance records for select air samplers, dosimeters, and composite water samplers to evaluate whether they demonstrated adequate operability of these components.

The inspectors assessed whether the licensee had initiated sampling of other appropriate media upon loss of a required sampling station.

The inspectors observed the collection and preparation of environmental samples from select environmental media to determine if environmental sampling was representative of the release pathways specified in the ODCM and if sampling techniques were in accordance with procedures.

The inspectors assessed whether the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the USAR, NRC Regulatory Guide 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” and licensee procedures. The inspectors assessed whether the meteorological data readout and recording instruments were operable.

The inspectors evaluated whether missed and/or anomalous environmental samples were identified and reported in the annual environmental monitoring report. The inspectors selected events that involved a missed sample, inoperable sampler, lost dosimeter, or anomalous measurement to determine if the licensee had identified the cause and had implemented corrective actions. The inspectors reviewed the licensee’s assessment of any positive sample results and reviewed any associated radioactive effluent release data that was the source of the released material.

The inspectors selected structures, systems, or components that involve or could reasonably involve a credible mechanism for licensed material to reach ground water, and assessed whether the licensee had implemented a sampling and monitoring program sufficient to detect leakage to ground water.

The inspectors evaluated whether records important to decommissioning, as required by 10 CFR 50.75(g), were retained in a retrievable manner.

The inspectors reviewed any significant changes made by the licensee to the ODCM as the result of changes to the land census, long-term meteorological conditions, or modifications to the sampler stations since the last inspection. The inspectors reviewed technical justifications for any changed sampling locations to evaluate whether the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors assessed whether the appropriate detection sensitivities with respect to the ODCM were used for counting samples. The inspectors reviewed the quality control program for analytical analysis.

The inspectors reviewed the results of the licensee’s inter-laboratory comparison program to evaluate the adequacy of environmental sample analyses performed by the licensee. The inspectors assessed whether the inter-laboratory comparison test included the media/nuclide mix appropriate for the facility. The inspectors reviewed the licensee’s determination of any bias to the data and the overall effect on the radiological environmental monitoring program.

These inspection activities constituted one sample as defined in IP 71124.07–05.

b. Findings

No findings were identified.

.2 Groundwater Protection Initiative Implementation

a. Inspection Scope

The inspectors reviewed monitoring results of the groundwater protection initiative to evaluate whether the licensee had implemented the program as intended and to assess whether the licensee had identified and addressed anomalous results and missed samples.

The inspectors evaluated the licensee’s implementation of the minimization of contamination and survey aspects of the groundwater protection initiative and the decommissioning planning rule requirements in 10 CFR 20.1406 and 10 CFR 20.1501.

The inspectors reviewed leak and spill events and 10 CFR 50.75 (g) records and assessed whether the source of the leak or spill was identified and appropriately mitigated.

The inspectors assessed whether unmonitored leaks and spills were evaluated to determine the type and amount of radioactive material that was discharged. The inspectors assessed whether the licensee completed offsite notifications in accordance with procedure.

The inspectors reviewed evaluations of discharges from onsite contaminated surface water bodies and the potential for ground water leakage from them. The inspectors assessed whether the licensee properly accounted for these discharges as part of the effluent release reports.

The inspectors assessed whether on-site ground water sample results and descriptions of any significant onsite leaks or spills into ground water were documented in the annual radiological environmental operating report or the annual radiological effluent release report.

The inspectors determined if significant new effluent discharge points were updated in the ODCM and the assumptions for dose calculations were updated as needed.

These inspection activities constituted one sample as defined in IP 71124.07–05.

b. Findings

No findings were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors assessed whether problems associated with the radiological environmental monitoring program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved the radiological environmental monitoring program.

These inspection activities constituted one sample as defined in IP 71124.07–05.

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

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for the period from the second quarter 2015 through the first quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, and NUREG–1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, issue reports, event reports and NRC Integrated IRs for the period of April 2015 through March 2016 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

The inspectors' reviews of this PI data constituted a single Safety System Functional Failure inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Emergency Alternating Current Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System PI for the period from the second quarter 2015 through the first quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated IRs for the period of April 2015 through March 2016 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

The inspectors' reviews of this PI data constituted a single MSPI – Emergency AC Power System inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - HPI Systems performance for the period from the second quarter 2015 through the first quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated IRs for the period of April 2015 to March 2016 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified.

The inspectors' reviews of this PI data constituted a single MSPI – HPI System inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity PI for the period from the second quarter 2015 through the first quarter 2016. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, CRs, licensee event reports (LERs), and NRC IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's CAP to determine whether any problems had been identified with the PI data collected or transmitted for this indicator, and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze an RCS sample.

The inspectors' reviews of this PI data constituted a single RCS Specific Activity inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.5 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/ODCM Radiological Effluent Occurrences PI for the period from the second quarter 2015 through the first quarter 2016. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee’s issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee’s methods for quantifying gaseous and liquid effluents and determining effluent dose.

The inspectors' reviews of this PI data constituted a single Radiological Effluent TS/ODCM Radiological Effluent Occurrences inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee’s CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue.

Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily CR packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review: Implementation of the Site Welding Program

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspectors CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the six-month period of January 1 through June 30, 2016, although examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, Quality Assurance audit/surveillance reports, self-assessment reports, and maintenance rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted a single semi-annual trend inspection sample as defined in IP 71152–05.

b. Observations

During the course of the review period for this inspection sample, the inspectors noted several examples of issues involving weaknesses in the implementation of the site's welding program. Specific examples associated with this trend included, but were not limited to:

- Failure to Perform Measurement of Weld Preheat Temperature: During the ISI inspection activities documented in Section 1R08 of this report, the inspectors identified that preheat temperatures were not being verified prior to the initiation of EFW system welding activities in accordance with applicable licensee procedures. A qualitative assessment by licensee personnel prior to welding indicated that the required temperature had been achieved, so the performance deficiency was assessed as being of minor safety significance and not subject to formal documentation as an inspection finding (CRs 2016–04301 and 2016–04595);

- Use of Incorrect Weld Rod/Filler Materials: During refuel outage repairs to the RCP No. 1–1 first stage seal package vent line (see Sections 1R13 and 4OA3), licensee personnel identified that the weld filler material that was used was incorrect for the intended application (i.e., carbon steel was used instead of the required stainless steel). Following identification of the deficiency, the repairs were performed over using the correct material. An extent of condition review by the licensee identified several additional instances where the incorrect weld rod/filler material had been used. All of these instances were corrected or evaluated as acceptable to use as-is prior to plant restart (CRs 2016–05270, 2016–05305, 2016–05306, and 2016–05461); and

- Failure of Completed Welds to Pass Radiographic Testing: Numerous welds, predominantly on activities associated with the licensee's new EFW system and Alloy 600 HPI nozzle activities, were rejected during radiographic testing during the licensee's refuel outage. Many of these welds also required multiple repair efforts before being able to successfully pass radiographic examination. (CR 2016–05527 and others)

Welding is one of only a very few special processes defined under Criterion IX of 10 CFR Part 50, Appendix B, giving indication as to the overall importance of implementing the activity correctly. Although individually these issues did not result in any safety consequences, a potential adverse trend associated with the license's implementation of their welding program is distinguishable and suggests that additional licensee attention and possibly corrective actions may be warranted. The inspectors noted that following completion of the site's RFO that the matter of welding program implementation was elevated as a concern by the licensee's internal Nuclear Oversight group (CR 2016–06789).

c. Findings

No findings were identified.

.4 Follow-Up Sample for In-Depth Review: Review of Issues Involving Operability of Safety-Related Equipment

a. Inspection Scope

Over the course of the past several months, the inspectors noted several issues related to the licensee's management of the operability of safety-related equipment when challenged with degraded or nonconforming conditions. During this most recent inspection period, the inspectors performed a review of the licensee's CAP and associated documents to identify issues and events involving challenges to the operability of safety-related equipment in order to specifically assess:

- Whether the licensee had completely and accurately identified the problem in a timely manner commensurate with the safety significance and ease of discovery;
- The licensee's consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- The licensee's classification and prioritization of the resolution of the problem commensurate with the safety significance;
- The adequacy of the licensee's causal analyses for the problem;
- Whether the licensee's corrective actions were appropriately focused to correct the problem;
- Whether the corrective actions were performed in a timely manner commensurate with the safety significance of the issue;
- Whether corrective actions taken to preclude repetition of significant conditions adverse to quality were effective;
- Whether the licensee had taken into account applicable operating experience and appropriately communicated lessons learned both internally and externally; and
- Whether or not the licensee had appropriately evaluated and dispositioned operability, functionality, and reportability issues.

This review constituted a single follow-up inspection sample for in-depth review as defined in IP 71152–05.

b. Observations

During the course of the review for this inspection sample, the inspectors noted that the licensee has been recently challenged by a number of issues involving the management of operability of safety-related equipment. Specific examples associated with the inspectors' observations included, but were not limited to:

- Operability of Power Range Nuclear Instruments (NIs). As discussed in NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), while operating at full power on January 29, 2016, the reactor tripped during normal periodic NI calibrations due to a blown fuse in the reactor protection system (RPS). Following a brief forced maintenance outage to complete repairs, the unit was restarted and the reactor made critical on January 31, 2016. During the ensuing power ascension, a channel check surveillance required by TS Surveillance Requirement (SR) 3.3.1.1 produced some indication that the NI Δ Flux parameter for NI No. 7 was out-of-tolerance at 8:41 a.m. on February 1, 2016. Instead of immediately declaring NI–7 inoperable, however, on-watch senior licensed operators in the control room sought confirmation of the condition from reactor engineers. When the NI–7 Δ Flux parameter was confirmed by reactor engineers as being out-of-tolerance, NI–7 was declared inoperable at 11:09 a.m. (CR 2016–01491);

- Operability of the Service Water Header While Using a Degraded No. 3 Component Cooling Water (CCW) Heat Exchanger (HX) SW Outlet Isolation Valve (SW37) for SW Header Pressure Control. See the discussion for URI 05000346/2016001–02 in Subsection c(1) below (CR 2016–02667); and

- Operability of AFW Train No. 2 Following Abnormal Flow Vibrations. On May 7, 2016, the licensee was conducting scheduled testing of AFW Train No. 2 to support unit restart from RFO 19. At approximately 2:45 p.m., plant operators heard a loud noise from the AFW pump recirculation line. A subsequent investigation revealed a number of degraded conditions potentially indicating that a water hammer or other serious flow event had occurred. Instrument tubing for the recirculation line differential pressure indicator (PDI2658) had sheared off; the valve handwheel for the manual recirculation line isolation valve (AF23) had fallen off; and the packing follower displaced and packing blown out on a closed recirculation line drain valve (AF29). While the licensee launched and conducted a thorough inspection into the event and determined that no damage had been done to adversely impact AFW Train No. 2 abilities to perform all required safety functions, documentation to support system/train operability was limited and entry of the event into the licensee's CAP did not occur until May 8, 2016, as described in the discussion for non-cited violation (NCV) 05000346/2016002–02 in Subsection c(2) below. (CRs 2016–06515 and 2016–06516).

The performance of the licensee's staff, particularly that of licensed SROs and senior members of the operations department staff, was weak with respect to the management of TS operability. In each case, the inspectors had to engage the licensee's staff to ensure that the basis for operability was well understood and properly documented and, in the case of AFW Train No. 2, that procedural CAP actions related to the determination and documentation of safety-related equipment operability were appropriately implemented. Taken together, these issues may indicate the need for additional licensee assessment and corrective actions with respect to how the licensee's staff manages operability issues for safety-related equipment.

c. Findings

(1) (Closed) URI 05000346/2016001-02: Service Water Header Operability While Using a Degraded No. 3 Component Cooling Water (CCW) Heat Exchanger SW Outlet Isolation Valve (SW37) for SW Header Pressure Control

As discussed in Section 1R15.1 of NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), during the colder months of the year the demand on the SW system is reduced. During these winter months, the licensee operates the system in a mode specifically intended to reduce header pressure to avoid any challenges to the SW header relief valves, as the reduced SW flow requirements would otherwise tend to cause SW header pressure to rise. In this "header pressure control" mode, the SW side of a spare HX is placed in service to allow flow to pass without cooling any loads, and the increased SW system flow subsequently reduces SW header pressure back down to a more nominal value.

Licensee operating crews frequently utilize the swing CCW HX No. 3 to perform this function, and its associated outlet valve (SW37) is throttled by procedure to accomplish this. SW37 had experienced a number of leakage issues, at least in part, as a result of this practice. Most recently, excessive through leakage on SW37 was identified in March of 2015 (CR 2015–03283).

Initially, the licensee's evaluation of the condition only evaluated the impact of the through leakage on the valve's isolation function. The evaluation concluded that the valve could be considered operable, but degraded, since an alternate means of isolation was available. The evaluation did not, however, assess the impact of the valve degradation on operation of the SW system if CCW HX No. 3 were to be placed in service or credited to be aligned to one of the CCW and SW headers in standby.

As licensee engineering and technical personnel were preparing for an upcoming SW system flow test, their analyses of the condition began to suggest that small changes in the resistance of SW37 as a result of continued valve degradation could impact SW flow and possibly challenge minimum SW design basis flow assumptions for certain accident scenarios. As a result, in January of 2016 the licensee prohibited use of CCW HX No. 3 as an in-service or standby HX (CR 2016–00438). However, the licensee's evaluation of the condition continued to permit CCW HX No. 3 and SW37 to be used for SW header pressure control.

In reviewing the issue, the inspectors noted that the licensee's evaluation, as documented in CR 2016–00438 and entered into their CAP, did not contain any technical justification for the continued use of SW37 in header pressure control mode. Field observations by the inspectors revealed that the licensee operations staff had attached a plant information tag to the SW37 valve hand wheel warning personnel of the degraded condition of the valve and the potential for rendering SW Header No. 1 inoperable if the valve position were to be altered. Given the unknown condition of the SW37 valve internals, the unknown extent of the degradation of the valve's liner/seat, and the unknown nature of the mechanism causing the degradation, the inspectors questioned how it could be possible for the licensee to conclude that use of SW37 in header pressure control mode would be acceptable.

On February 18, 2016, the inspectors raised their concerns on this matter to the licensee's operations supervisory staff, and asked to be provided with the licensee's technical basis for continued operability of the SW system with the degraded SW37 being utilized for header pressure control. After several days had passed without receiving an answer, the inspectors elevated the question to the Site Vice President on February 23, 2016. On February 24, 2016, licensee engineering and operations management informed the inspectors that CCW HX No. 3 and SW37 had been removed from SW header pressure control and would be precluded from further use in that manner pending additional licensee analysis.

The licensee had entered this issue into their CAP as CR 2016–02667, and the inspectors had documented the issue as unresolved pending their receipt and review of the licensee's completed engineering evaluation of the concern. On May 10, 2016, licensee engineering personnel completed a detailed analysis using flow modeling software that demonstrated that the SW system remained operable and capable of supporting all required safety functions with SW37's degraded condition. As a result, the inspectors determined that no findings were identified and URI 05000346/2016001–02 is closed.

(2) <u>Failure to Use the Corrective Action Program to Evaluate and Document Degraded Condition with Auxiliary Feedwater Train 2</u>

<u>Introduction</u>

An NRC-identified finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," were identified for the licensee's failure to have entered a degraded condition associated with AFW Train No. 2 into their CAP until challenged by the inspectors. Specifically, a flow transient that occurred on May 7, 2016, and that caused damage to components in the AFW recirculation line during AFW Train No. 2 testing was not entered into the licensee's CAP until May 8, 2016, following challenges from the inspectors. This omission on the part of the licensee's staff had the effect of bypassing certain features of the licensee's CAP associated with evaluating and documenting the operability of safety-related equipment.

<u>Description</u>

On May 7, 2016, the licensee was conducting planned testing on AFW Train No. 2 associated with their restart from RFO 19. At approximately 2:45 p.m., plant operators heard a loud noise from the AFW pump recirculation line. A subsequent investigation revealed a number of degraded conditions potentially indicating that a water hammer or other serious flow event had occurred. Instrument tubing for the recirculation line differential pressure indicator (PDI2658) had sheared off; the valve handwheel for the manual recirculation line isolation valve (AF23) had fallen off; and the packing follower displaced and packing blown out on a closed recirculation line drain valve (AF29).

The event was reported by plant operators to licensee management in their outage control center, who directed plant engineering personnel to conduct inspections to determine whether or not the AFW recirculation line or other system components had been damaged by the transient. The licensee conducted thorough inspections into the event and determined that no damage had been done to adversely impact AFW Train No. 2 abilities to perform all required safety functions. Repairs were initiated and completed to correct the damage that the licensee had observed on the various AFW recirculation line components. Documentation to support system/train operability was limited, however, and entries in the station's unit log associated with the event and the follow up actions stated that no entry into the licensee's CAP was required. Specifically, unit log entries on May 7, 2016, at 6:59 p.m., and again at 11:10 p.m., stated that no CR was necessary.

On May 8, 2016, the inspectors followed up with licensee management about the lack of documentation regarding the event and the licensee's actions in response, with particular focus on the licensee's failure to utilize their CAP. The inspectors specifically noted that the generation of a CR within the CAP provides the vehicle for documenting not only the degraded condition associated with safety-related equipment, but also the documentation associated with the licensee's reasonable assurance that the equipment is operable and capable of performing all specified safety functions. Furthermore, Section 3.4 of licensee reference document NORM–OP–1009, “SRO Review of Condition Reports,” calls for the establishment of a mode hold restraint for safety-related degraded conditions that are identified during outages to help ensure that plant operators do not enter a mode of operation as they are restoring from the outage work period without all required safety-related equipment being operable and fully capable of meeting all specified safety functions. In their discussions with plant management the inspectors noted that this important CAP tool had been circumvented by the licensee’s failure to generate a CR.

The physical event and equipment issues were entered into the licensee’s CAP as CR 2016–06515 on May 8, 2016, following prompting by the inspectors. Corrective actions taken by the licensee included repairs to all damaged equipment, detailed inspections of AFW Train No. 2, and an engineering analysis into why the event occurred. The matter of the licensee’s failure to enter the event into their CAP in a timely manner was documented as CR 2016–06516, with corrective actions including the coaching and counseling of personnel involved regarding the proper use of the CAP.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, “Power Reactor Inspection Reports.” The inspectors determined that the licensee’s failure to have documented the event associated with AFW Train No. 2 in a CR and evaluated the issue in accordance with their CAP constituted a performance deficiency that was reasonably within the licensee’s ability to foresee and correct and should have been prevented. This finding was associated with the Mitigating Systems Cornerstone of Reactor Safety and was determined to be of more than minor significance because it was associated with cornerstone attribute of equipment performance, and adversely affected the cornerstone objective: "To ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage)."

Even though the reactor was shut down, because the RCS was at normal operating temperature and pressure in preparation for reactor startup, the inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power.” Using Exhibit 2 – “Mitigating Systems Screening Questions,” the inspectors determined the finding to be of very low safety significance (Green) because it did not represent a deficiency affecting design or qualification of the AFW system; it did not represent a loss of system and/or function; it did not represent an actual loss of function for at least a single train for more than its TS allowed outage time; and it did not represent an actual loss of function of one or more non-TS trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program.

Using IMC 0310, "Aspects Within the Cross-Cutting Areas," the inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross-cutting aspect of “Identification” to the finding because the licensee’s staff failed to identify the issue with AFW Train No. 2 within their CAP completely, accurately, and in a timely manner in accordance with program requirements. (P.1)

Enforcement

The licensee's CAP procedure, NOP–LP–2001, "Corrective Action Program," Section 4.3, "CR Initiation," specifies that a CR shall be initiated upon discovery of any degraded conditions that affect safety-related equipment. Furthermore, Appendix B of 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that: "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Contrary to this requirement, the licensee's staff failed to enter a degraded condition associated with AFW Train No. 2 into their CAP on May 7, 2016, or evaluate that condition in accordance with CAP requirements, until the licensee was prompted by the inspectors on May 8, 2016.

Because this finding was of very low safety significance, had been entered into the licensee's CAP, and the licensee had established corrective actions under CRs 2016–06515 and 2016–06516, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000346/2016002–02)

.5 Follow-Up Sample for In-Depth Review: Review of Issues Involving 8-Hour Non-Emergency Reports Required By 10 CFR 50.72(b)(3)

a. Inspection Scope

During the past several months, the inspectors have noted issues related to the licensee's administration of certain emergency notification system (ENS) telephone reports to the NRC required under 10 CFR 50.72(b). Specifically, the licensee has been challenged by the 8-hour non-emergency reports called out in 10 CFR 50.72(b)(3). In conducting their review of this issue, the inspectors assessed:

- Whether the licensee had completely and accurately identified the problem in a timely manner commensurate with the safety significance and ease of discovery;
- The licensee's consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- The licensee's classification and prioritization of the resolution of the problem commensurate with the safety significance;
- Where applicable, the adequacy of the licensee's causal analyses for the problem;
- Whether the licensee's corrective actions were appropriately focused to correct the problem;
- Whether the corrective actions were performed in a timely manner commensurate with the safety significance of the issue;
- Whether the licensee had taken into account applicable operating experience and appropriately communicated lessons learned both internally and externally; and
- Whether or not the licensee had appropriately evaluated and dispositioned operability, functionality, and reportability issues.

This review constituted a single follow-up inspection sample for in-depth review as defined in IP 71152–05.

b. Observations

During the course of the review for this inspection sample, the inspectors noted that the licensee has been recently challenged by issues involving the handling and administration of non-emergency reports required to be made via ENS telephone to the NRC. In particular, the difficulties noted have been associated with the 8-hour reports required under 10 CFR 50.72(b)(3). Specific examples associated with the inspectors' observations included, but were not limited to:

- Reportability Associated with the Discovery of RCS Pressure Boundary Leakage. As discussed in Sections 4OA3.3 and 4OA3.4 of this report, on March 30, 2016, licensee engineers conducting a piping inspection for evidence of boric acid leakage identified evidence of a very minor (approximately ½ teaspoon of boric acid residue) RCS pressure boundary leak on the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line. The unit was shut down and in Mode 6 for refueling at the time the condition was identified, and the licensee made the determination that the issue was reportable under the requirements of 10 CFR 50.72(b)(3)(ii)(A) as a degraded condition ongoing at the time of discovery. The licensee's engineers initially identified the condition at approximately 12:13 p.m., and entered the issue into the CAP as CR 2016–04208. However, plant operations personnel ultimately responsible for performing the ENS notification to the NRC logged the event time as 5:15 p.m., and ultimately notified the NRC at 12:14 a.m. on March 31, 2016, some twelve hours after the engineers had identified the condition. As discussed in Section 2.2.1(c) of the NRC Enforcement Policy, the inspectors determined that the licensee's failure to have reported the issue within the 8-hour timeframe required under 10 CFR 50.72(b)(3) did not adversely impact the NRC's regulatory process, and as such constituted a violation of minor safety significance not subject to any formal enforcement; and

- Reportability Associated with the Shield Building Emergency Ventilation System (EVS) Being Degraded with Watertight Door No. 108 Inadvertently Left Open. See the discussion for URI 05000346/2016001–03 in Subsection c(1) below. (CR 2016–03694)

The inspectors have noted that the licensee's staff has been challenged recently on several occasions by the requirements to perform 8-hour reports required under 10 CFR 50.72(b)(3). Specific items noted by the inspectors include:

- The start time related to the 8-hour requirement to make reports under 10 CFR 50.72(b)(3) does not appear to be completely understood. Unlike the starting time for TS operability issues, the start time related to the 8-hour requirement to make reports under 10 CFR 50.72(b)(3) is not tied to the declaration of inoperability made by a licensed SRO, but the actual time the issue is identified by a knowledgeable member of the licensee's staff; and

- The actual reporting requirement under 10 CFR 50.72(b)(3) is to report "as soon as practical and in all cases within eight hours of the occurrence of any of the following..." The requirement does not imply that licensee should intentionally wait until near the end of the 8-hour time clock to complete required reports. The inspectors noted that the licensee is consistently diligent in pursuing exit from TS Limiting Condition for Operation (LCO) required action time clocks as soon as practical, yet in several instances has approached the management of reporting requirement time clocks with what appears to be an altogether different standard.

Individually, these issues may not rise to a level indicating the need for any additional actions on the part of the licensee. In sum, however, they may indicate the need for additional licensee assessment and corrective actions with respect to how the licensee's staff manages the reporting requirements under 10 CFR 50.72(b)(3).

c. Findings

(1) (Closed) URI 05000346/2016001–03: Shield Building Emergency Ventilation System Operability with Watertight Door No. 108 Inadvertently Left Open

As discussed in Section 1R15.1 of NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), the shield building EVS functions to collect and process potential leakage from the containment vessel to minimize environmental activity levels resulting from all sources of containment leakage following a design-basis accident. The EVS is required to maintain a negative pressure (a minimum of ¼ inch water gauge), with respect to outside atmosphere, within the annular space between the shield building and the containment vessel and in the penetration rooms following an accident. In addition, it is required to provide a filtered exhaust path from the shield building annulus and the penetration and pump rooms following an accident.

The EVS consists of two independent and redundant trains. Each train consists of a prefilter, a high efficiency particulate air filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The EVS boundary, consisting of various walls and doors within the plant's auxiliary building, must be intact and functional to ensure EVS operability. Door No. 108, "Emergency Core Cooling System Pump Room No. 115 to Detergent Waste Drain Tank to Clean Waste Receiver Tank," is one such plant door.

At approximately 7:53 p.m. on March 21, 2016, with the unit in Mode 1 and operating at power, operations personnel discovered a plant watertight door, Door No. 108, open and unattended. The operations personnel immediately secured the door and informed operations on-watch management of the issue. The on-watch operations shift manager determined that because the door was fully functional and closed when he was informed of the issue that neither the door nor the shield building EVS was inoperable. He then contacted the licensee's on-duty management team to discuss the issue. Collectively, the licensee's personnel concurred with the shift manager's operability decision and determined that the issue was not reportable under 10 CFR 50.72(b)(3)(v) as an "Event or Condition that Could Have Prevented Fulfillment of a Safety Function," since no SSCs had ever been declared inoperable. Subsequently, licensee engineering personnel reviewing the issue determined that based on exiting plant calculations and the area of the door that it was highly improbable that the EVS would be able to have met its specified safety function with Door No. 108 open and unattended. The licensee entered this issue into their CAP as CR 2016–03694. An investigation by the licensee into the issue identified that the door had been inadvertently left open by contractor workforce personnel approximately five minutes before it was discovered open by operations personnel.

During the next few days while conducting their routine review of the licensee's CAP entries, the inspectors took note of this issue and questioned the licensee regarding their decision not to report the matter under 10 CFR 50.72(b)(3)(v). Licensee management subsequently decided to perform a special test of the EVS with Door No. 108 in the open position (under the administrative control of a designated individual) to empirically determine the capability of the EVS in this condition. The test was performed during the afternoon/evening hours on March 25, 2016, and the test results indicated that the EVS passed, albeit by only 0.08 seconds.

Following documentation of this URI in Section 1R15.1 of NRC IR 05000346/2016001, the inspectors received the licensee's completed evaluation into the events surrounding the issue and performed a detailed review of the licensee's conclusions. Because the licensee had empirically demonstrated that the EVS was capable of performing its intended safety function with Door No. 108 in the open position, the inspectors determined that no findings were identified and URI 05000346/2016001–03 is closed.

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

.1 (Closed) Licensee Event Report (LER) 05000346/2016–001–00: Reactor Trip During Nuclear Instrumentation Calibrations, and Steam Feedwater Rupture Control System Actuation on High Steam Generator Level

a. Inspection Scope

On January 29, 2016, the licensee was performing normal periodic power range NI calibration activities. Each of the four RPS channels receives input from one of four separate and independent power range NIs. A manual bypass feature is provided on each RPS channel to remove that channel from the coincidence reactor trip logic in order to facilitate on line maintenance, such as periodic NI calibration. The RPS design precludes bypassing more than one RPS channel at a time.

Earlier in the current reactor operating cycle, the RCS hot leg temperature detector for RPS Channel No. 1 failed, and the licensee was required to declare RPS Channel No. 1 inoperable. Plant TS permit continued operation with one inoperable RPS channel indefinitely; however, TS LCO 3.3.1, Condition A, required the licensee to place the channel into the trip or bypassed condition. Since replacement of the temperature detector can only be done with the unit in the cold shutdown condition, the licensee had been operating with RPS Channel No. 1 bypassed during most normal operating conditions. However, during maintenance conditions requiring any of the other three remaining RPS channels to be bypassed, the licensee was forced to place RPS Channel No. 1 into the trip condition in order to allow the manual bypass to be used elsewhere. This was the case on January 29, 2016, at the time of the event.

At approximately 12:14 p.m., plant technicians completed calibration of the power range NI for RPS Channel No. 1 (NI–6), and the channel was manually tripped by plant operators to set up conditions for calibrations on the remaining three power range NIs. From approximately 12:20 p.m. to 1:03 p.m., RPS Channel No. 3 was bypassed to support calibration of its associated power range NI–8. At approximately 1:09 p.m., plant operators restored RPS Channel No. 3 to normal operation and placed RPS Channel No. 2 into the bypass condition to support calibration of its associated power range NI–5.

At approximately 1:21 p.m., a blown fuse associated with a power supply for RPS Channel No. 4 caused that channel to trip on the flux / Δ flux / flow protective function. With RPS Channel No. 1 already in the tripped condition to support the maintenance activities being performed, reactor trip coincidence logic was satisfied and a reactor trip ensued.

There were several anomalies associated with the plant trip. One of the two main generator electrical output circuit breakers (ACB 34561) failed to open rapidly enough (on the order of a tenth of a second), and a generator exciter lockout resulted. This subsequently caused one of the site’s four 345 KV offsite lines (the Bayshore Line) to be isolated due to protective relaying. Additionally, unrelated failures in the plant’s ICS caused SG No. 1 to experience a high water level condition, which resulted in an SFRCS actuation. This complicated the response of control room operators to the trip by removing MFW and initiating AFW to supply both SGs, and by removing the main condenser as the plant’s heat sink and forcing operators to vent steam to atmosphere. The licensee had entered the plant trip event into their CAP as CR 2016–01364.

The licensee determined that there were two root causes for the SFRCS actuation. The first root cause involved the failure of the plant’s ICS to properly respond to the reactor trip by automatically reducing MFW at a rapid rate, and was traced to less than sufficient work package documentation and instructions for an ICS module replacement that took place during the plant's 18th RFO in 2014. The second root cause involved the licensee’s failure to implement a technically correct software change associated with the SG/Reactor Demand ICS control station. Specifically, a known logic error within the plant’s ICS would cause the SG/Reactor Demand control station to trip to manual from automatic coincident with a reactor trip. The licensee had instituted compensatory operator actions for this condition but removed these actions in December 2015 when they implemented a software change to rectify the problem. However, the corrective actions were inadequate, and the SG/Reactor Demand ICS control station unexpectedly tripped to manual from automatic when the unit tripped on January 29, 2016. With the absence of any compensatory operator actions, the unexpected control station mode of operation change combined with the ICS failure to automatically reduce MFW flow at a rapid rate caused the SG No. 1 high level condition and the resultant SFRCS actuation.

The inspectors had previously reviewed the causes for the SG No. 1 high level condition and resultant SFRCS actuation. The results of that review and two findings of very low safety significance (Green – FIN 05000346/2016001–04; and Green – FIN 05000346/2016001–05) were previously documented in NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), Section 4OA3.1.

In addition to those actions previously performed, in response to receipt of this LER the inspectors completed additional reviews that included, but were not limited to:

- The potential for any generic issues, including those potentially requiring reporting under 10 CFR Part 21;
- The licensee's completed cause evaluation reports and additional corrective actions associated with the issues; and
- The accuracy of the information provided by the licensee in the LER.

Documents reviewed as part of this inspection are listed in the Attachment. This LER is closed.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153–05.

b. Findings

(1) Inadequate Evaluation of Trend Related to A25X Fuse Failures

Introduction

A self-revealed finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," were identified for the licensee's failure to have adequately addressed an identified adverse trend involving 10 and 15 ampere Gould – Shawmut A25X series fuses. Specifically, the licensee had identified adverse trends related to failures of A25X series fuses in 2005, and again in 2015, and had entered these adverse trends into their CAP as CRs 2005–05314 and 2015–03516. However, the evaluation performed under CR 2015–03516 did not recognize that the fuse failures were occurring much more frequently than originally anticipated and that the previously created PMs were not adequate to prevent failures. Additionally, the evaluation did not adequately incorporate industry experience that also identified a trend of failures with the A25X series fuses.

Description

On October 10, 2005, a blown Gould – Shawmut A25X series fuse caused RPS Channel No. 3 to trip on the flux/Δ flux /flow parameter. The licensee entered this event into their CAP as CR 2005–05314, and their subsequent evaluation identified an adverse trend of premature failures of A25X series fuses. The principle corrective action taken by the licensee involved the establishment of a series of preventative fuse replacements with a 15-year frequency. On March 17, 2015, the licensee's technical staff again identified an adverse trend in the premature failure of A25X series fuses. This trend was entered into the licensee's CAP, in this case as CR 2015–03516. Amongst the actions taken in response to the 2015 issue was an evaluation of the appropriateness of the 15-year preventative fuse replacement frequency established in 2005, which the licensee concluded to still be appropriate. Neither the 2005 CR evaluation nor the 2015 CR evaluation contained formal root or apparent cause evaluations.

As discussed in the Inspection Scope section above, with the reactor operating at full power on January 29, 2016, at approximately 1:21 p.m. the reactor tripped due to a blown A25X series fuse associated with a power supply for RPS Channel No. 4. The licensee entered the issue into their CAP as CR 2016–01364. Unlike the previous licensee CAP evaluations, the evaluation of CR 2016–01364 was performed at the full root cause level. Amongst the conclusions reached by the licensee as part of this evaluation was that the A25X series fuse failures were occurring much more frequently than originally anticipated and that the previously created 15-year preventative fuse replacements were not adequate to prevent failures. Additionally, the licensee's root cause evaluation also concluded that had the evaluation in CR 2015–03516 been sufficiently comprehensive and included a larger sample of fuse failures or a search of industry experience associated with the failed fuses, it may have been possible to identify that the previously created 15-year preventative fuse replacements were not effective in preventing failures of the A25X series fuses.

Corrective actions taken by the licensee in response to the CR 2016–01364 evaluation included replacement of the licensee's existing stock of uninstalled A25X series fuses with equivalent fuses of a different style and from a different manufacturer. In addition, the licensee identified the in-plant locations where A25X series fuses were still installed and developed a plan to replace these fuses in the near future. Approximately 133 A25X series fuses were identified by the licensee as part of this effort, almost all of which were in safety-related applications.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports." The inspectors determined that the licensee's failure to adequately evaluate the recent adverse trend of A25X series fuses in 2015 constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented. This finding was associated with the Initiating Events Systems Cornerstone of Reactor Safety and was determined to be of more than minor significance because it was associated with cornerstone attribute of equipment performance, and adversely affected the cornerstone objective: "To limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations."

In consultation with the NRC Region III Senior Reactor Analyst, the inspectors evaluated the finding using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." Using Exhibit 1 – "Initiating Events Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because it did not represent a deficiency that caused a reactor trip, as well as the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feedwater, etc.)

Using IMC 0310, "Aspects Within the Cross-Cutting Areas," the inspectors determined that the finding had a cross-cutting aspect in the area of problem identification and resolution. The inspectors assigned the cross-cutting aspect of "Resolution" to the finding because the licensee failed to take action to resolve the identified adverse trend associated with premature failures of A25X series fuses in a timely manner. (P.3)

Enforcement

Within the licensee's governing CAP procedure, NOP–LP–2001, "Corrective Action Program," adverse trends entered into the licensee's CAP are categorized as conditions adverse to quality. Appendix B of 10 CFR Part 50, Criterion XVI, "Corrective Action," states, in part, that: "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to this requirement, beginning with completion of the licensee's evaluation of the adverse trend associated with the premature failure of Gould – Shawmut A25X series fuses identified in CR 2015–03516 on April 16, 2015, and continuing through March 4, 2016, when the licensee completed their root cause evaluation for CR 2016–01364, the licensee failed to take adequate corrective action for the adverse trend.

Because this finding was of very low safety significance, had been entered into the licensee's CAP, and the licensee had established corrective actions under CR 2016–01364, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000346/2016002–03)

.2 (Closed) Licensee Event Report 05000346/2016–002–00: Unanticipated Steam and Feedwater Rupture Control System Actuation

a. Inspection Scope

On January 30, 2016, the licensee was performing recovery actions from the reactor trip that had occurred on January 29, 2016, (see Section 4OA3.1). With the reactor shutdown and in the hot standby condition, control room operators had restored the main condenser as the plant's heat sink and were preparing to restore MFW to the SGs with the motor-driven feedwater pump to permit shutdown of AFW. At approximately 1:23 a.m., control room operators received an unexpected SFRCS actuation on a reverse delta pressure (ΔP) signal for the No. 1 SG while attempting to restore MFW to that SG. This actuation isolated MFW and removed the main condenser as the plant's heat sink, once again forcing operators to vent steam to atmosphere to remove nuclear decay heat.

The licensee had entered this event into their CAP as CR 2016–01397, and the inspectors had previously reviewed the cause for this event. The results of that review and a finding of very low safety significance (Green – FIN 05000346/2016001–06) were documented in NRC IR 05000346/2016001 (ADAMS Accession No. ML16118A435), Section 4OA3.2.

In addition to those actions previously performed, in response to receipt of this LER the inspectors completed additional reviews that included, but were not limited to:

- The potential for any generic issues, including those potentially requiring reporting under 10 CFR Part 21;
- The licensee's completed cause evaluation reports and additional corrective actions associated with the issues; and
- The accuracy of the information provided by the licensee in the LER.

Documents reviewed as part of this inspection are listed in the Attachment. This LER is closed.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.3 Event Notification No. 51837: Degraded Condition Due to Discovery of Pressure Boundary Leakage

a. Inspection Scope

On March 30, 2016, licensee engineers conducting a piping inspection for evidence of boric acid leakage identified evidence of a very minor (approximately ½ teaspoon of boric acid residue) RCS pressure boundary leak on the flexible hose assembly (ASME Section III, Class 2) for the RCP No. 1–1 first stage seal cavity vent line. The unit was shut down and in Mode 6 for refueling at the time the condition was identified.

The inspectors observed and reviewed the licensee's response to the event, operator logs, computer and recorder data, and procedural requirements. Specific items associated with this event that were reviewed included, but were not limited to:

- Mitigating systems and fission product barriers performance and integrity;
- The performance of plant operators in response to the event;
- Event notifications made pursuant to 10 CFR 50.72;
- The potential for any generic issues, including those potentially requiring reporting under 10 CFR Part 21; and
- The licensee's completed root cause reports and corrective actions associated with the event.

The licensee entered this event into their CAP as CRs 2016–04208 and 2016–04287. Immediate corrective actions taken by the licensee included replacement of the leaking flexible hose assembly with a spare assembly, as well as extent-of-condition inspections of all similar components within the RCS. Documents reviewed as part of this inspection are listed in the Attachment.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.4 (Closed) Licensee Event Report 05000346/2016–003–00: Leak from Reactor Coolant Pump Seal Piping Flexible Hose Due to Undetected Manufacture Weld Defect

a. Inspection Scope

As discussed in Section 4OA3.3 above, On March 30, 2016, while the unit was shut down for a scheduled RFO, the licensee identified evidence of a very minor (approximately ½ teaspoon of boric acid residue) RCS pressure boundary leak on the flexible hose assembly (ASME Section III, Class 2) for RCP No. 1–1 first stage seal cavity vent line.

In addition to those actions previously performed in response to the licensee's initial event notification, following receipt of this LER the inspectors completed additional reviews that included, but were not limited to:

- The potential for any generic issues, including those potentially requiring reporting under 10 CFR Part 21;
- The licensee's completed cause evaluation reports and additional corrective actions associated with the issues; and
- The accuracy of the information provided by the licensee in the LER.

Documents reviewed as part of this inspection are listed in the Attachment. This LER is closed.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153–05.

b. Findings

(1) Inadequate Post-Manufacture Quality Control Inspections Performed for ASME Section III, Class 2 Reactor Coolant Pump Seal Cavity Vent Line Flexible Hose

Introduction

A self-revealed finding of very low safety significance (Green) and an associated NCV of TS 3.4.13, “RCS Operational Leakage,” were identified for the failure of the licensee’s vendor to have ensured that a replacement ASME Section III, Class 2 RCP seal cavity vent line flexible hose was subjected to adequate quality control testing following manufacture. Specifically, a manufacturing weld defect on the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line was not detected by post-manufacture testing, such that the hose developed a very minor leak during power operations for the reactor operating cycle occurring before the licensee’s spring 2016 RFO. As discussed in Section 1.2 of the NRC Enforcement Policy, it is NRC policy to hold licensees responsible for the actions of their vendors.

Description

During the unit’s 18th RFO in the spring of 2014, the licensee installed flexible hose assemblies in all three stages of the seal cavity vent lines on all four RCPs. This modification to the plant was performed as corrective action for a previously identified issue involving vibration-induced cracking of the socket welds on the originally installed hard pipe RCP seal cavity vent lines. The replacement flexible hose assemblies were newly manufactured specifically for the purpose and designed, constructed, and tested in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 2 components. The licensee procured a total of fifteen flexible hose assemblies from the same manufacturing batch; twelve were installed during the unit’s 18th RFO and three were retained as spares.

During the licensee’s 19th RFO, on March 30, 2016, licensee engineers conducting a VT for evidence of leakage from the RCS and other interconnected systems as part of the site’s boric acid corrosion control program identified a minute amount of dry boric acid residue (approximately ½ teaspoon) on the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line. The licensee concluded from this that the flexible hose assembly had experienced some minor pressure boundary leakage during the prior reactor operating cycle. The licensee entered the issue into their CAP as CRs 2016–04208 and 2016–04287, conducted follow on inspections of all installed similar flexible hose assemblies, and initiated a full root cause investigation of the issue.

The licensee's root cause evaluation concluded that the leakage was most likely due to a weld solidification crack that had occurred during the manufacture of the flexible hose assembly. Laboratory examination and analysis of the failed flexible hose assembly commissioned by the licensee was unable to confirm the pressure boundary leak path, but a partial depth weld solidification crack was observed in the pressure boundary at the hose/bellows tube to socket end of the assembly. From this, the licensee concluded that the root cause for the leakage was less than adequate quality control inspection following manufacture of the flexible hose assembly by the manufacturer, such that an extremely small pressure boundary defect was not detected by the hydrostatic testing required by, and conducted in accordance with, ASME Section III.

The licensee performed additional testing that added confirmation to their root cause evaluation conclusion. Prior to replacing the failed flexible hose assembly for RCP No. 1–1, the licensee subjected their three spare flexible hose assemblies to enhanced leakage detection testing consisting of both bubble and helium tracer probe leak testing at an offsite facility. All three uninstalled spare flexible hose assemblies passed the bubble test, but one assembly failed the helium tracer probe leak test and was subsequently discarded as a potential spare. The helium tracer probe leak testing was specified by the licensee due to its increased sensitivity for detection of extremely small leaks.

In addition to replacement of the failed RCP No. 1–1 flexible hose assembly with a spare assembly subjected to enhanced leakage testing, corrective actions taken by the licensee as a result of their root cause evaluation included revising the procurement requirements for subsequently ordered flexible hose assemblies to include enhanced helium tracer probe leak testing.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports." The inspectors determined that the licensee was responsible for the flexible hose assembly vendor's failure to have ensured that a replacement ASME Section III, Class 2 RCP seal cavity vent line flexible hose was subjected to adequate quality control testing following manufacture. This failure represented a performance deficiency that was reasonably within the vendor's and licenseee's ability to foresee and correct and should have been prevented. As discussed in Section 5.0 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," this finding was evaluated under the Initiating Events Systems Cornerstone of Reactor Safety as opposed to the Barrier Integrity Cornerstone of Reactor Safety because it was associated specifically with RCS leakage. The finding was determined to be of more than minor significance because it was associated with cornerstone attribute of equipment performance (i.e., RCS barrier integrity), and adversely affected the cornerstone objective: "To limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations."

The inspectors evaluated the finding using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power." Using Exhibit 1 – "Initiating Events Screening Questions," the inspectors determined the finding to be of very low safety significance (Green) because:

- Following the inspectors' reasonable assessment of the degradation, it was determined that the finding could not have resulted in exceeding the RCS leak rate for a small loss of coolant accident; and
- Following the inspectors' reasonable assessment of the degradation, it was determined that the finding could not have likely affected other systems used to mitigate a loss of coolant accident.

Using IMC 0310, "Aspects Within the Cross-Cutting Areas," the inspectors determined that the finding had a cross-cutting aspect in the area of human performance. The inspectors assigned the cross-cutting aspect of "Field Presence" to the finding because the licensee failed to ensure that their vendor performed adequate quality control testing following manufacture of a safety-related flexible hose assembly. (H.2)

Enforcement

With respect to RCS pressure boundary leakage, TS LCO 3.4.13 specifies that no leakage of this type is acceptable with the plant operating in Modes 1 – 4. Contrary to this requirement, the licensee operated the plant in Modes 1 – 4 for some period of time prior to their spring 2016 RFO with a very minor RCS pressure boundary leak in the flexible hose assembly for the RCP No. 1–1 first stage seal cavity vent line.

Because this finding was of very low safety significance, had been entered into the licensee's CAP, and the licensee had established corrective actions under CRs 2016–04208 and 2016–04287, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000346/2016002–04)

.5 Event Notification No. 52010: Unanalyzed Condition of Emergency Diesel Generators During Tornado Low Pressure

a. Inspection Scope

On June 16, 2016, licensee engineering personnel conducting a review of external operating experience from another utility identified an issue potentially impacting the site's EDGs during the low barometric pressure conditions associated with a tornado. The site's EDGs are equipped with a crankcase positive pressure trip that is bypassed during an automatic emergency start of the EDGs. During the course of their review, however, licensee engineering personnel determined that a design basis tornado could create a low pressure that would potentially actuate the crankcase positive pressure trip due to different vent paths between the EDG room and the EDG crankcase. If the crankcase pressure trip were to occur before the EDG start on an emergency signal, the crankcase pressure trip would cause an EDG lockout condition that would prevent normal and emergency starting of the EDG until lockout could be reset locally by plant operators. Thus, the condition could potentially affect both EDGs simultaneously.

The inspectors observed and reviewed the licensee's response to the event, operator logs, and procedural requirements. Specific items associated with this event that were reviewed included, but were not limited to:

- Mitigating systems performance;
- The performance of plant operators in response to the event;
- Event notifications made pursuant to 10 CFR 50.72;

- The potential for any generic issues, including those potentially requiring reporting under 10 CFR Part 21; and
- The licensee's initial corrective actions associated with the event.

The licensee entered this event into their CAP as CR 2016–07816 and commissioned a formal causal analysis, which was still in progress at the conclusion of the inspection period. Initial corrective actions taken by the licensee included establishment of compensatory actions to defeat the EDG crankcase pressure trips on each engine prior to the onset of severe weather in the area surrounding the site. Temporary modifications for each EDG were completed on June 25, 2016, prior to the onset of inclement weather in the local area; these actions disabled the EDG lockout associated with the EDG crankcase positive pressure trip. Documents reviewed as part of this inspection are listed in the Attachment.

This event follow-up review by the inspectors constituted a single inspection sample as defined in IP 71153–05.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 Spring 2016 Groundwater Sampling Results

a. Inspection Scope

The inspectors reviewed the results of a series of expanded groundwater samples taken from wells in the plant owner-controlled area. The sampling of wells was completed as part of the licensee's voluntary groundwater monitoring initiative and in response to the results obtained earlier, as discussed in Section 4OA5 of NRC IRs 05000346/2015001 (ADAMS Accession No. ML15113B387), 05000346/2015002 (ADAMS Accession No. ML15202A203), 05000346/2015003 (ADAMS Accession No. ML15295A107), 05000346/2015004 (ADAMS Accession No. ML16034A366), and 05000346/2016001 (ADAMS Accession No. ML16118A435). Several of the monitoring well locations sampled as part of the licensee's ongoing investigations indicated tritium levels above the 2,000 picocuries per liter (pCi/L) groundwater monitoring program threshold requiring courtesy notifications to state and local government officials and the NRC resident inspectors. The highest tritium concentration, approximately 10,527 pCi/L from a sample obtained on February 10, 2015, was located in a monitoring well, designated MW–22S, on the west side of the plant near the BWST. The formal reporting limit threshold for tritium in groundwater samples is 30,000 pCi/L, as documented in the licensee's ODCM.

The licensee continues to monitor wells in accordance with their groundwater monitoring program as tritium concentrations continue to lower. The inspectors have reviewed the licensee's compliance with their stated offsite agency reporting requirements and continue to track the licensee's corrective actions.

These routine reviews for samples to detect tritium in groundwater did not constitute any additional inspection samples. Instead, they were considered as part of the inspectors' daily plant status monitoring activities.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 14, 2016, the inspectors presented the inspection results to the Site Vice President, Mr. B. Boles, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed with the licensee the scope of material reviewed that was considered to be proprietary. Proprietary information reviewed by the inspectors was controlled in accordance with appropriate NRC policies regarding sensitive unclassified information, and has been denoted as "proprietary" in the attachment.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of Radiological Hazard Assessment and Exposure Controls; Radiation Monitoring Instrumentation; and the RCS Specific Activity PI Verification with the Radiation Protection Manager, Mr. Doug Noble, and other members of the licensee staff on April 8, 2016;
- The inspection results for the area of ISI with the Operations Manager, Mr. Brian Kremer, and other members of the licensee staff on April 8, 2016; and
- The inspection results for the areas of Radiological Environmental Monitoring; RCS Specific Activity; and the Radiological Effluent TS/ODCM Radiological Effluent Occurrences PI Verification with the Manager of Site Work Management, Mr. Randy Patrick, and other members of the licensee staff on June 16, 2016.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

4OA7 Licensee-Identified Violation

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section 2.3.2 of the NRC Enforcement Policy for being dispositioned as an NCV.

.1 Failure to Properly Assess and Implement Compensatory Measures for Fire Protection System Impairment

Plant TS 5.4.1(d), requires, in part, the licensee to establish, implement, and maintain applicable written procedures covering fire protection program implementation. The fire protection program was implemented, in part, by Davis-Besse Procedure DB–FP–00009, "Fire Protection Impairment and Fire Watch," Revision 21. Procedure DB–FP–00009, Step 6.1.2, states: "Upon request, notification, or plant condition that indicates a fire protection system/component impairment exists or will exist, the Shift Manager shall ensure an Impairment Initiation Work Sheet Sections 1, 2A, 2B as needed, 2C and 3 is completed for each impairment to assess the need for compensatory measures." Contrary to this requirement, from 7:30 p.m. on June 26, 2016, to 2:30 p.m. on June 28, 2016, the licensee failed to implement and adequately assess the need for compensatory measures in multiple locations of the turbine building when an applicable fire impairment existed based on plant conditions.

On June 26th, a section of underground fire protection piping was isolated for planned maintenance to repair FP355, South Underground Loop Sectionalizing Valve, due to the valve operator not functioning properly. A fire impairment initiation worksheet was completed for the pre-planned maintenance isolation boundary using the station's fire risk software program prior to the maintenance activity commencing. The assessment concluded that compensatory measures in the form of eight-hour roving fire watches were required for the SBODG building due to the sprinkler systems in those locations being isolated to support the FP355 maintenance activity. These compensatory measures were implemented at approximately 7:30 p.m.

On June 28, while preparing for future fire protection maintenance on FP40, East Underground Loop Sectionalizing Valve, the licensee recognized that FP40 had a pre-existing maintenance condition since November 2015, such that the valve was stuck in the closed position (normally open). This condition, when combined with the already in-progress FP355 valve maintenance activity, was not previously assessed and extended the pre-planned FP355 isolation boundary. As a result, the sprinkler systems in the following locations in the turbine building were isolated with no compensatory measures established:

- Lube Oil Storage Room (Room 249 – one–hour fire watch required);
- Oil Drum Storage Room (Room 337 – one–hour fire watch required);
- Turbine Generator Lube Oil Room (Room 432 – eight-hour fire watch required);
- Janitor Closet (Room 346 – eight-hour fire watch required);
- Lube Oil Filter Room (Room 347 – eight-hour fire watch required); and
- Main Tool Room (Room 341 – eight-hour fire watch required).

Upon identification and reassessment, the appropriate compensatory measures were immediately implemented around 2:30 p.m. on June 28.

The inspectors reviewed this violation using the guidance contained in Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports." The inspectors determined that the licensee's failure to properly implement plant procedures for assessing and establishing compensatory fire watches was a performance deficiency that was reasonably within the licensee's ability to foresee and correct and should have been prevented. This violation was associated with the Initiating Events cornerstone of reactor safety and was of more than minor significance because it was associated with the Initiating Events cornerstone attribute of Protection Against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the licensee failed to implement and adequately assess the need for compensatory measures in multiple locations of the turbine building when an applicable fire impairment existed based on plant conditions. Required fire watch patrols established as compensatory measures should have been performed for the duration of the impairment so that the site's ability to promptly detect and suppress a fire would be maintained.

The inspectors evaluated the violation using IMC 0609, Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings.” Because it involved fire protection, the inspectors transitioned to IMC 0609, Appendix F, “Fire Protection Significant Determination Process.” The violation was characterized according to IMC 0609, SDP, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The violation screened as of very low safety significance (Green), per Attachment 1, Question 1.3.1.A, because it did not affect the ability of the reactor to reach and maintain safe shutdown.

The licensee had entered this issue into their CAP as CR 2016–08266. Corrective actions include but are not limited to immediately establishing required compensatory measures upon identification of the issue and the performance of an apparent cause evaluation.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Boles, Site Vice President
K. Byrd, Director, Site Engineering
D. Blakely, Supervisor, Reactor Engineering
G. Cramer, Manager, Site Protection
J. Cuff, Manager, Training
J. Cunnings, Manager, Site Maintenance
A. Dawson, Manager, Chemistry
D. Hartnett, Superintendent, Operations Training
T. Henline, Manager, Site Projects
J. Hook, Manager, Design Engineering
B. Howard, Manager, Site Outage Management
D. Imlay, Director, Site Performance Improvement
B. Kremer, Manager, Site Operations
G. Laird, Manager, Technical Services Engineering
B. Matty, Manager, Plant Engineering
P. McCloskey, Manager, Site Regulatory Compliance
D. Noble, Manager, Radiation Protection
G. Nordlund, Superintendent, Radiation Protection
W. O'Malley, Manager, Nuclear Oversight
R. Oesterle, Superintendent, Nuclear Operations
R. Patrick, Manager, Site Work Management
D. Saltz, General Plant Manager
J. Sturdavant, Regulatory Compliance
L. Thomas, Manager, Nuclear Supply Chain
J. Vetter, Manager, Emergency Response
G. Wolf, Supervisor, Regulatory Compliance

U.S. Nuclear Regulatory Commission

J. Cameron, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000346/2016002–01	URI	Mispositioned Instrument Air Valves Result in Plant Transient (Section 1R11.2)
05000346/2016002–02	NCV	Failure to Use the Corrective Action Program to Evaluate and Document Degraded Condition with Auxiliary Feedwater Train 2 (Section 4OA2.4)
05000346/2016002–03	NCV	Inadequate Evaluation of Trend Related to A25X Fuse Failures (Section 4OA3.1)
05000346/2016002–04	NCV	Inadequate Post-Manufacture Quality Control Inspections Performed for ASME Section III, Class 2 Reactor Coolant Pump Seal Cavity Vent Line Flexible Hose (Section 4OA3.4)

Closed

05000346/2016001–02	URI	Service Water Header Operability While Using a Degraded No. 3 Component Cooling Water (CCW) Heat Exchanger (HX) SW Outlet Isolation Valve (SW37) for SW Header Pressure Control (Section 4OA2.4)
05000346/2016002–02	NCV	Failure to Use the Corrective Action Program to Evaluate and Document Degraded Condition with Auxiliary Feedwater Train 2 (Section 4OA2.4)
05000346/2016001–03	URI	Shield Building Emergency Ventilation System Operability with Watertight Door No. 108 Inadvertently Left Open (Section 4OA2.5)
05000346/2016–001–00	LER	Reactor Trip During Nuclear Instrumentation Calibrations, and Steam Feedwater Rupture Control System Actuation on High Steam Generator Level (Section 4OA3.1)
05000346/2016002–03	NCV	Inadequate Evaluation of Trend Related to A25X Fuse Failures (Section 4OA3.1)
05000346/2016–002–00	LER	Unanticipated Steam and Feedwater Rupture Control System Actuation (Section 4OA3.2)
05000346/2016–003–00	LER	Leak from Reactor Coolant Pump Seal Piping Flexible Hose Due to Undetected Manufacture Weld Defect (Section 4OA3.4)
05000346/2016002–04	NCV	Inadequate Post-Manufacture Quality Control Inspections Performed for ASME Section III, Class 2 Reactor Coolant Pump Seal Cavity Vent Line Flexible Hose (Section 4OA3.4)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

<u>1R01 Adverse Weather Protection</u>

Condition Reports:
- 2015–11519; Switchyard Foundation Conditions
- 2016–00413; System Monitoring Identified High Runtime Hours for Switchyard ACB34560 Compressor
- 2016–00823; Large Differential in 345KV Phase Voltages
- 2016–01366; Loss of Bayshore Line Following Reactor Trip
- 2016–01619; Rope Used In Switchyard and Material Not Actively Used Left in Switchyard Contrary to NOP–OP–1012 Requirements
- 2016–03924; ABS 34620 Stuck Closed During Shutdown
- 2016–06166; During DB–SC-03020 34561 Opened Unexpected After Closing
- 2016–06738; Startup Transformer X01 Oil Leak Trend
- 2016–08339; Backup Radio Communication To The System Dispatcher (SD), Referenced In DB–OP–01300, Does Not Exist

Procedures:
- DB–OP–01300; Switchyard Management; Revision 11
- DB–OP–02025; Davis-Besse 345 KV Switchyard Alarm Panel 25 Annunciators; Revision 18
- DB–OP–02521; Loss of AC Bus Power Sources; Revision 24
- DB–OP–02546; Degraded Grid; Revision 5
- DB–OP–06311; 345 KV Switchyard No. 1 (Main) Transformer, No. 11 (Auxiliary) Transformer, and Startup Transformers (01 and 02); Revision 44
- DB–OP–06441; Radio Communication System; Revision 17
- DB–SC–03023; Off-Site AC Sources Lined Up and Available; Revision22
- NG–DB–00001; On-Line Risk Management; Revision 14
- NOBP-CC-2008; Transformer, Switchyard, and Grid Reliability Design Interface and Control; Revision 1
- NOP–OP–1003; Grid Reliability Protocol; Revision 7
- NOP–OP–1007; Risk Management; Revision 23
- NOP–OP–1012; Material Readiness and Housekeeping Inspection Program; Revision 8

<u>1R04 Equipment Alignment</u>

Condition Reports:
- 2015–15339; Oil Leak from Containment Spray Pump 1 Motor Inboard Bearing
- 2016–01696; High Compressor Discharge Pressure on CREVS Train 1 in Air Cooled Mode
- 2016–05812; Scaffold Pole Contacts Tube Steel Supporting Pressure Indicator for #1 CREVS Pressure Indicator SW2925A
- 2016–06008; Control Room Emergency Ventilation Systems Compressor 1 Would Not Start During Monthly Test
- 2016–07526; Typographical Error (Copy / Paste) in DB–OP–06233, Auxiliary Feedwater Operating Procedure

Procedures:
- DB–OP–06013; Containment Spray System; Revision 26
- DB–OP–06334; Station Blackout Diesel Generator Operating Procedure; Revision 23
- DB–OP–06233; Auxiliary Feedwater System; Revision 39
- DB–OP–06505; Control Room Emergency Ventilation System Procedure; Revision 20
- DB–SS–03041; Control Room Emergency Ventilation System Train 1 Monthly Test; Revision 18
- DB–SS–03145; Control Room Emergency Ventilation Systems (CREVS) Refueling Interval or Special Test Train 1; Revision 12

Work Orders:
- 200681294; Troubleshoot / Repair CREVS Compressor S33–1

Notifications:
- 601022073; PI28011 (CREVS 1 Compressor Discharge Pressure) Overranged
- 601022104; High Compressor Discharge Pressure in Air Cooled Mode
- 601038727; CREVS 1 Compressor Did Not Start
- 601038983; Re-torque Spectacle Flange at CS20

Drawings:
- M–007B; Steam Generator Secondary System; Revision 61
- M–027A; Auxiliary Building Non-Radioactive Areas and Control Room; Revision 84
- M–060; Auxiliary Feedwater System; Revision 59
- OS–0005; Containment Spray System; Revision 14
- OS–0017A; Sheet 1; Auxiliary Feedwater System; Revision 34
- OS–0017A; Sheet 2; Auxiliary Feedwater System; Revision 4
- OS–0017B; Sheet 1; Auxiliary Feedwater Pumps and Turbines; Revision 25
- OS–0017B; Sheet 2; Auxiliary Feedwater Pumps and Turbines; Revision 9
- OS–0032B; Control Room Emergency Ventilation System; Revision 20
- OS–0041D; Station Blackout Diesel Generator Lube Oil and Jacket Water; Revision 14
- OS–0041E; Station Blackout Diesel Generator Air Start / Engine Air System; Revision 17
- OS–0041F; Station Blackout Diesel Generator Electrical Control and Fuel Oil Systems; Revision 5

Other:
- System 61 – 01: Containment Spray; Plant Health Report; January 29, 2016

1R05 Fire Protection

Condition Reports:
- 2016–03930; Broken Smoke Detector Found On Initial CTMT Entry
- 2016–03936; Door 103 Found Closed, Normally Held Open Magnetically
- 2016–04118; Low Level Trend Identified Challenges To Fire Protection Program/Systems
- 2016–04300; Smoke Detector DS 5608 Containment Exhaust Failed To Alarm During Testing
- 2016–04572; DS8676D Came Into Trouble Alarm Excessively Dirty in FDZ–410, The CTMT East Passage
- 2016–04585; Fire Detection Issues with DS8675M
- 2016–05084; Sparks Discovered Passing Below Platform During Welding Activity in Mechanical Penetration Room 1
- 2016–05263; DB–DS8676L Fire Detector In Containment Has Failed

- 2016–05309; DB EFW / FLEX Project – Flammable Compressed Gas Bottles Stored Beside Oxygen Bottles
- 2016–05311; Grinding Without Flame Retardant PCs

Procedures:
- DB–FP–00003; Pre-Fire Plan Guidelines; Revision 8
- DB–FP–00005; Fire Brigade; Revision 8
- DB–FP–00007; Control of Transient Combustibles; Revision 13
- DB–FP–00009; Fire Protection Impairment and Fire Watch; Revision 21
- DB–FP–00018; Control of Ignition Sources; Revision 12
- DB–OP–02501; Serious Station Fire; Revision 25
- DB–OP–02529; Fire Procedure; Revision 9

Pre-Fire Plans:
- PFP–AB–101; Pipe Tunnel and Equipment Pipe Chase, Rooms 100 and 101, Fire Area B; Revision 5
- PFP–AB–105; ECCS Pump Room 1–1, Room 105, Fire Area AB; Revision 8
- PFP–AB–208; No. 1 Mechanical Penetration Room, Rooms 202, 208 and 208DC, Fire Area AB; Revision 6
- PFP–AB–402; No. 1 Electrical Penetration Room, Room 402, Fire Area DG; Revision 5
- PFP–AB–236; No. 2 Mechanical Penetration Room, Room 236, Fire Area A; Revision 4
- PFP–CB–214; Core Flooding Tank Area, Room 214, Fire Area D; Revision 5
- PFP–CB–215; Let Down Coolers Area, Room 215, Fire Area D; Revision 5
- PFP–CB–216; Steam Generator West D Ring Area, Room 216, Fire Area D; Revision 5
- PFP–CB–218; Steam Generator East D Ring Area, Room 218, Fire Area D; Revision 5
- PFP–CB–220; Incore Instrument Trench Area, Room 220, Fire Area D; Revision 5
- PFP–CB–316; Core Flooding Tank Area, Room 316, Fire Area D; Revision 7
- PFP–CB–317; Containment Air Cooler Area, Room 317, Fire Area D; Revision 7
- PFP–CB–410; East Elevation 603’ and Valve Room Elevation 636’, Room 410, Fire Area D; Revision 4
- PFP–CB–EL565; North Area 565’ Elevation, Rooms 213, 217, and Normal Sump Area, Fire Area D; Revision 7
- PFP–CB–EL603; Fuel Transfer Pool and North and West 603’ Elevation; Rooms 219, 407, and 410A; Revision 4
- PFP–CB–EL653; Entire 653’ Elevation, Rooms 700 and 701, Fire Area D; Revision 4
- PFP–CB–PSV1; Pressurizer, Partial Room 218, Fire Area D; Revision 5
- PFP–CB–RCP1–1; Reactor Coolant Pump 1–1, Partial Room 216, Fire Area D; Revision 5
- PFP–CB–RCP1–2; Reactor Coolant Pump 1–2, Partial Room 216, Fire Area D; Revision 5
- PFP–CB–RCP2–1; Reactor Coolant Pump 2-1, Partial Room 218, Fire Area D; Revision 5
- PFP–CB–RCP2–2; Reactor Coolant Pump 2-2, Partial Room 218, Fire Area D; Revision 5
- PFP–Diesel Generator 1–1 Room, Rooms 318 and 318UL, Fire Area K; Revision 7
- PFP–Diesel Generator 1–2 Room, Rooms 319 and 319A, Fire Area J; Revision 7

Drawings:
- A–0221F; Fire Protection General Floor Plan El. 545’-0” and 555’-0”; Revision 9
- A–0222F; Fire Protection General Floor Plan El. 565’-0”; Revision 18
- A–0223F; Fire Protection General Floor Plan El. 585’-0”; Revision 25
- A–0224F; Fire Protection General Floor Plan El. 603’-0”; Revision 26

Other:
- Fire Hazard Analysis Report; Revision 26
- GEN–SAF–0001; Generation Personal Safety Manual; Revision 2

1R07 Heat Sink Performance

Condition Reports:
- 2016–05172; Less than Minimum Wall Thickness Measured on CCW No. 1 Shell per Order 200486506
- 2016–05353; Eddy Current Results – CCW Heat Exchanger E22–1

Procedures:
- NOP–ER–2006; Service Water Reliability Management Program; Revision 3

Work Orders:
- 200423683; PM 6584: (E 22–1) Eddy Current Testing
- 200486506; PM 0076: (E 22–1) Clean – Inspect CCW Heat Exchanger No. 1

Other:
- NOP–ER–2006, Attachment 1; Heat Exchanger Inspection Report for E 22–1; 4/11/2016
- SWRPM; NRC Generic Letter 89–13 Service Water Reliability Program Manual; Revision 1

1R08 Inservice Inspection Activities

Condition Reports:
- 2014–02485; Non-Consequential Foreign Material Observed on a Fuel Assembly
- 2014–03021; FME – Paint Chip Found on the Reactor Vessel Plenum Just Prior to the Interim Placement of the Reactor Vessel Head
- 2014–03597; Ultrasonic Recordable Indications Detected in Feedwater Reweld FW–7–EBB–3–128–SW67A
- 2014–03533; FME Event – South Cold Leg of 1–2 Steam Generator
- 2014–03613; Ultrasonic Recordable Indications Detected in Feedwater Reweld FW–7–EBB–3–128–SW61A
- 2014–06371; FENOC Replicate CR for Bechtel CR 864 – SG 1–1 Hot Leg Foreign Object Search and Retrieval
- 2014–07810; Visual Examination of Containment Vessel Exterior Moisture Barrier-Recordable Conditions
- 2014–15560; Order 2002877208 Can be Voided due to Change in Program Requirements
- 2014–16291; Equipment Equivalency Issues with AREVA RPV Procedure
- 2016–02476; Radiographic Exam for FLEX Train 1 Decay Heat
- 2016–04082; East Hot Leg High Point Vent Line Pipe Gouge
- 2016–04301; Base Metal Temperature not Verified Prior to Welding
- 2016–04528; Minor Maintenance Items Noticed by NRC Inspector
- 2016–04595; Additional NRC Question Regarding Weld Metal Temperature

Procedures:
- NA–QC–05560; Visual Examination Procedure for VT–1, VT–3, and General Visual Examinations; Revision 11
- EN–DP–01501; Boric Acid Corrosion Control Inspections; Revision 17
- NOP–ER–2001; Boric Acid Corrosion Control Program; Revision 12
- NOP–LP–2001; Corrective Action Program; Revision 32

Work Orders:
- 200511892; Aux Feedwater Pipe-to-Pipe Weld
- 200532106; FW34, Main Steam Pipe-to-Elbow
- 200511889; Main Feedwater Pipe-to-Elbow Weld

Drawings:
- ISIM2–203A; Inservice Inspection Isometric, Main Steam System; Revision 3
- M–0203A; Main Steam System; Revision 22
- M–0203B; Main Steam System; Revision 18
- M–0207C; Main Feedwater System; Revision 18

Non-Destructive Examination (NDE) Reports:
- 18–UT–053; UT Examination NDE Report for Valve-to-Elbow Weld,
- MS–3A–EBB–1–30–FW15A; 2/8/2014
- 18–UT–078; UT Examination NDE Report for Tee-to-Reducer Weld, FW–7–EBB–3–127–SW61A; 2/21/2014
- 18–UT–032; UT Examination NDE Report for Pipe-to-Elbow Weld, FW–7–EBB–3–128–SW67A; 2/1/2014
- 18–UT–079; UT Examination NDE Report for Pipe-to-Elbow Weld, FW–7–EBB–3–128–SW67A; 2/21/2014
- Vendor–011; Ultrasonic Examination NDE Report for Valve-to-Elbow Weld, MS–3A–EBB–1–30–FW15A; 3/5/2014

Other:
- 55–GWP01–016; ASME Code and Safety Related Applications for Welding; Revision 16
- 54–ISI–240–047; Visible Solvent Removable Liquid Penetrant Examination Procedure; Revision 47
- 54–ISI–864–003; Manual Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds; Revision 3
- 54–ISI–369–003; VT–1, VT–3, General and Detailed Visual Examinations; Revision 3
- 54–ISI–615–001; Procedure for Manual and Semi-Automated Phased Array Ultrasonic Examination of Socket Fillet Welds; Revision 1
- 54–ISI–836–014; Ultrasonic Examination of Austenitic Piping Welds; Revision 14
- 54–ISI–840–008; Straight Beam Ultrasonic Examination of Studs and Bolts; Revision 8
- 54–ISI–864–005; Manual Phased Array Procedure for Weld Overlaid Similar and Dissimilar Metal Welds; Revision 5
- PO 45249949; W256, Auxiliary FDW Riser Stub to Header Pipe Weld; 8/21/2013
- WPS A1–3–1; WPS for Manual GTAW/SMAW P1 to P1 Material; Revision 1
- PQR 001; PQR for WPS A1–3–1; 3/14/1977
- PQR 015; PQR for WPS A1–3–1; 5/17/1979
- PQR 064; PQR for WPS A1–3–1; Revision 1
- WPS P1–AT–Lh; WPS for Manual GTAW/SMAW P1 to P1 Material; Revision 0
- PQR 695; PQR for WPS P1–AT–Lh; 6/27/1979
- PQR 1173A; PQR for WPS P1–AT–Lh; 12/3/1997
- PQR 1259; PQR for WPS P1–AT–Lh; 7/24/2000
- PQR 1310; PQR for WPS P1–AT–Lh; 6/28/2001
- GWS–1; Bechtel General Welding Standard; Revision 3
- M–452Q; Specification for Operational Phase for Procurement of Pipe, Tubing, Fittings, Bolting, Bars and Plates; Revision 3

1R11 Licensed Operator Requalification Program and Licensed Operator Performance

Condition Reports:
- 2016–07228; Blown Fuse in DEHC for CIV 2
- 2016–07235; ARTS Channel 4 +48V Power Supply LED Light Out
- 2016–07282; Inadvertent Technical Specification 3.4.1, Condition A, Entry
- 2016–07286; Plant Transient Due to Possible Feedwater Issue
- 2016–07337; Misposition Event: SP6A Valves Found Out of Position
- 2016–08363; Operating Crew Performance Critique for Feedwater Transient on 5/31/16

Procedures:
- DB–MI–03211; Channel Functional Test of SFRCS Actuation Channel 1 Logic for Mode 1; Revision 19
- DB–NE–03212; Zero Power Physics Testing; Revision 11
- DB–NE–06202; Reactivity Balance Calculations; Revision 10
- DB–OP–02526; Primary to Secondary Heat Transfer Upset; Revision 4
- DB–OP–06011; High Pressure Injection System; Revision 31
- DB–OP–06012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 66
- DB–OP–06014; Core Flooding System Procedure; Revision 28
- DB–OP–06023; Fill, Drain, and Purification of the Refueling Canal; Revision 19
- DB–OP–06202; Turbine Operating Procedure; Revision 28
- DB–OP–06224; Main Feed Pump and Turbine; Revisions 37 – 38
- DB–OP–06301; Generator and Exciter Operating Procedure; Revision 28
- DB–OP–06401; Integrated Control System Operating Procedure; Revision 24
- DB–OP–06402; Control Rod Drive Operating Procedure; Revisions 26 – 27
- DB–OP–06900; Plant Heatup; Revision 63
- DB–OP–06901; Plant Startup; Revision 38
- DB–OP–06902; Power Operations; Revisions 54 – 56
- DB–OP–06904; Shutdown Operations; Revisions 46 – 47
- DB–OP–06912; Approach to Criticality; Revision 18
- DB–PF–03083; High Pressure Injection Pump 2 Baseline Test; Revision 3
- DB–SC–03272; Control Rod Exercising Test; Revision 5
- DB–SS–04150; Main Turbine Stop Valve Test; Revision 13
- DB–SS–04151; Main Turbine Control Valve Test; Revision 15
- DB–SS–04152; Main Turbine Combined Intermediate Valve Test; Revision 10
- DB–SS–04163; Main Turbine Overspeed Trip Test; Revision 10
- NOP–OP–1004; Reactivity Management; Revision 13
- NOP–TR–1200; Conduct of Training; Revision 3

Other:
- Evolution Specific Reactivity Plan; Cycle 20 Zero Power Physics Testing and Escalation to 100% Full Power; Revision 0
- Periodic Reactivity Plan; Cycle 20 Reactor Operator Guidance 0 Effective Full Power Days (EFPD) to 33 EFPD; Revision 0

1R12 Maintenance Effectiveness

Condition Reports:
- 2014–018484; System Monitoring: TERC3B2 Has a Degraded (High Resistance) Wire or Connection Inside Containment or the Containment Penetration

- 2015–04686; Reactor Protection System Channel 1 Declared Inoperable Due to Erroneous Loop 1 Hot Leg Temperature Indication
- 2015–05053; Step Change in T721 Narrow Range Hot Leg Temperature, RPS Channel 1
- 2015–07652; System Monitoring Identified Decreasing Trend in Indication for TERC3B2
- 2015–08549; System Monitoring Identified Increasing Trend in Indication for TERC3B2 "RC Loop 1 Hot Leg Narrow Range Temperature Element"
- 2015–09237; RPS Cannel 1 Hot Leg Narrow Range Temperature Indication Lower Following Checks in Order 200647179
- 2015–10750; DB–TERC3B2 Reading Erratic
- 2016–04587; TERC3B2/3B5 Insulation Burnt Off
- 2016–05287; TDTRC8A Found Out of Tolerance During Performance of WO 200606992 and DB–MI–04229
- 2016–05782; New Hot Leg RTD Assembled Incorrectly
- 2016–05792; Incorrectly Assembled Conax Connectors

Work Orders:
- 200678040; Replace TERC3A1/3A2
- 200678041; Replace TERC3A4/3A6
- 200678042; Replace TERC3B2/3B5
- 200678043; Replace TERC3B3/3B4
- 200678045; Replace TERC3B1/3B6

Other:
- Davis-Besse Nuclear Power Plant Design Basis Assessment Report; Second Half 2015
- Davis-Besse Plant Health Report; Second Half 2015
- MRPM; Maintenance Rule Program Manual; Revision 35

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

Condition Reports:
- 2005–05686; ES 370 Failed to Return to Its Original Position During the NRV Performance Test
- 2014–14586; ES278 NRV Three-Way Test Valve Failed in the Test Position
- 2016–04208; Reactor Coolant Pressure Boundary Leakage at Flexible Hose Assembly for RCP 1–1 First Stage Seal Cavity Vent
- 2016–04287; 19R BACC: P36–1 RCP 1st Stage Seal Vent Line Flex Hose Leakage
- 2016–05270; Wrong Weld Rod Filler Material Used on Flex Hose Installation
- 2016–05350; Debris Found in Borated Water Storage Tank During Diving Operations
- 2016–05416; License Renewal – 1R19 Borated Water Storage Tank Inspection Results 200424017
- 2016–06594; 19R Order Not Completed as Written
- 2016–06600; Three Extraction Steam System Solenoids Were Installed in an Improper Configuration During 1R19
- 2016–06617; Feed Water Heater 1–4 Trip as a Result of Work on SV298A
- 2016–07282; Inadvertent Technical Specification 3.4.1, Condition A, Entry
- 2016–07286; Plant Transient Due to Possible Feedwater Issue
- 2016–07337; Misposition Event: SP6A Valves Found Out of Position

Procedures:
- DB–MI–03211; Channel Functional Test of SFRCS Actuation Channel 1 Logic for Mode 1; Revision 19

- DB–MI–09009; Maintenance of ASCO Solenoid Valves; Revision 3
- DB–MN–00006; Control of Lifting and Handling of Heavy Loads; Revision 17
- DB–OP–02526; Primary to Secondary Heat Transfer Upset; Revision 4
- DB–OP–06228; Deaerator System Operating Procedure; Revision 4
- DB–OP–06229; High Pressure Feedwater Heater System Operation
- DB–OP–06401; Integrated Control System Operating Procedure; Revision 24
- DB–SS–04090; Extraction Steam Non-Return Valves; Revision 6
- NOP–OP–1007; Risk Management; Revision 22
- NOP–OP–4105; Diving in Contaminated Systems; Revision 2
- NOP–WM–1001; Order Planning Process; Revision 22
- NOP–WM–4300; Order Execute Process; Revision 12
- NOP–WM–5003; Rigging, Lifting, and Load Handling; Revision 5
- NG–DB–00001; On-Line Risk Management; Revision 14

Business Practices:
- DBBP–OPS–0003; On-Line Risk Management Process; Revision 12
- DBBP–OPS–0011; Protected Equipment Posting; Revision 9
- NOBP–OP–0007; Conduct of Infrequently Performed Tests or Evolutions; Revision 5
- NOBP–WM–1004; FENOC Planning Writers' Guide; Revision 2
- NORM–ER–3707; Tanks; Revision 1

Work Orders:
- 200424017; PM 9995: (T 10) Borated Water Storage Tank Internal Inspection
- 200510612; ECP 15–0361 Solenoid Valve SV377A (Extraction Steam Solenoid Valve) Replacement
- 200682412; Re-install SV298A (Extraction Steam Solenoid Valve) Correctly
- 200682779; Troubleshoot/Repair/Replace NNI Modules for Feedwater Flow

Underwater Engineering Services, Inc. UT Wall Thickness Inspection Records for BWST:
- NUC2016118–UT–001 through NUC2016118–UT–022; BWST Floor Plates 1 through 22; 4/15/2016

Notifications:
- 601041203; Solenoid Valve 298A is Installed Backwards

Drawings:
- E–0046B, Sheet 20; Steam and Condensate Deaerator Heater Low Pressure Turbine Extraction Valves; Revision 7
- OS–013, Sheet 3; Extraction Steam System; Revision 19

Other:
- GEN–SAF–0003; Generation Confined Space Entry Program; Revision 1

<u>1R15 Operability Determinations and Functionality Assessments</u>

Condition Reports:
- 2012–01332; API for Control Rod 7–3 Erratic
- 2012–01620; ODMI: Contingency for a Potential Control Rod 7–3 Asymmetric Position Indication Condition
- 2014–18484; System Monitoring: TERC3B2 Has a Degraded (High Resistance) Wire or Connection Inside Containment or the Containment Penetration

- 2015–04686; Reactor Protection System Channel 1 Declared Inoperable Due to Erroneous Loop 1 Hot Leg Temperature Indication
- 2015–05053; Step Change in T721 Narrow Range Hot Leg Temperature, RPS Channel 1
- 2015–07652; System Monitoring Identified Decreasing Trend in Indication for TERC3B2
- 2015–08549; System Monitoring Identified Increasing Trend in Indication for TERC3B2
- 2015–09237; RPS Channel 1 Hot Leg Narrow Range Temperature Indication Lower Following Checks in Order 200647179
- 2015–10750; DB–TERC3B2 Reading Erratic
- 2016–04587; TERC3B2/3B5 Insulation Burnt Off
- 2016–05782; New Hot Leg RTD Assembled Incorrectly
- 2016–05792; Incorrectly Assembled Conax Connectors
- 2016–07289; Erratic API/RPI Comparison Reading on Rod 7–3 (H12)
- 2016–07315; CRD Abnormal Procedure Entry Due to Rod 7–3 Degraded API/RPI Indication
- 2016–08249; Rod 7–3 (H12) Indication Drift

Procedures:
- DB–OP–00018; Inoperable Equipment Tracking Log; Revision 19
- DB–OP–02516; CRD Malfunctions; Revision 15
- DB–OP–06402; CRD Operating Procedure; Revision 27

Business Practices:
- DBBP–OPS–0018; Non-Control Room Assigned Operator Coordination During Abnormal and Emergency Operations; Revision 2

Work Orders:
- 200684075; Troubleshoot / Resolve Rod 7–3 API Drift

Other:
- Operations List of Control Room Deficiencies, Work Arounds, and Operator Burdens; 6/20/2016
- Operations Loss of Function List; 6/20/2016

<u>1R18 Plant Modifications</u>

Condition Reports:
- 2014–08772; SG Operate Levels High
- 2016–05315; DB–MM–09382 Step 8.4.10.c – SG Orifice Plate Locking Screw Binding
- 2016–05679; DCRDCS Project - Position Indication Field Cable Connectors Pin Movement
- 2016–06044; Flow Indicator Found Out of Service During Testing of HPI Pump No. 2
- 2016–06051; High Pressure Injection Pump 2 (HPI) Baseline Test DB–PF–03083 Design Basis Review
- 2016–06083; DCRDCS Project – DB–TP–10405 DCRDCS Post Installation Testing for ECP 12–0272 Test Deficiency
- 2016–06213; Three Reactor Trip Breakers (B,C, and D) Opened Unexpectedly During Performance of DB–MI–03012

Procedures:
- DB–MI–03012; Channel Functional Test of Reactor Trip Breaker A, RPS Channel 2 Reactor Trip Module Logic, and ARTS Channel 2 Output Logic; Revision 36
- DB–MM–09382; ROTSG Secondary Handhole Maintenance and Orifice Plate Adjustment; Revision 2

- DB–OP–06011; High Pressure Injection System; Revision 31
- DB–OP–06402; Control Rod Drive Operating Procedure; Revisions 26 – 27
- DB–PF–03083; High Pressure Injection Pump 2 Baseline Test; Revision 3
- DB–TP–10405; Digital Control Rod Drive Control System (DCRDCS) Post Installation Testing for ECP 12–0272; Revision 0
- DB–TP–10406; Digital Control Rod Drive Control System (DCRDCS) Power Ascension Testing for ECP 12–0272; Revision 0

Work Orders:
- 200604700; Update ROTSG Orifice Plate Setting
- 200617857; ECP 12–0272–006 Replacement of the Inductrol Voltage Regulator Control Panel
- 200626410; OTSG 1B Orifice Plate Adjustment
- 200626411; OTSG 2A Orifice Plate Adjustment
- 200635524; ECP 12–0272–000 CRD Testing – DB–TP–10406 – Power Ascension Testing
- 200635748; ECP 12–0272–000 CRD Testing – DB–TP–10405 – Post Installation Testing
- 200648129; Adjust Steam Generator 2 Orifice Plate

Engineering Change Packages (ECPs):
- 12–0272–000; Replacement of the Control Rod Drive System with a Digital Control Rod Drive System; Revision 0
- 14–0396–000; Update Steam Generator Orifice Plate Setting; Revision 0
- 14–0396–001; Update SG 1–1 Orifice Plate Setting; Revision 0
- 14–0396–002; Update SG 1–2 Orifice Plate Setting; Revision 0
- 15–0187–000; Replacement of Train 2 High Pressure Injection Pump Motor (MP58–2); Revision 0

Other:
- M–515–00068; AREVA Digital Control Rod Drive System Instruction Manual for Davis-Besse; Revision 1 [PROPRIETARY]

<u>1R19 Post Maintenance Testing</u>

Condition Reports:
- 2012–02470; 2012 CDBI Self-Assessment: Battery Connection Resistance
- 2016–04977; 1P/1N Batteries Washers Installed Wrong
- 2016–05570; FLEX CWRT Charging and Booster Pump Interlock Not Functional
- 2016–05734; FLEX BWST Charging Pump Suction Connection Leak
- 2016–05770; 19R BACC: DH 210 Active Borated Water Leakage
- 2016–06044; Flow Indicator Found Out of Service During Testing of HPI Pump No. 2
- 2016–06051; High Pressure Injection Pump 2 (HPI) Baseline Test DB–PF–03083 Design Basis Review
- 2016–06083; DCRDCS Project – DB–TP–10405 DCRDCS Post Installation Testing for ECP 12–0272 Test Deficiency
- 2016–06120; Service Water Discovered Lined Up to #1 AFP Suction
- 2016–06149; DB–OP–06233 - AFW Gas Voids Detected
- 2016–06239; BACC – Dry Boric Acid Found Under Vessel and Normal Sump Area
- 2016–06299; FI4630 Indicates 0 gpm
- 2016–06337; DB–OP–06233 - AFW Gas Voids Detected - CR 2016–06149 Followup
- 2016–06471; 19R BACC: RC226 Swagelok Cap
- 2016–06474; 19R BACC – A Packing Leak was Found on MU248
- 2016–06475; 1R19 BACC – PTRC2B2 Packing Leak Identified

- 2016–06476; 1R19 BACC – Containment Spray Spectacle Flange Leak Identified
- 2016–06478; 1R19 BACC – FTRC1A3 Leak Identified
- 2016–06479; 1R19 BACC – A Packing Leak was Found on MU262
- 2016–06482; 1R19 BACC – FTRC1B1 Packing Leak Identified
- 2016–06483; BACC – DR19 Potentially Borated Water Active Leak
- 2016–06485; Hydraulic Fluid Leak from Snubber SNC257
- 2016–06488; 19R BACC – A Swagelock Fitting Leak was Found on RC13B
- 2016–06491; CRD Insertion Time Test DB–SC–03270 Anomalies
- 2016–06525; Cycle 20 Zero Power Physics Testing – Failure to meet Measured Total Worth Deviation Acceptance Criteria

Procedures:
- DB–ME–03001; Station Batteries Quarterly Surveillance; Revision 22
- DB–ME–03002; Station Battery Service and Performance Discharge Test; Revision 20
- DB–ME–03004; Station Battery Monthly Surveillance; Revision 6
- DB–MM–05003; Vibration Monitoring; Revision 11
- DB–MM–09059; Packing Valves; Revision 20
- DB–ME–09200; Station Battery Maintenance Guidelines; Revision 17
- DB–MM–09245; General Welding Procedure (ASME/ANSI Applications); Revisions 9 - 10
- DB–NE–03212; Zero Power Physics Testing; Revision 11
- DB–NE–04382; Core Alignment Verification; Revisions 5 – 6
- DB–NE–06101; Fuel/Control Component Shuffle; Revisions 25 – 26
- DB–NE–06202; Reactivity Balance Calculations; Revision 10
- DB–PF–03010; RCS Leakage Test; Revision 15
- DB–PF–03065; System Leakage Tests; Revision 14
- DB–PF–03083; High Pressure Injection Pump 2 Baseline Test; Revision 3
- DB–OP–06011; High Pressure Injection System; Revision 31
- DB–OP–06233; Auxiliary Feedwater System; Revision 39
- DB–OP–06402; Control Rod Drive Operating Procedure; Revisions 26 – 27
- DB–SC–03270; Control Rod Assembly Insertion Time Test; Revision 14
- DB–SP–03152; AFW Train 1 Level Control, Interlock, and Flow Transmitter Test; Revision 30
- DB–SP–03165; AFW Train 1 and AFW Train 2 Flow Path to SG Verification; Revision 0
- DB–TP–10405; Digital Control Rod Drive Control System (DCRDCS) Post Installation Testing for ECP 12–0272; Revision 0
- DB–TP–10406; Digital Control Rod Drive Control System (DCRDCS) Power Ascension Testing for ECP 12–0272; Revision 0
- DB–TP–12421; Testing of the Flex Charging Pumps for ECP 13–0463; Revision 0
- DB–TP–12143; Supplying Power to Flex Charging Pumps for Testing During 1R19; Revision 0
- NOP–ER–2001; Boric Acid Corrosion Control Program; Revision 12
- NOP–OP–4016; Control of Radiography Operations; Revision 4

Work Orders:
- 200558138; ME3001: 1N Station Battery Yearly
- 200558139; ME3001: 1P Station Battery Yearly
- 200592216; SP–3152 AFW Train 1 Level Control, Interlock, and Flow Transmitter Test
- 200593009; ME3004: 1N Station Battery Monthly
- 200593010; ME3004: 1P Station Battery Monthly
- 200595349; ME3001: 1N Station Battery Quarterly
- 200595401; ME3001: 1P Station Battery Quarterly
- 200598574; ME3004: 1N Station Battery Monthly
- 200598575; ME3004: 1P Station Battery Monthly

- 200606979; PM 4234 1P Replace Station Batteries
- 200606981; PM 4233 1N Replace Station Batteries
- 200610316; ECP 13–0196–003 Tie-in 6” EFW to AFW Train 1
- 200610319; Flex Mod Welds Mechanical Penetration Rooms 3 and 4
- 200610320; 3” EBD–14 Pipe to 3” WOL Inlet (Mechanical Penetration Room 4)
- 200612158; Zero Power Physics Test Refueling Outage
- 200612163; Core Alignment Verification
- 200629114; ME3002: 1N Performance Test
- 200629115; ME3002: 1P Performance Test
- 200634926; SP3165 AFW Train 1 Cold Shutdown
- 200635524; ECP 12–0272–000 CRD Testing – DB–TP–10406 – Power Ascension Testing
- 200635748; ECP 12–0272–000 CRD Testing – DB–TP–10405 – Post Installation Testing
- 200646541; EF2 Valve Radiography
- 200663416; ECP 13–0463–006: Install Flex Charging Pumps 1–1 & 2–1 in Room 100
- 200663417; ECP 13–0463–006: Install Flex Charging Pumps 1–2 & 2–2 in Room 124
- 200674351; ECP 13–0463–006: Test Flex Charging Pump P296–1
- 200674352; ECP 13–0463–006: Test Flex Charging Pump P296–2

Drawings:
- M–0060; Auxiliary Feedwater System; Revision 59
- OS–017A; Auxiliary Feedwater System; Revision 33

Vendor Manuals:
- E–018Q–00017–09; GNB Station Batteries; 2/2011

Engineering Change Packages:
- ECP 13–0463–006; Install Flex RCS Charging Components (Not Connected to System); Revisions 3 – 4

Other:
- 03–9254353; Radiography Plan for Flex and NPS; Revision 2 [PROPRIETARY]
- Cycle 20 Core Fuel Assembly Verification Video
- Davis-Besse Cycle 19 Core Map; 4/20/2014
- Davis-Besse Cycle 20 Core Map; 4/25/2016
- Evolution Specific Reactivity Plan; Cycle 20 Zero Power Physics Testing and Escalation to 100% Full Power; Revision 0
- FS1-0026897; AREVA Assessment of D-B Cycle 20 ZPPT CRG Worth Measurement; Revision 1 [PROPRIETARY]
- GNB Industrial Power Specifications; General Purpose Flooded Batteries; Section 33.40 2007–01
- Mode 3 Walkdown Plan, Davis-Besse ISI Program; 5/7/2016
- Mode 3 VT–2 Examination Pre-Job Brief Power Point Presentation; 5/72016
- Periodic Reactivity Plan; Cycle 20 Reactor Operator Guidance 0 Effective Full Power Days (EFPD) to 33 EFPD; Revision 0

<u>1R20 Outage Activities</u>

Condition Reports:
- 2016–04453; Nuclear Fuel – 19R: Fuel to Baffle Interaction Fuel Rod Wear Observed on Fuel Assembly
- 2016–04454; Nuclear Fuel – 19R: Unusual Crud Indications Overserved on Fuel Assembly

- 2016–04492; CWRT 2 Level Slowly Rising, CWRT 1 Level Slowly Lowering
- 2016–04513; Confined Space Rescue Plan and Team Not in Place
- 2016–04517; Human Performance Event During 1R19 Core Offload
- 2016–05791; 1R19 BACC: During Fill of Refueling Canal Leakage Was Noted Coming from the Electrical Penetrations in the Swiss Cheese Area
- 2016–05922; 19RFO – AREVA: Near Miss – Employee Fall into Flooded Refueling Canal
- 2016–06080; DB 1R19 Stud Elongation Measurement Tooling Failure Results in Rework

Procedures:
- DB–MM–09234; Equipment Hatch Removal and Reinstallation; Revision 8
- DB–NE–03212; Zero Power Physics Testing; Revision 11
- DB–NE–04382; Core Alignment Verification; Revisions 5 – 6
- DB–NE–06202; Reactivity Balance Calculations; Revision 10
- DB–OP–03013; Containment Daily Inspection & Containment Closeout Inspection; Revision 10
- DB–OP–06000; Filling and Venting the Reactor Coolant System; Revision 29
- DB–OP–06002; RCS Draining and Nitrogen Blanketing; Revision 21
- DB–OP–06005; RC Pump Operation; Revision 32
- DB–OP–06011; High Pressure Injection System; Revision 31
- DB–OP–06012; Decay Heat and Low Pressure Injection System Operating Procedure; Revision 66
- DB–OP–06014; Core Flooding System Procedure; Revision 28
- DB–OP–06023; Fill, Drain, and Purification of the Refueling Canal; Revision 19
- DB–OP–06202; Turbine Operating Procedure; Revision 28
- DB–OP–06224; Main Feed Pump and Turbine; Revisions 37 and 38
- DB–OP–06301; Generator and Exciter Operating Procedure; Revision 28
- DB–OP–06401; Integrated Control System Operating Procedure; Revision 24
- DB–OP–06402; Control Rod Drive Operating Procedure; Revisions 26 - 27
- DB–OP–06900; Plant Heatup; Revision 63
- DB–OP–06901; Plant Startup; Revision 38
- DB–OP–06902; Power Operations; Revisions 54 - 56
- DB–OP–06904; Shutdown Operations; Revisions 46 - 47
- DB–OP–06912; Approach to Criticality; Revision 18
- DB–PF–03083; High Pressure Injection Pump 2 Baseline Test; Revision 3
- NOP–OP–1004; Reactivity Management; Revision 13
- NOP–OP–1005; Shutdown Defense in Depth; Revision 15
- NOP–OP–1007; Risk Management; Revision 22
- NG-DB–00117; Shutdown Defense in Depth Assessment; Revision 17

FENOC Business Practices:
- NOBP-OP–0007; Conduct of Infrequently Performed Tests or Evolutions; Revision 5

Other:
- 19RFO Shutdown Defense in Depth Report; Revision 0
- ALARA Plan
- Containment Leakage Rate Testing Program; Revision 11
- Evolution Specific Reactivity Plan for Cycle 19 End-of-Cycle Shutdown While Restoring RCS Average Coolant Temperature to 582 °F; Revision 0

1R22 Surveillance Testing

Condition Reports:
- 2010–74444; 16RFO Local Leak Rate Test Failure of DR2012A
- 2012–08574; As-Left LLRT on Penetration 13, Valve DR2012A Failed to Pass the Acceptance Limit
- 2012–08972; As-Left LLRT on Penetration 13, Valve DR2012A Failed to Pass the Administrative Acceptance Limit During Troubleshooting under CR 2012–08574
- 2016–04242; Abnormal Trend of FLRMs (LLRT equipment) Being Received From Calibration With Performance Issues
- 2016–04913; P101 Module P2L4G Exceeds LLRT Leakage Criteria
- 2016–04916; P101 Module P2L4G Nitrogen Supply Check Valve NN1036 Exceeds IST Leakage Criteria
- 2016–05338; Packing Leak on CV5079 Impacts LLRT Results on Penetration 8J
- 2016–05441; DR2012A (Containment Normal Sump Discharge) Exceeds LLRT Leakage Criteria
- 2016–05587; Unexpected Restraining Wire on SS165 during As Left LLRT of SS235A/B
- 2016–05653; Seat Leakage on CV5071 Impacts LLRT Results on Penetration 8B Inboard Flange
- 2016–05654; Seat Leakage on CV5073 Impacts LLRT Results on Penetration 8D Inboard Flange
- 2016–05952; ODMI to Address Completing No. 2 EDG Monthly Test (8 hour run) While at Reduced Inventory (Flange Level)
- 2016–06048; EDG 2 DC Turbo Oil Pump (P147-6) Failed to Start When AC Turbo Oil Pump Was Turned Off
- 2016–06302; Clamps on Top of EDG 2 Not Latched
- 2016–06643; Configuration of Oil Metering Tube Hoods Within EDG Intake Air Filters
- 2016–06923; EDG No. 2 Logs Lost When Uploading
- 2016–06933; Low Air Start Pressure on DA 45
- 2016–07228; Blown Fuse in DEHC for CIV 2
- 2016–07235; ARTS Channel 4 +48V Power Supply LED Light Out

Procedures:
- DB–OP–06011; High Pressure Injection System; Revision 31
- DB–OP–06316; Diesel Generator Operating Procedure; Revisions 57 – 58
- DB–MM–05003; Vibration Monitoring; Revision 11
- DB–PF–00201; Inservice Testing of Pumps and Valves; Revision 12
- DB–PF–00205; Containment Leakage Test Program; Revision 7
- DB–PF–06704; Pump Performance Curves; Revision 36
- DB–PF–03008; Containment Local Leakage Rate Tests; Revision 20
- DB–PF–03083; High Pressure Injection Pump 2 Baseline Test; Revision 3
- DB–SC–03071; Emergency Diesel Generator 2 Monthly Test; Revision 34
- DB–SC–03272; Control Rod Exercising Test; Revision 5
- DB–SS–04150; Main Turbine Stop Valve Test; Revision 13
- DB–SS–04151; Main Turbine Control Valve Test; Revision 15
- DB–SS–04152; Main Turbine Combined Intermediate Valve Test; Revision 10
- NG–DB–00117; Shutdown Defense in Depth Assessment; Revisions 15 – 17
- NOP–OP–1005; Shutdown Defense in Depth; Revision 15
- NOP–OP–1007; Risk Management; Revision 22
- NOP–OP–1010; Operational Decision-Making; Revision 6

Work Orders:
- 200676652; PF–3008–064 Containment Vessel LLRT Penetration 68A

Notifications:
- 601038853; EDG 2 DC Turbo Oil Pump Failed to Start

Other:
- ISTB4; Pump and Valve Basis Document, Volume IV – Maximum Allowable Leakage Rate (MALR) Basis; Revisions 18 – 20

<u>2RS1 Radiological Hazard Assessment and Exposure Controls</u>

Condition Reports:
- 2016–04867; NRC Inspector Potential Concern for Unmonitored Dose When Dose Gradient Exists
- 2016–04717; RP NRC Inspection Identified RWP Not in Alignment with Air Sampling Procedure
- 2016–00806; Workers Observed Wearing Electronic Alarming Dosimeter on Inside Protective Clothing

Procedures:
- NOP–OP–4010; Determination if Radiological Risk; Revision 8
- NOP–OP–4101; Access Controls for Radiologically Controlled Areas; Revision 11
- NOP–OP–4104; Job Coverage; Revision 6
- NOP–OP–4204; Special External Exposure Monitoring; Revision 8
- DB–HP–01115; OTSG Entries; Revision 14
- DB–HP–01152; Performance of High Exposure Work; Revision 18
- DB–HP–01702; Transfer, Handling, and Storage of Radioactive Material; Revision 24

FENOC Business Practices:
- NOBP–NF–3102; Control of Non-Special Nuclear Material in the Fuel Pools; Revision 1

Radiation Work Permits:
- 2016–5114; Incore Cutting/Cask Transfer/Incore Tank Decontamination; Revision 0
- 2016–5104; Reactor Head Disassembly/Reassembly Work Activities; Revision 0
- 2016–5202; Alloy 600 Mitigation on all Four High Pressure Injection Nozzles; Revision 0
- 2016–5206; Removal and Replacement of Reactor Coolant Pump Motors (RCPMs) and Seals; Revision 0
- 2016–5302; Installation/Removal of Roger and Ranger in the Steam Generator Bowls; Revision 0

Other:
- Alpha Area Level Assessment, Davis-Besse; 04/19/2015

<u>2RS2 Occupational As-Low-As-Reasonably-Achievable Planning and Controls</u>

Condition Reports:
- 2016–04524; 1R–19 Containment Dose Rates Higher Than Expected

Procedures:
- NOP–OP–4005; ALARA Program; Revision 4

- NOP–OP–4012; Outage Radiological Goal Setting; Revision 0
- NOP–OP–4107; Radiation Work Permit (RWP); Revision 15

Radiation Work Permits and Associated ALARA Files:
- 2016–5114; Incore Cutting/Cask Transfer/Incore Tank Decontamination; Revision 0
- 2016–5104; Reactor Head Disassembly/Reassembly Work Activities and Associated ALARA Files; Revision 0
- 2016–5202; Alloy 600 Mitigation on all Four High Pressure Injection Nozzles; Revision 0
- 2016–5206; Removal and Replacement of Reactor Coolant Pump Motors (RCPMs) and Seals; Revision 0
- 2016–5302; Installation/Removal of Roger and Ranger in the Steam Generator Bowls; Revision 0

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

Procedures:
- NOP–OP–4702; Air Sampling; Revision 5
- NOP–OP–4703; Determination of Alpha Monitoring Levels; Revision 3

2RS7 Radiological Environmental Monitoring Program

Procedures:
- DB–CN–00015; Radiological Environmental Monitoring Program; Revision 2
- DB–CN–03004; Radiological Monitoring Quarterly, Semiannual and Annual Sampling; Revision 7
- DB–CN–03005; Radiological Monitoring Weekly, Semimonthly and Monthly Sampling; Revision 4
- DB–CN–03023; Annual Land Use Census; Revision 2
- DB–CN–04060; A Priori Minimum Detectable Activity for HPGE Gamma Spectrometers; Revision 1
- DB–CN–10101; Radiological Environmental Monitoring Program Enhancement Sampling; Revision 4
- NOP–OP–2012; Groundwater Monitoring; Revision 9
- EN–DP–00103; Meteorological Monitoring Program; Revision 5
- DB–HP–01452; Air Sampler Calibration; Revision 8

Calibration Data Sheets:
- DB–0560–1; Air Sampler LI# ECP–1; November 19, 2015
- DB–0560–1; Air Sampler LI# ECP–5; May 17, 2016
- DB–0560–1; Air Sampler LI# ECP–8; February 23, 2016
- DB–0560–1; Air Sampler LI# ECP–11; January 5, 2016
- DB–0560–1; Air Sampler LI# ECP–19; March 8, 2016
- DB–0560–1; Air Sampler LI# ECP–20; January 19, 2016

Other:
- Offsite Dose Calculation Manual; Revision 32
- 2014 Annual Radiological Environmental Operating Report, May 2015
- 2015 Annual Radiological Environmental Operating Report, May 2016
- Evaluation Report; QSL #10114; Environmental Inc. – Midwest Labs; January 12, 2016.
- Contaminated Soil History at Davis-Besse Nuclear Power Station Pursuant to 10 CFR 50.75(g)

4OA1 Performance Indicator Verification

Condition Reports:
- 2016–07765; Incorrect DEI-131 Value Reported on February 2016 RCS Performance Indicator

FENOC Business Practices:
- NOBP–LP–4012; NRC Performance Indicators; Revision 5

Forms:
- NOBP–LP–4012–45; Safety System Functional Failures; Revision 0; Completed Forms for April 2015 through March 2016
- NOBP–LP–4012–46; MSPI Emergency AC Power System; Revision 1; Completed Forms for April 2015 through March 2016
- NOBP–LP–4012–47; MSPI High Pressure Injection System; Revision 1; Completed Forms for April 2015 through March 2016
- NOBP–LP–4012–52; Reactor Coolant System Specific Activity; Revision 0; October 2015 through March 2016
- NOBP–LP–4012–58; RETS/ODCM Radiological Effluent Occurrence; Revision 0; July 2015 through March 2016

Procedures:
- DB–CH–01815; Dose Equivalent IodiNE–131 Determination; Revision 3

Other:
- Davis-Besse Nuclear Power Station Reactor Oversight Program Mitigating System Performance Index Basis Document; Revision 4
- MRPM; Maintenance Rule Program Manual; Revision 35
- Select Operator Logs covering the period of April 2015 through March 2016

4OA2 Problem Identification and Resolution

Condition Reports:
- 2014–09117; SW 37 Shows Significant Leakby
- 2014–09480; SW 37 Leakby Causes Larger than Anticipated SW Header Pressure Swing
- 2014–13172; Spur Gear Box Housing Cracked During Disassembly of SW 37 Valve Limittorque
- 2014–13288; Half of SW 37 Valve Liner Was Discovered Missing When Disassembled (Unrecovered FME)
- 2014–13293; CR Not Initiated in Timely Fashion for SW 37 Valve Liner
- 2014–17657; Flange Leak on SW 37 CCW Heat Exchanger 3 Outlet Isolation
- 2014–17776; SW 37 Possible Rework
- 2014–17828; SW 37 Degraded Flange Leakage
- 2015–03283; SW 37 Has Excessive Leakby
- 2016–00438; Degraded SW37 Effect on Operation of Service Water System with CCW HX No. 3 In Service Not Evaluated
- 2016–01491; RPS Channel 4 NI-7 Differential Flux Indicates Low
- 2016–02667; Impact of SW 37 Used for Primary Header Pressure Control with Degraded Seat Liner
- 2016–03466; Liner Torn in SW 37
- 2016–03694; Door 108 Found Open and Unattended

- 2016–04208; Reactor Coolant Pressure Boundary Leakage at Flexible Hose Assembly for RCP 1–1 First Stage Seal Cavity Vent
- 2016–04287; 19R BACC: P36–1 RCP 1st Stage Seal Vent Line Flex Hose Leakage
- 2016–04301; Base Metal Temperature Not Verified Prior to Welding
- 2016–04432; Socket Weld Indication on FSK–M–HCB–34–7 Elbow No. 19
- 2016–04595; Additional NRC Questions Regarding Weld Base Metal Temperature
- 2016–05028; Incomplete Fusion in Weld WJ–68–2 Safe End to Nozzle Weld of HPI Location 1–2
- 2016–05084; Sparks Discovered Passing Below Platform During Welding Activity In Mechanical Penetration Room 1
- 2016–05181; Overgrinding on HPI Nozzle 1–2 Mirror of AREVA CR 2016–2586
- 2016–05186; HPI Location No. 1–2 Weld WJ–68–2 (Safe End to Nozzle Weld) Repair OD Surface Not Suitable for UT
- 2016–05188; HPI Nozzle No. 1–2 Nozzle to New Safe End Weld – During a Localized Weld Repair the ID Was Cleaned Up Causing a Localized Area with Reduced Wall Thickness
- 2016–05204; EFW / FLEX - Rejectable Radiograph Indications in Two AFW Tie In Welds
- 2016–05250; Alloy 600 Minimum Wall Thickness Required for UT of Piping Weld Unclear
- 2016–05270; Wrong Weld Rod Filler Material Used on Flex Hose Installation
- 2016–05271; Radiography Performance Errors During Alloy 600 Exams
- 2016–05275; 54–ISI–829–011 Did Not Provide Adequate Guidance to Accept or Reject a Reflector Indication Found During UT Inspection of a Piping Weld
- 2016–05305; Extent of Condition Wrong Weld Rod Used
- 2016–05306; Filler Metal Material Not Verified Against Welding Procedure Specification
- 2016–05309; DB EFW / FLEX Project – Flammable Compressed Gas Bottles Stored Beside Oxygen Bottles
- 2016–05324; EFW / FLEX ECP13–0196–003 Tie-In 6-Inch Weld SWC EFW to AFW Train1 Downstream of AF608: Rejectable Radiograph Indications in 6-Inch Pipe to 6-Inch Elbow
- 2016–05411; EFW Weld SWC Indication Found During Radiography
- 2016–05461; Wrong Welding WPS and Filler Material Used for Tack Weld on HD4 Disk to Disk Nut
- 2016–05488; RT Rejected Weld 1–1 HPI Elbow to Pipe Weld
- 2016–05515; RT Rejected Weld Repair of Weld SWD-R2 and SWA–R1 on AF Tie to EFW
- 2016–05516; CAC No. 3 Inlet Flange Bolts Finger Tight
- 2016–05519; Alloy 600 – Weld Failed Radiography Test for HPI Nozzle 2-1
- 2016–05520; Alloy 600 – Weld Failed Radiography Test for HPI Nozzle 2-2
- 2016–05527; High Percentage of EFW Piping Welds Failed Radiographic Testing
- 2016–05587; Unexpected Restraining Wire on SS165 during As Left LLRT of SS235A/B
- 2016–05630; Mop Handle Identified In Cable Tray
- 2016–05706; FLEX Decay Heat Valve DH210 Found Not Torqued In Accordance with Order Requirement
- 2016–05753; DB EFW / FLEX Project - Valve DH210 Over-Pressurized and Over-Torqued
- 2016–05770; 19R BACC: DH210 Active Borated Water Leakage
- 2016–05793; DB EFW / FLEX Project: Documentation of Extent of Condition Review of FLEX / EFW Valve Torqueing Procedure References Based on CR 2016–05706 and 05753 – DH210 Valve Torqueing Issues
- 2016–05922; 19RFO – AREVA: Near Miss – Employee Fall into Flooded Refueling Canal
- 2016–05991; FW170 Indication Found in Field Weld B During Radiography
- 2016–06103; Indication Found During Radiography at Field Weld 27A
- 2016–06302; Clamps on Top of EDG 2 Not Latched
- 2016–06515; Vibration Felt and Heard when Additional Portion of AFW Recirculation Line Placed in Service

- 2016–06516; Untimely Documentation of Issue on AFW Train No. 2 Testing
- 2016–06756; Reactor Coolant Pump High Oil Level - Lower Bearing, Computer Point Received With No Other Abnormal Indications
- 2016–06789; PA–DB–16-03: Elevation – Weld Program Implementation
- 2016–06881; Delayed Documentation of Adverse Conditions on Condition Reports
- 2016–06927; Consistent Negative Values Observed on RCS Unidentified Leakage
- 2016–07155; EFW Storage Tank, T89, Found With Duct Tape Left on Tank Walls
- 2016–07526; Typographical Error (Copy / Paste) in DB–OP–06233, Auxiliary Feedwater Operating Procedure
- 2016–08339; Backup Radio Communication To The System Dispatcher (SD), Referenced In DB–OP–01300, Does Not Exist

Procedures:
- NOP–ER–1001; Continuous Equipment Performance Improvement; Revision 4
- NOP–LP–2001; Corrective Action Program; Revisions 37 and 38
- NOP–OP–1009; Operability Determinations and Functionality Assessments; Revision 6
- NA–QC–00191; Liquid Penetrant Examination; Revision 8
- NA–QC–00193; Magnetic Particle Examination; Revision 3
- NA–QC–00194; Radiographic Examination; Revision 1
- NA–QC–00252; Quality Control Review of Brazing and Weld Travelers; Revision 5
- NA–QC–00253; ASME/ANSI Visual Weld Inspection; Revisions 3 – 4
- DB–MM–09245; General Welding Procedure (ASME/ANSI Applications); Revision 9
- DB–MN–00011; Control of Brazing and Weld Travelers; Revision 6

FENOC Business Practices:
- NOBP–LP–2001; FENOC Self-Assessment / Benchmarking; Revision 24
- NOBP–LP–2003; Employee Concerns Program; Revision 4
- NOBP–LP–2008; FENOC Corrective Action Review Board; Revision 18
- NOBP–LP–2011; FENOC Cause Analysis; Revision 18
- NOBP–OP–1009; Prompt Operability Determination and Functionality Assessment Preparation Guide; Revision 6

FENOC Policy Statements and Reference Materials:
- NOPL–LP–2003; Safety Conscious Work Environment (SCWE); Revision 2
- NOPL–LP–2007; Corrective Action Program; Revision 1
- NORM–OP–1009; SRO Review of Condition Reports; Revision 5

Calculations:
- C–NSA–011.01–026; Past Operability Analysis of SW37 Leakage with SW37 in Service; Revision 0

Other:
- Control Room Unit Logs for February 1, 2016
- Control Room Unit Logs for May 7–8, 2016

4OA3 Followup of Events and Notices of Enforcement Discretion

Condition Reports:
- 2005–05314; Flux / Delta Flux / Flow Trip of RPS Channel 3
- 2014–18484; System Monitoring: TERC3B2 Has a Degraded (High Resistance) Wire or Connection Inside Containment or the Containment Penetration

- 2015–03516; Trend - Fuse Failures in 2013 and 2014
- 2015–04686; Reactor Protection System Channel 1 Declared Inoperable Due to Erroneous Loop 1 Hot Leg Temperature Indication
- 2015–05053; Step Change in T721 Narrow Range Hot Leg Temperature, RPS Channel 1
- 2015–07652; System Monitoring Identified Decreasing Trend in Indication for TERC3B2
- 2015–08549; System Monitoring Identified Increasing Trend in Indication for TERC3B2
- 2015–09237; RPS Channel 1 Hot Leg Narrow Range Temperature Indication Lower Following Checks in Order 200647179
- 2015–10750; DB–TERC3B2 Reading Erratic
- 2016–01364; Reactor Trip During NI Calibrations
- 2016–01397; SFRCS Reverse D/P Trip Received During Recovery From an Earlier SFRCS High Level Trip
- 2016–04208; Reactor Coolant Pressure Boundary Leakage at Flexible Hose Assembly for RCP 1–1 First Stage Seal Cavity Vent
- 2016–04587; TERC3B2/3B5 Insulation Burnt Off
- 2016–04287; 19R BACC: P36-1 RCP 1st Stage Seal Vent Line Flex Hose Leakage
- 2016–05782; New Hot Leg RTD Assembled Incorrectly
- 2016–05792; Incorrectly Assembled Conax Connectors
- 2016–07816; Potential Impact of a Tornado on the Emergency Diesel Generators (EDGs) via the Crankcase Pressure Switch

Procedures:
- NOP–LP–2001; Corrective Action Program; Revisions 37 and 38

Work Orders:
- 200685069; ECP 16-0340-001: Disable EDG 1 Crankcase Pressure Switch
- 200685070; ECP 16-0340-002: Disable EDG 2 Crankcase Pressure Switch

Engineering Change Packages:
- ECP 16–0340–000; Install Temporary Modification to Defeat EDGs Crank Case Pressure Trip; Revision 0
- ECP 16–0340–001; Temporary Modification to Disable Trip from Crank Case Pressure Switch for EDG 1; Revision 0
- ECP 16–0340–002; Temporary Modification to Disable Trip from Crank Case Pressure Switch for EDG 2; Revision 0

4OA5 Other Activities

Condition Reports:
- 2015–00214; Groundwater Tritium Concentration in Monitoring Well (MW–37S) Above 2,000 pCi/liter
- 2015–01455; Elevated Tritium Concentrations in Seven Groundwater Monitoring Wells
- 2015–01639; Water Containing 1 Million pCi/L Tritium on the Floor in the Borated Water Storage Tank Pit
- 2015–02108; Groundwater Tritium Results Greater Than Courtesy Notification Level of 2000 pCi/l
- 2015–03642; Several Davis-Besse March Groundwater Well Tritium Samples Over 2,000 pCi/liter
- 2015–07189; Fourteen of Thirty-One Groundwater Samples Over 2,000 PicoCuries/Liter (pCi/L) Tritium
- 2015–12043; Review Impact of Elimination of Monitoring Well (MW) 22 S/D

Procedures:
- NOP–OP–1015; Event Notifications; Revision 2
- NOP–OP–2012; Groundwater Monitoring; Revision 9
- NOP–OP–4705; Response to Contaminated Spills/Leaks; Revision 8

Business Practices:
- NOBP–OP–1015; Event Notifications; Revision 3

Other:
- Groundwater Monitoring Well Data covering the period of January 2014 through June 2016

4OA7 Licensee-Identified Violations

Condition Reports:
- 2016–08266; Isolated Fire Protection Loop

Procedures:
- DB–FP–00009; Fire Protection Impairment and Fire Watch; Revision 21
- NG–DB–00302; DBNPS Fire Protection Program; Revision 10

Notifications:
- 601008850; FP40 Stuck Shut

Work Orders:
- 200373273; FP355: Repair/Replace FP Curb Box Valve

Drawings:
- OS–0047A, Sheet 2; Fire Protection System; Revision 20
- OS–0047A, Sheet 4; Fire Protection System; Revision 12
- OS–0047B, Sheet 3; Fire Suppression System; Revision 5
- OS–0047B, Sheet 4; Fire Suppression System; Revision 20
- OS–0047B, Sheet 6; Fire Suppression System; Revision 12

Clearances:
- ODB–SUB013–017; FP355 Repair/Replace Curb Box/Operator
- SUB013–03–CT–00014; FP40 East Underground Loop Isolation

Other:
- DB–0100–2; Fire System Impairment Control System – Initiation Work Sheet
- Fire Hazard Analysis Report; Revision 26

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BWST	Borated Water Storage Tank
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CREATCS	Control Room Emergency Air Temperature Control System
CREVS	Control Room Emergency Ventilation System
CS	Containment Spray
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
ENS	Emergency Notification System
EPRI	Electric Power Research Institute
EVS	Emergency Ventilation System
HPI	High Pressure Injection
HX	Heat Exchanger
IA	Instrument Air
ICS	Integrated Control System
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Testing
KV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MFW	Main Feedwater
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NI	Nuclear Instrument
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OWA	Operator Workaround
PARS	Publicly Available Records System
pCi/L	picocuries per liter
PI	Performance Indicator
PM	Preventative Maintenance
PMT	Post-Maintenance Testing
psig	Pounds Per Square Inch Gauge
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage

RPS	Reactor Protection System
RWP	Radiation Work Permit
SBODG	Station Blackout Diesel Generator
SDP	Significance Determination Process
SFRCS	Steam and Feedwater Rupture Control System
SG	Steam Generator
SR	Surveillance Requirement
SRO	Senior Reactor Operator
SSC	System, Structure, and Component
SW	Service Water
TS	Technical Specification
TSO	Transmission System Operator
URI	Unresolved Item
USAR	Updated Safety Analysis Report
UT	Ultrasonic
VT	Visual Examination
WO	Work Order

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Sincerely,

/RA/

Jamnes L. Cameron, Chief
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