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

February 11, 2016

EA-16-024

Mr. Kevin Davison
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: REVISED PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2–NRC INTEGRATED INSPECTION REPORT 05000282/2015004; 05000306/2015004

Dear Mr. Davison:

On December 31, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on January 7, 2016, with you, and other members of your staff. Please note that this report is a revision to the one issued on February 8, 2016, which inadvertently did not reference the appropriate Enforcement Action number associated with the licensee identified violation documented in Section 4OA7 of the report.

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 violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. Additionally, a licensee-identified violation for which enforcement discretion was granted is listed in Section 4OA7 of this report.

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

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 Inspector at the Prairie Island Nuclear Generating Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (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
Branch 2
Division of Reactor Projects

Docket Nos. 50–282; 50–306; 72–010
License Nos. DPR–42; DPR–60;
SNM–2506

Enclosure:
IR 05000282/2015004; 05000306/2015004

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50–282; 50–306; 72–010
License Nos:	DPR–42; DPR–60; SNM–2506
Report No:	05000282/2015004; 05000306/2015004
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	Welch, MN
Dates:	October 1 through December 31, 2015
Inspectors:	L. Haeg, Senior Resident Inspector P. LaFlamme, Resident Inspector K. Barclay, Resident Inspector G. Edwards, Health Physicist J. Cassidy, Senior Health Physicist S. Bell, Health Physicist A. Shaikh, Reactor Inspector M. Garza, Emergency Preparedness Inspector
Approved by:	K. Riemer, Chief Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY

Inspection Report 05000282/2015004, 05000306/2015004; 10/01/2015–12/31/2015; Prairie Island Nuclear Generating Plant, Units 1 and 2; Inservice Inspection Activities; Radiation Monitoring Instrumentation.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings were identified by the inspectors. The findings involved Non-Cited Violations (NCVs) of U.S. Nuclear Regulatory Commission (NRC) requirements. The significance of inspection findings was 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 were determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated 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," dated February 2014.

NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance (Green), and an associated NCV of Title 10 of the *Code of Federal Regulations* (CFR) Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to incorporate the American National Standards Institute (ANSI) N14.6–1978, Section 5.3.1 required testing frequency for the reactor vessel head and reactor vessel internals lifting devices into the controlling preventive maintenance procedure. Compliance with the ANSI standard was documented in the Safety Evaluation Report (SER) for the licensee's control of heavy loads. The licensee documented the issue in the corrective action program (CAP) as CAP 01497779 and performed testing on the reactor vessel head and internals lifting devices during the outage.

  The inspectors determined the licensee's failure to comply with ANSI N14.6–1978, Section 5.3.1, for the continued use testing of special lifting devices was a performance deficiency (PD). The PD was determined to be more-than-minor and a finding because the PD was associated with the Initiating Events Cornerstone attribute of design control, and adversely affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, compliance with ANSI N14.6–1978, Section 5.3.1 ensured safe load handling of heavy loads over the reactor core, and/or over safety-related systems through established testing for the continued functionality of the special lifting devices. The failure to perform the required frequency of testing on special lifting devices could increase the likelihood of a load drop and could decrease the load handling reliability of the lifting device if the device were returned to service with potentially unacceptable flaws. The inspectors determined the finding could be evaluated using the Significance Determination Process in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown conditions, the inspectors used Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process."

The inspectors determined that none of the conditions constituting a loss of control were met, as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRs [Pressurized Water Reactors] and BWRs [Boiling Water Reactors]," for this finding, and neither a Phase II nor a Phase III analysis was required. Therefore, the inspectors determined that this finding was of very low safety significance (Green). The inspectors determined that this finding has a cross-cutting aspect in the area of Human Performance, Resources, for the licensee's failure to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety. Specifically, the licensee staff evaluated NRC Information Notice (IN) 2014–12, "Crane and Heavy Lift Issues Identified during NRC Inspections," in corrective action program (CAP) document 01457469. However, in CAP 01457469, the licensee concluded that issues identified in IN 2014–12 related to other licensees not performing testing in accordance with ANSI N14.6 requirements were not applicable to the licensee at the Prairie Island Nuclear Generating Plant. Therefore, the inspectors determined that there was a recent missed opportunity for the licensee to have reasonably identified that the current preventive maintenance procedure for special lifting devices was not in accordance with the ANSI N14.6–1978 requirements, as referenced in the SER. [H.1] (Section 1R08)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of TS 5.5.1.a for the failure to comply with the Offsite Dose Calculation Manual (ODCM) for not using calibration sources that were traceable to the National Institute of Standards and Technology (NIST) or equivalent during the calibration of station effluent monitors. The licensee entered the issues into the CAP as CAPs 01490581 and 01500149. Immediate corrective actions included the re-calibration of impacted monitors and the performance of an extent of condition evaluation for other radiation monitor calibrations.

  The PD was determined to be of more than minor safety significance in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the plant facilities/equipment and instrumentation attribute of Public Radiation Safety and it adversely impacted the cornerstone objective of ensuring adequate protection of public health and safety due to failure to properly calibrate certain effluent monitors. Subsequent calibrations of the monitors determined that the monitor efficiency was previously overstated. The inspectors also reviewed IMC 0612, Appendix E, "Examples of Minor Issues," dated August 11, 2009, but did not identify any similar examples. The finding was assessed using IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," dated, February 12, 2008, and determined to be of very low safety significance (Green), because it was associated with the effluent release program but was not a failure to implement an effluent program, public dose did not exceed Appendix I criteria, and the limits in Title 10 CFR 20.1301(e) were not exceeded. A cross-cutting aspect was not assigned as this issue occurred numerous years ago. The station has since performed monitor calibrations with radioactive sources with known quality. (Section 2RS5)

Licensee-Identified Violations

- Violations of very low safety or security significance or Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or

planned by the licensee have been entered into the licensee's corrective action program (CAP). These violations and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at full power for the entirety of the inspection period, with the exception of brief down-power maneuvers to accomplish planned surveillance testing activities.

Unit 2 began the inspection period at full power. On October 17, 2015, operations personnel shut down the Unit 2 reactor to perform Refueling Outage (RFO) 2R29. Major activities completed during the RFO included replacement of the main generator, main transformer, containment fan coil unit (FCU) components, 21 reactor coolant pump (RCP) seal, and also performed steam generator (SG) tube integrity testing. Operations personnel returned the Unit 2 reactor to operation on December 3, 2015. The main generator was synchronized with the electrical grid on December 5, 2015.

On December 17, 2015, the Unit 2 main turbine automatically tripped due to a detected electrical fault within the main generator. This resulted in a reactor trip from 100 percent power. The licensee began Forced Outage 2F2901HS to address the main generator issue and remained shutdown in Mode 3 at the end of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Documents reviewed were listed in the Attachment to this report. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- D5 and D6 emergency diesel generators (EDGs) and cooling water (CL) systems.

This inspection constituted one winter seasonal readiness preparations 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 Walkdowns

a. Inspection Scope

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

- 12 motor driven auxiliary feed water (AFW) system;
- Unit 2 train A component cooling (CC) system; and
- Unit 1 caustic addition system.

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, the USAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These inspections constituted three quarterly partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Zone 42, Unit 2 reactor building, El. 697' 6";
- Fire Zone 54, Unit 2 reactor building Unit 2, El. 755';
- Fire Zone 88, Unit 2 rod control room, El. 735'; and
- Fire Zone 87, Unit 1 rod control room, El, 735'.

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 (IPEEE) 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. Using the documents listed in the Attachment to this report, 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. Documents reviewed are listed in the Attachment to this report.

These inspections constituted four quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the USAR, engineering calculations, and abnormal operating procedures to identify licensee's commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee's 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 the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following plant areas to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- Unit 1 and 2 AFW system rooms.

Documents reviewed are listed in the Attachment to this report. This inspection constituted one internal flooding sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the Unit 2 train B CC heat exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

This inspection constituted one annual heat sink performance sample as defined in IP 71111.07–05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From October 19, 2015, through November 25, 2015, the inspectors conducted a review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the Unit 2 reactor coolant system (RCS), emergency feedwater systems, risk-significant piping and components, and containment systems.

The reviews described in Sections 1R08.1 through 1R08.5 below constituted one inservice inspection activities inspection sample as defined in IP 71111.08–05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors reviewed records of the following Non-Destructive Examinations (NDE) required by the American Society of Mechanical Engineers (ASME) Section XI Code, and/or Title 10 of the *Code of Federal Regulations* (CFR) Part 50.55a to evaluate compliance with the ASME Code, Section XI and V requirements, and if any indications and defects were detected to determine whether these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement:

- Magnetic Particle Examination of Reactor Vessel Internals Lift Fixture Welds;
- Magnetic Particle Examination of Reactor Vessel Head Lift Rig Welds;
- Ultrasonic Examination of Safety Injection Elbow-to-Pipe Weld W–6;
- Ultrasonic Examination of RH Pipe-to-Elbow Weld W-5;
- Eddy Current Testing (ET) of SG Tubes;
- Visual Examination (VT-3) of AFWH–64 Sway Strut/Clamp; and
- Visual Examination (VT-3) of AFWH–79 Sway Strut/Clamp.

The licensee had not identified any recordable indications during non-destructive surface and/or volumetric examinations performed since the last RFO. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed records of the following risk-significant pressure boundary ASME Code Section XI Class 2 welds fabricated since the beginning of the last refuelling outage to determine if the licensee: followed the welding procedure; applied appropriate weld filler material; and implemented the applicable Section XI or Construction Code NDEs and acceptance criteria. Additionally, the inspectors reviewed the following welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of Construction Code and the ASME Code Section XI:

- Class 1–WO 00406128; Remove and Replace Valve 2RC–7–2, Loop A to Pressurizer CV–31228 BY–PASS.

b. Findings

Failure to Meet American National Standards Institute N14.6, Section 5.3.1 Requirements

Introduction: The inspectors identified a finding of very low safety significance (Green), and an associated NCV of Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to incorporate American National Standards Institute (ANSI) N14.6–1978, Section 5.3.1 required testing frequency for the reactor vessel head and reactor vessel internals lifting devices into the controlling preventive maintenance procedure. Compliance with the ANSI standard was documented in the Safety Evaluation Report (SER) for the licensee’s control of heavy loads.

Description: The reactor vessel head and reactor vessel internals lifting devices are classified as safety-related components at Prairie Island. The SER for the “Control of Heavy Loads Phase 1 at Prairie Island Nuclear Generating Plant, Units 1 and 2,” dated June 6, 1983, classified the reactor vessel head and reactor vessel internals lifting devices as special lifting devices and provided documentation on how compliance with ANSI N14.6–1978, “Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials,” was met. Specifically, Section 2.1.5 of the SER stated, in part, “the Licensee has indicated that the reactor vessel head and reactor vessel internals special lifting devices are inspected prior to use in accordance with the requirements of ANSI N14.6–1978. Such inspection will include NDE of welds and other critical components.” ANSI N14.6–1978, Section 5.3.1 stated, in part, “each special lifting device shall be subjected annually (period not to exceed 14 months) to either of the following:

- A load test equal to 150 percent of the maximum load to which the device is to be subjected; and/or
- Dimensional testing, visual examination, and NDE of major load-carrying welds and critical areas. If the device has not been used for a period exceeding one year, this testing shall not be required. However, in this event, the test shall be applied before returning the device to service."

The licensee did not perform load testing on these special lifting devices. Further, the licensee had last performed NDE on these special lifting devices on June 3, 2005, despite having used them during several outages since.

The inspectors reviewed the licensee's preventive maintenance (PM) procedure PM 3560–52, "Reactor Head Lifting Rig Spreader & Connection Legs Assembly Inspection," Revision 13, and identified that the procedure specified conducting NDE once during each 10-year interval. The inspectors identified that this site procedure requirement was contrary to the plant licensing basis as documented in the SER and ANSI N14.6–1978. The inspectors questioned the licensee's basis for decreasing the required frequency of NDE on the special lifting devices. Specifically, the inspectors were concerned that failure to perform NDE on the special lifting devices' load-carrying welds at the ANSI N14.6 required frequency potentially challenged continued functionality of the devices.

The licensee documented the inspector's concerns in CAP 01497779. As part of its immediate corrective actions, the licensee performed NDE on the reactor vessel head and reactor vessel internals lifting devices prior to lifting the head and internals during the outage. As part of additional corrective actions, the licensee intended to revise procedure PM 3560–52 to correctly translate ANSI N14.6–1978, Section 5.3.1 testing frequency requirements.

Analysis: The inspectors determined the licensee's failure to comply with ANSI N14.6–1978, Section 5.3.1 for the continued use testing of special lifting devices was a performance deficiency (PD). The PD was determined to be more-than-minor, and a finding, because the PD was associated with the Initiating Events Cornerstone attribute of design control, and adversely affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations. Specifically, compliance with ANSI N14.6–1978, Section 5.3.1, ensured safe load handling of heavy loads over the reactor core, and/or over safety-related systems through established testing for the continued functionality of the special lifting devices. The failure to perform the required frequency of testing on special lifting devices could increase the likelihood of a load drop and could decrease the load handling reliability of the lifting device if the device were returned to service with potentially unacceptable flaws.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that none of the conditions constituting a loss of control were met, as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRs and BWRs," for this

finding, and neither a Phase II nor a Phase III analysis was required. Therefore, the inspectors determined that this finding was of very low safety significance (Green).

The inspectors determined that this finding has a cross-cutting aspect in the area of Human Performance, Resources, for the licensee's failure to ensure that personnel, equipment, procedures, and other resources are available and adequate to support nuclear safety. Specifically, the licensee's staff evaluated NRC Information Notice (IN) 2014–12, "Crane and Heavy Lift Issues Identified during NRC Inspections," in CAP 01457469. However, within CAP 01457469, the licensee concluded that issues identified in IN 2014–12 related to other licensees not performing testing in accordance with ANSI N14.6 requirements were not applicable to the licensee at the Prairie Island Nuclear Generating Plant. Therefore, the inspectors determined that there was a recent missed opportunity for the licensee to have reasonably identified that the current preventive maintenance procedure for special lifting devices (PM 3560–52) was not in accordance with the ANSI N14.6–1978 requirements as referenced in the SER. [H.1]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that "measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR Part 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."

Contrary to the above, since June 3, 2005, the licensee failed to correctly translate its licensing design basis standard for the control of heavy loads into its PM procedure used for controlling testing of special lifting devices. Specifically, the licensee failed to translate the ANSI N14.6–1978 (as required by SER, dated June 6, 1983) testing frequency requirements for special lifting devices into its controlling PM procedure for special lifting devices.

The licensee subsequently took immediate corrective actions, which included NDE of the reactor vessel head and reactor vessel internals lifting devices' welds prior to lifting. Because this violation was of very low safety significance, and it was entered into the licensee's CAP as CAP 01497779, it is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy
(NCV 05000306/2015004–01, Failure to Meet ANSI N14.6 Section 5.3.1 Requirements).

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 reactor vessel head, no examinations (visual or non-visual) were required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D) requirements. Therefore, no examination was conducted by the licensee and no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors independently walked down the Unit 2 reactor coolant system loop piping, including the reactor coolant pumps, pressurizer, and emergency core cooling systems within containment to identify boric acid leakage. The inspectors then reviewed the walkdown performed by the licensee to ensure that components with boric acid deposits were identified and entered into the CAP. The inspectors observed these examinations to determine whether the licensee focused on locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of components with boric acid deposits to determine if the affected components were documented and properly evaluated in the corrective action system. Specifically, the inspectors evaluated the following CAP documents to determine if degraded components met the component Construction Code and/or the ASME Section XI Code:

- CAP 01450417; Boric Acid Corrosion Control (BACC) Evaluation for ISI Indication on CV–31325;
- CAP 01450480; BACC Evaluation for ISI Indication on RC–19–1;
- CAP 01450476; BACC Evaluation for ISI Indication on 135–011; and
- CAP 01427328; BACC Evaluation for Leak Identified in Unit 2 Containment 21 Vault.

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

- CAP 01412727; Leak From Capped Drain Downstream of 2RC–8–19;
- CAP 01431405; Boric Acid Leak was Found on 2RC–8–31;
- CAP 01469111; Boric Acid Packing Leak on 2SI–35–6;
- CAP 01492989; Boric Acid Built Up Below 21 Safety Injection Pump; and
- CAP 01445383; 22 Safety Injection Pump IB/OB Mechanical Seal Leakage.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition of ET data, interviewed ET data personnel, and reviewed documentation related to the SG ISI program to determine if:

- in-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) TR–107620, “Steam Generator In-Situ Pressure Test Guidelines,” and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;

- the numbers and sizes of SG tube flaws/degradation identified were bound by the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TSs and EPRI 1003138, "Pressurized Water Reactor SG Examination Guidelines;"
- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- the licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee's primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons-per-day or the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for ET Examination" of EPRI 1003138, "Pressurized Water Reactor SG Examination Guidelines;"
- the licensee performed secondary side SG inspections for location and removal of foreign materials; and
- the licensee implemented repairs for SG tubes damaged by foreign material.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no NRC review was completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

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

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

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On October 6, 2015, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. 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. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–05 and satisfied the inspection program requirement for the resident inspectors to observe a portion of an in-progress annual requalification operating test during a training cycle in which it was not observed by the NRC during the biennial portion of this IP.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Observation during Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On October 16, 2015, the inspectors observed the Unit 2 shutdown activities in preparation for the Unit 2 RFO. These were activities that required heightened

awareness or were related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.3 Resident Inspector Quarterly Observation during Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On December 3, 2015, the inspectors observed the Unit 2 control room startup activities. These were activities that required heightened awareness or were related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

The performance in these areas was compared to pre-established operator action expectations, procedural compliance and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk 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 the following:

- Unit 1 CC system; and
- Title 10 CFR 50.65(a)(3) periodic evaluation.

The inspectors reviewed events such as where ineffective equipment maintenance had 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 structures, systems, and components (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 that maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two quarterly maintenance effectiveness samples 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:

- loss of Unit 1 & 2 emergency response computer system (ERCS) on November 17 & 18, 2015;
- failure of a re-heat drain tank valve resulting in a thermal power increase on November 19, 2015; and
- emergent examination of reactor special lift devices and resulting extended duration at reduced inventory.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. 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.

Documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted three samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issue:

- CAP 01500184, Unit 1 AFW Recirculation Line Operability Evaluation.

The inspectors selected this potential operability issue based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluation to ensure that TS operability was properly justified and the subject components or systems 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 evaluation to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluation. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluation. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted one sample as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- Unit 2 turbine building crane load capacity uprate.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the USAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected/impacted systems. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one permanent plant modification sample as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- cooling water supply to 121 safeguards traveling water screen solenoid valve replacement;
- 21 RCP testing following seal replacement; and
- Unit 2 emergency core cooling system (ECCS) venting following outage maintenance activities.

These activities were selected based upon the structure, system, or component'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 post-maintenance tests 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. Documents reviewed are listed in the Attachment to this report.

These inspections constituted three post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 2 RFO conducted October 17, 2015, through December 5, 2015, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below:

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

- reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- controls over activities that could affect reactivity;
- maintenance of secondary containment as required by TS;
- licensee fatigue management, as required by 10 CFR 26, Subpart I;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- licensee identification and resolution of problems related to RFO activities.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RFO sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

.2 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage that began on December 17, 2015, and continued through the remainder of the inspection period. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed portions of the reactor trip and associated action taken in response the Unit 2 main generator trip, which caused an automatic turbine and subsequent reactor trip. Additionally, the inspectors observed and reviewed outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, and identification and resolution of problems associated with the outage. As of December 31, 2015, the cause of the main generator trip and resultant automatic turbine and reactor trip was still under investigation. Because the unscheduled outage was ongoing at the end of the inspection period, this inspection did not constitute a complete other outage activities 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 test results for the following 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:

- SP 2083A, "Unit 2 Integrated SI [Safety Injection] Test with a Simulated Loss of Offsite Power Train A," Revision 4 (inservice testing).

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;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- 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;
- test data and results were accurate, complete, within limits, and valid;
- 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 code, and reference values were consistent with the system design basis;
- where applicable, test results that did not meet acceptable 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, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and

- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one inservice test sample as defined in IP 71111.22, Sections–02 and–05.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

.1 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Regional inspectors performed an in-office review of the latest revisions to the Emergency Plan, Emergency Action Levels (EAL), and EAL Bases document to determine if these changes decreased the effectiveness of the Emergency Plan. The inspectors also performed a review of the licensee's 10 CFR Part 50.54(q) change process, and Emergency Plan change documentation to ensure proper implementation for maintaining Emergency Plan integrity.

The NRC review was not documented in an SER and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment to this report.

This inspection constituted one EAL and Emergency Plan change sample as defined in Inspection Procedure 71114.04–05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Training Observation

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on October 6, 2015, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors

reviewed the scenario package and other documents listed in the Attachment to this report.

This inspection constituted one training evolution with emergency preparedness drill aspects sample as defined in IP 71114.06–06.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

These inspection activities supplement those documented in IR 05000282/2015002; 05000306/2015002, and constituted one complete radiological hazard assessment and exposure controls sample as defined in IP 71124.01–05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators (PIs) for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that resulted in significant new radiological hazards for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard(s).

The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard(s).

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- 2R29–resistance temperature detector (RTD) replacement project;
- 10-year ISI/corrosion inspection–2R29;
- scaffold standard work–U2 outage; and
- primary steam generator activities–U2 outage.

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranic and/or other hard-to-detect radioactive materials (this evaluation may have included licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee had established a means to inform workers of changes that could have significantly impacted their occupational dose; and
- severe radiation field dose gradients that could have resulted in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers that held non-exempt, licensed radioactive materials that could have caused unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, "Labeling Containers," or met the requirements of 10 CFR 20.1905(g), "Exemptions To Labeling Requirements".

The inspectors reviewed the following radiation work permits (RWPs) used to access high-radiation areas and evaluated the specified work control instructions or control barriers:

- RWP 152500; 2R29–RTD Replacement Project;

- RWP 155021, 10-Year ISI/Corrosion Inspection–2R29;
- RWP 152055, Scaffold Standard Work–U2 Outage; and
- RWP 152300, Primary Steam Generator Activities.

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policies. The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issues were included in the CAP and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitored potentially contaminated material leaving radiological controlled areas and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicated the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee had established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high-noise areas as high-radiation area monitoring devices.

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

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

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures:

- RWP 152500; 2R29–RTD Replacement Project;
- RWP 155021; 10-Year ISI/Corrosion Inspection–2R29;
- RWP 152055; Scaffold Standard Work–U2 Outage; and
- RWP 152300; Primary Steam Generator Activities.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, and entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (i.e., nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected high-radiation areas and very-high radiation areas to verify conformance with the occupational PI.

b. Findings

No findings were identified.

.6 Risk-Significant High-Radiation Area and Very-High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed the controls in place for special areas that had the potential to become very-high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations required communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed performance of radiation workers with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological-related CAPs since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions that were planned or that were taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological-related CAPs since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors

evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and whether they were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involved radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience at the facility.

b. Findings

No findings were identified.

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

These inspection activities supplement those documented in IR 05000282/2014002; 05000306/2014002, and constitute one complete occupational as-low-as-reasonably-achievable (ALARA) planning and controls sample as defined in IP 71124.02–05.

.1 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance:

- RWP 152500; 2R29–RTD Replacement Project;
- RWP 155021; 10–Year ISI/Corrosion Inspection–2R29;
- RWP 152055; Scaffold Standard Work–U2 Outage; and
- RWP 152300; Primary Steam Generator Activities.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features, considered alternate mitigation features, and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessments had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors

determined whether the licensee's work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and RWP documents.

The inspectors compared the results achieved (dose rate reductions and person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity and failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

The inspectors determined whether post-job reviews were conducted and if identified, problems were entered into the licensee's CAP.

b. Findings

No findings were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors reviewed the assumptions and bases (including dose rate and man-hour estimates) for the current annual collective exposure estimate for reasonable accuracy for select ALARA work packages. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

b. Findings

No findings were identified.

.3 Source Term Reduction and Control (02.04)

a. Inspection Scope

The inspectors used licensee records to determine the historical trends and current status of significant tracked plant source terms known to contribute to elevated facility aggregate exposure. The inspectors assessed whether the licensee had made allowances or developed contingency plans for expected changes in the source term as the result of changes in plant fuel performance issues or changes in plant primary chemistry.

b. Findings

No findings were identified.

.4 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed the performance of radiation workers and radiation protection technicians during work activities within in radiation areas, airborne radioactivity areas, and/or high-radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers were familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas), and whether there were any procedure compliance issues (e.g., workers were not complying with work activity controls). The inspectors observed the performance of radiation workers to assess whether training and skill levels were sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

2RS5 Radiation Monitoring Instrumentation (71124.05)

These inspection activities supplement those documented in IR 05000282/2014002; 05000306/2014002 and constituted one complete radiation monitoring instrumentation sample as defined in IP 71124.05–05.

.1 Calibration and Testing Program (02.03)

Process and Effluent Monitors

a. Inspection Scope

The inspectors selected effluent monitor instruments (such as gaseous and liquid) and evaluated whether channel calibration and functional tests were performed consistent with radiological effluent TS/ODCM requirements. The inspectors assessed whether: (a) the licensee calibrated its monitors with National Institute of Standards and Technology (NIST) traceable sources; (b) the primary calibrations adequately represented the plant nuclide mix; (c) when secondary calibration sources were used, the sources were verified by the primary calibration; and (d) the licensee's channel calibrations encompassed the instrument's alarm setpoints.

The inspectors assessed whether the effluent monitor alarm setpoints were established as provided in the ODCM and station procedures.

For changes to effluent monitor setpoints, the inspectors evaluated the basis for changes to ensure that an adequate justification existed.

b. Findings

Failure to Adequately Calibrate Liquid Effluent Monitors

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of TS 5.5.1.a for the failure to comply with the ODCM for not using calibration sources, which were traceable to the NIST or equivalent during the calibration of station effluent monitors.

Description: The inspectors reviewed the primary calibration records for various station effluent monitors. These calibrations included the Unit 2 SG blowdown effluent monitor (2R–19) and the waste effluent liquid monitor (R–18), which were both performed on April 7, 1993. Primary calibrations are normally performed after monitor installation or major maintenance. The purpose of primary calibrations is to determine the in-situ or installed effluent monitor efficiency. Subsequent secondary calibrations are then performed periodically, as specified by the ODCM, to ensure monitor response is unchanged. During the inspection, the inspectors determined that the radioactive sources used for these calibrations did not contain any quality information, such as NIST or equivalent traceability. This issue of concern was then entered into the license's CAP on August 20, 2015. Subsequently, the licensee performed new primary calibrations with a set of radioactive sources with an established quality. The new calibrations resulted in a reduced efficiency when compared to the previous calibrations, which was outside of the station's acceptance criteria. Although the new calibration efficiency was lower, this did not require changes to the monitor alarm setpoints.

Analysis: The inspectors determined that not utilizing NIST traceable calibration sources (or equivalent) during the primary calibration of the station effluent monitors was a PD, the cause of which was reasonably within the licensee's ability to foresee and correct, and should have been prevented. The finding was not subject to traditional enforcement since the incident did not result in a significant safety consequence, did not impact the NRC's ability to perform its regulatory function, and was not willful.

The PD was determined to be of more than minor safety significance in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the plant facilities/equipment and instrumentation attribute of Public Radiation Safety and it adversely impacted the objective of ensuring adequate protection of public health and safety due to failure to properly calibrate certain effluent monitors. Subsequent calibration of the monitors determined that the monitor efficiency was previously overstated. The inspectors also reviewed IMC 0612, Appendix E, "Examples of Minor Issues," dated August 11, 2009, but did not identify any similar examples. The finding was assessed using IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," dated February 12, 2008, and determined to be of very low safety significance (Green) because it was associated with the effluent release program but was not a failure to implement an effluent program, public dose did not exceed Appendix I criteria, and the limits in 10 CFR 20.1301(e) were not exceeded. A cross-cut aspect was not assigned as this issue occurred numerous years ago. The station has since performed monitor calibrations with radioactive sources with known quality. An example was the Unit 1 SG blowdown effluent monitor, which was calibrated on August 25, 2015.

Enforcement: Technical Specification 5.5.1.a states, in part, that "the ODCM shall contain the methodology and parameters used in the calculation of offsite doses." The ODCM Table 2.3, "Radioactive Liquid Effluent Monitoring Instrumentation, Surveillance Requirements," specifies, in part, that "initial channel calibrations be performed with NIST certified or NIST traceable sources."

Contrary to the above, on April 7, 1993, 2R–19 and R–18 were not calibrated with NIST certified, NIST traceable, or other suitable quality radioactive source(s). Corrective actions included the re-calibration of these monitors and an extent of condition on additional effluent monitors. Since the finding was of very low safety significance

(Green) and was entered into the licensee's CAP as CAPs 01490581 and 01500149, this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000282/2015004–02, Failure to Adequately Calibrate Liquid Effluent Monitors)**.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index—Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI)–Emergency AC Power System PI, Units 1 and 2, for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. 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, MSPI derivation reports, issue reports, event reports and NRC integrated IRs for the period of October of 2014 through September of 2015 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI emergency AC power system samples 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 – RHR System PI, Units 1 and 2, for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. 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, MSPI derivation reports, issue reports, event reports and NRC integrated IRs for the period of October of 2014 through September of 2015 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI residual heat removal system samples 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–CL Systems PI, Units 1 and 2, for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. 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, MSPI derivation reports, issue reports, event reports and NRC integrated IRs for the period of October of 2014 through September of 2015 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. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two MSPI cooling water system samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity PI for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's RCS chemistry samples, technical specification requirements, issue reports, event reports and NRC Integrated IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to

determine if any problems had been identified with the PI data collected or transmitted for this indicator. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a RCS sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two RCS specific activity samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate, accumulated dose alarms, and dose reports, and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very-high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151–05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors sampled licensee's submittals for the Radiological Effluent Technical Specification (RETS)/ODCM Radiological Effluent Occurrences PI for the period from the 4th quarter of 2014 through the 3rd quarter of 2015. The inspectors used PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's CAP database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as

unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RETS/ODCM radiological effluent occurrences 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 inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2, licensee trending efforts, and licensee's human performance results. The inspectors' review nominally considered the 6-month period of July of 2015 through December of 2015, although some examples expanded beyond those dates where the scope of the trend warranted.

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

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

b. Findings

No findings were identified.

.4 Annual Follow-up of Selected Issues: CAP 01494532; Safety-Related Relay for Reactor Protection System Service Life Evaluation

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors noted that CAPs 01493179 and 01493183 documented safety related relays installed in the reactor protection system had surpassed the vendor qualified life of 20 years. In their review, the inspectors identified that the CAPs listed above did not address the impact of exposure to low humidity conditions present during the late fall and winter months. In response, the licensee performed a detailed evaluation under CAP 01494532 to document the impact of low humidity on electrical equipment. The inspectors reviewed the associated evaluation and also the procedures associated with humidity monitoring and noted that per CAP 01495083, issued on September 29, 2015, the service building computer room containing the ERCS reached the low humidity alarm set point and annunciated. The inspector discussed this noted condition with operations staff and it

was recognized that the same alarm had annunciated in the fall of 2014. Based on the above information, the inspectors reviewed associated CAPs and work requests that had been generated since September of 2014 that addressed low humidity indications present during the late fall and winter months in 2014 and 2015, respectively. The inspectors determined that the evaluation performed under CAP 01494532 adequately addressed the impact of low humidity on electrical components and noted that the licensee planned to replace all applicable relays during the next RFOs for each Unit. The inspectors noted that the associated relays addressed in CAP 01494532 remained operable but non-conforming and therefore remained capable of performing their required safety functions.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152–05.

b. Findings

No findings were identified.

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

.1 Unit 2 Automatic Reactor Trip and Notice of Unusual Event

a. Inspection Scope

On December 17, 2015, Unit 2 experienced an automatic reactor trip resulting from an automatic turbine shutdown caused by a main generator lockout. The inspectors responded to the control room and monitored the operator actions taken to address the event. Following the trip, the control room received unexpected fire alarms within the Unit 2 containment. A Notice of Unusual Event was declared but subsequently exited after the licensee verified that no fire existed within containment. The inspectors reviewed the procedures used during this event to determine whether the control room operators responded properly. Documents reviewed are listed in the Attachment to this report.

This review constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report 05000282/2014–002–00 and–01: Emergency Diesel Generators Declared Inoperable Due to Not Meeting High Energy Line Break Requirements

a. Inspection Scope

On August 4, 2014, the licensee submitted the above Licensee Event Report (LER) to the NRC to document a condition that could have prevented the fulfillment of the D1 and D2 EDGs' safety function. The condition, identified on June 3, 2014, was associated with a calculated turbine building (TB) high energy line break (HELB) heat-up analysis temperature that exceeded the maximum supply and exhaust fan blade positioner temperatures for the D1 and D2 EDGs. The licensee declared both D1 and D2

inoperable, entered the applicable TS action statements, and implemented compensatory measures to bypass the supply and exhaust fan blade positioners to full cooling mode allowing the station to exit the applicable TS action statements.

The licensee initiated a CAP and root cause evaluation that was still in progress at the LER submittal deadline, therefore, on January 30, 2015, the licensee submitted supplement–01 to the above LER which described the final root cause and corrective actions.

Following submittal of the above LER supplement, the licensee received testing data from a third party vendor that demonstrated acceptable operation of the supply and exhaust fan blade positioners at the elevated temperatures identified within the TB HELB calculation. Therefore, on July 23, 2015, the licensee submitted a cancellation letter to NRC for LERs 05000282/2014–002–00 and–01 since the original condition did not result in the prevention of the fulfillment of the D1 and D2 EDGs' safety function.

The inspectors reviewed the revised analysis and the cancellation letter. No concerns were identified. Documents reviewed are listed in the Attachment to this report. This LER is closed.

This review constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.3 (Closed) LER 05000282/2015–001–00: 14 Fan Coil Unit Leak

a. Inspection Scope

On January 16, 2015, the licensee submitted the above LER to the NRC to document a condition that could have prevented the fulfillment of the Unit 1 containment safety function. The condition, identified on November 20, 2014, with Unit 1 in Mode 3, was associated with a cooling water leak from the 14 containment fan coil unit (FCU) that impacted containment integrity. The licensee declared the Unit 1 containment inoperable, entered the applicable TS LCO statement, and isolated the 14 containment FCU within the Unit 1 containment TS action completion time allowing the station to exit the applicable TS action statement. Repairs were conducted shortly thereafter and the 14 containment FCU was returned to service.

Following submittal of the above LER, the licensee performed an engineering evaluation that demonstrated that containment leakage past the auxiliary building special ventilation zone and shield building would have remained less than the available containment leakage margin. Therefore, since the 14 containment FCU leak did not represent a condition that could have prevented the fulfillment of the Unit 1 containment safety function, the licensee submitted a cancellation letter to NRC for LER 05000282/2015–001–00 on September 3, 2015.

The inspectors reviewed the revised analysis and the cancellation letter. No concerns were identified. Documents reviewed are listed in the Attachment to this report. This LER is closed.

This review constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.4 (Closed) LER 05000282/2015–002–00: 14 Fan Coil Unit Leak (Lower Head)

a. Inspection Scope

On April 10, 2015, the licensee submitted the above LER to the NRC to document a condition that could have prevented the fulfillment of the Unit 1 containment safety function. The condition, identified on February 10, 2015, with Unit 1 in Mode 3, was associated with a cooling water leak from the 14 containment FCU that impacted containment integrity. The licensee declared the Unit 1 containment inoperable, entered the applicable TS action statement, and implemented repairs to the 14 containment FCU within the Unit 1 containment TS action completion time allowing the station to exit the applicable TS action statement.

Following submittal of the above LER, the licensee performed an engineering evaluation (see Section 4OA3.3) that demonstrated that containment leakage past the auxiliary building special ventilation zone and shield building would have remained less than the available containment leakage margin. Therefore, since the 14 containment FCU leak did not represent a condition that could have prevented the fulfillment of the Unit 1 containment safety function, the licensee submitted a cancellation letter to NRC for LER 05000282/2015–002–00 on September 3, 2015.

The inspectors reviewed the revised analysis and the cancellation letter. No concerns were identified. Documents reviewed are listed in the Attachment to this report. This LER is closed.

This review constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

No findings were identified.

.5 (Closed) LER 05000282/2015–003–00: Unanalyzed Condition Due to Non-Compliance with 10 CFR 50 Appendix R

a. Inspection Scope

The inspectors reviewed information provided by the licensee regarding the April 19, 2015, identification of inadequate procedure steps within procedure F5 Appendix B, “Control Room Evacuation (Fire)”, Revision 31. Specifically, during a National Fire Protection Association (NFPA) 805 transition process review of Engineering Change 25405, “12 Reactor Cooling Pump Seal Face Replacement”, the licensee identified that F5 Appendix B did not contain required procedural steps to open direct current (DC) knife switches for the 11, 12, 21, and 22 RCP breakers prior to evacuation of the control/relay and cable spreading rooms during a postulated fire event in those areas. The unanalyzed condition was associated with the potential for fire induced circuit damage resulting in RCP(s) restarting without adequate seal cooling

restored, leading to seal failure and a small break loss of coolant accident. The unanalyzed condition only impacted the 12 RCP since the 11, 21, and 22 RCPs had appropriately modified seals to allow time for restoring seal cooling prior to seal failure.

During the inspection, the inspectors reviewed the fire protection program documents, licensee's CAPs, the apparent cause evaluation, immediate corrective actions (F5 Appendix B procedure change), and longer term corrective actions. Documents reviewed are listed in the Attachment to this report. This LER is closed.

This review constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

One finding and NCV for which the NRC exercised enforcement discretion was identified during the review of this LER. The inspectors determined that the finding and NCV associated with the unanalyzed condition was best characterized as a licensee identified finding and violation. As a result, the inspectors documented information regarding this issue in Section 4OA7 of this inspection report.

4OA5 Other Activities

.1 Institute of Nuclear Power Operations Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the Institute of Nuclear Power Operations (INPO) plant evaluation conducted in September and October of 2015. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee's performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 7, 2016, the inspectors presented the inspection results to Mr. K. Davison, Site Vice President, and other members of the staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results for the areas of radiological hazard assessment and exposure controls; occupational ALARA planning and controls; and RCS specific activity, occupational exposure control effectiveness, and RETS/ODCM radiological effluent occurrences PI verification with Mr. K. Klotz, Acting Radiation Protection Manager, on November 6, 2015;
- The results of the ISI inspection with Mr. M. Pearson, Regulatory Affairs Manager, and other members of the licensee staff on November 25, 2015;
- The inspection results for the area of radiation monitoring instrumentation via teleconference, with Mr. D. Gauger, Chemistry Manager, on December 28, 2015; and
- The Annual Review of EAL and Emergency Plan Changes with the Licensee's Emergency Preparedness Manager, Mr. B. Carberry, Emergency Preparedness Manager, via telephone on December 21, 2015.

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 safety significance (Green) was identified by the licensee. The NRC is not taking enforcement action for this violation because it met the criteria of the NRC Enforcement Policy, "Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48)," as described below:

- Title 10 CFR 50.48(b)(2) requires, in part, that “all nuclear power plants licensed to operate before January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O.” Appendix R, Section III.G.3 of 10 CFR Part 50, requires, in part, that “alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, or zone under consideration should be provided where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section. In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration.” The details of this issue were documented in Section 4OA3.5 of this report.

  Contrary to the above, on April 19, 2015, the licensee failed to ensure that alternative or dedicated shutdown capability and its associated circuits were independent of cables in the area. Specifically, procedure F5 Appendix B, “Control Room Evacuation (Fire),” Revision 31, did not contain actions to isolate the RCP breaker circuits to prevent restarting due to a fire induced loss of remote trip and loss of RCP seal cooling water that could lead to an increased rate of seal degradation and a small break loss of coolant accident. These actions were required to achieve and maintain safe shutdown in the event of a fire that resulted in functional loss and/or evacuation of the control/relay and cable spreading rooms.

Section 9.1 of the NRC Enforcement Policy allows the NRC to exercise enforcement discretion for certain fire protection related non compliances identified as a result of a licensee's transition to the new risk informed, performance based fire protection approach included in 10 CFR 50.48(c), and for certain existing non compliances that reasonably may be resolved by compliance with 10 CFR 50.48(c) as long as certain criteria are met. This risk informed, performance based approach is referred to as NFPA 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

The licensee is in transition to NFPA 805 and therefore the licensee-identified violation was evaluated in accordance with the criteria established by Section 9.1(a) of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48) for a licensee in NFPA 805 transition. The inspectors determined that for this violation: (1) the licensee would have identified the violation during the scheduled transition to 10 CFR 50.48(c); (2) the licensee had established adequate compensatory measures (see Section 4OA3.5) within a reasonable time frame following identification and would correct the violation as a result of completing the NFPA 805 transition; (3) the violation was not likely to have been previously identified by routine licensee efforts; and (4) the violation was not willful. The finding also met additional criteria established in section 12.01.b of IMC 0305, "Operating Assessment Program." In addition, in order for the NRC to consider granting enforcement discretion the violation must not be associated with a finding of high safety significance (i.e., Red).

The licensee performed risk evaluation V.SPA.15.012, Revision 3, dated December 18, 2015, and determined that this issue was not associated with a finding of high safety significance. A region III senior reactor analyst (SRA) reviewed the evaluation and concluded that the result was reasonable and that the finding was less than Red and eligible for enforcement discretion. The dominant core damage sequence from the licensee's evaluation involved an electrical cabinet fire in the relay room involving the cables that could cause spurious operation of the RCPs and that would lead to alternate shutdown. The licensee identified several conservative assumptions in the analysis. The SRA agreed that some were conservative, notably that any fire affecting the cables in the relay room that could cause a spurious start of an RCP would also result in a loss of all seal cooling due to fire damage. The SRA used IMC 0609, Appendix F, "Fire Protection Significance Determination Process," to review the results of the licensee's evaluation. The relay room is similar to a cable spreading room with electrical cabinets. The fire frequency for this room in Appendix F is 6E–3/yr. The probability of non-suppression was estimated to be 2E–2 and the spurious operation probability was assumed to be 0.6. The product of these values (7.2E–5/yr) represents a bounding relay room fire scenario delta core damage frequency (CDF) for this finding. Since the bounding result is consistent with the licensee's conclusion, the SRA determined that the delta core damage frequency for the finding was less than 1E–4/yr, which is less than Red.

In addition, the licensee entered this issue into their corrective action program as CAP 01475242. As a result, the inspectors concluded that the violation met all four criteria established by Section 9.1(a) and that the NRC was exercising enforcement discretion to not cite this violation in accordance with the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Davison, Site Vice President
T. Conboy, Director of Site Operations
E. Blondin, Engineering Director
W. Paulhardt, Plant Manager
J. Boesch, Maintenance Manager
T. Borgen, Operations Manager
B. Boyer, Radiation Protection Manager
H. Butterworth, Business Support Manager
B. Carberry, Emergency Preparedness Manager
J. Corwin, Security Manager
D. Gauger, Chemistry/Environmental Manager
S. Martin, Performance Assessment Manager
M. Pearson, Regulatory Affairs Manager
P. Wildenborg, Sr. Health Physicist
S. Redner, Project Manager, Engineering Programs
P. Brunsgaard, Manager, Engineering Programs
T. Downing, ISI Engineering
J. Wren, NDE Level III
G. Carlson, Senior Licensing Engineer, Regulatory Affairs
E. Baker, Chemist
J. Callahan, Fleet EP Manager
P. Oleson, Regulatory Analyst

U.S. Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2
T. Beltz, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000306/2015004–01	NCV	Failure to Meet ANSI N14.6 Section 5.3.1 Requirements (Section 1R08.1)
05000282/2014005–02	NCV	Failure to Adequately Calibrate Liquid Effluent Monitors (Section 2RS5.1)

Closed

05000306/2015004–01	NCV	Failure to Meet ANSI N14.6 Section 5.3.1 Requirements (Section 1R08.1)
05000282/2014005–02	NCV	Failure to Adequately Calibrate Liquid Effluent Monitors (Section 2RS5.1)
05000282/2014002–00	LER	Diesel Generators Declared Inoperable Due to Not Meeting High Energy Line Break Requirements (Section 4OA3.2)
05000282/2014002–01	LER	Diesel Generators Declared Inoperable Due to Not Meeting High Energy Line Break Requirements (Section 4OA3.2)
05000282/2015001–00	LER	Fan Coil Unit Leak (Section 4OA3.3)
05000282/2015002–00	LER	Fan Coil Unit Leak (Lower Head) (Section 4OA3.4)
05000282/2015003–00	LER	Unanalyzed Condition Due to Non-Compliance with 10 CFR 50 Appendix R (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.

1R04 Equipment Alignment (71111.04)

- 1C28.1 AOP4; Restarting Unit 1 AFWP After Low Suction/Discharge Pressure Trip; Revision 6
- CAP 01002789; AF System Components with Incorrect Quality Level; November 3, 2005
- CAP 01271632; MRB-028 AF System Components are Incorrect Quality Level; February 18, 2011
- C28–2; Auxiliary Feedwater System Unit 1; Revision 52
- CAP 01500181; QF1142 in WO–472499 Needs More Info; November 4, 2015
- Condition Report Search from January 1, 2010 to October 1, 2015
- DBD SYS–28B; Prairie Island Nuclear Generating Plant Design Bases Document; Revision 8
- Drawing B–15300; Min. Flow Orifice Assembly; August 27, 1970
- Drawing NF–39222; Feedwater & Aux Feedwater Unit 1; Revision 83
- Letter from Pacific Pumps Dresser to Northern States Power Company; Minimum Flow-Nuclear Service Pumps Comments to NRC Bulletin 88–04; September 16, 1988
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- SWI Eng–30 Addendum–A; Prairie Island Nuclear Generating Plant Q–List Downgrade Resolution Project Position Papers; Revision 1
- Temporary Change Request 084A for SP 1100 12; Motor Driven AFW Pump Monthly Test; Revision 84
- USAR Appendix I; Prairie Island Updated Safety Analysis Report; Revision 33
- CAP 01500373; Question by NRC in Regards to Drawings in SharePoint; November 5, 2015
- CAP 01500324; Safety Function for RCP Inlet and Outlet MV's Evaluation; November 5, 2015
- SP 2251; Caustic Addition Valve Quarterly Test; Revision 13
- H5; Motor Operated Valve Program; Revision 19
- H10.1.B; Inservice Testing Program Component Basis Document; Revision 3
- CAP 01502434; 21 Caustic Standpipe Level Spiked Low to Low Alarm Set-point; November 18, 2015
- DBD SYS–8D; Containment Spray System Design Basis Document; Revision 4
- B18D; Containment Spray System Bases Document; Revision 9
- NF–39252; Caustic Addition System Unit 1 & 2 Flow Diagram; Revision 83
- B18D; Containment Spray System Description; Revision 9

1R05 Fire Protection (71111.05)

- NF–39228-1; Fire Protection and Screen Wash System Unit 1 & 2 Flow Diagram; Revision 91
- F5 Appendix A; Fire Area 71 Requirements; Revision 13
- F5 Appendix F; Fire Hazards Analysis Matrix; Revision 30
- NF–39228–3; Sprinkler Fire Protection System Turbine U 2–Screen-house Unit 1 & 2; Revision 78

- CAP 01504212; Unexpected FP Annunciator in Unit 2 Feedwater Pump Area; December 1, 2015
- CAP 01504521; NRC Question: Fire Strategies Shows Containment Fire Extinguishers; December 3, 2015

1R06 Flooding (71111.06)

- 1C28.1 AOP2; Loss of Condensate Supply to Aux Feed Water Pump Suction; Revision 0
- 5AWI 8.9.0; Internal Flooding Drainage Control; Revision 15
- C31 AOP1; Fire Protection Line Break; Revision 3
- Calculation 1067–0022–001; Determination of Flow Path Input for Floor Drains; Revision 0
- Calculation ENG–ME–586; Effects of Flooding in the AFW Pump Room from a Postulated Pipe Rupture; June 9, 2005
- Calculation ENG–ME–759; Gothic Internal Flooding Calculation for the Turbine Building; Revision 1D
- Drawing NF–38213; Turbine Room–Concrete Plan of Base Slab Floor Drains–Class I Area; Revision G
- H36; Plant Flooding; Revision 10
- TP 1398; Verify Physical Inputs to Internal Flooding Evaluations; Revision 5
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1R07 Annual Heat Sink Performance (71111.07A)

- WO 498516–01; 22 CC HX South Internal Inspection; Revision 0

1R08 Inservice Inspection Activities (71111.08)

- Safety Evaluation Related to the Control of Heavy Loads; Dated 06/06/83
- ANSI N14.6; American National Standards for Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds or More for Nuclear Materials; 1978 Edition
- Calculation ENG–CS–361; Evaluation of the Acceptability of the Reactor Vessel Head Lift Rig, Reactor Vessel Internals Lift Rig, Load Cell and Linkage to the Requirements of NUREG 0612; Revision 0
- NDE Report Nos. BOP–MT–15–040–051,021–023; Magnetic Particle Examination of Reactor Vessel Internals Lift Fixture Welds; Dated 10/27/15
- WO 00457321; Perform NDE Magnetic Particle Examination of Reactor Head Lifting Rig; Dated 10/23/15
- NDE Report No. 2005M008; Magnetic Particle Examination of Reactor Head Lift Fixture; Dated 06/03/05
- WO 457197–07; Perform NDE Exams of Reactor Vessel Internals Lifting Device; Dated 10/24/15
- CAP 01457469; Operating Experience (IN 2014–02) Item Evaluated: Crane and Heavy Lift Issues Identified; Dated 01/12/15
- Engineering Evaluation EC 26341; Reactor Vessel Internal Lift Rig Torque Tube Weld Evaluation; Dated 10/26/15
- WO 00436724; SP 1392 Unit 1 RCS System Bolting Inspection; Dated 08/05/14
- CAP 01450417; BACC Evaluation for ISI Indication on CV-31325; Dated 10/11/14
- CAP 01450480; BACC Evaluation for ISI Indication on RC-19-1; Dated 10/11/14
- CAP 01450476; BACC Evaluation for ISI Indication on 135-011; Dated 10/11/4
- CAP 01427328; BACC Evaluation for Leak Identified in Unit 2 Containment 21 Vault; Dated 04/17/14

- CAP 01498446; Functionality Assessment of Reactor Vessel Internals Lifting Device; Dated 10/29/15
- CAP 01412727; Leak From Capped Drain Downstream of 2RC–8–19; Dated 12/30/13
- CAP 01431405; Boric Acid Leak was Found on 2RC–8–31; Dated 05/20/14
- CAP 01469111; Boric Acid Packing Leak on 2SI–35–6; Dated 03/06/15
- CAP 01492989; Boric Acid Built Up Below 21 SI Pump; Dated 09/11/15
- CAP 01445383; 22 Safety Injection Pump IB/OB Mechanical Seal Leakage; Dated 09/04/14
- WO 00406128; Remove and Replace Valve 2RC–7–2, Loop A to Pressurizer CV 31228 BY–PASS; Dated 09/28/13
- H2; Boric Acid Corrosion Control Program; Revision 25
- NDE Report No. 2015V011; VT–3 of AFWH–79 Sway Strut/Clamp; Dated 10/21/15
- FP–PE–NDE–530; Visual Examination, VT–3; Revision 8
- CAP 01499213; Internals Lift Rig Question; Dated 10/29/15
- CAP 01497779; NRC Question in Regards to ISI Exam of Reactor Head Lift Rig; Dated 10/21/15
- CAP 01497774; PM 3560–52 Needs to be Revised for Better Work Execution; Dated 10/21/15
- PM 3560–52; Reactor Head Lifting Rig Spreader & Connection Legs Assembly Inspections; Revision 13
- CAP 01452946; ISI Indication on Hanger 1RSIH–415; Dated 10/25/14
- CAP 01414257; Non-Conformance with ASME Section XI; Dated 01/13/14
- CAP 01429434; Potential ISI Issue for ANII Procedure Reviews; Dated 05/05/14
- CAP 01452105; Metallic Item Found on the Reactor Vessel Flange; Dated 10/20/14
- NDE Report No. 2015U023; Ultrasonic Examination of SI Elbow-to-Pipe Weld W–6; Dated 11/01/15;
- NDE Report No. 2015U044; Ultrasonic Examination of RH Pipe-to-Elbow Weld W–5; Dated 11/05/15
- NDE Report No. 2015V013; VT–3 of AFWH–64 Sway Strut/Clamp; Dated 10/21/15
- Procedure FP–PE–NDE–402; Ultrasonic Examination of Austenitic Pipe Welds–Supplement 2; Revision 5
  SWI–NDE–ET–1; Bobbin Coil Data Analysis; Revision 6
- SWI–NDE–ET–3; Rotating Coils Data Analysis; Revision 6
- SWI–NDE–ET–6; Array Coil Data Analysis; Revision 1

<u>1R11 Licensed Operator Requalification Program</u> (71111.11)

- 2C1.3–M3; Unit 2 Shutdown to Mode 3; Revision 5
- 2C1.2–M2; Unit 2 Startup to Mode 2; Revision 4
- 2C1.2–M1; Unit 2 Startup to Mode 1; Revision 1
- D30; Post Refueling Startup Testing; Revision 61

<u>1R12 Maintenance Effectiveness</u> (71111.12)

- CAP 01198723; 11 RCP Motor Had High Breakaway Torque Reading; September 21, 2009
- CAP 01490741; IST and MOV Program Discrepancies; August 21, 2015
- WO 532146–16; 21 RCP Seal Replacement; November 30, 2015
- CAP 01500324; Unit 1 and 2 RCP CC Inlet and Outlet MV Operability Evaluation; November 6, 2015
- CAP 01250606; HELB Interaction in Aux Building Overstress CC Piping; September 21, 2010
- NF–39216–1; Cooling Water–Screen-house Unit 1 & 2; Revision 89
- EC 13000; High Energy Line Break Evaluation for Component Cooling Water Piping in the Turbine Bldg; Revision 0

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

- Shift Manager Logs and Control Room Logs Units 1 & 2; November 17–19, 2015

1R15 Operability Determination and Functional Assessments (71111.15)

- CAP 01504216; AFW Recirc Line Recommended Procedure Change; December 2, 2015
- QF0739; AFWP Recirculation Line Evaluation NRC Response Form; December 2, 2015
- CAP 01500184; AFWP Recirculation Line Seismic Evaluation; November 14, 2015
- CAP 01501764; Additional NRC Question Related to AFW Recirc Line and HELB; November 13, 2015
- NF–39222; Flow Diagram Feedwater and Aux Feedwater Unit 1; Revision 83
- Operating Information 15–63; Auxiliary Feedwater Recirculation Flow Min-flow Requirements; November 20, 2015

1R19 Post-Maintenance Testing (71111.19)

- H36; Plant Flooding; Revision 10
- CAP 01501977; Six of 97 FCU Pipe Flanges Required Re-torque During PMT; November 16, 2015
- WO 519240–03; Replace SV–33133; October 2, 2015
- CAP 01495575; SV–33133 Stroked Outside Ref. Range During PMT
- CAP 01495499; SV–33133 Failed PMT Testing Under WO 519240; October 2, 2015
- WO 519240-02; SV–33133 Stroke Time Outside Ref. Range; October 2, 2015
- SP 1151A; Train A Cooling Water System Quarterly Test; Revision 20
- CAP 01501698; 21 RCP did not Rotate as Expected During Alignment; November 13, 2015
- WO 492354–06; Mechanical Troubleshooting of 21 RCP; November 17, 2015

1R20 Outage Activities (71111.20)

- CAP 01500324; Safety Function for RCP Inlet and Outlet MVs; November 20, 2015
- Unit Two Refueling Outage October 2015 Shutdown Safety Assessment; Revision 0
- 2C1.4; Unit 2 Power Operation; Revision 56
- 2C1.3–M2; Unit 2 Shutdown to Mode 2; Revision 3
- 2C19.1; Containment Unit 2 Plant Operation Requirements; Revision 23
- CAP 01504150; Identified Issues During Final Unit 2 Containment Walk Through; December 2, 2015
- CAP 01503211; U2 Rx Vsl Support Fan Motor FLA Question; November 24, 2015
- NF–39220; Unit 1 Condensate System Flow Diagram; Revision 79
- SP 2177; Core Inventory Verification; November 16, 2015
- CAP 01506302; Water Found on 755 Level of U2 CTMT Following Rx Trip; December 17, 2015
- CAP 01506286; 25B Tube Side Relief Valve Lifted Prior to Taking 22 FWP OOS; December 17, 2015
- CAP 01504099; Ceiling Leak in Unit 2 Rod Drive Room; December 1, 2015
- CAP 01499287; Leaching was Identified on the 21 RCP Seal Faces; October 29, 2015
- CAP 01499105; MV–32197 as Found Configuration; October 29; 2015
- CAP 01491555; Ball Valve PMs Suspended Due to Parts Unavailability; August 29, 2015
- CAP 01505839; Eddy Current Heating on Unit 2 Generator Bushing Box IPB Duct; December 14, 2015

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

- Prairie Island Emergency Plan; Revisions 50 and 51
- PINGP–1576; Emergency Action Level Matrix; Revisions 7 and 8
- F3-2.1; Emergency Action Level Technical Bases; Revision 10
- FP–R–EP–02; 10 CFR 50.54(q) Review Process, Revision 11
- QF–0724; 10 CFR 50.54(q) Review Form, Revision 6
- CAP 01506257; NRC ID Wrong Reference Used in 10 CFR 50.54(q) Evaluation; Dated 12/17/15

1EP6 Drill Evaluation (71111.06)

- P9116SE–0101; LOR Cycle 16A Simulator Evaluation; October 1, 2015

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- RPIP 1331; Radioactive Material Control; Revision 2
- RPIP 1120; Posting of Restricted Areas; Revision 40
- RPIP 1123; Alpha Characterization Smears; Revision 2
- Technical Basis Document; 14–001; Alpha Radiation Protection Program; Revision 0
- FP–RP–AM–01; Alpha Monitoring Program; Revision 5
- RPIP 1135; RWP Coverage; Revision 35
- RPIP 1204; Evaluation of Airborne Radioactivity; Revision 20
- RPIP 1202; Gaseous Airborne Radioactive Monitoring; Revision 9
- RPIP 1331; Radioactive Material Control; Revision 2
- RPIP 1300; Control and Tagging of Radioactive Material; Revision 23

2RS2 Occupational ALARA Planning and Controls (71124.02)

- Prairie Island Nuclear Generating Plant 2R28 Radiation Protection Department Outage Manual; Date Not Provided
- Prairie Island Nuclear Generating Plant; Dose Excellence Plan; 2013–2017; Revision 0
- Prairie Island Nuclear Generating Plant; 2R28 Radiation Protection Department Outage Report; Steam Generator Project; Dated 02/03/14
- Prairie Island; 1R29 Radiation Protection Department Outage Report; Dated 02/12/15
- FP–RP–SEN–02; Radiological Work Planning and Controls; Revision 3
- RWP and Associated ALARA Files; RWP 152500; 2R29–RTD Replacement Project; Various Dates
- RWP and Associated ALARA Files; RWP 155021; 10-Year ISI/Corrosion Inspection–2R29; Various Dates
- RWP and Associated ALARA Files; RWP 152055; Scaffold Standard Work–U2 Outage; Various Dates
- RWP and Associated ALARA Files; RWP 152300; Primary SG Activities–U2 Outage; Various Dates

2RS5 Radiation Monitoring Instrumentation (71124.05)

- CAP 01490581; Missing Documents for Radiation Monitor Primary Calibrations; August 20, 2015
- CAP 01494632; Cal of Process Rad Monitors Dose Meet H4 ODCM Standard; September 25, 2015
- CAP 01500149; Additional Eff Rad Monitors Require Evaluation; November 4, 2015

- Offsite Dose Calculation Manual (ODCM); Revision 29
- 1R19; Unit 1 Steam Generator Blowdown Monitor Calibration; August 25, 2010
- 2R19; Unit 2 Steam Generator Blowdown Monitor Calibration; April 7, 1993
- 2R19; Unit 2 Steam Generator Blowdown Monitor Calibration; October 15, 2015
- R–18; Waste Effluent Liquid Monitor Calibration; April 7, 1993
- R–18; Waste Effluent Liquid Monitor Calibration; October 2, 2015

4OA1 Performance Indicator Verification (71151)

- FP–R–PI–01; Preparation of NRC Performance Indicators; Revision 3
- FP–R–PI–01; Preparation of NRC Performance Indicators; Attachment 6; RCS Specific Activity; Various Dates
- FP–R–PI–01; Preparation of NRC Performance Indicators; Attachment 9; Occupational Exposure Control Effectiveness; Various Dates
- FP–R–PI–01; Preparation of NRC Performance Indicators; Attachment 10; RETS/ODCM Radiological Effluent Occurrence; Various Dates

4OA2 Identification and Resolution of Problems (71152)

- C37.9; Control, Relay, and Computer Room Ventilation; Revision 28
- C37.14; Service Building Ventilation System; Revision 13
- CAP 01469452; NRC Questioned Relative Humidity Level in Multiple Areas; March 10, 2015
- CAP 01468498; NRC Identified–Question CR Humidity Requirements; March 3, 2015
- CAP 01447443; Computer Room Low Humidity; September 21, 2014
- CAP 01498739; C37.14 Rev. 13 (Secondary); November 17, 2015
- CAP 01276040; GL–08–01 TI–177 Drawing Error on NF–39252; March 18, 2011
- PI–21.3B.002; Namco Limit Switch Series Qualification H, K; Revision 1
- EC 00026393; Relative Humidity Impact(s) on Electrical Components; Revision 0
- CAP 01495083; ERCS Alarm for Low Humidity; September 29, 2015
- CAP 01494719; Control Room Humidifiers Should be Repaired or Abandoned; September 25, 2015
- CAP 01495292; C37.9, Revision 28, Relay, and Computer Room Ventilation; October 1, 2015
- WO 531436; ERCS Alarm for Low Humidity; October 1, 2015
- CAP 01501381; 31 Namco Position Switches are Past Qualified Life; November 11, 2015

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

- CAP 01506299; NUE HU2.1 Declared Following Fire Alarm in U2 CTMT; December 17, 2015
- CAP 01506281; Flakes Found on U2 CTMT A/S Particle Filter; December 17, 2015
- CAP 01506285; Unit 2 Reactor Trip Due to Turbine Trip; December 17, 2015

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feed Water
ALARA	As-Low-As-Is-Reasonably-Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CC	Component Cooling
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
CL	Cooling Water
DC	direct current
EAL	Emergency Action Level
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ERCS	Emergency Response Computer System
ET	Eddy Current
FCU	Fan Cooling Unit
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IN	Information Notice
IP	Inspection Procedure
IPEEE	Individual Plant Examination of External Events
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OSP	Outage Safety Plan
PARS	Publicly Available Records System
PD	Performance Deficiency
PI	Performance Indicator
PM	Planned or Preventative Maintenance
PWR	Pressurized Water Reactor
RCA	Radiologically Controlled Area
RCP	Reactor Cooling Pump
RCS	Reactor Coolant System
RETS	Radiological Effluent Technical Specifications
RFO	Refueling Outage

RTD	Resistance Temperature Detector
RWP	Radiation Work Permit
SER	Safety Evaluation Report
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
SRA	Senior Risk Analyst
SSC	Structures, Systems, Components
TB	Turbine Building
TS	Technical Specifications
USAR	Updated Safety Analysis Report
VT-3	Visual Examination
WO	Work Order

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Sincerely,

/RA/

Kenneth Riemer
Branch 2
Division of Reactor Projects

Docket Nos. 50–282; 50–306; 72–010
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