UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, IL 60532-4352

November 4, 2016

Mr. Brian D. Boles
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449–9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION—NRC INTEGRATED INSPECTION REPORT 05000346/2016003

Dear Mr. Boles:

On September 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 October 6, 2016, with you and other members of your staff.

Based on the results of this inspection, the NRC has identified two 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. These NCVs are described in the subject inspection report. 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 Enforcement Policy.

If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to: (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 Kenneth Riemer Acting for/

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

Docket No. 50–346
License No. NPF–3

Enclosure:
Inspection Report 05000346/2016003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–346
License No:	NPF–3
Report No:	05000346/2016003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Davis-Besse Nuclear Power Station
Location:	Oak Harbor, OH
Dates:	July 1, 2016, through September 30, 2016
Inspectors:	D. Kimble, Senior Resident Inspector T. Briley, Resident Inspector J. Cassidy, Senior Health Physicist M. Garza, Emergency Preparedness Inspector J. Rutkowski, Project Engineer N. Valos, Senior Reactor Analyst
Approved by:	J. Cameron, Chief Branch 4 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY

Inspection Report (IR) 05000346/2016003; 7/1/16 – 9/30/16; Davis-Besse Nuclear Power Station; Operability Determinations and Functionality Assessments; 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. Two Green findings were identified. Both 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 Inspection Manual Chapter (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: Mitigating Systems

- Green. A self-revealed 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 provide adequate instructions to correctly assemble electrical conductor seal assemblies (ECSAs) used to provide an environmental barrier for resistance temperature detectors (RTDs). Specifically, the midlock ferrules inside two ECSAs were installed backwards during the 18th refueling outage (RFO) in 2014 which rendered multiple post accident monitoring system (PAMS) indications required by Technical Specification (TS) 3.3.17 inoperable. This issue was entered into the licensee's corrective action program (CAP). Corrective actions by the licensee included, but were not limited to, replacement of the two dual element RTDs impacted and their associated ECSAs during the 2016 RFO, performance of an extent of condition review, development of enhanced procedural guidance, and implementation of additional training on ECSA components.

  This finding was of more than minor significance because it was associated with the 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)." The inspectors determined the finding to be of very low safety significance because it did not represent a deficiency affecting design or qualification of a mitigating system, structure, and component (SSC); 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. The inspectors determined that the finding had a

cross-cutting aspect in the area of human performance. Specifically, the cross-cutting aspect of “Training” was assigned to the finding because a job task analysis was performed prior to the 2014 RFO and determined that the procedural guidance to correctly assemble the ECSAs was adequate; thus no training or procedural changes were required. But the as-found condition of the RTDs during the 2016 RFO identified that a knowledge gap and procedure deficiency existed. (H.9) (Section 1R15.1)

- 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 III, “Design Control,” were identified for the licensee’s failure to have adequately prepared and implemented a permanent plant modification associated with steam generator (SG) replacement during the unit’s 18th RFO in 2014. Specifically, in conjunction with SG replacement the licensee had also replaced a significant amount of reactor coolant system (RCS) piping and instrumentation, including all RCS hot leg resistance temperature detectors (RTDs). The RTD housings were improperly insulated during the modification, such that over the ensuing reactor operating cycle the RTD wiring insulation degraded to the extent that nearly all the RTDs were rendered inoperable. This issue was entered into the licensee’s CAP. Corrective actions by the licensee included replacement of the degraded RTDs.

  This finding was of more than minor safety significance because it affected the attribute of design control 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 RPS. Specifically, the inspectors determined that the licensee’s failure to have properly designed and implemented the insulation packages for the RTD housings ultimately resulted in the overheating and degradation of the RTD wiring insulation and inoperability of the RTDs associated with the RCS high temperature and RCS pressure/temperature reactor trips. The finding was determined to be of very low safety significance based on a detailed risk analysis that yielded a change in core damage frequency (CDF) of less than 1E–7 events per year. 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’s SG replacement project management team failed to reinforce the importance of close communication between responsible engineers with overlapping and interfacing modification packages, and did not adequately promote effective work execution through the use of clearly defined work documents that were written and structured to minimize the likelihood for human error. (H.2) (Section 4OA3.4)

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

The unit began the inspection period operating at full power. On September 10, 2016, with heavy rainfall occurring in the area the reactor tripped due to a lockout of the main electrical generator. The event was complicated by the isolation of the steam generators (SGs), which was required due to a high water level condition in SG No. 1 following the reactor trip (see Section 4OA3.2). As the licensee was preparing for plant restart on September 11, 2016, operators identified a significant seat leakage condition associated with a pressurizer code safety valve (RC13B). The plant was subsequently taken to cold shutdown conditions to facilitate repairs. Following a forced maintenance outage to complete repairs on various components, the unit was restarted and the reactor made critical on September 20, 2016 (see Section 1R20.1). The main generator was synchronized to the electrical power grid on September 22, 2016, and full power was achieved on September 23, 2016. The unit continued to operate at or near full power through the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

During the course of the inspection period, the inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also checked underground bunkers/manholes subject to flooding that contained multiple train or multiple function risk-significant cables. Additionally, the inspectors reviewed the licensee's off-normal procedure for mitigating flooding to ensure it could be implemented as written.

These reviews by the inspectors constituted a single external flooding 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's electric fire pump while the station's diesel fire pump was out of service for corrective maintenance involving an engine coolant leak during the week ending July 16, 2016;
- Emergency Diesel Generator (EDG) No. 2 while the station blackout diesel generator was out of service for preventative maintenance involving injector rocker arm replacements, general cleaning and inspections during the week ending July 16, 2016; and
- The station's motor driven feedwater pump while Auxiliary Feedwater (AFW) Train No. 1 was out of service for various preventative maintenance activities during the week ending July 23, 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, USAR, technical specification (TS) requirements, outstanding work orders (WOs), condition reports (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 corrective action program (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 weeks ending September 3, 2016, through September 24, 2016, the inspectors performed a complete system alignment inspection of the station's newly commissioned emergency feedwater (EFW) system to verify the functional capabilities of the system. This system was selected because EFW 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:

- The EFW facility (Fire Area EF) during the weeks ending July 16, 2016, through July 30, 2016;
- Electrical Penetration Room No. 1 (Room 402 – Fire Area DG) during the week ending August 13, 2016;
- Electrical Penetration Room No. 2 (Room 427 – Fire Area DF) during the week ending August 13, 2016; and
- Low Voltage Switchgear Room No. 2 (Rooms 428, 428A, 428B – Fire Area X) during the week ending September 10, 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 four quarterly fire protection zone inspection tour samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors conducted internal flooding reviews for:

- Auxiliary building elevations 585'-0" and areas below susceptible to flooding from the spent fuel pool cooling system during the week ending August 20, 2016; and
- Emergency Core Cooling System (ECCS) Room No. 1 (Room 105) and ECCS Room No. 2 (Room 115) during the week ending August 20, 2016.

The inspectors reviewed flood analyses and design documents, including the USAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or cooling tower makeup systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed physical inspections of the above noted plant areas to assess the adequacy of watertight boundaries/barriers and verify drains and sumps were clear of debris and were operable, and that the licensee had complied with applicable commitments.

The inspectors' reviews constituted two internal flooding inspection samples as defined in IP 71111.06–05.

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

On July 26, 2016, the inspectors observed a crew of licensed operators in the plant's simulator during a graded simulator scenario. 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 to ensure that the integrity of the graded scenario was being protected from being compromised. 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; and
- The ability of the crew to identify and implement appropriate TS actions and emergency plan 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 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:

- AFW Pump No. 1 issues with its governor setting while performing normal periodic functional testing during the week ending August 20, 2016;

- The main electric generator reactive power 5-year capability test during the week ending August 20, 2016;
- Periodic at-power control rod exercise testing, main turbine valve testing, and associated plant power maneuvers on Sunday, September 4, 2016;
- Crew response to a main generator electrical lockout, reactor trip, and steam and feedwater rupture control system (SFRCS) actuation due to high SG post-trip water level on Saturday, September 10, 2016;
- RCS cool down and depressurization into cold shutdown conditions for a forced maintenance outage during the week ending September 17, 2016; and
- Reactor approach to criticality and plant startup during the week ending September 24, 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 senior reactor operators 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

No findings were identified.

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:

- Pressurizer code safety valve performance and seat tightness.

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 systems, structures, 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 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:

- Troubleshooting and corrective actions associated with flow vibrations experienced on the recirculation test line during AFW Train No. 1 periodic testing during the week ending July 23, 2016;
- Interim corrective actions for problematic position indication on Control Rod 7–3 during the week ending September 3, 2016;
- Various maintenance issues associated with Low Pressure Injection Train No. 2 during the week ending September 10, 2016; and
- Emergent repair and replacement activities associated with seat leakage on a pressurizer code safety valve (RC13B) during the week ending September 17, 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 four 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 EDG No. 1 and No. 2 following identification of low cetane index values in the EDG fuel oil storage tanks, as documented in CR 2016–10167;
- The operability and functionality of RCS hot leg temperature instrumentation following identification of the installation of improperly configured instrument cable connectors, as documented in CR 2016–05792; and
- The operability and functionality of borated water storage tank (BWST) level indication associated with the safety features actuation system (SFAS) following the identification of various maintenance issues, as documented in CR 2016–08419.

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 three inspection samples as defined in IP 71111.15–05.

b. Findings

(1) Inadequate Instructions to Correctly Assemble Electrical Conductor Seal Assemblies

Introduction

A self-revealed 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 provide adequate instructions to correctly assemble electrical conductor seal assemblies (ECSAs) used to provide an environmental barrier for resistance temperature detectors (RTDs). Specifically, the midlock ferrules inside two ECSAs were installed backwards during the 2014 refueling outage (RFO) which rendered multiple PAMS indications required by TS 3.3.17 inoperable.

Description

As part of SG replacement activities during the 2014 RFO, all six dual element RTDs on the reactor coolant system hot legs were replaced. Each RTD element (twelve in total) provides an input to RPS, PAMS, and/or the non-nuclear instrumentation (NNI) system. Each dual element RTD has two ECSAs, one for each element. An ECSA provides an environmentally qualified barrier for the conductors exiting each RTD element to protect the device from propagation of steam or water through the conduit into the instrument. The ECSAs consist of a seal body, midlock ferrule, and midlock cap. The midlock ferrule is directional and is located in between the seal body and midlock cap which screw together to form the seal.

On March 26, 2014, work order (WO) 200574536 for RTD temperature element TERC3B2 (feeds RPS Channel 1 high temperature and pressure-temperature indication as well as remote shutdown panel indication) and TERC3B5 (feeds PAMS RCS temperature loop 1B indication and hot leg level monitoring system (HLLMS) loop 1B instrumentation) directed the installation and termination of the ECSAs to the associated RTD element in the shop prior to field installation. The WO instructions did not specifically discuss the orientation of the midlock ferrule within the seal body and midlock cap and only provided a note that referenced one particular section of the Conax ECSA vendor manual that did not contain the appropriate level of detail to determine the orientation of the midlock ferrule. The vendor manual included a drawing that showed the correct orientation of the midlock ferrule but that particular drawing was not referenced in the WO or procedure DB–ME–09501, "Installation of Connectors," Revision 7 which were used to assemble and torque the ECSAs. Consequently, the midlock ferrule for TERC3B5 was installed backwards by site maintenance personnel and was not identified at the time of installation.

On April 8, 2014, WO 200574538 for RTD temperature element TERC3A4 (feeds RPS Channel 2 high temperature and pressure-temperature indication as well as remote shutdown panel indication) and TERC3A6 (feeds PAMS RCS temperature loop 2A indication) similarly directed the installation and termination of the ECSA to the

associated RTD in the shop prior to field installation. The WO instructions and procedural guidance, again, did not specifically discuss the orientation of the midlock ferrules. Consequently, the midlock ferrule for TERC3A4 was installed backwards by site maintenance personnel and was not identified at the time of installation.

During the 2016 RFO, the licensee had determined that several dual element RTDs needed to be replaced due to wiring degradation (see section 4OA3.4 for details). While evaluating Conax ECSAs that had been removed from the field for potential reuse, the licensee identified on April 23, 2016, that two ECSAs installed during the 2014 RFO were assembled with the midlock ferrule installed backwards which were associated with TERC3B5 and TERC3A4. Condition report 2016–05792 was written and a full apparent causal evaluation was performed. The licensee's causal evaluation concluded that the apparent cause of the incorrectly assembled Conax ECSAs was less than adequate instruction (i.e. training and guidance documents) to support successful Conax ECSA assembly/installation.

The licensee performed a past operability review which included TERC3B5 (shares same RTD head as TERC3B2) and TERC3A6 (shares same RTD head as TERC3A4). The past operability review concluded that these four temperature elements could not maintain their environmental qualifications with the midlock ferrules installed backwards. Consequently, the PAMS RCS hot leg temperature instruments (one out of two required for each loop by TS 3.3.17) would not be expected to perform their function under accident conditions and hence were inoperable for the duration of the previous operating cycle (May 2014 – March 2016). Additionally, one of two required channels of the reactor coolant HLLMS per TS 3.3.17 was also inoperable for the duration of the previous operating cycle. Environmental qualification does not impact the RPS or remote shutdown panel functions of the affected instruments.

Corrective actions taken by the licensee included, but were not limited to, replacement of the two dual element RTDs impacted and their associated ECSAs during the 2016 RFO, performance of an extent of condition review, development of enhanced procedural guidance, and implementation of additional training on ECSA components.

Analysis

The inspectors reviewed this finding using the guidance contained in Appendix B, "Issue Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." The inspectors determined that the licensee's failure to provide adequate instructions to correctly assemble ECSAs used to provide an environmental barrier for RTDs was a performance deficiency that was reasonably within the licensee's ability to foresee and correct, and that 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 the 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)."

Although the issue was identified with the reactor shutdown for refueling, the inspectors evaluated the finding using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," because the condition involved the operability of the PAMS system while the unit was operating at power during the past reactor operating cycle. 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 a mitigating SSC; 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 human performance. Specifically, the cross-cutting aspect of "Training" was assigned to the finding because a job task analysis was performed prior to the 2014 RFO and determined that the procedural guidance needed to correctly assemble the ECSAs was adequate, thus no training or additional procedure guidance was required. But the as-found condition of the RTDs during the 2016 RFO identified that a knowledge gap and procedure deficiency existed. (H.9)

Enforcement

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. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.*

Contrary to the above, during the licensee's 2014 refueling outage, activities affecting quality were not prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Specifically, the WO instructions and procedural guidance for installation of ECSAs on RTDs did not provide sufficient instruction to ensure the ECSAs were assembled correctly.

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 2016–05792, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000346/2016003–01)**

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:

- Operational and functional testing of AFW Train No. 1 following a periodic maintenance work window during the week ending July 23, 2016;
- Operational and functional testing of the station's EFW pump and associated equipment during an extended endurance run following commissioning during the week ending July 30, 2016;
- Operational and functional testing of EDG No. 1 following a periodic maintenance work window during the week ending July 30, 2016;
- Functional testing of the main electrical generator voltage regulator following forced outage repairs during the week ending September 24, 2016;
- Functional testing of Control Rod 7–3 position indication following forced outage repairs during the weeks ending September 24, 2016, and October 1, 2016; and
- Leakage testing for a pressurizer code safety valve (RC13B) at normal RCS pressure and temperature following forced outage replacement during the week ending September 24, 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 six PMT inspection samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 September 2016 Forced Maintenance Outage

a. Inspection Scope

The inspectors evaluated outage activities for a forced maintenance outage that began with an automatic reactor trip at approximately 3:43 a.m. on September 10, 2016, that resulted from a lockout protective action on the unit's main electrical generator. (See Section 4OA3.2 for event details.) A leaking pressurizer code safety valve (RC13B) identified on September 11, 2016, caused the licensee to bring the unit to the cold shutdown condition to facilitate replacement of the valve. During this time, the licensee also performed other repairs to the facility, including corrective actions for a main

condenser tube leak and replacement of a suspect wiring connector associated with the position indication for one control rod assembly. Following completion of repairs to the plant, operators restarted the reactor on September 20, 2016, and the unit returned to full power on September 23, 2016.

The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule. Outage equipment configuration, risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, personnel fatigue management, startup activities, and identification and resolution of problems associated with the outage were reviewed and selectively observed by the inspectors.

These observations and reviews by the inspectors constituted a single other (i.e., non-refueling) outage 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:

- Monthly periodic testing of EDG No. 2 during the week ending August 13, 2016 (routine);
- Periodic channel functional testing of SFRCS actuation channel No. 1 logic and SG level inputs during the weeks ending August 27, 2016 and September 3, 2016 (routine); and
- Periodic at-power turbine valve testing during the week ending September 10, 2016 (routine).

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 inservice testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel 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 for safety-related instrument control surveillance tests, that reference setting data were accurately incorporated in the test procedure;
- 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.

Documents reviewed are listed in the Attachment to this report.

These activities conducted by the inspectors constituted three routine surveillance testing inspection samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP2 Alert and Notification System Evaluation (71114.02)

.1 Alert and Notification System Evaluation

a. Inspection Scope

The inspectors reviewed documents and held discussions with Emergency Preparedness (EP) staff regarding the operation, maintenance, and periodic testing of the primary and backup Alert and Notification System (ANS) in the plume pathway Emergency Planning Zone. The inspectors reviewed monthly trend reports and siren test failure records from October 2014 through September 2016. Information gathered during document reviews and interviews were used to determine whether the ANS equipment was maintained and tested in accordance with Emergency Plan commitments and procedures. Documents reviewed are listed in the Attachment to this report.

This ANS evaluation inspection constituted one sample as defined in IP 71114.02–06.

b. Findings

No findings were identified.

1EP3 Emergency Response Organization Staffing and Augmentation System (71114.03)

.1 Emergency Response Organization Staffing and Augmentation System

a. Inspection Scope

The inspectors reviewed and discussed with plant EP management and staff the Emergency Plan commitments and procedures that addressed the primary and alternate methods of initiating an Emergency Response Organization (ERO) activation to augment the on-shift staff as well as the provisions for maintaining the plant's ERO team and qualification lists. The inspectors reviewed reports and a sample of CAP records of unannounced off-hour augmentation drills and pager tests, which were conducted from October 2014 through September 2016, to determine the adequacy of the drill critiques and associated corrective actions. The inspectors also reviewed a sample of the training records of approximately six ERO personnel, who were assigned to key and support positions, to determine the status of their training as it related to their assigned ERO positions. Documents reviewed are listed in the Attachment to this report.

This ERO augmentation testing inspection constituted one sample as defined in IP 71114.03–06.

b. Findings

No findings were identified.

1EP5 Maintenance of Emergency Preparedness (71114.05)

.1 Maintenance of Emergency Preparedness

a. Inspection Scope

The inspectors reviewed the nuclear oversight staff's 2015 and 2016 audit of the Davis-Besse Nuclear Power Station's Emergency Preparedness Program to determine that the independent assessments met the requirements of 10 CFR 50.54(t). The inspectors reviewed samples of CAP records associated with the 2015 biennial exercise, as well as various EP drills conducted from October 2014 through September 2016, in order to determine whether the licensee fulfilled drill commitments and to evaluate the licensee's efforts to identify and resolve identified issues. The inspectors reviewed a sample of EP items and corrective actions related to the station's EP program, and activities to determine whether corrective actions were completed in accordance with the site's CAP. Documents reviewed are listed in the Attachment to this report.

This maintenance of emergency preparedness inspection constituted one sample as defined in IP 71114.05–06.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observations

a. Inspection Scope

The inspectors evaluated the conduct of the following planned licensee full scale integrated EP drill:

- August 9, 2016.

The inspectors observed emergency response operations in the site's technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures, and to identify any weaknesses or deficiencies in classification, notification, or protective action recommendation development activities. The inspectors also attended the licensee drill critique to compare any inspector-observed weaknesses with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of their inspection activities, the inspectors reviewed the drill package for the scenario and other documents listed in the Attachment to this report.

The inspectors' review of this EP drill scenario and other related activities constituted a single inspection sample as defined in IP 71114.06–06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

CORNERSTONES: OCCUPATIONAL RADIATION SAFETY AND PUBLIC RADIATION SAFETY

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

.1 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed procedural guidance for use of ventilation systems, and assessed whether the systems were used, to the extent practicable, during high-risk activities to control airborne radioactivity and minimize the use of respiratory protection. The inspectors assessed whether installed ventilation airflow capacity, flow path, and filter/charcoal unit efficiencies for selected systems were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable. The inspectors also evaluated whether selected temporary ventilation systems used to support work in contaminated areas were consistent with licensee procedural guidance and as-low-as-reasonably-achievable (ALARA).

These inspection activities supplemented those documented in IR 05000346/2016002 and constituted one complete sample as defined in IP 71124.03–05.

.2 Use of Respiratory Protection Devices

a. Inspection Scope

The inspectors assessed whether the licensee provided respiratory protection devices for those situations where it was impractical to employ engineering controls such that occupational doses were ALARA. For select instances where respiratory protection devices were used, the inspectors assessed whether the licensee concluded that further engineering controls were not practical. The inspectors also assessed whether the licensee had established means to verify that the level of protection provided by the respiratory protection devices was at least as good as that assumed in the work controls and dose assessment.

The inspectors reviewed records of air testing for supplied-air devices and self-contained breathing apparatus (SCBA) bottles to assess whether the air used met or exceeded Grade D quality. The inspectors evaluated whether plant breathing air supply systems satisfied the minimum pressure and airflow requirements for the devices.

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 the physical condition of respiratory protection devices ready for issuance and reviewed records of routine inspection for selected devices. The inspectors reviewed records of maintenance on the vital components for selected devices and assessed whether on-site personnel assigned to repair vital components received vendor-provided training.

These inspection activities supplemented those documented in IR 05000346/2016002 and constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.3 Self-Contained Breathing Apparatus for Emergency Use

a. Inspection Scope

The inspectors reviewed the status and surveillance records for select SCBAs.
The inspectors evaluated the licensee’s capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions.

The inspectors assessed whether control room operators and other emergency response and radiation protection personnel were trained and qualified in the use of SCBAs and evaluated whether personnel assigned to refill bottles were trained and qualified for that task.

The inspectors assessed whether appropriate mask sizes and types were available for use. The inspectors evaluated whether on-shift operators had no facial hair that would interfere with the sealing of the mask and that appropriate vision correction was available.

The inspectors reviewed the past two years of maintenance records for selected in-service SCBA units used to support operator activities during accident conditions. The inspectors assessed whether maintenance or repairs on an SCBA unit's vital components were performed by an individual certified by the manufacturer of the device to perform the work. The inspectors evaluated the on-site maintenance procedures governing vital component work to determine whether there was any inconsistencies with the SCBA manufacturer's recommended practices. The inspectors evaluated whether SCBA cylinders satisfied the hydrostatic testing required by the U.S. Department of Transportation.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors assessed whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by the licensee at an appropriate threshold and were properly addressed for resolution. Additionally, the inspectors evaluated the appropriateness of the corrective actions for selected problems involving airborne radioactivity documented by the licensee.

These inspection activities constituted one complete sample as defined in IP 71124.03–05.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

.1 Source Term Characterization

a. Inspection Scope

The inspectors evaluated whether the licensee had characterized the radiation types and energies being monitored and that the characterization included gamma, beta, hard-to-detects, and neutron radiation.

The inspectors assessed whether the licensee had developed scaling factors for including hard-to-detect nuclide activity in internal dose assessments.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.2 External Dosimetry

a. Inspection Scope

The inspectors evaluated whether the licensee's dosimetry vendor was National Voluntary Laboratory Accreditation Program accredited and if the approved irradiation test categories for each type of personnel dosimeter used were consistent with the types and energies of the radiation present and the way the dosimeter was being used.

The inspectors evaluated the on-site storage of dosimeters before their issuance, during use, and before processing/reading. For personal dosimeters stored on-site during the monitoring period, the inspectors evaluated whether they were stored in low-dose areas with control dosimeters. For personal dosimeters that are taken offsite during the monitoring period, the inspectors evaluated the guidance provided to individuals with respect to care and storage of the dosimeter.

The inspectors evaluated the calibration of active dosimeters. The inspectors assessed the bias of the active dosimeters compared to passive dosimeters and the correction factor used. The inspectors also assessed the licensee's program for comparing active and passive dosimeter results, investigations for substantial differences, and recording of dose. The inspectors assessed whether there were adverse trends for active dosimeters.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.3 Internal Dosimetry

a. Inspection Scope

The inspectors reviewed procedures used to assess internal dose using whole body counting equipment to evaluate whether the procedures addressed methods for differentiating between internal and external contamination, the release of contaminated individuals, the route of intake and the assignment of dose. The inspectors assessed whether the frequency of measurements was consistent with the biological half-life of the nuclides available for intake. The inspectors reviewed the licensee's evaluation for use of portal radiation monitors as a passive monitoring system to determine if instrument minimum detectable activities were adequate to detect internally deposited radionuclides sufficient to prompt additional investigation. The inspectors reviewed whole body counts and evaluated the equipment sensitivity, nuclide library, review of results, and incorporation of hard-to-detect radionuclides.

The inspectors reviewed procedures used to determine internal dose using in-vitro analysis to assess the adequacy of sample collection, determination of entry route and assignment of dose.

The inspectors reviewed the licensee's program for dose assessment based on air sampling, as applicable, and calculations of derived air concentration. The inspectors determined whether flow rates and collection times for air sampling equipment were adequate to allow lower limits of detection to be obtained. The inspectors also reviewed the adequacy of procedural guidance to assess internal dose if respiratory protection was used.

The inspectors reviewed select internal dose assessments and evaluated the monitoring protocols, equipment, and data analysis.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.4 Special Dosimetric Situations

a. Inspection Scope

The inspectors assessed whether the licensee informs workers of the risks of radiation exposure to the embryo/fetus, the regulatory aspects of declaring a pregnancy, and the specific process to be used for declaring a pregnancy. The inspectors selected individuals who had declared pregnancy during the current assessment period and evaluated whether the monitoring program for declared pregnant workers was technically adequate to assess the dose to the embryo/fetus. The inspectors assessed results and/or monitoring controls for compliance with regulatory requirements.

The inspectors reviewed the licensee's methodology for monitoring external dose in non-uniform radiation fields or where large dose gradients exist. The inspectors evaluated the licensee's criteria for determining when alternate monitoring was to be implemented. The inspectors reviewed dose assessments performed using multi-badging to evaluate whether the assessment was performed consistently with licensee procedures and dosimetric standards.

The inspectors evaluated the licensee’s methods for calculating shallow dose equivalent from distributed skin contamination or discrete radioactive particles. The inspectors reviewed select shallow dose equivalent dose assessments for adequacy.

The inspectors evaluated the licensee’s program for neutron dosimetry, including dosimeter types and/or survey instrumentation. The inspectors reviewed select neutron exposure situations and assessed whether dosimetry and/or instrumentation was appropriate for the expected neutron spectra, there was sufficient sensitivity, and neutron dosimetry was properly calibrated. The inspectors also assessed whether interference by gamma radiation had been accounted for in the calibration and whether time and motion evaluations were representative of actual neutron exposure events.

For the special dosimetric situations reviewed in this section, the inspectors assessed how the licensee assigns dose of record. This included an assessment of external and internal monitoring results, supplementary information on individual exposures, and radiation surveys and/or air monitoring results when dosimetry was based on these techniques.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution

a. Inspection Scope

The inspectors assessed whether problems associated with occupational dose assessment 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 involving occupational dose assessment.

These inspection activities constituted one complete sample as defined in IP 71124.04–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index—Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) – Heat Removal System performance indicator (PI) for the period from the third quarter 2015 through the second 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, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports, MSPI derivation reports, and NRC Integrated IRs for the period of July 2015 through June 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 constituted a single MSPI – Heat Removal System PI inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI – Residual Heat Removal System PI for the period from the third quarter 2015 through the second quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the 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 July 2015 through June 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 constituted a single MSPI – Residual Heat Removal System PI inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI – Cooling Water Systems performance for the period from the third quarter 2015 through the second quarter 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the 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 July 2015 through June 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 constituted a single MSPI – Cooling Water Systems PI inspection sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Drill and Exercise Performance

a. Inspection Scope

The inspectors sampled licensee submittals for the Drill and Exercise Performance (DEP) Indicator for the period from the second quarter of 2015 through the second quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment PI Guideline," Revision 7, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the DEP indicator, in accordance with relevant procedures and NEI guidance. Specifically, the inspectors reviewed licensee records and processes, including procedural guidance on assessing opportunities for the PI; assessments of PI opportunities during pre-designated control room simulator training sessions; performance during the 2015 biennial exercise; and performance during other drills. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one DEP sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.5 Emergency Response Organization Drill Participation

a. Inspection Scope

The inspectors sampled licensee submittals for the ERO Drill Participation PI for the period from the second quarter of 2015 through the second quarter of 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 PI Guideline," Revision 7, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the indicator, in accordance with relevant procedures and NEI guidance. Specifically, the inspectors reviewed licensee records and processes, including procedural guidance on assessing opportunities for the PI; participation during the 2015 biennial exercise and other drills; and revisions of the roster of personnel assigned to key ERO positions. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one ERO drill participation sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.6 Alert and Notification System Reliability

a. Inspection Scope

The inspectors sampled licensee submittals for the ANS PI for the period from the second quarter of 2015 through the second quarter of 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 PI Guideline," Revision 7, were used. The inspectors reviewed the licensee's records associated with the PI to verify that the licensee accurately reported the indicator, in accordance with relevant procedures and NEI guidance. Specifically, the inspectors reviewed licensee records and processes, including procedural guidance on assessing opportunities for the PI and results of periodic ANS operability tests. Documents reviewed are listed in the Attachment to this report.

This inspection constitutes one ANS 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 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.

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

.1 Event Notification No. 52079: Failure to Perform Technical Specification Required Shutdown

a. Inspection Scope

On June 30, 2016, at 11:42 p.m., control room operators received panel alarms indicating that SFAS Channel No. 2 had experienced a partial loss of power. Operators declared several parameters associated with SFAS Channel No. 2 inoperable and entered TS 3.3.5, "SFAS Instrumentation," Condition A, for each of the parameters. One of these parameters was for the BWST low-low level condition.

At approximately 2:45 a.m. on July 1, 2016, licensee personnel concluded that TS 3.3.5, Condition B, also was applicable for the BWST low-low level condition parameter in addition to Condition A. This conclusion was based on the identification by licensee personnel that maintenance being performed on the BWST level transmitter associated with SFAS Channel No. 1 was not complete, and that the BWST low-low level condition parameter for SFAS Channel No. 1 had remained inoperable at the end of the normal work day on June 30, 2016. Condition B of TS 3.3.5 requires, in part, that the unit be shut down and in the hot standby condition (Mode 3) within six hours.

On July 1, 2016, at 3:30 a.m., licensee operators inappropriately declared the BWST level transmitter associated with SFAS Channel No. 1 operable, but degraded, and exited TS 3.3.5, Condition B. Ultimately, the BWST level transmitter for SFAS Channel No. 1 was repaired, restored to operation, and declared operable on July 1, 2016, at 1:51 p.m. A faulty power supply was identified as the cause for the issues associated with SFAS Channel No. 2. That power supply was replaced on July 1, 2016, and at approximately 6:00 p.m. SFAS Channel No. 2 was declared operable and all remaining TS 3.3.5 actions exited.

In retrospect, the licensee concluded that they should have initiated and completed a plant shutdown to Mode 3 within six hours of 11:42 p.m. on June 30, 2016. While conducting an internal review of the actions associated with this event, licensee senior leadership concluded that a four-hour non-emergency report would have been made under 10 CFR 50.72(h)(2)(i) had they correctly initiated the plant shutdown required by TS 3.3.5, Condition B. As a result, the licensee reported the event on July 10, 2016.

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:

- SFAS 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 CRs 2016-08402, 2016–8415, 2016–08419, 2016–08699, 2016–08700, and 2016–08765, and commissioned a formal root cause analysis. Initial corrective actions taken by the licensee included completion of repairs to SFAS Channels 1 and 2, and the reinforcement of expectations for licensed operators and licensee duty team personnel regarding TS operability and the decision-making process.

The licensee's causal evaluations associated with this event were still in progress at the conclusion of the inspection period. This event remains under NRC review.

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

b. Findings

The event remained under NRC review at the end of the inspection interval.

.2 Event Notification No. 52232: Automatic Unit Trip Due to Main Generator Lockout

a. Inspection Scope

In the early morning hours on September 10, 2016, the local area in the vicinity of the facility experienced periods of heavy rain. A roof vent on the station's turbine building was stuck open and the licensee erected a catch basin on the turbine deck to collect rainwater intruding into the building through the open vent. During one particularly heavy period of rainfall, the catch basin overflowed and water migrated through a turbine deck floor plug seam and found its way into a control cabinet for the main electrical generator exciter and voltage regulator on the level below. At approximately 3:43 a.m., with the unit operating at full power, the main electrical generator suffered a lockout and an associated reactor trip.

Coincident with the reactor trip, an unrelated failure in the plant's integrated control system caused SG No. 1 to experience a high water level condition; operators initiated a SFRCS actuation to isolate the SGs and mitigate the condition. This complicated the response of control room operators to the trip by removing main feedwater and initiating AFW to supply both SGs, and by removing the main condenser as the plant's heat sink, forcing operators to vent steam to the atmosphere.

NRC inspectors responded to the site immediately following the reactor trip and remained on station in the site's control room providing independent assessment of the event until it was determined that the plant was stable and that the licensee was able to move forward with recovery operations. 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 realignment of plant equipment in response to the trip and SFRCS actuation;
- The performance of plant operators in the control room and in the field;
- 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;
- The licensee's termination from their trip response procedures and transition to normal shutdown plant operations; and
- The licensee's initial investigations and corrective actions associated with the event.

The licensee's causal evaluations associated with this event were still in progress at the conclusion of the inspection period. This event remains under NRC review.

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

b. Findings

The event remained under NRC review at the end of the inspection interval.

.3 Event Notification No. 52247: Essential Buses Not Aligned to Power Transformers During Plant Startup

a. Inspection Scope

On September 16, 2016, the licensee was preparing to start up the unit from a forced maintenance outage. At approximately 4:57 p.m. control room operators transitioned the RCS from cold shutdown (Mode 5) to hot shutdown (Mode 4). In the hot shutdown condition, many conditions of the plant's operating license required for operation at power become effective, including the requirements for essential electric power. At approximately 5:10 p.m., control room operators identified that the station's 480 Vac essential electrical buses (E1 and F1) were still aligned to their respective shutdown power sources, contrary to the requirements of TS 3.8.9, "Distribution Systems – Operating."

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 2016-11004 and commissioned a formal causal analysis. Initial corrective actions taken by the licensee included immediately realigning buses E1 and F1 to the sources required for operation of the plant at power. This action was completed at approximately 5:33 p.m. on September 16, 2016, or approximately 36 minutes later than required by plant TS.

The licensee’s causal evaluations associated with this event were still in progress at the conclusion of the inspection period. This event remains under NRC review.

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

b. Findings

The event remained under NRC review at the end of the inspection interval.

.4 (Closed) Licensee Event Reports 05000346/2016–004–00 and 05000346/2016–004–01: Reactor Coolant System Hot Leg Resistance Temperature Detector Wire Insulation Degradation

a. Inspection Scope

On April 5, 2016, with the unit in cold shutdown for its 19th scheduled RFO, plant personnel identified that all six dual-element RTDs on both RCS hot legs had experienced wire insulation degradation during the preceding reactor operating cycle. The cause of the RTD insulation degradation was determined to have been accelerated aging due to high temperatures as a result of improper configuration of piping insulation on the RTD enclosures during the unit’s previous RFO in 2014.

Although pertaining to the same RTDs, an unrelated issue identified at the same time involving the conductor seal assemblies for several of the RTDs was also discussed by the licensee within these licensee event reports (LERs). A finding of very low safety significance (Green) and an associated NCV for that condition are discussed in Section 1R15.1 of this report.

In response to receipt of these LERs the inspectors completed 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 LERs.

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

b. Findings

.5 Inadequate Modification Design Control Measures Result in Reactor Protection System Inoperability

Introduction

A self-revealed finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," were identified for the licensee's failure to have adequately prepared and implemented a permanent plant modification associated with SG replacement during the unit's 18th RFO in 2014. Specifically, in conjunction with SG replacement the licensee had also replaced a significant amount of RCS piping and instrumentation, including all RCS hot leg RTDs. The RTD housings were improperly insulated during the modification, such that over the ensuing reactor operating cycle the RTD wiring insulation degraded to the extent that nearly all the RTDs were rendered inoperable.

Description

As part of the licensee's SG replacement project during the 2014 RFO, one of the associated permanent plant modifications was the replacement of the all six dual-element RCS Hot Leg RTDs and RTD cables.

On October 8, 2014, the licensee identified that the RTD associated with RPS Channel No. 1 was exhibiting some erratic indications, albeit still within allowable tolerances. The licensee continued to monitor the instrument. Ultimately, on July 17, 2015, the licensee was no longer able to maintain the instrument in calibration and it was declared inoperable. The associated RPS Channel No. 1 was also declared inoperable and maintained in the tripped or bypassed state to comply with plant TS. The licensee determined through troubleshooting that repairs could not be made to the RTD with the plant operating at power, and scheduled the repair activities for the next RFO, which was scheduled for the spring of 2016.

On April 5, 2016, with the unit in cold shutdown for its 19th scheduled RFO, the RTD for RPS Channel No. 1 was inspected by the licensee. The inspection identified that the RTD wire insulation was severely degraded due to apparent heat damage. All six dual-element RTDs on both RCS hot legs were subsequently inspected and observed to have varying degrees of wire insulation degradation, up to total wastage.

The licensee entered the condition into their CAP as CR 2016–04587 and commissioned a formal causal analysis. Even though each appeared to remain functional, the RTDs associated with RPS Channel Nos. 2 through 4 were determined by the licensee through engineering analysis to have been rendered inoperable by the condition during the preceding reactor operating cycle and, thus, incapable of being credited to meet their applicable plant TS requirements. In addition, RTDs utilized for the unit's post-accident

monitoring system (PAMS) and NNI system were also adversely impacted. However, the inspectors determined that these issues were of minor safety significance, and not subject to formal enforcement in accordance with Section 2.3.1 of the NRC's Enforcement Policy.

Due to the size and scope of the 2014 SG replacement project, the licensee utilized a sizable amount of contract engineering support. The licensee's formal causal analysis determined that the design of the insulation package for the new RTDs had been performed by contractor engineering personnel, and that communications and other human errors had occurred with the development of the insulation modification and insulation installation WO. Comments and statements pertaining to insulation requirements from one engineer responsible for RTD environmental qualifications were not fully addressed and not passed on to the engineer responsible for the insulation package design.

In addition to the licensee's formal causal analysis, corrective actions taken included replacing five of the six RCS hot leg RTDs. The sixth RCS hot leg RTD was evaluated by licensee engineering and determined to be suitable for continued operation. The licensee currently plans to replace it during their next RFO, which is scheduled for the spring of 2018.

<u>Analysis</u>

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 developed and implemented an appropriate insulation package for their RCS hot leg RTD housings during the unit's 2014 SG replacement outage constituted a performance deficiency that was reasonably within the licensee's ability to foresee and correct, and that 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 design control, 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)."

Although the issue was identified with the reactor shutdown for refueling, the inspectors evaluated the finding using IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” because the condition involved the operability of the RCS hot leg RTDs while the unit was operating at power during the past reactor operating cycle. Using Exhibit 2 – “Mitigating Systems Screening Questions,” the inspectors determined that a detailed risk analysis by the NRC Region III Senior Reactor Analyst (SRA) was required since the finding involved the inoperability of more than one channel of RPS for greater than the TS allowed outage time.

The SRA used the Davis-Besse Standardized Plant Analysis Risk (SPAR) Model, Version 8.19, and Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE), Version 8.1.4, for the calculation of the change in core damage frequency (CDF) for the issue.

The following assumptions were made in the analysis:

- The exposure time for the issue was assumed to be one year, which is the maximum time allowed by the risk assessment standardization project handbook;
- The RPS reactor trips that were assumed to be unavailable as a result of the inoperability of the RCS hot leg RTDs were the RCS high temperature and RCS pressure/temperature reactor trips, since RCS hot leg temperature provides inputs to the circuitry for these reactor trips;
- Per the TS Bases, Section B3.3.1, "RPS Instrumentation," the RCS high temperature and RCS pressure/temperature reactor trips are not credited for transient protection in the USAR; and
- Though not credited for transient protection, it was conservatively assumed that the RCS high temperature and RCS pressure/temperature reactor trips provided backup protection to all the reactor trips, such that with these two trips available the probability of malfunction of all the automatic reactor trip instrumentation functions was reduced from its nominal low value of 1.40E–5 (in the SPAR model) to zero.

The result was a change in CDF of less than 1E–7 events per year. The dominant core damage sequence was a transient initiating event with a failure of the automatic RPS circuitry to trip to allow the control rods to insert into the core, along with the failure of plant operators to manually trip the reactor, and along with a failure of plant operators to initiate emergency RCS boration. Based on the detailed risk evaluation, the inspectors determined that the finding was of very low safety-significance (Green).

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's SG replacement project management team failed to reinforce the importance of close communication between responsible engineers with overlapping and interfacing modification packages, and did not adequately promote effective work execution through the use of clearly defined work documents that were written and structured to minimize the likelihood for human error. (H.2)

<u>Enforcement</u>

Appendix B of 10 CFR Part 50, Criterion III, "Design Control," states, in part, that:

> *Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.*

Contrary to this requirement, during the licensee's 2014 SG replacement outage the permanent plant modification package developed for the replacement of the plant's six dual-element RTDs on both RCS hot legs did not correctly translate the RTD insulation requirements into the specifications, drawings, procedures, and instructions for those components.

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 2016–04587, the associated violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000346/2016003–02)**

.6 <u>(Closed) Licensee Event Report 05000346/2016–005–00: Plant Startup with Anticipatory Reactor Trip System in Main Turbine Bypass</u>

On May 10, 2016, at approximately 5:28 a.m., the unit was at 53 percent and increasing power following the 19th RFO. A routine check of control room indications by the Operations Manager, who was providing senior leadership oversight of plant startup activities, identified that all four anticipatory reactor trip system (ARTS) instrumentation channels were in bypass for both the main turbine and SFRCS/main feed pump trip functions. The main turbine function is required by plant TS to be in the normal/enabled state when the unit is operating above 45 percent power; the SFRCS/main feed pump trip function is required by plant TS to be in the normal/enabled state when the unit is operating at power in Mode 1. The main turbine and SFRCS ARTS functions were restored to the normal/enabled state and all applicable TS actions exited at approximately 5:52 a.m. Based on a log of the unit's power history, the licensee determined that the unit had been operating in a condition prohibited by plant TS for the main turbine function for the approximately 54 minutes, and for the SFRCS / main feed pump trip function since 3:24 p.m. on May 9, 2016, when the unit entered Mode 1.

The ARTS consists of four channels located in individual locked cabinets in the rear area of the plant's control room. The purpose of the ARTS is to initiate a reactor trip on a loss of main feedwater or a main turbine trip with the unit operating at higher power levels to reduce the magnitude of RCS pressure and temperature transients from these events. If any two ARTS channels transmit channel trip signals, the logic trip module in each channel actuates to remove power from its associated control rod drive (CRD) trip breaker. While beneficial, the ARTS is not credited to mitigate the consequence of any accident described in the USAR.

The licensee's investigation into the mispositioning of the main turbine and SFRCS ARTS function switches identified that the switches were most probably inadvertently left in the bypass condition following testing associated with the unit's new digital CRD system, which had been installed during the refuel outage. Additionally, the licensee's investigation determined that plant operators failed to adequately implement applicable unit operating procedures, which called for verification of the ARTS channels being in the normal/enabled state prior to reaching approximately 40 percent power. Licensee corrective actions included the issuance of an operations standing order to require periodic walk downs of all control room panels by on-watch control room operators in pairs to ensure a comprehensive understanding of plant status awareness, as well as enhancements to applicable procedures.

The inspectors' review of this event determined that the operation of the unit above 45 percent power with the main turbine and SFRCS ARTS functions bypassed on all four channels constituted a licensee-identified violation of TS 3.3.16, which was of very low safety significance. Further details of this licensee-identified violation are discussed in Section 4OA7.1 of this report. The licensee had entered this event into their CAP as CR 2016–06563.

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

4OA5 Other Activities

.1 Summer 2016 Groundwater Sampling Results

a. Inspection Scope

The inspectors reviewed the results of a series of groundwater samples taken from seven 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), 05000346/2016001 (ADAMS Accession No. ML16118A435), and 05000346/2016002 (ADAMS Accession No. ML16207A600). 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 a part of the inspectors' daily plant status monitoring activities.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 6, 2016, the inspectors presented the inspection results to Mr. B. Boles, the Site Vice President, 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 results of the Emergency Preparedness Program inspection with Mr. D. Imlay, General Plant Manager, conducted at the site on September 29, 2016; and

- The inspection results for the Radiation Safety Program review with Mr. D. Imlay, General Plant Manager, on September 29, 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 Violations

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 Operation of the Plant Above Allowed Power Level with All Four Anticipatory Reactor Trip System Channels Bypassed

Plant TS 3.3.16, "Anticipatory Reactor Trip System (ARTS) Instrumentation," requires that three ARTS channels for the main turbine trip function be maintained operable with the unit operating in Mode 1 above 45 percent power, and three ARTS channels for the SFRCS / main feed pump trip function be maintained operable with the unit operating in Mode 1 at any power. While this TS provides actions and allowed outage time for a single inoperable ARTS channel, there are no provisions for more than a single ARTS channel being simultaneously inoperable. The provisions of TS Limiting Condition for Operation 3.0.3, therefore, apply when more than one ARTS channel is inoperable at the same time, and require that actions be initiated within 1 hour from the onset of the condition to:

- Be in Mode 3 within 7 hours;
- Be in Mode 4 within 13 hours; and
- Be in Mode 5 within 37 hours.

As discussed in Section 4OA3.5 of this report, contrary to the requirements of TS 3.3.16, all four ARTS channels were bypassed and inoperable for both the main turbine and SFRCS functions for a period of approximately 15 hours on May 9–10, 2016.

The objective of the Mitigating Systems Cornerstone of Reactor Safety is to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Key attribute associated with

this objective are human performance and configuration control. In accordance with NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the violation was of more than minor significance in that it had a direct impact on this cornerstone objective. Specifically, plant operators in failing to adequately implement applicable operating procedures allowed the unit to enter into a mode of operation with less that the required three channels of ARTS operable and available. Using Exhibit 2 – "Mitigating Systems Screening Questions," the inspectors determined that a detailed risk analysis by the NRC Region III SRA was required since the issue involved the inoperability of more than one channel of ARTS, a condition for which there is no allowed outage time specified in TS 3.3.16.

The SRA used the Davis-Besse SPAR Model, Version 8.19, and SAPHIRE, Version 8.1.4, for the calculation of the change in CDF for the issue.

The following assumptions were made in the analysis:

- The exposure time for the issue was conservatively assumed to be 15 hours, from 3:24 p.m. on May 9, 2016, when the unit entered Mode 1 and the TS 3.3.16 for the ARTS became applicable to 5:52 a.m. on May 10, 2016, when the ARTS bypass switches were returned to the normal/enabled state; and
- With the ARTS SFRCS function bypassed, the SFRCS input to the ARTS to provide a reactor trip signal was bypassed. Since the ARTS is not modeled in the SPAR model, it was very conservatively assumed that the RPS automatic trips were bypassed during the 15–hour exposure time, and only a manual reactor trip was available.

The result was a change in CDF of 7.6E–7 events per year. The dominant core damage sequence was a transient initiating event with a failure of plant operators to manually trip the reactor, along with a failure of plant operators to initiate emergency RCS boration. Based on the detailed risk evaluation, the inspectors determined that the violation was of very low safety-significance (Green).

As discussed in Section 4OA3.5 of this report, the licensee had entered this issue into their CAP as CR 2016–06563. In addition to the commissioning of a formal root cause evaluation, licensee corrective actions included the issuance of an operations standing order to require periodic walk downs of all control room panels by on-watch control room operators in pairs to ensure a comprehensive understanding of plant status awareness and enhancements to applicable operating procedures.

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, General Plant Manager
J. Kraus, Supervisor, Radiation Protection Services
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, Director, Site Performance Improvement
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

05000346/2016003–01	NCV	Inadequate Instructions to Correctly Assemble Electrical Conductor Seal Assemblies (Section 1R15.1)
05000346/2016003–02	NCV	Inadequate Modification Design Control Measures Result in Reactor Protection System Inoperability (Section 4OA3.4)

Closed

05000346/2016003–01	NCV	Inadequate Instructions to Correctly Assemble Electrical Conductor Seal Assemblies (Section 1R15.1)
05000346/2016003–02	NCV	Inadequate Modification Design Control Measures Result in Reactor Protection System Inoperability (Section 4OA3.4)
05000346/2016–004–00 05000346/2016–004–01	LER	Reactor Coolant System Hot Leg Resistance Temperature Detector Wire Insulation Degradation (Section 4OA3.4)
05000346/2016–005–00	LER	Plant Startup with Anticipatory Reactor Trip System in Main Turbine Bypass (Section 4OA3.5)

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.

1R01 Adverse Weather Protection

- RA-EP-02810; Tornado or High Winds; Revision 12
- RA-EP-02830; Flooding; Revision 4
- RA-EP-02880; Internal Flooding; Revision 4
- L-14-104; FirstEnergy Nuclear Operating Company (FENOC) Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Flooding Aspects of Recommendation 2.1 of the Near-Term Task Force (NTTF) Review of Insights from the Fukushima Dai-ichi Accident; March 11, 2014
- Individual Plant Examination of External Events for the Davis-Besse Nuclear Power Station Section 5.3 and 5.4; High Winds and Tornadoes and Floods; December 1996

1R04 Equipment Alignment

Condition Reports:
- 2013-09452; Oil Leak on Motor Driven Feed Water Pump Inboard Bearing P241
- 2016-05518; Gasket Leaking above E184-2 Motor Driven Feed Pump Seal Water Cooler
- 2016-06048; EDG 2 DC Turbo Oil Pump (P147-6) Failed to Start When AC Turbo Oil Pump was turned off
- 2016-07174; EFP Oil Level Discrepancies
- 2016-07578; HIS105A Green Indicating Light for the Electric Fire Pump was Found Out
- 2016-07864; FP2984 (EFP Discharge Relief Valve) Leak By
- 2016-08421; Valves EF46, EF31 and EF47 cannot be Manipulated Due to Incorrect Installation of Hand Operator
- 2016-08422; Leak on Threaded Pipe Connection Downstream of EF27
- 2016-08564; EDG A/C #2 Failed Post Maintenance Charging Test
- 2016-08601; DB FLEX/EFW Project – Foreign Material Identified in EFW Pump Bearing Housing
- 2016-08604; DB FLEX / EFW Project – Damaged Ball Identified in EF31 Ball Valce
- 2016-08669; EFW Facility Diesel Oil Storage Tank T14 Level Indicator LTEF106 Periodically Displays a Level Drift
- 2016-08681; EFWF Battery Charger D5005A High Voltage Alarm in Without Actual High Voltage Condition
- 2016-09005; CAT Engine Engine Control System for K310 Engine Did Not Retain Engine Rating
- 2016-09158; EFW Storage Tank, T89, Exceeded FLEX Specification Maximum Temperature per NORM-LP-7202 During 24 Hour Run of EFW Pump P310
- 2016-09159; EFW Fuel Oil Day Tank Makeup Pump Threaded Connection Leaking Past Gasket
- 2016-09234; EFW Pump Mechanical Seal has a Small Leak
- 2016-09535; EFW Overhead Door and Its Missile Shield Found Open
- 2016-10003; Questioning Attitude Prevented Improper Return to Service of the New EFW Diesel Engine

- 2016-10039; Higher Than Expected System Pressure During EF12 Check Valve Testing
- DB-FP-04048; Electric Fire Pump Test; Revision 12
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 59
- DB-OP-06234; Emergency Feedwater System; Revision 0
- DB-OP-06235; EFW Facility Electrical and Support Systems Procedure; Revision 0
- DB-OP-06601; Station Fire Suppression Water System; Revision 34
- DB-PF-06705; Tank Level Calibration Curves; Revision 12
- DB-SC-03071; Emergency Diesel Generator 2 Monthly Test; Revision 35
- DB-SS-03090; Motor Driven Feed Pump Monthly Valve Verification; Revision 11
- DB-SS-03091; Motor Driven Feed Pump Quarterly Test; Revision 16
- DB-SS-04202; Emergency Feedwater Pump Monthly Test; Revision 0 and 1
- NORM-LP-7202; Davis-Besse Specifications for Flex Equipment Out of Service; Revision 2
- 200684576; Electric Fire Pump Stop Indication Out
- 600828707; Oil Leak Electric Fire Pump Thermocouple
- 601016012; EFP Panel Batteries Need Replaced
- 601035887; Gasket Leaking Above E184-2 Motor Driven feed Pump Seal Water Cooler
- 601045817; Electric Fire Pump Stop Indication Out
- 601055007; EFWF MCC Positions and Blue Tape Issues
- 601058481; Flashing Red Light on EFW MCC Supply Breaker
- M-0006D; Auxiliary Feedwater System; Revision 59
- M-0007B; Steam Generator Secondary System; Revision 61
- M-0016A; Station Fire Protection System; Revision 56
- M-0016B; Station Fire Protection System; Revision 52
- M-0017A; Diesel Generators; Revision 19
- M-0017B; Diesel Generators Air Start; Revision 47
- M-0017C; Fuel Oil; Revision 30
- M-0052; Emergency Feedwater System; Revision 2
- OS-0010, Sheet 1; Condensate System; Revision 23
- OS-0012A, Sheet 1; Main Feedwater System; Revision 26
- OS-0012A, Sheet 2; Main Feedwater System; Revision 33
- OS-0017A, Sheet 1; Auxiliary Feedwater System; Revision 34
- OS-0017B, Sheet 1; Auxiliary Feedwater Pumps and Turbines; Revision 25
- OS-0041A, Sheet 1; Emergency Diesel Generator Systems; Revision 33
- OS-0041A, Sheet 2; Emergency Diesel Generator Systems; Revision 32
- OS-0041B; Emergency Diesel Generator Air Start / Engine Air System; Revision 42
- OS-0041C; Emergency Diesel Generator Diesel Oil System; Revision 16
- OS-0047A; Station Fire Protection System; Revision 25
- OS-0062; Emergency Feedwater System; Revision 2
- OS-0062, Sheet 1; Emergency Feedwater System; Revision 1

<u>1R05 Fire Protection</u>

- 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-MS-09270; Fire Door Installation; Revision 3
- DB-OP-02501; Serious Station Fire; Revision 26
- DB-OP-02529; Fire Procedure; Revision 10
- PFP-AB-402; No. 1 Electrical Penetration Room – Room 402, Fire Area DG; Revision 5

- PFP-AB-427; No. 2 Electrical Penetration Room – Room 427, Fire Area DF; Revision 4
- PFP-AB-428; Low Voltage Switchgear Room F-Bus – Room 428, Fire Area X; Revision 4
- PFP-AB-428A; Battery Room B – Room 428A, Fire Area X; Revision 4
- PFP-AB-428B; No. 1 Electrical Isolation Room – Room 428B, Fire Area X; Revision 4
- A-0223F; Fire Protection General Floor Plan Elevation 585'-0"; Revision 25
- A-0224F; Fire Protection General Floor Plan Elevation 603'-0"; Revision 26
- A-0226F; Fire Protection General Floor Plan Elevation 643'-0"; Revision 14
- A-0232F; Emergency Feedwater Facility Fire Protection General Floor Plan; Revision 0
- Fire Hazard Analysis Report; Revision 26
- Fire Hazard Analysis Report: Change Notice 15-089; Emergency Feedwater Facility; 6/28/2016
- GEN-SAF-0001; Generation Personal Safety Manual; Revision 2

1R06 Flood Protection Measures

- 2016-09710; Spent Fuel Pool Leakage Sample Results
- 2016-09784; ECCS No. 1 Sump Pumps Not Alternating
- 2016-09986; Flapper Gate on Drain in No. 1 MPR Found Partially Open
- DB-MM-05003; Vibration Monitoring; Revision 11
- RA-EP-02830; Flooding; Revision 4
- RA-EP-02880; Internal Flooding; Revision 4
- A-0211; Post-Accident Radiation Zones, Containment and Auxiliary Building Plan Elevation 545'-0"; Revision 2
- A-0212; Post-Accident Radiation Zones, Containment and Auxiliary Building Plan Elevation 565'-0"; Revision 1
- A-0213; Post-Accident Radiation Zones, Containment and Auxiliary Building Plan Elevation 585'-0"; Revision 3
- 200668462; Repair/Replace Sump Pumps P77-1A and 1B – PERP 1227

1R11 Licensed Operator Requalification Program and Licensed Operator Performance

- 2016-10723; SG/RX Demand Response Following Trip
- 2016-10724; RFR Response Following Reactor Trip
- 2016-10725; Reactor Trip Due to Water Intrusion into Automatic Voltage Regulator Cabinet
- 2016-10726; Broken Roof Vent on Northwest Corner of Turbine Building Allowed Rain Intrusion into Exciter Cabinet on 603 Turbine Building
- 2016-10733; Post Transient Assessment of ARTS
- 2016-10741; MS101, Main Steam Isolation Valve No. 1, Exceeds Transient Assessment Program Specified Time Requirement
- 2016-10745; Pressurizer Level During Reactor Trip
- 2016-10748; Unplanned LCO 3.8.1 Entry Due to Low 345 KV Bus Voltages
- 2016-10760; Q233, Control Rod Drive 5 Vdc Power Supply B Fault, Indicates Not Normal
- 2016-10762; Pressurizer Code Safety Leakage
- 2016-10765; SP7B/A Not Functioning Properly During Cooldown
- 2016-10774; BACC - Active Leak from Pressurizer Code Safety Valve
- 2016-10883; Mispositioning Plant Status Control Level 3 - RC1719A Containment Vent Header Isolation Found Closed
- 2016-11004; E1 & F1 on Transformers with Shutdown Tap Settings While in Mode 4
- 2016-11025; Pressurizer Spray Line Temperature Differential Exceeded 300°F During RCS Heatup
- 2016-11034; Pressurizer Code Safety Leakage

- 2016-11051; Pressurizer Spray Line and Spray Nozzle Differential Temperature
- 2016-11068; Pressurizer Code Safety Leakage
- 2016-11138; Main Turbine Trip During Startup - Misposition Event
- DB-NE-06202; Reactivity Balance Calculations; Revision 10
- DB-OP-02000; RPS, SFAS, SFRCS Trip, or Steam Generator Tube Rupture; Revision 29
- DB-OP-02520; Load Rejection; Revision 7
- DB-OP-02526; Primary to Secondary Heat Transfer Upset; Revision 4
- DB-OP-02546; Degraded Grid; Revision 4
- DB-OP-06000; Filling and Venting the Reactor Coolant System; Revision 29
- DB-OP-06002; RCS Draining and Nitrogen Blanketing; Revision 23
- DB-OP-06003; Pressurizer Operating Procedure; Revision 31
- DB-OP-06004; Quench Tank; Revision 11
- 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-06202; Turbine Operating Procedure; Revision 28
- DB-OP-06224; Main Feed Pump and Turbine; Revision 38
- DB-OP-06301; Generator and Exciter Operating Procedure; Revision 28
- DB-OP-06401; Integrated Control System Operating Procedure; Revision 25
- DB-OP-06402; Control Rod Drive Operating Procedure; Revision 28
- DB-OP-06900; Plant Heatup; Revision 64
- DB-OP-06901; Plant Startup; Revision 38
- DB-OP-06902; Power Operations; Revision 56
- DB-OP-06903; Plant Cooldown; Revision 48
- DB-OP-06904; Shutdown Operations; Revision 47
- DB-OP-06912; Approach to Criticality; Revision 18
- DB-SC-03272; Control Rod Exercising Test; Revision 5
- DB-SC-04029; Davis-Besse Overexcited Capability Test; Revision 1
- DB-SP-04150; AFP 1 Monthly Test; Revision 16
- DB-SS-04150; Main Turbine Stop Valve Test; Revision 14
- DB-SS-04151; Main Turbine Control Valve Test; Revision 16
- DB-SS-04152; Main Turbine Combined Intermediate Valve Test; Revision 11
- NOP-OP-1002; Conduct of Operations; Revision 11
- NOP-OP-1013, Control of Time Critical Operator Actions, Revision 1
- NOP-TR-1010; Licensed Operator Requalification Exam Development; Revision 2
- NOP-TR-1200; Conduct of Training; Revision 3
- NOP-TR-1280; FENOC Simulator Configuration Management; Revision 0
- NG-DB-00319; Control of the Emergency Operating Procedure and Technical Bases; Revision 4
- NG-NT-00600; Training and Qualification; Revision 6
- NG-NT-00601; Control of the Plant-Referenced Simulator; Revision 3
- NT-OT-7001; Training and Qualification of Operations Personnel; Revision 14
- DBBP-TRAN-0014; License Requirements for Licensed Individuals; Revision 11
- DBBP-TRAN-0021; Simulator Configuration Control; Revision 5
- DBBP-TRAN-0502; Continuing Training Simulator Evaluations; Revision 11
- NOBP-OP-0007; Conduct of Infrequently Performed Tests or Evolutions; Revision 5
- NOBP-TR-1112; FENOC Conduct of Simulator Training and Evaluation; Revision 3
- NOBP-TR-1151; Operating Crew Performance Critique; Revision 1
- NOBP-TR-1200; Operator Fundamentals; Revision 1
- DBBP-OPS-1013; Control of Time Critical Actions; Revision 2

- DBBP-OPS-1113; Control of Time Sensitive Operator Actions; Revision 0
- NORM-OP-1002; Conduct of Operations Handbook; Revision 5

1R12 Maintenance Effectiveness

- 2016-10762; Pressurizer Code Safety Leakage
- 2016-10774; BACC - Active Leak from Pressurizer Code Safety Valve
- 2016-11034; Pressurizer Code Safety Leakage
- 2016-11068; Pressurizer Code Safety Leakage
- DB-PF-03000; Pressurizer Code Safety Valve Testing; Revision 3
- DB-MI-03743; Channel Functional/Calibration of Pressurizer PORV and Safety Relief Valve Position Indicators, Channel 1; Revision 8
- DB-MI-03744; Channel Functional/Calibration of Pressurizer PORV and Safety Relief Valve Position Indicators, Channel 2; Revision 8
- DB-MI-09015; Maintenance of TEC/ENDEVCO Model 2273AM1 Accelerometer and TEC Model 504A Charge Converter; Revision 3
- DB-MM-09299; Pressurizer Code Relief Valve Removal and Installation; Revision 6
- 200682856; RC13B Forced Outage Replacement
- 14-272; Valve Serial N54891-00-0002; P.O. No. 45455222; March 2016
- 14-273; Valve Serial N56264-00-0005; P.O. No. 45455222; March 2016
- 16-224; Valve Serial N59303-00-0001; P.O. No. 45495735; June 2016
- 16-236; Valve Serial N54891-00-0001; P.O. No. 45495735; September 2016
- Davis-Besse Nuclear Power Plant Design Basis Assessment Report; First Half 2016
- Davis-Besse Plant Health Report; Second Half 2015
- MRPM; Maintenance Rule Program Manual; Revision 35

1R13 Maintenance Risk Assessments and Emergent Work Control

- 2012-01332; API for Control Rod 7-3 Erratic
- 2012-01620; ODMI: Contingency for a Potential Control Rod 7-3 Asymmetric Indication Condition 2012-02873; ODMI: Revision 01 for Contingency for a Potential Control Rod 7-3 Asymmetric Indication Condition
- 2016-06515; Vibration Felt and Heard When Additional Portion of AFW Recirculation Line Placed In Service
- 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
- 2016-08956; AFPT 1 Abnormal Vibration on the Flow Test Line During Quarterly Test
- 2016-08958; Abnormal flow noise experienced when AFW Pump 1 Recirculation Was Realigned to the CST
- 2016-08970; Walkdown Findings of AFW Train 1
- 2016-08986; PDI2658 Oscillations During AFW Train 1 Quarterly Pump Test
- 2016-10298; Rod 7-3 (H12) Indication Fluctuation
- 2016-10633; Gas Void Detected Upstream of DH158 (DH Pump 2 Discharge Line Vent)
- 2016-10480; Control Rod 7-3 API to RPI Difference Exceeds Allowable Tolerance
- DB-PF-03000; Pressurizer Code Safety Valve Testing; Revision 3
- DB-PF-03010; RCS Leakage Test; Revision 15
- DB-PF-03065; System Leakage Tests; Revision 14
- DB-MI-03743; Channel Functional/Calibration of Pressurizer PORV and Safety Relief Valve Position Indicators, Channel 1; Revision 8
- DB-MI-03744; Channel Functional/Calibration of Pressurizer PORV and Safety Relief Valve Position Indicators, Channel 2; Revision 8

- DB-MI-09015; Maintenance of TEC/ENDEVCO Model 2273AM1 Accelerometer and TEC Model 504A Charge Converter; Revision 3
- DB-MM-09299; Pressurizer Code Relief Valve Removal and Installation; Revision 6
- DB-OP-02005; Primary Instrumentation Alarm Panel 5 Annunciators; Revision 19
- DB-OP-02516; CRD Malfunctions; Revision 15
- DB-SP-03151; AFP 1 Quarterly Test; Revision 24
- NA-QC-05560; Visual Examination Procedure for VT-1, VT-3, and General Visual Examinations; Revision 11
- NOP-WM-4001; Foreign Material Exclusion; Revision 13
- DBBP-OPS-0003; On-Line Risk Management Process; Revision 12
- DBBP-OPS-0011; Protected Equipment Posting; Revision 9
- 200684075; Troubleshoot / Resolve Rod 7-3 API Drift
- 200682856; RC13B Forced Outage Replacement
- 200688536; PDI2658 Aux Feed Pump Discharge Test Flow Instrument 3 Point Check
- 601058857; Rod H12 Fluctuating Indication
- M-0006D; Auxiliary Feedwater System; Revision 59
- M-0033A; High Pressure Injection, Revision 48
- M-0033B; Decay Heat Train 1; Revision 57
- M-0033C; Decay Heat Train 2; Revision 29
- OS-0004, Sheet 1; Decay Heat Removal / Low Pressure Injection System; Revision 57
- Cycle 20 Reactor Operating Guidance 78 EFPD to 129 EFPD; Revision 0

1R15 Operability Determinations and Functionality Assessments

- 2014-06073; Conax Seals Were Installed and Torqued Outside the Crimp Bands for TE-RC3B2 and TE-RC3A4 While Prepping Them on the Bench Prior to Installation in the Plant
- 2014-07256; TERC3B5 Conax Damaged
- 2016-05782; New Hot Leg RTD Assembled Incorrectly
- 2016-05792; Incorrectly Assembled Conax Connectors
- 2016-07643; New Conax ECSAs Assembled Incorrectly
- 2016-08385; Evidence of Water in O'Brien Box for DB-LT1525A
- 2016-08402; SFAS Channel 2 +15V Power Supply Failure
- 2016-08415; LCO 3.3.5 Parameter 5 BWST Level Low-Low Operability
- 2016-08419; Performance Review of LCO 3.3.5 Application During LT1525A Maintenance
- 2016-08699; PA-DB-16-03: Crew Briefing Performance Shortfall
- 2016-08700; PA-DB-16-03: Delayed Request for Prompt Operability Determination
- 2016-08765; Restoration of SFAS Channel 1 (LT-1525A) - Assessment of Organizational Response to Extended Work Window
- 2016-10167; EDG 1 and 2 Fuel Oil Storage Tank Cetane Index
- DB-CH-03023; Emergency Diesel Generator Fuel Oil Storage Tank 2 Analysis; Revision 15
- DB-CH-03024; Emergency Diesel Generator Fuel Oil Storage Tank 1 Analysis; Revision 15
- DB-CH-06900; Operational Chemical Control Limits; Revision 57
- DB-ME-09501; Installation of Connectors; Revisions 7 and 8
- DB-OP-03006; Miscellaneous Instrument Shift Checks; Revision 55
- DB-SC-03180; Remote Shutdown, Post Accident Monitoring Instrumentation Monthly Channel Check; Revision 14
- 200574536; ECP 10-0466-009 Replace TERC3B2/3B5
- 200574538; ECP 10-0466-008 Replace TERC3A4/3A6
- 200678041; Replace TERC3A4/3A6
- 200678042; Replace TERC3B2/3B5
- E-1039; EQ Master Report; Revision 19

- M-030A; Reactor Coolant System; Revision 72
- M-030B; Reactor Coolant System Instrumentation; Revision 26
- G-E-00058; Conax Nuclear Installation Manual for Electric Conductor Seal Assemblies With Longbody for Pipe Thread Equipment Inferface; Revision 4 [Proprietary]
- BETA Laboratory Analytical Report; Davis-Besse EDG 1 Fuel Oil Storage Tank Monthly; 5/12/2016, 6/21/2016, and 8/29/2016
- BETA Laboratory Analytical Report; Davis-Besse EDG 2 Fuel Oil Storage Tank Monthly; 5/12/2016, 6/21/2016, and 8/29/2016
- BETA Laboratory Analytical Report; Davis-Besse EDG 1 Fuel Oil Storage Tank New Fuel; 4/27/2016 and 7/20/2016
- BETA Laboratory Analytical Report; Davis-Besse EDG 2 Fuel Oil Storage Tank New Fuel; 5/4/2016
- BETA Laboratory Analytical Report; Davis-Besse EDG 1 Fuel Oil Storage Tank Quarterly; 4/18/2016 and 7/22/2016
- BETA Laboratory Analytical Report; Davis-Besse EDG 2 Fuel Oil Storage Tank Quarterly; 4/18/2016 and 7/22/2016
- SGS Herguth Laboratories Certificate of Analysis Lab Number V5021521; T153-1 Emergency Diesel Generator FOST 1-1; 8/26/2016
- SGS Herguth Laboratories Certificate of Analysis Lab Number V5021521; T153-2 Emergency Diesel Generator FOST 1-2; 8/26/2016
- Specification Number M-182Q; Technical Specification for Operational Phase for Diesel Fuel Oil Testing for The Toledo Edison Company Davis-Besse Nuclear Power Station; Revision 0

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

- 2014-07972; Small Bore Piping Vibration Readings in Excess of 0.5 in/sec
- 2016-06515; Vibration Felt and Heard When Additional Portion of AFW Recirculation Line Placed In Service
- 2016-06786; Turbo Charger Inlet Filters Missing on EFW CAT Diesel
- 2016-07315; CRD Abnormal Procedure Entry Due to Rod 7-3 Degraded API/RPI Indication]
- 2016-07809; Arc Strike Found on EFW Diesel Engine, K310
- 2016-07844; EFW Diesel, K310, Received With Glycol Not Used at Davis-Besse
- 2016-07862; Smoke Observed Coming from EFW Diesel, K310, Exhaust Piping Insulation During Initial Loaded Run for Testing
- 2016-08040; EFW Diesel Engine, K310, Does Not Run at 1800 RPM Isochronous After Receiving Start Signal
- 2016-08045; EFW Diesel Engine, K310, Fuel Lines Lose Prime Over Time
- 2016-08099; During Operation of the EFW Pump, P310, the Pump Outboard Thrust Bearing Began to Smoke Coincident with EFW Diesel Engine, K310, RPM Drop
- 2016-08209; Failure of EF8, EFW Flow Control Valve, to Control During Testing
- 2016-08213; K310, EFW Diesel Engine, Oil Filter Housing Has Small Crack
- 2016-08759; EFW Diesel Engine, K310, Is Not Capable of Maintaining Rated Speed, 1800 RPM, at Rated Load
- 2016-08956; AFPT 1 Abnormal Vibration on the Flow Test Line During Quarterly Test
- 2016-08958; Abnormal flow noise experienced when AFW Pump 1 Recirculation Was Realigned to the CST
- 2016-08970; Walkdown Findings of AFW Train 1
- 2016-08986; PDI2658 Oscillations During AFW Train 1 Quarterly Pump Test
- 2016-09004; EFW Diesel, K310, Procured Software for Diesel Delivered with Wrong Software License
- 2016-09005; CAT Engine Control System for K310 Engine Did Not Retain Engine Rating

- 2016-09006; Air Intrusion Observed on K310 EFW Diesel Fuel Supply
- 2016-09007; Fuel Line Supply Line for K310 EFW Diesel Does Not Meet CAT Specifications
- 2016-09033; Procedure Not Followed for Connecting Flow Transducer to Auxiliary Feedwater System
- 2016-09158; EFW Storage Tank, T89, Exceeded FLEX Specification Maximum Temperature per NORM-LP-7202 During 24 Hour Run of EFW Pump 310
- 2016-09159; EFW Fuel Oil Day Tank Makeup Pump Threaded Connection Leaking Past Gasket
- 2016-09161; Leak Identified Upstream EF330 on Welded Connection
- 2016-09234; EFW Pump Mechanical Seal Has a Small Leak
- 2016-09334; DB FLEX / EFW Project – EFW Pump Diesel Engine Cranking Battery Rack Does Not Meet Design Requirements
- 2016-10762; Pressurizer Code Safety Leakage
- 2016-10774; BACC - Active Leak from Pressurizer Code Safety Valve
- 2016-11034; Pressurizer Code Safety Leakage
- 2016-11068; Pressurizer Code Safety Leakage
- DB-OP-06234; Emergency Feedwater System; Revision 0
- DB-OP-06301; Generator and Exciter Operating Procedure; Revision 28
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 59
- DB-OP-06900; Plant Heatup; Revision 64
- DB-PF-03065; System Leakage Tests; Revision 14
- DB-PF-06703; Miscellaneous Operation Curves; Revision 23
- DB-SC-03070; Emergency Diesel Generator 1 Monthly Test; Revision 38
- DB-SC-03270; Control Rod Assembly Insertion Time Test; Revision 14
- DB-SP-03151; AFP 1 Quarterly Test; Revision 24
- DB-TP-12425; Post Installation Test Diesel Driven EFW Pump System; Revision 1
- NORM-LP-7202; Davis-Besse Specifications for FLEX Equipment Out of Service; Revision 1
- 200602747; PM 1710 MV3870 (Auxiliary Feedwater Pump 1 Discharge to SG 1) Inspection
- 200605942; PM 1709 MV3869 (Auxiliary Feedwater Pump 1 to SG 2 Stop) Inspection
- 200615793; AF3 Add 2x1 Fillet Weld at Valve (Auxiliary Feedwater Pump 1 Cooling Water Return Line)
- 200637626; PM 5264 Replace Agastat Relays (Auxiliary Feedwater Pump 1)
- 200692440; Troubleshoot / Repair Automatic Voltage Regulator Cabinet
- 200674354; ECP 13-0196-02: Test EFW Diesel Pump P310
- 200680153; AC113 Relay 2X/2X1 Need to be Retested
- 200684534; API Bulkhead Adapter Replacement for Rod 7-3
- 200686244;Troubleshoot / Repair/ Calibrated EDG 1 RPM Indication SI6222
- 200688536; PDI2658 Auxiliary Feed Pump Discharge Test Flow Instrument 3 Point Check
- 200688542; AFP 1 Quarterly
- 200692440; Troubleshoot Automatic Voltage Regulator Water Intrusion
- OS-0001A, Sheet 2; Reactor Coolant System; Revision 29
- M-0006D; Auxiliary Feedwater System; Revision 59
- OS-0017A, Sheet 1; Auxiliary Feedwater System; Revision 34
- ECP-13-0196-002; Install EFW Piping and Instrumentation; Revision 15
- ECP-16-0439-001; Temporary Modification Installation – Cap Code Safety Drain Line; Revision 0
- ECP-16-0439-002; Temporary Modification Restoration – Cap Code Safety Drain Line; Revision 0
- BOP-VT-16-183; VT-2 of Pressurizer Code Safety Valve RC13B; 9/19/2016

1R20 Outage Activities

- 2016-10723; SG/RX Demand Response Following Trip
- 2016-10724; RFR Response Following Reactor Trip
- 2016-10725; Reactor Trip Due to Water Intrusion into Automatic Voltage Regulator Cabinet
- 2016-10726; Broken Roof Vent on Northwest Corner of Turbine Building Allowed Rain Intrusion into Exciter Cabinet on 603 Turbine Building
- 2016-10733; Post Transient Assessment of ARTS
- 2016-10741; MS101, Main Steam Isolation Valve No. 1, Exceeds Transient Assessment Program Specified Time Requirement
- 2016-10745; Pressurizer Level During Reactor Trip
- 2016-10748; Unplanned LCO 3.8.1 Entry Due to Low 345 KV Bus Voltages
- 2016-10755; As Found Condition of 40 Device Main Generator Loss of Field Relay
- 2016-10760; Q233, Control Rod Drive 5 Vdc Power Supply B Fault, Indicates Not Normal
- 2016-10762; Pressurizer Code Safety Leakage
- 2016-10765; SP7B/A Not Functioning Properly During Cooldown
- 2016-10774; BACC - Active Leak from Pressurizer Code Safety Valve
- 2016-10883; Mispositioning Plant Status Control Level 3 - RC1719A Containment Vent Header Isolation Found Closed
- 2016-10883; Impact of Modification to Secure Turbine Building Roof Vents on Turbine Building HELB Analyses
- 2016-11004; E1 & F1 on Transformers with Shutdown Tap Settings While in Mode 4
- 2016-11025; Pressurizer Spray Line Temperature Differential Exceeded 300°F During RCS Heatup
- 2016-11034; Pressurizer Code Safety Leakage
- 2016-11040; NRC Resident Found Active BACC Leak FG4839 During OP-3013 Mode 3 Walkdown
- 2016-11051; Pressurizer Spray Line and Spray Nozzle Differential Temperature
- 2016-11068; Pressurizer Code Safety Leakage
- 2016-11138; Main Turbine Trip During Startup - Misposition Event
- 2016-11173; Feedwater Leak at FW802 Vent and FW780 Body
- DB-OP-02500; Turbine Trip; Revision 14
- DB-OP-02513; Pressurizer System Abnormal Operation; Revision 11
- DB-OP-03013; Containment Daily Inspection and 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 23
- DB-OP-06003; Pressurizer Operating Procedure; Revision 31
- DB-OP-06004; Quench Tank; Revision 11
- 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-06202; Turbine Operating Procedure; Revision 28
- DB-OP-06224; Main Feed Pump and Turbine; Revision 38
- DB-OP-06301; Generator and Exciter Operating Procedure; Revision 28
- DB-OP-06401; Integrated Control System Operating Procedure; Revision 25
- DB-OP-06402; Control Rod Drive Operating Procedure; Revision 28
- DB-OP-06900; Plant Heatup; Revision 64
- DB-OP-06901; Plant Startup; Revision 38
- DB-OP-06902; Power Operations; Revision 56

- DB-OP-06903; Plant Cooldown; Revision 48
- DB-OP-06904; Shutdown Operations; Revision 47
- DB-OP-06912; Approach to Criticality; Revision 18
- NG-DB-00117; Shutdown Defense in Depth Assessment; Revision 17
- NOP-OP-1005; Shutdown Defense in Depth; Revision 15
- NOP-OP-3502; FENOC Shutdown Chemistry Program; Revision 1
- DBBP-ESAF-1015; Industrial Safety Requirements for Containment Entry; Revsion 18
- NOBP-OP-0007; Conduct of Infrequently Performed Tests or Evolutions; Revision 5
- 1FOAC1 Restart Readiness Review Package; 9/11/2016
- Evolution Specific Reactivity Plan; Escalation to 100%FP After September 2016 Forced Outage; Revision 0
- Pressurizer Code Safety Forced Outage Shutdown Defense in Depth Report; Revisions 0 and 1

1R22 Surveillance Testing

- DB-MI-03211; Channel Functional Test of SFRCS Actuation Channel 1 Logic For Mode 1; Revision 19
- DB-MI-03245; Channel Functional Test and Device Calibration of SFRCS Steam Generator Level Inputs 83C-ISLSP9A6, A7, B8, and B9 to Actuation Channel 1; Revision 17
- DB-OP-06316; Diesel Generator Operating Procedure; Revision 59
- DB-PF-06705; Tank Level Calibration Curves; Revision 12
- DB-SC-03071; Emergency Diesel Generator 2 Monthly Test; Revision 35
- DB-SC-03272; Control Rod Exercising Test; Revision 5
- DB-SS-04150; Main Turbine Stop Valve Test; Revision 14
- DB-SS-04151; Main Turbine Control Valve Test; Revision 16
- DB-SS-04152; Main Turbine Combined Intermediate Valve Test; Revision 11
- 200615304; MI-03211 SFRCS Actuation Ch. 1 Logic Functional
- 200615305; MI-03245 SFRCS Actuation Ch. 1 Steam Generator Level Functional
- E-0018, Sheet 1; SFRCS Logic Diagram Logic Channels 1 & 3 Actuation Channel 1; Revision 7
- E-0018; Sheet 3; SFRCS Logic Diagram Miscellaneous Circuits; Revision 7
- M-0017A; Diesel Generators; Revision 19
- M-0017B; Diesel Generators Air Start; Revision 47
- M-0017C; Fuel Oil; Revision 30
- OS-0041A, Sheet 1; Emergency Diesel Generator Systems; Revision 33
- OS-0041A, Sheet 2; Emergency Diesel Generator Systems; Revision 32
- OS-0041B; Emergency Diesel Generator Air Start / Engine Air System; Revision 42
- OS-0041C; Emergency Diesel Generator Diesel Oil System; Revision 16

1EP2 Alert and Notification System Evaluation

- 2014-15857; Trending Alert and Notification System (Sirens); Dated October 17, 2014
- 2015-06485; Trending Alert and Notification System (Sirens); Dated May 6, 2015
- 2016-01979; Davis-Besse Siren 401 Indication of Rotate Failure; Dated February 10, 2016
- DBEP-028-07; Davis-Besse 2014 Annual Siren Preventive Maintenance Results; Dated December 31, 2014
- NOP-LP-5005-05; Davis-Besse 2015 Annual Siren Preventive Maintenance Results; Dated December 30, 2015
- NOP-LP-5005; FENOC Siren Testing and Maintenance; Dated October 1, 2015

- First Energy Prompt Notification System Design Report; Dated March 26, 2014
- Notification 600871077; 2014 Siren Maintenance

1EP3 Emergency Response Organization Staffing and Augmentation System

- 2015-09997; Self-Contained Breathing Apparatus Qualification Lapsed; Dated July 24, 2015
- 2015-14811; SAM Engineer ERO Position Not Filled During October 30th Unannounced Drive-In Drill; Dated October 30, 2015
- SN-SA-2015-0366; 2015 Unannounced Drive-In Drill; Dated December 17, 2015
- SN-SA-2016-0744; First Quarter 2016 Augmentation Drill; Dated February 23, 2016
- SA/BN-2016-0038; Second Quarter 2016 Unannounced Drill; Dated June 17, 2016
- Davis-Besse Nuclear Power Station Emergency Plan; Revision 27 and Revision 30
- EPIP RA-EP-02110; Emergency Notification; Revision 14
- Qualification Records for select ERO staff; Dated September 28, 2016
- On-shift staff schedules for Chemistry, I&C, and Radiation Protection Departments
- Davis-Besse Nuclear Power Station Emergency Plan Telephone Directory; Dated June 14, 2016

1EP5 Maintenance of Emergency Preparedness

- 201507143; Aggregate Review of Response to the Automated Notification System (CANS) During May 9, 2015, Unusual Event; Dated May 18, 2015
- 2015-06807; Periodic Update Notification Form Missing Page 4 (Dose Assessment) During Unusual Event on May 9, 2015; Dated May 12, 2015
- 2015-06757; Communication Issues for Unusual Event; Dated May 11, 2015
- 2015-06802; Missed Pages During the May 9th Unusual Event; Dated May 12, 2015
- 2015-07021; Deactivation Report (form DBEP062) Was Not Completed During May 9th Unusual Event; Dated May 15, 2015
- 2015-14968; Emergency Response Organization Call In; Dated November 2, 2015
- 2015-14425; All Clear Message at the Termination of the October 20 Unusual Event; Dated October 22, 2015
- MS-C-14-11-24; Fleet Oversight Audit Report; Dated December 5, 2014
- MS-C-15-11-24; Fleet Oversight Audit Report; Dated December 4, 2015
- SA/BN-2016-0051; Pre-NRC Biennial Emergency Preparedness Inspection Assessment; Dated July 19, 2016
- SN-SA-2015-0154; May 9, 2015 Unusual Event-Explosion in the Protected Area Due to Steam Leak; Dated September 2, 2015
- SN-SA-2015-0347; October 20, 2015, Unusual Event – Security Condition in the OCA; Dated December 17, 2015
- Control Room Unit Logs dated May 9, 2015 – May 10, 2015
- NOBP-TR-1122; Operating Crew Performance Critique Form; Dated May 9, 2015

1EP6 Drill Evaluation

- 2016-09690; EP Drill – CAS/SAS Environment
- 2016-09699; ERO Drill – Non Licensed Operator (NLO) Radios Did Not Work in the OSC
- 2016-09702; ERO Drill – August 2016 – Inconsistent Data Being Supplied by the Plant Status Board in the OSC
- 2016-09729; EP Drill 8/9/2016 – Emergency Operating Facility Not Activated Timely
- 2016-09730; EP Drill 8/9/2016 – Emergency Director Turnover Not Completed in a Timely Manner

- 2016-09761; EP Drill – Missed Emergency Classification and Drill and Exercise Performance (DEP) Opportunity
- 2016-09764; EP Drill – Control Room Periodic Update Deficiencies
- 2016-09766; EP Drill – Radio Communication Challenges
- 2016-09769; EP Drill – 10CFR50.54(x) Declaration by the EAPM
- 2016-09789; EP Drill – RMT Radio in the EOF Did Not Function
- 2016-09799; EP Drill – Green Bridge Phone Line Nonfunctional at Start of 08/09/2016 Integrated Drill
- 2016-09801; OSC Procedure Performance Issue During the 08/09/2016 Integrated Drill
- 2016-09861; EP Drill – 08/09/2016 Joint Information Center Emergency Response Facility Critique Summary
- 2016-09902; EP Drill – Missed Drill Objective for ERO Activation
- RA-EP-00200; Emergency Plan Drill and Exercise Program; Revision 12
- RA-EP-00520; Emergency Response Organization; Revision 11
- RA-EP-01500; Emergency Classification; Revision 15
- RA-EP-01600; Unusual Event; Revision 8
- RA-EP-01700; Alert; Revision 8
- RA-EP-01800; Site Area Emergency; Revision 7
- RA-EP-02010; Emergency Management; Revision 18
- RA-EP-02110; Emergency Notification; Revision 14
- RA-EP-02310; Technical Support Center Activation and Response; Revision 13
- DBRM-EMER-1500A; Davis-Besse Emergency Action Level Basis Document; Revision 7
- DBRM-EMER-1500B; Hot EAL Wall Board, Revision 1
- DBRM-EMER-1500B; Cold EAL Wall Board, Revision 1
- DBRM-EMER-1500C; Davis-Besse Emergency Action Level Reference Manual; Revision 0
- Davis-Besse Emergency Preparedness August 9, 2016 Integrated Drill Manual

<u>2RS3 In-Plant Airborne Radioactivity Control and Mitigation</u>

- 2015-09997; Self Contained Breathing Apparatus Qualification Lapsed; July 24, 2015
- 2016-04717; RWP Not in Alignment with Air Sampling Procedure; Dated April 6, 2016
- AR 02557223; Accumulated Dose Alarm; Dated September 18, 2015
- DB-HP-01301; Use of Respiratory Protection; Revision 9
- DB-HP-01308; Respiratory Protection Equipment and Maintenance; Revision 16
- DB-HP-06000; Operation of Air Compressors; Revision 9
- NOP-OP-4206; Bioassay Program; Revision 3
- NOP-OP-4301; Respiratory Protection Program; Revision 7
- NOP-OP-4302; Issuing Respiratory Protection; Revision 4
- NOP-OP-4310; FireHawk M7 Self Contained Breathing Apparatus; Revision 7
- NOP-OP-4330; Use of Non-Face Sealing Respirators; Revision 1
- NOP-OP-4331; Use of Powered Air-Purifying Respirators; Revision 1
- NOP-OP-4703; Determination of Alpha Monitoring Levels; Revision 3
- SA/BN-2016-0088; Respiratory Protection Program Benchmarking of Top Performing Station; Dated July 11, 2016
- SN-SA-2015-0238; Fleet Respiratory Protection Program Review; Dated December 4, 2015
- SN-SA-2016-0764; Airborne Radioactivity Control & Mitigation, Occupational Dose Assessment, Performance Indicators; Dated August 31, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer SN: 26937; Dated April 22, 2015
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer SN: 26937; Dated July 21, 2015

- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer Air Compressor; Dated October 23, 2015
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer RSP; Dated January 26, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer RSP; Dated June 29, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Bauer Air Compressor; Dated August 10, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated April 22, 2015
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated July 21, 2015
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated October 23, 2015
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated January 26, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated June 29, 2016
- Air/Gas Quality Report & Certificate; Davis-Besse Nuclear Plant; Station Air Compressor; Dated August 10, 2016
- MSA Regulator Service Report; Regulator Serial Number; LAA271081; Dated August 11, 2015
- MSA Regulator Service Report; Regulator Serial Number; LAA271081; Dated August 16, 2016
- MSA Regulator Service Report; Regulator Serial Number; LAA271097; Dated August 11, 2015
- MSA Regulator Service Report; Regulator Serial Number; LAA271097; Dated August 16, 2016
- MSA Regulator Service Report; Regulator Serial Number; LAA333250; Dated August 12, 2015
- MSA Regulator Service Report; Regulator Serial Number; LAA333250; Dated August 16, 2016

2RS4 Occupational Dose Assessment

- 2016-08735; Level 2 Personnel Contamination Event; Dated July 13, 2016
- 2016-09732; SN-SA-2016-0764 Declaration of Pregnancy; Dated July 25, 2016
- 2016-11549; Declared Pregnant Worker Dose Allowance; Dated September 28, 2016
- NOP-OP-4201; Routine External Exposure Monitoring; Revision 2
- NOP-OP-4202; Declared Pregnant Workers; Revision 0
- NOP-OP-4205; Dose Assessment; Revision 7
- NOP-OP-4206; Bioassay Program; Revision 3
- NOP-OP-4204; Special External Exposure Monitoring; Revision 8
- NOP-OP-4503; Personnel Contamination Monitoring; Revision 11
- DB-HP-01320; Operation of Whole Body Counters; Revision 11
- DB-HP-01322; Body Counter Calibration and Performance Testing; Revision 6
- DB-HP-01436; DMC 90/100/2000 Calibration and Use; Revision 3
- RP-92-031; Neutron Dose and Energy Spectral Measurements inside DB Reactor Containment Building; December 1992
- SN-SA 2016-0764; Airborne Radioactivity Control & Mitigation, Occupational Dose Assessment, Performance Indicators; Dated August 31, 2016
- On-Site Assessment Report; Mirion Technologies (GDS), Inc.; Dated November 2, 2015
- Notification 601056275; Declared Pregnant Workers; Dated September 28, 2016

4OA1 Performance Indicator Verification

- 2015-08041; Acceptance Criteria 5.7 and 5.8 Failed During Performance of DB-PF-04736
- 2015-13269; Pipe Plug Steam Leak on ICS38D, AFP 2 Trip Throttle Valve
- 2015-16479; Thickness Measurements Below Work Order Acceptance Criteria on Service Water Piping
- 2016-02481; System Monitoring – Service Water Pump No. 3 Motor Trend Does Not Support Refurbishment Time
- 2016-03566; Failed Acceptance Criteria for DB-PF-04736, ECCS Room Cooler Monitoring Test
- 2016-05172; Less than Minimum Wall Thickness Measured on CCW #1 Shell per Order 200486506
- 2016-05353; Eddy Current Results – CCW Heat Exchanger E22-1
- 2016-05799; Service Water Pump 3 Strainer Did Not Appear to Start on High Pressure
- 2016-06300; AFP 2 Governor Does Not Function Correctly on First Attempt
- 2016-10633; Gas Void Detected Upstream of DH158
- NOBP-LP-4012-48; MSPI Heat Removal System (AFW); Completed Forms for July 2015 through June 2016
- NOBP-LP-4012-49; MSPI Residual Heat Removal System (LPI); Completed Forms for July 2015 through June 2016
- NOBP-LP-4012-50; MSPI Support Cooling System, Component Cooling Water; Completed Forms for July 2015 through June 2016
- NOBP-LP-4012-51; MSPI Support Cooling System, Service Water; Completed Forms for July 2015 through June 2016
- NOBP-LP-4012; NRC Performance Indicators; Revision 5
- Select Operator Logs covering the period of July 2015 through June 2016
- Davis-Besse Nuclear Power Station Reactor Oversight Program Mitigating System Performance Index Basis Document; Revision 4
- MRPM; Maintenance Rule Program Manual; Revision 35
- ANS Siren Data; Second quarter of 2015 through second quarter of 2016
- DEP Opportunity Data; Second quarter of 2015 through second quarter of 2016
- ERO Personnel Participation Data; Second quarter of 2015 through second quarter of 2016

4OA2 Problem Identification and Resolution

- NOP-ER-1001; Continuous Equipment Performance Improvement; Revision 5
- NOP-LP-2001; Corrective Action Program; Revision 38
- NOP-OP-1009; Operability Determinations and Functionality Assessments; Revision 6
- 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 20
- NOBP-LP-2011; FENOC Cause Analysis; Revision 19
- NOBP-OP-1009; Prompt Operability Determination and Functionality Assessment Preparation Guide; Revision 6
- 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

4OA3 Followup of Events and Notices of Enforcement Discretion

- 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 "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
- 2016-06563; ARTS Test Trip Bypass Left in Bypass for SFRCS and Main Turbine During Startup – Plant Status Control Misposition Event
- 2016-08385; Evidence of Water in O'Brien Box for DB-LT1525A
- 2016-08402; SFAS Channel 2 +15V Power Supply Failure
- 2016-08415; LCO 3.3.5 Parameter 5 BWST Level Low-Low Operability
- 2016-08419; Performance Review of LCO 3.3.5 Application During LT1525A Maintenance
- 2016-08699; PA-DB-16-03: Crew Briefing Performance Shortfall
- 2016-08700; PA-DB-16-03: Delayed Request for Prompt Operability Determination
- 2016-08765; Restoration of SFAS Channel 1 (LT-1525A) - Assessment of Organizational Response to Extended Work Window
- 2016-10723; SG/RX Demand Response Following Trip
- 2016-10724; RFR Response Following Reactor Trip
- 2016-10725; Reactor Trip Due to Water Intrusion into Automatic Voltage Regulator Cabinet
- 2016-10726; Broken Roof Vent on Northwest Corner of Turbine Building Allowed Rain Intrusion into Exciter Cabinet on 603 Turbine Building
- 2016-10733; Post Transient Assessment of ARTS
- 2016-10741; MS101, Main Steam Isolation Valve No. 1, Exceeds Transient Assessment Program Specified Time Requirement
- 2016-10745; Pressurizer Level During Reactor Trip
- 2016-10748; Unplanned LCO 3.8.1 Entry Due to Low 345 KV Bus Voltages
- 2016-10755; As Found Condition of 40 Device Main Generator Loss of Field Relay
- 2016-10760; Q233, CRD 5 Vdc Power Supply B Fault, Indicates Not Normal
- 2016-11004; E1 & F1 on Transformers with Shutdown Tap Settings While in Mode 4
- DB-OP-02000; RPS, SFAS, SFRCS Trip, or Steam Generator Tube Rupture; Revision 29
- DB-OP-02520; Load Rejection; Revision 7
- DB-OP-02526; Primary to Secondary Heat Transfer Upset; Revision 4
- DB-OP-02546; Degraded Grid; Revision 4
- DB-OP-06003; Pressurizer Operating Procedure; Revision 31
- DB-OP-06004; Quench Tank; Revision 11
- DB-OP-06202; Turbine Operating Procedure; Revision 28
- DB-OP-06301; Generator and Exciter Operating Procedure; Revision 28
- DB-OP-06401; Integrated Control System Operating Procedure; Revision 25
- DB-OP-06402; Control Rod Drive Operating Procedure; Revision 28

- NA-QC-00356, Transient Assessment Program; Revision 5
- Thursday, June 30, 2016, Day Shift
- Friday, July 1, 2016, Night Shift
- Friday, July 1, 2016, Day Shift

4OA5 Other Activities

- 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
- NOP-OP-1015; Event Notifications; Revision 2
- NOP-OP-2012; Groundwater Monitoring; Revision 9
- NOP-OP-4705; Response to Contaminated Spills/Leaks; Revision 8
- NOBP-OP-1015; Event Notifications; Revision 3
- Groundwater Monitoring Well Data Covering the Period of January 2014 through September 2016

4OA7 Licensee-Identified Violations

- 2016-06563; ARTS Test Trip Bypass Left in Bypass for SFRCS and Main Turbine During Startup – Plant Status Control Misposition Event

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Reasonably-Achievable
ANS	Alert and Notification System
ARTS	Anticipatory Reactor Trip System
ASME	American Society of Mechanical Engineers
BWST	Borated Water Storage Tank
CAP	Corrective Action Program
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DEP	Drill and Exercise Performance
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
ECSA	Electrical Conductor Seal Assemblies
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EP	Emergency Preparedness
ERO	Emergency Response Organization
HLLMS	Hot Leg Level Monitoring System
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
LER	Licensee Event Report
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NNI	Non-Nuclear Instrumentation
NRC	U.S. Nuclear Regulatory Commission
PAMS	Post Accident Monitoring System
PARS	Publicly Available Records System
PI	Performance Indicator
PMT	Post-Maintenance Testing
RCS	Reactor Coolant System
RFO	Refueling Outage
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
SAPHIRE	Systems Analysis Programs for Hands-On Integrated Reliability Evaluations
SCBA	Self-Contained Breathing Apparatus
SFAS	Safety Features Actuation System
SFRCS	Steam and Feedwater Rupture Control System
SG	Steam Generator
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SSC	Systems, Structures, and Components

TS	Technical Specification
USAR	Updated Safety Analysis Report
Vac	Volts Alternating Current
WO	Work Order

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Sincerely,

/RA Kenneth Riemer Acting for/

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

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