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

April 28, 2017

EA–17–072

Mr. Scott D. Northard
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota
1717 Wakonade Drive East
Welch, MN 55089–9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000282/2017001 and 05000306/2017001

Dear Mr. Northard:

On March 31, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Prairie Island Nuclear Generating Plant, Units 1 and 2. On April 6, 2017, the NRC inspectors discussed the results of this inspection with you and other members of your staff. The enclosed report represents the results of this inspection.

Based on the results of this inspection, the NRC has identified one issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that a violation was associated with this issue. Because the licensee initiated condition reports to address this issue, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the Enforcement Policy. The NCV is described in the subject inspection report. Further, the inspectors documented a licensee-identified violation that was determined to be of very low safety significance in this report. The NRC is treating this violation as a NCV consistent with Section 2.3.2.a of the Enforcement Policy.

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

This letter, its enclosure, and your response, (if any), will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding.”

Sincerely,

/RA Karla Stoedter Acting for/

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 05000282/2017001; 05000306/2017001

cc: Distribution via LISTSERV®

Letter to Scott D. Northard from Kenneth Riemer dated April 28, 2017

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2—NRC INTEGRATED INSPECTION REPORT 05000282/2017001 and 05000306/2017001

DISTRIBUTION:
Jeremy Bowen
RidsNrrPMPrairieIsland Resource
RidsNrrDorlLpl3
RidsNrrDirsIrib Resource
Cynthia Pederson
Darrell Roberts
Richard Skokowski
Allan Barker
Carole Ariano
Linda Linn
DRPIII
DRSIII
ROPreports.Resource@nrc.gov

ADAMS Accession Number: ML17118A218

OFFICE	RIII	RIII	RIII	
NAME	JMancuso:tt	RSkokowski	KStoedter for KRiemer	
DATE	4/26/2017	4/26/2017	4/28/2017	

OFFICIAL RECORD COPY

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:	05000282/2017001; 05000306/2017001
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant, Units 1 and 2
Location:	Welch, MN
Dates:	January 1 through March 31, 2017
Inspectors:	L. Haeg, Senior Resident Inspector P. LaFlamme, Resident Inspector S. Bell, Health Physicist J. Bozga, Reactor Inspector
Approved by:	K. Riemer, Chief Branch 2 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY

Inspection Report 05000282/2017001, 05000306/2017001; January 1, 2017, through March 31, 2017; Prairie Island Nuclear Generating Plant, Units 1 and 2. Other Activities.

This report covers a 3–month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding was identified by the inspectors. The finding involved a Non-Cited Violation (NCV) of the U.S. Nuclear Regulatory Commission (NRC) requirements. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," dated July 2016.

NRC–Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Severity Level IV/Green. The inspectors identified a Green finding and associated Severity Level IV Violation of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59(d)(1), for the licensee's failure to perform a written evaluation which provided the bases for the determination that a change in the NRC–approved Westinghouse methodology referenced in the Updated Safety Analysis Report (USAR) for evaluating the acceptability of reactor pressure vessel internals baffle former bolting distributions did not require a license amendment. This finding was entered into the licensee's Correction Action Program (CAP) as CAP documents 1539487, "Documentation Missing in 50.59 Screening 4443," dated October 26, 2016; 1552331, "BFB Screen Referenced Eval for SER Limitation 4 Non-Existent," dated March 6, 2017; and 1552314, "BFB Screening Lacks Documentation for SER Limitation 3," dated March 6, 2017. The licensee performed an operability determination and determined the baffle bolts were operable. The inspectors reviewed the operability determination and no performance deficiencies were identified in this determination.

The inspectors determined that the licensee's failure to perform a written evaluation, providing the bases for the determination that a change in the NRC–approved Westinghouse methodology for evaluating the acceptability of baffle former bolting distributions did not require a license amendment, was a performance deficiency. This finding was also evaluated using traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. The performance deficiency was determined to be more-than-minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, compliance with the NRC–approved methodology of WCAP–15029–P–A ensured the baffle former assembly maintained its structural integrity, avoiding a failure or excessive deflection of the baffle plates, and hence the primary concern of ensuring the emergency core cooling system could continue to perform its design function of cooling the reactor core. The inspectors determined the finding could be evaluated using the Significance

Determination Process (SDP) in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," for the Mitigating Systems cornerstone. The finding screened as having very-low safety significance (Green) because the emergency core cooling system maintained its operability, specifically with respect to performing its safety function of ensuring adequate core cooling. As such, the finding corresponded to a Severity Level IV Violation in accordance with Example 6.1.d.2 of the NRC Enforcement Policy. The inspectors did not identify a cross cutting aspect because the performance deficiency was from 2013, and hence the issue did not represent current performance. (Section 4OA5.2(1))

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 CAP. The inspectors documented one licensee-identified violation for which enforcement discretion was granted. This violation and CAP tracking number is 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. On March 30, 2017, the licensee reduced power to 40 percent to perform surveillance testing, main condenser maintenance, and to investigate elevated temperatures associated with the 11 main feedwater pump motor. Unit 1 remained at 40 percent power at the end of the inspection period.

Unit 2 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. On January 16, 2017, the licensee reduced power to 15 percent to investigate performance issues associated with the 22 main feedwater regulating valve. Unit 2 returned to full power on January 19, 2017.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition-High Wind Conditions

a. Inspection Scope

Since high winds were forecast in the vicinity of the facility for February 14–17, 2017, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On February 14, 2017, the inspectors walked down the offsite power sources in the substation, in addition to the licensee's emergency alternating current (AC) power systems, because their safety-related functions could be affected or required as a result of high winds or tornado-generated missiles or the loss of offsite power. The inspectors evaluated the licensee staff's preparations against the site's procedures and determined that the staff's actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the USAR and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. The inspectors also reviewed a sample of CAP items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in 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:

- D5 emergency diesel generator (EDG);
- Bus 15 and 111 safeguards electrical systems;
- D1 EDG; and
- 21 125V safeguards battery.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, 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 four partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

During the week of January 4, 2017, the inspectors performed a complete system alignment inspection of the Unit 1 and 2 control room safeguards chilled water systems to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one complete system walkdown sample as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Quarterly 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 Area 101; D5 EDG room;
- Fire Area 22; unit 2 bus 121 room;
- Fire Area 33; 11 safeguards battery room;
- Fire Area 41A; screenhouse diesel driven cooling water pump room; and
- Fire Areas 35 & 36; battery rooms 21 & 22.

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 five quarterly fire protection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

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 commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or 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 area 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:

- Intake screenhouse.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one internal flooding sample as defined in IP 71111.06–05. In addition, the inspectors did not identify a history of cable degradation or failure due to submergence at the site. The underground vaults inspection sample was not performed as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On February 2, 2017, 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 (if applicable);

- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan (EP) 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.

b. Findings

No findings were identified.

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

a. Inspection Scope

During the week of January 16, 2017, the inspectors observed operator performance in the control room during a Unit 2 reduction in power to 15 percent for feedwater regulating valve troubleshooting activities. This was an activity that required heightened awareness or was 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 (if applicable);
- correct use and implementation of procedures;
- control board (or equipment) manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and EP actions and notifications (if applicable).

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 (71111.12Q)

a. Inspection Scope

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

- Nuclear instrumentation; and
- 11 and 12 safeguards battery systems.

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 maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These inspections constituted two routine quarterly evaluation 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:

- Unit 2 power reduction to 15 percent and subsequent 22 main feedwater regulating valve maintenance with D2 EDG unavailable for planned maintenance (PM);
- 10 bank transformer unavailability during substation maintenance in parallel with power range channel 1N41 inoperable for PM;
- Work Week 1709 activities;
- Letdown flow control valve maintenance and control room annunciator system replacement activities;
- Unit 1 8H17 345 kilovolt (kV) main generator output breaker troubleshooting & maintenance repairs following low pressure alarms; and
- Unit 1 control room annunciator panel replacement maintenance activities and an associated inadvertent 12 inverter trip.

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 are listed in the Attachment to this report.

These inspections constituted six maintenance risk assessments and emergent work control samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15)

.1 Operability Determinations

a. Inspection Scope

The inspectors reviewed the following issues:

- CAP 1548138; Non-Conservative Stress Limits in BFB Analysis;
- CAP 1547382; D1 Air Start System Non-Conformance Evaluation;
- CAP 1548243; D2 Emergency Diesel Generator Lube Oil Warming System Malfunction Evaluation;
- CAP 1547796; NRC Identified Safeguards Bus 111 Overhead Scaffold Configuration Evaluation;
- CAP 1548783; Preliminary Battery DC Calculation Results Evaluation; and
- CAP 1546496; NRC Identified 122 Control Room Particle Absolute and Charcoal Filter Scaffold Proximity Evaluation.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TSs and USAR to the licensee's evaluations 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 evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

These inspections constituted six operability determination samples 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 modifications:

- Annunciator replacement project engineering change (EC) evaluations 24410, 24411, 24412 and 24413.

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 modifications did not affect the operability or availability of the affected 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 modifications with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modifications in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one 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:

- 4–CL–2 cooling water pipe thru-wall leak repair post-maintenance testing;
- 21 main feedwater regulating valve control board replacement testing;
- D2 EDG testing following 2–year overhaul PM; and
- 2N42 power range nuclear instrument high voltage power supply testing following replacement 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 four post-maintenance testing samples as defined in IP 71111.19–05.

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 2089A; Train A Residual Heat Removal Pump and Suction Valve from the RWST Quarterly Test In-service Test (IST);
- SP 2093; D5 Monthly Surveillance Test (Routine); and
- SP 1100; 12 Motor-Driven Auxiliary Feedwater Pump Quarterly Flow and Valve Test (Routine); and
- SP 2003; Analog Protection Functional Test (Routine).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were clearly stated, demonstrated operational readiness, and were consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- 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 in-service testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers (ASME) code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, 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.

These inspections constituted three routine surveillance testing samples and one IST sample as defined in IP 71111.22, Sections–02 and–05.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on March 2, 2017, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room simulator, technical support center, operations support center, and emergency offsite facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)

.1 Radioactive Material Storage (02.02)

a. Inspection Scope

The inspectors selected areas where containers of radioactive waste are stored, and evaluated whether the containers were labeled in accordance with Title 10 CFR, Part 20.1904, or controlled in accordance with 10 CFR 20.1905.

The inspectors assessed whether the radioactive material storage areas were controlled and posted in accordance with the requirements of 10 CFR Part 20. For materials stored or used in the controlled or unrestricted areas, the inspectors evaluated whether they were secured against unauthorized removal and controlled in accordance with 10 CFR 20.1801 and 10 CFR 20.1802.

The inspectors evaluated whether the licensee established a process for monitoring the impact of low-level radioactive waste storage that was sufficient to identify potential unmonitored, unplanned releases or nonconformance with waste disposal requirements.

The inspectors evaluated the licensee's program for container inventories and inspections. The inspectors selected containers of stored radioactive material, and assessed for signs of swelling, leakage, and deformation.

This inspection constituted one complete radioactive solid waste processing and radioactive material handling, storage, and transportation sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.2 Radioactive Waste System Walkdown (02.03)

a. Inspection Scope

The inspectors walked down accessible portions of select radioactive waste processing systems to assess whether the current system configuration and operation agreed with the descriptions in plant and/or vendor manuals.

The inspectors reviewed the adequacy of changes made to the radioactive waste processing systems since the last inspection. The inspectors evaluated whether changes from what is described in the USAR were reviewed and documented in accordance with 10 CFR 50.59 or that changes to vendor equipment were made in accordance with vendor manuals. The inspectors also assessed the impact of these changes on radiation doses to occupational workers and members of the public.

The inspectors evaluated whether tank recirculation procedures provided sufficient mixing.

The inspectors assessed whether the licensee's process control program correctly described the current methods and procedures for dewatering and waste stabilization.

This inspection constituted a partial radioactive waste system walkdown sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.3 Waste Characterization and Classification (02.04)

a. Inspection Scope

For select waste streams, the inspectors assessed whether the licensee's radiochemical sample analysis results were sufficient to support radioactive waste characterization as required by 10 CFR Part 61. The inspectors evaluated whether the licensee's use of scaling factors and calculations to account for difficult-to-measure radionuclides was technically sound and based on current 10 CFR Part 61 analysis.

The inspectors evaluated whether changes to plant operational parameters were taken into account to: (1) maintain the validity of the waste stream composition data between the sample analysis update; and (2) assure that waste shipments continued to meet the requirements of 10 CFR Part 61.

The inspectors evaluated whether the licensee had established and maintained an adequate quality assurance program to ensure compliance with the waste classification and characterization requirements of 10 CFR 61.55 and 10 CFR 61.56.

This inspection constituted one complete waste characterization and classification sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.4 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors reviewed the technical instructions presented to workers during routine training. The inspectors assessed whether the licensee's training program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities. The inspectors assessed whether shippers were knowledgeable of the shipping regulations and demonstrated adequate skills to accomplish package preparation requirements. The inspectors evaluated whether the licensee was maintaining shipping procedures in accordance with current regulations. The inspectors assessed whether the licensee was meeting the expectations in U.S. NRC Bulletin 79–19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR Part 172, Subpart H, "Training."

The inspectors evaluated whether the requirements for Type B shipment Certificates of Compliance had been met. The inspectors determined whether the user was a registered package user and had an U.S. Nuclear Regulatory Commission approved quality assurance program. The inspectors assessed whether procedures for cask loading and closure were consistent with vendor procedures.

The inspectors assessed whether non–Type B shipments were made in accordance with the package quality documents.

The inspectors assessed whether the receiving licensee was authorized to receive the shipment packages.

This inspection constituted one complete shipment preparation sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.5 Shipping Records (02.06)

a. Inspection Scope

The inspectors reviewed select shipments to evaluate whether the shipping documents indicated the proper shipper name; emergency response information and a 24–hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and UN number. The inspectors assessed whether the shipment marking, labeling, and placarding was consistent with the information in the shipping documentation.

This inspection constituted one complete shipping records sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

.6 Identification and Resolution of Problems (02.07)

a. Inspection Scope

The inspectors assessed whether problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and were properly addressed for resolution. Additionally, the inspectors evaluated whether the corrective actions were appropriate for a selected sample of problems documented by the licensee that involve radioactive waste processing, handling, storage, and transportation.

This inspection constituted one complete sample as defined in IP 71124.08–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator (PI), Units 1 and 2, for the period from the first quarter of 2016 through the fourth quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Integrated Inspection Reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

These inspections constituted two unplanned scrams per 7000 critical hours samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Power Changes per 7000 Critical Hours performance indicator, Units 1 and 2, for the period from the first quarter of 2016 through the fourth quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports and NRC Integrated Inspection Reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issued 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.

These inspections constituted two unplanned power changes per 7000 critical hours samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications performance indicator, Units 1 and 2, for the period from the first quarter of 2016 through the fourth quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Integrated Inspection Reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

These inspections constituted two unplanned scrams with complications samples as defined in IP 71151–05.

b. Findings

No findings were identified.

.4 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator, Units 1 and 2, for the period from the first quarter of 2016 through the fourth quarter of 2016. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, and NUREG–1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC Integrated Inspection Reports for the period of January 1, 2016, through December 31, 2016, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

These inspections constituted two safety system functional failures samples 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 CAP

a. Inspection Scope

As 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 corrective action program at an appropriate threshold, adequate attention was being given to timely corrective actions, and adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action program as a result of the inspectors' observations; however, they are not discussed in 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.

b. Findings

No findings were identified.

.2 Annual Follow-Up of Selected Issues: Adverse Condition Monitoring Program

a. Inspection Scope

The inspectors selected the following for in-depth review:

- Implementation of the adverse condition monitoring program (ACMP).

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for CAPs associated with ACMP assignments and other related condition reports:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;
- classification and prioritization of the resolution of the problem commensurate with safety significance;
- identification of the root and contributing causes of the problem;
- identification of corrective actions, which were appropriately focused to correct the problem;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

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

b. Findings

No findings were identified.

.3 Annual Follow-Up of Selected Issues: Trend in Offsite Power Issues

a. Inspection Scope

The inspectors selected the following condition reports for in-depth review:

- CAP 1545635; Adverse Trend in Offsite Power Challenges.

As appropriate, the inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of the extent of condition, generic implications, common cause, and previous occurrences;
- evaluation and disposition of operability/functionality/reportability issues;
- classification and prioritization of the resolution of the problem commensurate with safety significance;
- identification of the root and contributing causes of the problem;
- identification of corrective actions, which were appropriately focused to correct the problem;
- completion of corrective actions in a timely manner commensurate with the safety significance of the issue;
- effectiveness of corrective actions taken to preclude repetition; and
- evaluate applicability for operating experience and communicate applicable lessons learned to appropriate organizations.

The inspectors discussed the corrective actions and associated evaluations with licensee personnel.

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

b. Findings

No findings were identified.

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

.1 (Closed) Licensee Event Report 5000282/2016004–00: Missing Fire Barriers Between Fire Area 59 and 85 / Fire Hazard Analysis Drawings Do Not Match Boundary Description

a. Inspection Scope

The inspectors reviewed information provided by the licensee regarding the April 21, 2016, identification of fire barriers with unsealed combustible pathway penetrations between fire area (FA) 59 (Auxiliary Building Mezzanine Level Unit 1) and FA 85 (Holdup Tank Area/Demineralizer Area); and fire barrier penetration seals with non-fire rated materials between FA 68 (Unit 1 Annulus) and FA 60 (Auxiliary Building Operating Level Unit 1), FA 68 and 61A (Auxiliary Building Hatch Area), FA 72 (Unit 2 Annulus) and FA 75 (Auxiliary Building Operating Level Unit 2), and FA 72 and 61A.

Specifically, during walk downs to support a National Fire Protection Association (NFPA) 805 transition project, the licensee identified the above unanalyzed conditions (potential loss of train separation for safe shutdown equipment during a postulated fire in the area(s)) that were not in accordance with the license bases. Upon discovery of the conditions, the licensee documented the issues in CAP 1519659, implemented compensatory measures (hourly fire watches), performed an extent of condition review, and performed an apparent cause evaluation.

The inspectors reviewed the License Event Report (LER), apparent cause evaluation, fire protection program documents, immediate corrective actions, and longer term corrective actions. The inspectors also reviewed a Fire Protection Engineering Evaluation performed in April 2017, which determined that only postulated fires propagating from FA 85 to adjacent FAs 59 or 74 (Auxiliary Building Mezzanine Level Unit 2) could have prevented safe shutdown of Prairie Island Units 1 and 2 respectively. 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 Non-Cited Violation (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.

.2 (Closed) Licensed Event Report 5000282/2016006–00: 121 Motor Driven Cooling Water Pump Auto Start

a. Inspection Scope

The inspectors reviewed information provided by the licensee regarding the December 18, 2016, automatic start of the 121 Motor Driven Cooling Water Pump (MDCLP). Following an unexpected lockout of the Bus 2 in the substation with the 11 cooling water pump in service, the 11 cooling water pump tripped on loss of power and resulted in a low cooling water header pressure. The 121 MDCLP automatically started, as designed. Since low pressure actually existed in the cooling water header (valid actuation signal) the licensee submitted an LER for this event/condition based on 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in automatic actuation of an emergency service water system that does not normally run and serves as an ultimate heat sink.

The inspectors reviewed licensee CAP 1545369 which was generated as a result of the automatic actuation of the 121 MDCLP, the causal evaluation, and corrective actions. No issues were identified. Documents reviewed are listed in the Attachment to this report. This LER is closed.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/192, "Inspection of the Licensee's Interim Compensatory Measures Associated with the Open Phase Condition Design Vulnerabilities in Electric Power Systems"

a. Inspection Scope

The objective of this performance based Temporary Instruction (TI) is to verify implementation of interim compensatory measures associated with an Open Phase Condition (OPC) design vulnerability in electric power system for operating reactors. The inspectors conducted an inspection to determine if the licensee had implemented the following interim compensatory measures. These compensatory measures are to remain in place until permanent automatic detection and protection schemes are installed and declared operable for OPC design vulnerability. The inspectors verified the following: The licensee had identified and discussed with plant staff the lessons-learned from the OPC events at the US operating plants including the Byron station OPC event and its consequences. This includes conducting operator training for promptly diagnosing, recognizing consequences, and responding to an OPC event;

- The licensee had updated plant operating procedures to help operators promptly diagnose and respond to OPC events on off-site power sources credited for safe shutdown of the plant;
- The licensee had established and continue to implement periodic walkdown activities to inspect switchyard equipment such as insulators, disconnect switches, and transmission line and transformer connections associated with the offsite power circuits to detect a visible OPC; and
- The licensee had ensured that routine maintenance and testing activities on switchyard components have been implemented and maintained. As part of the maintenance and testing activities, the licensee assessed and managed plant risk in accordance with 10 CFR 50.65(a) (4) requirements.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 05000282/2016004–01; 05000306/2016004–01: Baffle Bolt Acceptance Criteria

a. Inspection Scope

During the 2016 In-service Inspection (ISI), the inspectors reviewed the analysis that demonstrated the design adequacy of the reactor pressure vessel internals baffle former bolting under design and licensing basis loading conditions. The inspectors identified an unresolved item involving using acceptance criteria for the analysis of the baffle former bolting different than what was reviewed and approved by the NRC.

During the follow-up inspection activities to the unresolved item, the licensee identified that baffle bolts yield and ultimate strength used for the top level of the baffle assembly were not in accordance with the design and licensing basis requirements. The licensee initiated CAP document 1548138, "Non-Conservative Stress Limits in BFB Analysis," dated January 20, 2017, to address the concern. The licensee performed Prompt

Operability Determination (POD) 1548138–01; dated January 27, 2017, to address the nonconforming condition. The inspectors reviewed the operability determination and no performance deficiencies were identified with respect to that evaluation. The inspectors did not address this licensee identified issue from an enforcement standpoint, because it was identified pursuant and incident to follow-up of NRC concerns resulting in the NRC finding below and would just be another example of that same finding.

Subsequently, the inspectors reviewed additional information provided by the licensee and identified the finding described below.

b. Findings

(1) Failure to Evaluate Changes to NRC–Approved Methodology

Introduction: The inspectors identified a Green finding and associated Severity Level IV violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a written evaluation which provided the bases for the determination that a change in the NRC–approved Westinghouse methodology referenced in the Updated Safety Analysis Report (USAR) for evaluating the acceptability of reactor pressure vessel internals baffle former bolting distributions did not require a license amendment.

Description: The Prairie Island USAR, Section 3.6, describes the reactor vessel internals and states, in part, "The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the incore instrumentation support structure… The major containment and support member of the reactor internals is the lower core support structure. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate, and the bottom support plate which is welded to the core barrel… Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core." The axial baffle and former plates which comprise the baffle former assembly are connected with baffle former bolts. The baffle former bolts hold together the baffle former assembly to maintain the fuel assembly structural integrity to ensure that the control rods insert, maintain a coolable core geometry, and ensure a core configuration that supports long-term reactor shutdown. The coolable geometry must be maintained throughout the postulated accident, including for long term core cooling.

Section 3.8 of the USAR incorporated by reference WCAP–15029–P–A, Revision 1, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999. This WCAP described the Westinghouse methodology for evaluating the acceptability of baffle-former barrel bolting distributions under faulted conditions. This WCAP methodology had been accepted by the NRC via a Safety Evaluation Report (SER) included in WCAP–15029–P–A. The SER stated that the use of the WCAP–15029 methodology guidance is acceptable in accordance with the following limitations:

- The bolt loading should be determined by analysis for a class of plants and a specific break;
- The noding to be used in the representation of the loading is demonstrated to be adequate by performing nodalization sensitivity studies or by some other acceptable methodology;

- The methodology should not be used to assess existing bolting without demonstration of adequate conservatism in projected bolting material properties (i.e., yield and ultimate strength) to ensure that sufficient ductility is present in existing irradiated stainless steel bolting materials; and
- The use of the methodology for existing irradiated stainless steel bolting should account for limitation in available ISI methods with regard to the probability of detection characteristics.

The discussion section in the SER portion of WCAP–15029–P–A provided the NRC staff basis for the establishment of Limitations Nos. 3 and 4 which stated, in part, "The ISI methods used for inspecting the baffle bolts will have certain sensitivity and probability of detection characteristics. In using irradiated tensile strength for calculating safety margins, the bolt load-bearing cross-section area should be reduced based upon the sensitivity of the ISI methods and the potential for degraded bolts to escape detection by ISI. In addition, since the available data for irradiated stainless steel indicates that the ductility (and presumably the fracture toughness) of irradiated bolts is severely degraded, a fracture mechanics approach would be necessary to demonstrate that the degraded bolts exhibit adequate toughness with a postulated flaw size undetected by ISI… The determination of adequate conservatism of bolting material properties and characteristics, resides with the licensee in applying appropriate stress and deflection limits to the baffle assembly under faulted conditions. This determination should include consideration of the subject plant's historical operating conditions and current licensing bases. The determination should be based on conservative yield strength and ultimate strength values representative of the plant's existing bolting material properties."

The inspectors identified that, as a result of degraded baffle former bolting that the licensee planned to leave in place, the licensee evaluated the baffle former bolting, and in so doing, used acceptance limits different than what was reviewed and approved by the NRC. The inspectors reviewed WCAP 17586–P, "Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2," Revision 0, which provides analyses of baffle former bolting failure patterns in accordance with USAR methodology of WCAP–15029–P–A. The stress allowable used in WCAP 17586–P was based on American Society for Mechanical Engineers Section III, Appendix F, specifically minimum of (0.9 times Su (ultimate strength of the baffle bolt material), maximum of (0.67 times Su, Sy + 1/3 (Su – Sy))). The WCAP–15030–NP–A was the non-proprietary version of WCAP–15029–P–A. In WCAP–15030–NP–A, Section 4.3.2 stated that the stress allowable for primary membrane and bending of irradiated bolt material is taken to 0.9 times Sy (yield strength of the baffle bolt material) for the faulted load condition. The use of the stress allowable in WCAP 17586–P in lieu of NRC approved stress allowable in WCAP–15030–NP–A, Section 4.3, was a change from what was incorporated in the USAR by reference to the WCAP.

The inspectors reviewed the licensee's 50.59 Screening No. 4443, "Determination of Acceptable Baffle-Barrel Bolting," dated January 24, 2013, and determined that the change to use ASME Section III, Appendix F, acceptance criteria was not explicitly addressed. Further, Limitations Nos. 3 and 4 in the SER portion of WCAP–15029–P–A, specifically with respect to Prairie Island's site specific parameters including historical operating conditions and material data, were also not explicitly addressed in the screening or WCAP 17586–P, Revision 0. In particular, the licensee did not evaluate whether sufficient ductility was present in existing site specific baffle bolting material properties. The screening also left Limitation No. 4, in particular, whether ISI methods established adequate safety margin based on the probability of detection of indications, to be addressed as part of an inspection specific engineering evaluation. The inspector was unable to identify the existence of such an evaluation. Hence, the licensee failed to perform a written evaluation in accordance with 10 CFR 50.59 (d)(1) that addressed whether the above change in the evaluation methodology required a license amendment. The licensee performed an operability determination as discussed above, POD 1548138–01 and determined the baffle bolts were operable. The inspectors reviewed the operability determination and no performance deficiencies were identified in this determination.

In response to the inspector's concern, the licensee initiated CAP documents 01539487, "Documentation Missing in 50.59 Screening 4443," dated October 26, 2016, 1552331, "BFB Screen Referenced Eval for SER Limitation 4 Non-Existent," dated March 6, 2017, and 1552314, "BFB Screening Lacks Documentation for SER Limitation 3," dated March 6, 2017.

Analysis: The inspectors determined that the licensee's failure to perform a written evaluation, providing the bases for the determination that a change in the NRC–approved Westinghouse methodology for evaluating the acceptability of baffle former bolting distributions did not require a license amendment, was a performance deficiency.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significant Determination Process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. The performance deficiency was determined to be more-than-minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, compliance with the NRC–approved methodology of WCAP–15029–P–A ensured the baffle former assembly maintains its structural integrity, avoiding a failure or excessive deflection of the baffle plates, and hence the primary concern of ensuring the emergency core cooling system can continue to perform its design function of cooling the reactor core.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," for the Mitigating Systems cornerstone. The finding screened as having very-low safety significance (Green) because the emergency core cooling system maintained its operability, specifically with respect to performing its safety function of ensuring adequate core cooling. As such, the finding corresponded to a Severity Level IV violation in accordance with Example 6.1.d.2 of the NRC Enforcement Policy. The inspectors did not identify a cross cutting aspect because the performance deficiency was from 2013, and hence the issue did not represent current performance.

Enforcement: Title 10 CFR 50.59, "Changes, Tests and Experiments," Paragraph d(1) states, the licensee shall maintain records of changes in the facility, made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change does not require a license amendment pursuant to Paragraph (c)(2) of this section. Paragraph (c)(1) states, a license may make changes in the facility as described in the final safety analysis report (as updated) without obtaining a licensee amendment only if a change to technical specifications (TS) is not required or the change does not meet any of the criteria in (c)(2).

The USAR, Section 3.8, incorporates by reference WCAP–15029–P–A, Revision 1. This WCAP described the NRC–approved Westinghouse methodology for evaluating the acceptability of baffle-former barrel bolting distributions.

Contrary to the above, on December 24, 2013, the licensee changed the facility as described in USAR Section 3.8, specifically with respect to the methodology it employed to evaluate the acceptability of baffle former bolting distribution, and did not include a written evaluation which provided the bases for determining the change did not require a license amendment. In particular, the licensee did not address a change in acceptance criteria or in limitations described in WCAP–15029–P–A.

Because this finding is of very-low safety significance and was entered into the CAP as 1539487, “Documentation Missing in 50.59 Screening 4443,” dated October 26, 2016, 1552331, “BFB Screen Referenced Eval for SER Limitation 4 Non-Existent,” dated March 6, 2017, and 1552314, “BFB Screening Lacks Documentation for SER Limitation 3,” dated March 6, 2017, this violation is being treated as a Non-Cited Violation in accordance with Section 2.3.2 of the Enforcement Policy.
(NCV 05000282/2017001–01; 05000306/2017001–01; Failure to Evaluate Changes to NRC Approved Methodology)

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 6, 2017, the inspectors presented the overall inspection results to Mr. S. Northard, Site Vice President, and other members of the licensee 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:

- On March 3, 2017, the inspectors presented the inspection results of the Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation inspections to Mr. S. Northard, Site Vice President. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.
- On March 6, 2017, the inspector presented the results to the Site Vice President, Mr. S. Northard, and other members of the licensee’s staff. The licensee personnel acknowledged the inspection results presented.

4OA7 Licensee-Identified Violations

The following violation was identified by the licensee. The NRC is not taking enforcement action for these violations because they 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 Part 50, Appendix R, Section III.G.2 requires, in part, that "where cables or equipment of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one means of ensuring that one of the redundant trains is free of fire damage shall be provided."

  Contrary to the above, up until April 21, 2016, the licensee failed to ensure that where cables or equipment of redundant trains of systems necessary to achieve and maintain hot shutdown conditions were located within the same fire area outside of primary containment, one means of ensuring that one of the redundant trains is free of fire damage was provided. Specifically, the requirement was to provide "separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3–hour rating." However, fire barriers with unsealed combustible pathway penetrations existed between FA 85 (Holdup Tank Area/Demineralizer Area) and adjacent FAs 59 (Auxiliary Building Mezzanine Level Unit 1) and FA 74 (Auxiliary Building Mezzanine Level Unit 2) for Units 1 and 2 respectively.

  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 identified the violation during the scheduled transition to 10 CFR 50.48(c); (2) the licensee had established adequate compensatory measures 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 provided the Fire PRA Multi-Compartment Analysis Notebook (FPRA–PI–MCA) for review, and concluded that this issue was not associated with a finding of high safety significance. An NRC Region III Senior Reactor Analyst (SRA) reviewed the evaluation and discussed it with licensee staff. The evaluation documents the results of fire modeling that concludes the fire scenarios screen from further consideration because a damaging hot gas layer that could affect both compartments is not generated. The SRA concluded that the licensee's result was reasonable and that the finding was less than Red and eligible for Enforcement Discretion.

In addition, the licensee entered this issue into their CAP as 1519659. As a result, the inspectors concluded that the violation met all four criteria established by Section 9.1(a) of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues 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

S. Northard, Site Vice President
T. Conboy, Director of Site Operations
W. Paulhardt, Plant Manager
S. Sharp, Director of Performance Improvement
J. Bjorseth, Engineering Director
H. Butterworth, Business Support Manager
J. Boesch, Maintenance Manager
J. Kivi, Regulatory Affairs Manager
T. Borgen, Operations Manager
A. Chladil, Nuclear Oversight Manager
B. Boyer, Radiation Protection Manager
B. Carberry, Emergency Preparedness Manager
B. Truckenmiller, Chemistry & Environmental Manager
D. Lapcinski, Assistant Operations Manager
S. Martin, Human and Organizational Performance Manager
S. Lappegaard, Production Planning Manager
P. Johnson, Regulatory Affairs Analyst
F. Sienczak, Senior Licensing Engineer
P. Wildenborg, Health Physicist
D. Allison, RP Supervisor—Monticello
G. Sherwood, Fleet Area Engineering Programs Manager
D. Hanson, Manager Programs Engineering
T. Dendinger, Manager Design Engineering
T. Downing, ISI Program Engineer
L. Drenth, Materials and Inspection Engineer

U.S. Nuclear Regulatory Commission

K. Riemer, Chief, Reactor Projects Branch 2
R. Kuntz, Project Manager, Office of Nuclear Reactor Regulation

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000282/2017001–01; 05000306/2017001–01	NCV	Failure to Evaluate Changes to NRC Approved Methodology (Section 4OA5.2(1))

Closed

05000282/2016004–01 05000306/2016004–01	URI	Baffle Former Bolting Acceptance Criteria (Section 4OA5.2)
05000282/2017001–01; 05000306/2017001–01	NCV	Failure to Evaluate Changes to NRC Approved Methodology (Section 4OA5.2(1))
05000282/2016004–00; 05000306/2016004–00	LER	Missing Fire Barriers Between Fire Area (FA) 59 and 85 / Fire Hazard Analysis Drawings Do Not Match Boundary Description (Section 4OA3.1)
05000282/2016006–00; 05000306/2016006–00	LER	121 Motor Driven Cooling Water Pump Auto Start (Section 4OA3.2)
2515/192	TI	Inspection of the Licensee’s Interim Compensatory Measures Associated with the Open Phase Condition Design Vulnerabilities within Electric Power Systems (Section 4OA5.1)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- AB-2; Tornado/Severe Thunderstorm/High Winds; Revision 42
- CAP 1550564; 8H14 B Phase Control Panel Top Cover Not Installed Properly; February 14, 2017

1R04 Equipment Alignment

- CAP 1549703; D5 and D6 Lube Oil Make-Up Tanks Need to be Filled; February 6, 2017
- CAP 1549689; D5 Engine #2–Very Minor Fuel Oil Leak at Return Fitting; February 6, 2017
- CAP 1549700; 21 D5 FO STG TNK Strainer High DP; February 6, 2017
- CAP 1549698; 23 D5 FO STG TNK Strainer High Alarm; February 6, 2017
- CAP 1547431; NRC Question Concerning Combustible Loading in D5/D6 Rooms; January 13, 2017
- XH–1–7; Flow Diagram Reactor Coolant System Unit 1; Revision 92
- CAP 1547908; Housekeeping Issue From D1 DSL Gen L/O Strn Drain Line; January 18, 2017
- B38A; Unit 1 Diesel Generators; Revision 12
- WO 448111–04; Repair 11 Cooling Water Strain ER Drain Piping, Pin Hole Leak; January 12, 2017
- B37A; Plant Normal Ventilation System; Revision 15
- CAP 1546215; Door 159 Latch Failure Prevented Isolation of 121 CR Chiller; January 2, 2017
- CAP 1546323; 121 CR Chiller Work Does Not Have a Complex Work Plan; January 4, 2017
- CAP 1312803; 123 CR Humidifier Blower Motor Seized; November 10, 2014
- CAP 1546843; NRC Question Regarding 123 CR Humidifier; January 7, 2017
- C37.11; Chilled Water Safeguard System Operation; Revision 29
- CAP 1479560; FBD Completed in Error; 123 CR Humidifier Blower Motor Seize; May 18, 2015
- CAP 154853; Restoration of 121 CR Chiller Delayed; January 7, 2017
- CAP 1546851; No CAP Available for CL–20–43 (121 CR Chlr Out Line Drain);
- January 7, 2017
- SP 2412; 21 Battery Charger Load Test; Revision 7
- CAP 1546260; 21 Battery Inspection; January 3, 2017
- NR–39603–4; Flow Diagrams—Cooling & Chilled Water Systems & Fire Prot. For Vent Filters in Aux. & Containment Bldgs.; Revision 80
- WO 446385–01; Determ 123 CR Humidifier Blower Motor and Re-Power Circuit; June 13, 2012

1R05 Fire Protection

- F5 Appendix E; Fire Protection Safe Shutdown Analysis Summary; Revision 17
- F5 Appendix D; Impact of Fire Outside Control/Relay Room Zone 75 (Fire Area 41A); Revision 14
- F5 Appendix F; Fire Hazard Analysis; Revision 32
- F5 Appendix K; Fire Protection Systems Functional Requirements; Revision 22
- CAP 1546262; Oily Rags Not Stored IAW Combustible Control Procedure; January 3, 2017

- FP–PE–CC–01; Combustible Control; Revision 3

1R06 Flood Protection

- ENG-ME-203; Evaluation of Screenhouse Internal Flooding at El. 670'; Revision 2
- H36; Plant Flooding; Revision 10
- 5AWI 8.9.0; Internal Flooding Drainage Control; Revision 19
- AB-4; Flood; Revision 50

1R11 Licensed Operator Requalification Program

- SEG 96116ST-0812; Cycle 16H Simulator Session #2; Revision 0
- CAP 1549409; Simulator Event Inadvertently Entered Early During Training; February 2, 2017

1R12 Maintenance Effectiveness

- H66; High Energy Line Break Program; Revision 4
- SP 1412; 11 Battery Charger Load Test; Revision 0
- ENG–ME–716; Center Aisle Input For Turbine Building Gothic Analyses; Revision 1
- H32; Predictive Maintenance Program; Revision 1
- H24; Maintenance Rule Program; Revision 21
- CAP 1549882; 2N42 Power Range High Voltage Power Supply Failed; February 7, 2017
- CAP 1493179; Safety Related NBFD65NR Relays Surpassed the Qualified Life; September 14, 2015

1R13 Maintenance Risk Assessment and Emergent Work Control

- SP 1307; D2 Diesel Generator 6 Month Fast Start Test; Revision 50
- CAP 1548241; During D2 Run the Local Tachometer Did Not Read Correctly; January 21, 2017
- SP 1306; D2 Diesel Generator Relay Functional Test; Revision 17
- CAP 1548131; D2 PMT Lessons to Learn; January 20, 2017
- CAP 1548359; Letdown Pressure CV Erratic After Removal of Second Orifice; January 23, 2017
- CAP 1550306; Further Clarification Needed for ETAP Sensitivity Analysis; February 12, 2017
- 2C1.4; Unit 2 Power Operation; Revision 58
- WO 557702–03; B SG Water Level Oscillating Excessively With B FRV in Auto.; January 24, 2017
- CAP 1549287; NRC Prompting Was Required for OPS Status Notes; February 1, 2017
- GEN–PI–026; Appendix C412A Part 1-Compliance Assessment Summary; Revision 4
- CAP 1552940; 480V Bus 211 and 212 Temperature Margin During PM; March 14, 2017
- 1C12.1; Letdown, Charging, and Seal Water Injection – Unit 1; Revision 27
- CAP 1552700; CV-31203, 11 Letdown Pressure Control Valve Oscillating; March 10, 2017
- CAP 1554214; Found 12 Inverter CB 401 Tripped; March 28, 2017
- 1C20.8; Instrument AC Distribution System; Revision 26
- 1C20.8 AOP1; Abnormal Operation, Instrument AC Inverters; Revision 12

1R15 Operability Determinations and Functionality Assessments

- Framatome Report No. 51–5003385–00; Evaluation of Point Beach Unit 2 Baffle Bolt Tensile Test Result; September 1999

- CAP 1547796; NRC Question—Scaffold Build Above BUS 111 No CAP Was Generated to Answer NRC Question on the Scaffold Build Above BUS 111; January 17, 2017
- CAP 1549161; Improper Ladder Storage in Plant Screenhouse; January 31, 2017
- CAP 1548225; Door 225 Not Sealing Fully When Closed; January 21, 2017
- CAP 1547686; Backup Air Supply to 121 CR Chiller <4000 psig For Eight Days; January 16, 2017
- CAP 1546496; Scaffold too Close to 122 CR Pac Filter 069–242; January 5, 2017
- CAP 1547818; D2 EDG Gear-Train Backlash Measurement Issue; January 18, 2017
- CAP 1548243; D2 Standby LO Pump Tripped Off—Keep Warm Failure; January 21, 2017
- CAP 1546123; D2 Standby Jacket Coolant Pump Not Running with D2 Shutdown; December 29, 2016
- CAP 1547514; 1T2817A, 11 FW PMP Mtr IBRG Temp, Reading 179 Deg F; January 14, 2017
- CAP 1546496; Scaffold in 122 CR PAC Filter 069–242; January 5, 2017
- 2C28.2; Unit 2 Feedwater System; Revision 37
- NF–39605–3; Flow Diagram—Aux. Bldg. Hot Water Heating System; Revision 79
- H28; Control Room Habitability Program; Revision 13
- SP 1449; Tracer Gas Test of Control Room; Revision 3
- D80; Scaffolding, Ladders and Cable Trays Platforms; Revision 33
- 1C20.7; D1/D2 Diesel Generators; Revision 48
- CAP 1548783; Preliminary Battery Calculation Results; January 27, 2017

1R18 Plant Modifications

- EC 24411; Unit 2 BOP SER and Control Room Annunciator Replacement; Revision 0
- 50.59 Screening 5153; U1 and U2 BOP and NSSS Annunciator Replacement; Revision 0
- C47.0 AOP1; Annunciator System Malfunction; Revision 18

1R19 Post-Maintenance Testing

- CAP 1546842; 11 CNTMT FCU Fan Inbrd Vibr Det in Alert Range; January 7, 2017
- CAP 1288985; Replace Power Range Digital Indicators; June 3, 2011
- CAP 1550005; Subject: 2N42–N1302 Failed When Drawer Powered Up; February 8, 2017
- CAP 1549882; 2N42 Power Range High Voltage Power Supply Failed; February 7, 2017
- CAP 1287932; 2N42 Detector A Went Blank and Then Came Back High; May 27, 2011
- WO 559396–01; Replace 2N42 High Voltage Power Supply; February 8, 2017
- SP 2318.3; NIS Power Range Channel Calibration; February 7, 2017
- WO 543179–01; 2N42 Detector B Display Failing; February 7, 2017
- NSP 4–CL–28–2; Flow Diagram Cooling Water—Screenhouse Unit 1 & Unit 2; Revision 93
- CAP 1547825; B SG Water Level Oscillating Excessively With B FRV in Auto; January 18, 2017
- WO 448111–04; Repair 11 Cooling Water Strain ER Drain Piping, PIN Hole Leak; January 12, 2017
- H27; Control of Steam Exclusion Boundaries; Revision 16
- CAP 1549947; Subject: 2N42 Work Caused 47513–0107; February 8, 2017
- WP 520775–06; ELEC: Perform PMT for Relay Replacement; January 21, 2017
- CAP 1548108; Unexpected Alarms During WO 520775–06; January 20, 2017

1R22 Surveillance Testing

- SP 2089A; Train A RHR Pump and Suction Valve from the RWST Quarterly Test; Revision 30
- H10.1; ASME Inservice Testing Program; Revision 38

- SP 1101; 12 Motor-Driven Auxiliary Feedwater Pump Quarterly Flow and Valve Test; Revision 66
- CAP 1551794; MV–32382 12 MD AFW Pump Discharge to 12 ASG MV Noise; February 28, 2017
- SP 2003; Analog Protection Functional Test; Revision 76
- SP 2002A; Analog Protection System Calibration; Revision 47
- 5 AWI 3.12.4; Post-Maintenance Testing; Revision 23
- CAP 1552926; 2TM-402V Found Not Responding Correctly to Changing Input; March 13, 2017
- CAP 1550117; SP 2093, D5 Diesel Generator Monthly Slow Start (WO#547276); February 9, 2017
- CAP 1547181; Build Up of Oil on Brackets Outboard of D5 Eng 1; January 11, 2017
- CAP 1546272; D5 Wire Shielding Broke; January 3, 2017
- CAP 1546258; U2, D5, Engine #1 Crankcase Pressure High; January 3, 2017
- WO 547276–01; SP 2093 D5 Diesel Generator Monthly Slow Start; February 6, 2017
- SP 2093; D5 Diesel Generator Monthly Slow Start Test; February 6, 2017
- CAP 1549163; 2CL–39–I Remote Handwheel Sheath Interaction With Structural; January 31, 2017
- CAP 1546882; D1 Day Tank Level Low During SP 1093; January 8, 2017

1EP6 Drill Evaluation

- Prairie Island Nuclear Generating Plant Emergency Plan Drill March 2, 2017; Revision 2
- CAP 1551964; 2017 Mar Drill–Communication Issues in TSC LL; March 2, 2017
- CAP 1552013; PI EP Drill; March 2, 2017
- CAP 1552015; EP Drill: Pagers Alarmed Well After Drill Started; March 2, 2017
- CAP 1552027; EP Drill: Dosimetry Equipment Failures; March 2, 2017
- CAP 1552035; EP Drill: DC Not Listed as Assembly Point in PINGP 573; March 2, 2017
- CAP 1552042; EP Drill March 2, 2017: Incomplete or Erroneous TSC Communications; March 2, 2017
- CAP 1552044; ERO Drill Critique—Early Release of Maintenance Personnel; March 2, 2017
- CAP 1552047; Drill February 3, 2017 Relief Valve RS–21–1 Scenario Encountered Problem; March 2, 2017
- CAP 1552049; Classification During Emergency Plan Drill; March 2, 2017
- CAP 1552050; EP Drill—Erroneous SG Rad Levels Comm During Drill; March 2, 2017

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

- Prairie Island Response to IE Bulletin 79–19; September 21, 1979
- Satellite RCA Area #19; Radiological Survey; February 13, 2017
- Satellite RCA Area #19; Container Inventory and Inspections; February 13, 2017
- C21.1.3.7; Spent Resin; Revision 18
- D59; Process Control Program; Revision 11
- D20.20; Sluicing Resin from 121 ADT IX to a Resin Shipping Liner; Revision 21
- 5AWI 13.1.0; Radioactive Material Packaging and Shipment; Revision 4
- RPIP 1302; Packaging of Radioactive Material for Shipment; Revision 9
- RPIP 1320; Monitoring of Radwaste in Interim Storage; Revision 9
- FP–RP–RW–02; Radioactive Shipping Procedure; Revision 18
- CAP 1415642; Barrel Yard Housekeeping; January 22, 2014
- CAP 1441158; Weight of Sealand Containers Estimated Versus Measured; July 31, 2014

- CAP 1478121; Filter Drum Selected for Shipment Found Degraded; May 7, 2015
- CAP 1479625; Rad Shipping Software Error; May 18, 2015
- CAP 1506117; High Integrity Container Dewatering Complication; December 16, 2015
- CAP 1544298; Determine the Frequency of Container Internal Inspections; December 7, 2016
- CAP 1551916; Torque Values Not Included in Cask Shipping Documents; March 1, 2017
- CAP 1552002; NRC RP Inspector Request Concerning Radwaste Equipment; March 2, 2017
- Radioactive Waste Shipment; 15–05; Cartridge Filters; December 8, 2015
- Radioactive Waste Shipment; 16–027; Mixed Bed Resin; September 6, 2016
- Radioactive Waste Shipment; 16–029; Mixed Bed Resin; September 20, 2016
- Radioactive Material Shipment; 16–048; RCP Impeller; October 29, 2016
- L60733; 10CFR61 Resin Analysis; October 2, 2014
- L67668; 10CFR61 Resin Analysis; March 9, 2016
- Radionuclide Radioactive Waste Trending Data; 1984 to Present
- Hazardous Material Transportation Training Lesson Plans; Various Documents
- Hazardous Material Training Records; Various Records

4OA1 Performance Indicator Verification

- FP-R-PI-01; Preparation of NRC Performance Indicators; Revision 6
- FP-PA-PI-02; NRC/INPO/WANO Performance Indicator Reporting; Revision 12

4OA2 Identification and Resolution of Problems

- FP–PA–HU–05; Decision Making; September 26, 2016
- FP–OP–ACM–01; Adverse Condition Monitoring Plan Process; November 29, 2016
- FP–OP–OL–01; Operability/Functionality Determination; December 2, 2016
- CAP 1549046; NRC Question on Ops Adverse Condition Monitoring Plans; January 30, 2017
- CAP 1545643; Adverse Trend in Offsite Power Challenges; December 20, 2016

4OA3 Follow-Up of Events and Notices of Enforcement Discretion

- CAP 1519659; Missing Fire Barrier Between FA 59 and 85; April 21, 2016
- CAP 1506561; Appendix R MOV Hot Short Concern; December 21, 2015
- F5 Appendix F; Fire Hazards Analysis; Revision 32

4OA5 Other Activities

- Safety Evaluation by the Office of Nuclear Reactor Regulation WCAP-15029, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions; November 10, 1998
- Updated Safety Analysis Report Section 3.6; Revision 33
- Updated Safety Analysis Report Section 3.8; Revision 34
- Westinghouse, WCAP-15030-NP-A, Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions; March 2, 1999
- WCAP 17586-P; Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2; Revision 0
- Framatome Report No. 51-5003385-00; Evaluation of Point Beach Unit 2 Baffle Bolt Tensile Test Result; September 1999
- Prompt Operability Determination 1548138-01; January 27, 2017
- CAP 1548138; Non Conservative Stress Limits in BFB Analysis; January 20, 2017 [NRC-Identified]

- CAP 1539487, “Documentation Missing in 50.59 Screening 4443,” October 26, 2016 [NRC-Identified]
- CAP 1552331, “BFB Screen Referenced Eval for SER Limitation 4 Non-Existent”, March 6, 2017 [NRC-Identified]
- CAP 1552314, “BFB Screening Lacks Documentation for SER Limitation 3”, March 6, 2017
- CAP 1324369-01; Design Vulnerability on 4.16KV Bus DV/UV Scheme; Revision 2
- PM 4910 Thermographic Inspection of Prairie Island Components; Revision 7

4OA7 Licensee-Identified Violations

- FPRA–PI–MCA–1.0; Multi-Compartment Analysis Notebook; Revision 1
- FPEE–12–006; Fire Area 85 Boundaries and F5 Appendix K Barriers; Revision 2

LIST OF ACRONYMS USED

AC	Alternating Current
ACMP	Adverse Condition Monitoring Plan
ADAMS	Agencywide Document Access Management System
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
EC	Engineering Change
EDG	Emergency Diesel Generator
EP	Emergency Plan
FA	Fire Area
IMC	Inspection Manual Chapter
IPEEE	Individual Plant Examination of External Events
ISI	In-service Inspection
IST	In-service Test
kV	Kilovolt
LER	Licensee Event Report
MDCLP	Motor Driven Cooling Water Pump
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
OPC	Open Phase Condition
PI	Performance Indicator
PM	Planned Maintenance
POD	Prompt Operability Determination
SDP	Significance Determination Process
SER	Safety Evaluation Report
SRA	Senior Reactor Analyst
SSCs	Structures, Systems, and Components
TI	Temporary Instruction
TS	Technical Specification
USAR	Updated Safety Analysis Report
WO	Work Order