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

February 27, 2014

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

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2, STEAM GENERATOR REPLACEMENT INSPECTION REPORT 05000306/2013011

Dear Mr. Davison:

On January 16, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed a Steam Generator Replacement Inspection at your Prairie Island Nuclear Generating Plant, Unit 2. The enclosed inspection report documents the inspection results which were discussed on January 16, 2014, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, two NRC-identified findings of very low safety significance were identified. One of the findings involved a violation of NRC requirements. However, because of its very low safety significance and because the issue was entered into your Corrective Action Program, the NRC is treating this issue as a Non-Cited Violation (NCV) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

If you contest the subject or severity of any NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant.

If you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Prairie Island Nuclear Generating Plant.

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

Sincerely,

/RA/

David E. Hills, Branch Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-306
License No. DPR-60

Enclosure:
Inspection Report 05000306/2013011
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

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| Docket No.: | 50-306 |
| License No.: | DPR-60 |
| Report No: | 05000306/2013011(DRS) |
| Licensee: | Northern States Power Company, Minnesota |
| Facility: | Prairie Island Nuclear Generating Plant, Unit 2 |
| Location: | Welch, MN |
| Dates: | August, 7, 2013, through January 16, 2014 |
| Inspectors: | A. Shaikh, Senior Reactor Inspector (Lead)<br>J. Bozga, Reactor Inspector<br>N. Egan, Reactor Inspector<br>K. Stoedter, Senior Resident Inspector<br>E. Sanchez, Acting Resident Inspector<br>M. Phalen, Senior Reactor Inspector |
| Approved by: | David E. Hills, Chief<br>Engineering Branch 1<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

Inspection Report 05000306/2013011(DRS); 08/7/2013 – 01/16/2014; Prairie Island Nuclear Generating Plant, Unit 2; Steam Generator Replacement Inspection.

This report covers a five month announced infrequently performed inspection on steam generator replacements. The inspection was conducted by Region III based engineering, radiological, and security inspectors and the site resident inspectors. Two findings were identified by the inspectors. One of the findings was considered a Non-Cited Violation (NCV) of NRC regulations and the other finding did not have an associated violation of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be (Green) or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

### A. NRC-Identified and Self-Revealed Findings

**Cornerstone: Initiating Events**

Green. The inspectors identified a finding of very low safety significance (Green) involving the licensee's failure to meet the requirements of the American Institute of Steel Construction (AISC) specification. Specifically, the licensee did not use the specified minimum yield strength of the outside lift system (OLS) girder material to establish an appropriate factor of safety to qualify the allowable loads that can be safely handled by the OLS girder. The AISC factor of safety to failure ensured the OLS girder would maintain structural integrity (no permanent deformation or structural failure) when subjected to the applied loads (lifted load, wind load, design basis earthquake load). This issue was entered into the licensee's Corrective Action Program (CAP) as CAP 1404203, "OLS calculation used actual material strength rather than ASTM." The licensee performed a functionality assessment to demonstrate that there was reasonable assurance the OLS girder remained capable of performing its intended design functions.

The inspectors determined the finding to be more than minor because the finding was associated with the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown. Specifically, the load handling reliability of the OLS girder inherently decreased when the AISC requirements were not met. The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 -- Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown (defueled) conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that none of the conditions constituting a loss of control were met as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRS and BWRS," for this finding and no Phase II or Phase III analysis was required. Therefore, the inspectors determined that this finding was of very low safety significance. No violation of regulatory requirements is associated with this finding. The inspectors identified that there was a Human Performance, Design Margin (H.6) cross-cutting aspect associated with this finding for the licensee failure to ensure the OLS girder reflected the intended design margins (Section 4OA5.2).

**Cornerstone: Barrier Integrity**

Green. The inspectors identified a finding of very low safety significance and associated NCV of Title 10 of the Code of Federal Regulations Part 50, Appendix B, Criterion III, "Design Control," for the failure to provide adequate design control measures for the steam generator blowdown (SGBD) pipe supports 8D-2SGB-1A, 2-RBDH-5294, 2-RBDH-606, 2-RBDH-363, 2-RBDH-350, 2-RBDH-349, 2-RBDH-339, and 2-RBDH-358. Specifically the SGBD pipe supports design was non-conservative with respect to Class I requirements as defined in Updated Safety Analysis Report (USAR) Section 12, "Plant Structures and Shielding", and referenced specifications. The licensee documented the violation in its CAP as CAPs 1405404 and 1412225 and performed an evaluation to demonstrate that there was reasonable assurance that the SGBD pipe supports remained capable of performing their safety functions.

The inspectors determined the finding was more than minor because the finding adversely affected the barrier integrity cornerstone and the associated cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee's calculations were not sufficient to demonstrate that the pipe supports were capable of properly supporting SGBD piping and isolation valves during design basis events, and hence ensure containment integrity. The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, "The Significance Determination Process (SDP) for Findings At-Power," Appendix A, Exhibit 3 (Section B). The inspectors determined that this finding was very low safety significance (Green) because each of the screening questions was answered "no." Specifically, the SGBD pipe supports were subsequently determined to be capable of performing their safety function. The inspectors identified a Human Performance, Documentation (H.7) cross-cutting aspect associated with this finding for the licensee's failure to ensure complete, accurate, and, up-to-date design documentation. Specifically, the licensee failed to provide adequate oversight of design calculations and documentation of as-built conditions during the SGBD pipe support re-analysis conducted to support the steam generators replacement (Section 4OA5.1).

## B. Licensee-Identified Violations

No violations of significance were identified.

# REPORT DETAILS

## 4. OTHER ACTIVITIES

**Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity**

4OA5 Other Activities

.1 Design Changes and Modifications to Systems, Structures, and Components

a. Inspection Scope

The inspectors reviewed engineering changes associated with the replacement steam generators, secondary side piping, and steam generator support systems. During these reviews, the inspectors focused on key design aspects and modifications of the replacement steam generators and verified that changes to the facility as described in the Updated Safety Analysis Report (USAR) were reviewed and documented in accordance with 10 CFR 50.59. The inspectors used IP 71111.17, “Evaluation of Changes, Tests and Experiments, and Permanent Plant Modifications” as guidance, as suggested in IP 50001, to complete these reviews.

The inspectors reviewed design calculations associated with the design changes to the steam generator blowdown piping, external recirculation piping and main steam piping. The inspectors also reviewed the design specification and design stress report for the replacement steam generators. The inspectors reviewed design calculations associated with the changes to the steam generator vertical column supports. The inspectors performed walkdowns of select main steam pipe supports and steam generator vertical column supports.

b. Findings

Steam Generator Blowdown Pipe Support Anchorages Failure to Meet Design Requirements

Introduction: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to ensure the adequacy of the design of several safety-related Steam Generator Blowdown (SGBD) pipe supports. Specifically the design was non-conservative with respect to Class I requirements as defined in Updated Safety Analysis Report (USAR) Section 12, “Plant Structures and Shielding,” and referenced specifications.

Description: The SGBD pipe supports which are upstream of the SGBD isolation valves are required to be Class I per USAR, Table 12.2-1, “Structures, Systems, and Components Classification.” The USAR Section 12.2.1.1 defines Class 1 as “Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of substantial amounts of radioactivity, and those structures and components vital to safe shutdown and isolation of the reactor.” The USAR Sections 5.2.2.1.1 and 9.2.2 delineated that the safety-related function of the SGBD system was to isolate during a steam generator tube rupture event or main steam line break event and to maintain the containment boundary. The inspectors identified the following two representative examples in which the licensee failed to demonstrate that the SGBD pipe supports met Class 1 requirements:

1) The inspectors reviewed Calculation Book No. 225, “Unit 2 Reactor Building Steam Generator Blowdown System Pipe Rupture Restraints,” Revision 0. This calculation was to demonstrate Class 1 compliance for SGBD Pipe Support 8D-2SGB-1A. The inspectors identified that the calculation did not validate the load path and structural integrity of the anchor bolts and anchor plate, which support and are part of Pipe Support 8D-2SGB-1A. The licensee documented these deficiencies in CAP 1405404, "Legacy calculation has missing pages", dated November 7, 2013.

2) The inspectors reviewed Calculation No. 020781-02-PISGR-45DK-0031, "Design Calculation – Blowdown Support Analysis-SGR U2," Revision 4. This calculation was to demonstrate Class 1 compliance of the SGBD pipe supports affected by the replacement of the steam generators. Pipe Supports 2-RBDH-5294, 2-RBDH-606, 2-RBDH-363, 2-RBDH-350, 2-RBDH-349, 2-RBDH-339, and 2-RBDH-358 used anchor bolt allowables from Engineering Specification 3.2.1.8, “Specification for Concrete Expansion Anchors,” Revision 3. The allowable loads that the anchor bolts can withstand during design bases events are specified based on a required minimum spacing and minimum edge distance. The minimum spacing is defined as centerline-to-centerline distance between adjacent anchors. The minimum edge distance is defined as the distance between the centerline of an anchor and a free edge where no concrete exists. The inspectors asked the licensee whether the minimum spacing and edge distances were met for the anchor bolts that comprised the anchorage of the aforementioned supports. The licensee subsequently performed walkdowns and discovered that they had not met the minimum spacing and edge distance for the anchorage for each of the aforementioned supports and therefore, the licensee did not appropriately reduce the allowable load that the anchor bolts can withstand during a design basis event. The licensee documented these deficiencies in CAP 1412225 “Deficiency in Analysis of U2 Blowdown Supports,” dated December 22, 2013.

Although these were existing, older calculations, the licensee did not identify the deficiencies in the calculations during the use of these calculations while performing a re-analysis of the SGBD pipe supports for the steam generator replacement project. As a result of the inspectors’ concerns, the licensee performed an evaluation of the aforementioned SGBD pipe supports to address the various deficiencies and demonstrate that there was reasonable assurance that the pipe supports remained capable of performing their intended safety functions. The inspectors reviewed the licensee’s evaluation and concluded there was sufficient rationale to support the final conclusion.

<u>Analysis</u>: The inspectors determined the licensee’s failure to perform adequate evaluations to demonstrate Class I compliance of the SGBD pipe supports was contrary to the design control measures requirements per 10 CFR Part 50, Appendix B, and was a performance deficiency. Specifically, compliance with Class I requirements for the SGBD pipe supports ensures structural integrity of these pipe supports and associated piping during Class 1 design basis events. In accordance with IMC 0612, “Issue Screening,” Appendix B, the inspectors determined the performance deficiency affected the Barrier Integrity Cornerstone. The performance deficiency was determined to be more than minor, and a finding, because it adversely affected the associated cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee’s calculations were not sufficient to demonstrate that the pipe supports were capable of properly supporting SGBD piping and isolation valves during design basis events, and hence ensure containment integrity.

The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, "The Significance Determination Process (SDP) for Findings At-Power," Appendix A, Exhibit 3 (Section B). The inspectors determined that this finding was very low safety significance (Green) because each of the screening questions was answered "no." Specifically, the SGBD pipe supports were subsequently determined to be capable of performing their safety function. The inspectors identified a Human Performance, Documentation (H.7) cross-cutting aspect associated with this finding for the licensee's failure to ensure complete, accurate, and, up-to-date design documentation. Specifically, the licensee failed to provide adequate oversight of design calculations and documentation of as-built conditions during the SGBD pipe support re-analysis conducted to support the steam generators replacement.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above,

1) As of November 7, 2013, in Calculation Book No. 225, "Unit 2 Reactor Building Steam Generator Blowdown System Pipe Rupture Restraints," Revision 0, the licensee's design control measures failed to verify adequacy of the SGBD anchor plate and anchor bolts for SGBD Pipe Support 8D-2SGB-1A. Specifically, no calculation was performed to verify the anchor bolts and anchor plate could withstand the applied loads.

2) As of December 22, 2013, in Calculation No. 020781-02-PISGR-45DK-0031, "Design Calculation – Blowdown Support Analysis-SGR U2", Revision 4, the licensee's design control measures failed to ensure adequacy of the Pipe Supports 2-RBDH-5294, 2-RBDH-606, 2-RBDH-363, 2-RBDH-350, 2-RBDH-349, 2-RBDH-339, and 2-RBDH-358 design. Specifically, the anchor bolts allowables were not based on the spacing and edge distances that existed in the plant.

The licensee entered this violation into their Corrective Action Program as CAPs 1405404 and 1412225, and performed a corrective action of evaluating the SGBD pipe supports operability. The licensee determined that the SGBD were operable but nonconforming and the inspectors reviewed the licensee's evaluation and did not have any further questions. Because this violation was of very low safety significance (Green), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000306/2013011-01).

.2 Engineering Design, Modification, Testing, and Analysis Associated with Steam Generator Lifting and Rigging

a. Inspection Scope

The inspectors reviewed the adequacy of the lifting program for lifts of the steam generators inside and outside containment assuring that it was prepared in accordance with regulatory requirements, appropriate industrial codes, and standards; and verified that the lifting and rigging equipment, Outside Lift System (OLS), laydown areas, equipment hatch transfer system, containment polar crane and containment polar crane support structure were adequate to withstand the maximum anticipated loads to be lifted.

The inspectors reviewed engineering changes (EC) 16803 and EC 16804 that were associated with the rigging and lifting of the old and replacement steam generators inside and outside containment. The inspectors selected and reviewed samples of design specifications, corrective actions, change requests, and design calculations to confirm that the engineering changes were in compliance with applicable codes and standards. In addition, the inspectors reviewed inspection records for the containment polar crane and OLS that were done before and after each lift of the steam generator. The inspectors reviewed test records of the OLS for the maximum lifted load.

The inspectors reviewed the adequacy of the haul route evaluation and the documentation of haul route for load testing and transport of the steam generators. The inspectors verified that they had been prepared in accordance with regulatory requirements and appropriate industrial codes and standards. The inspectors also discussed the transport path load testing with the licensee's steam generator replacement project (SGRP) engineering personnel and performed a walkdown of the haul route.

b. Findings

Outside Lift System Girder Failure to Meet American Institute of Steel Construction Requirements

Introduction: The inspectors identified a finding of very low safety significance (Green) involving the licensee's failure to meet the requirements of the American Institute of Steel Construction (AISC) specification. Specifically, the licensee did not use the specified minimum yield strength of the outside lift system OLS girder material to qualify the allowable loads that can be safely handled by the OLS girder.

Description: The OLS was a non-safety-related structure that was erected during the Unit 2 Prairie Island Steam Generator Replacement to move parts of the old and replacements SGs inside and outside of containment. The OLS was comprised of a main horizontal steel girder supported at each end by vertical steel members.

Design Bases Calculation No. C-3218-12, Revision 2, "Outside Lift System Structural Analysis and Interface Loads" evaluated the OLS for the applied loads due to lifting the old and replacement steam generators concurrent with other design loads such as impact, dead load of OLS itself, and wind loads.

Design Basis Calculation No. PI-996-111-007-S01, "Analysis of Outside Lift System for DBE Loads," Revision 1 evaluated the OLS for the applied loads due to lifting the old and replacement steam generators concurrent with other design loads such as dead lead of OLS itself, and design basis earthquake loads.

In both calculations, the licensee used the AISC specification to establish the structural adequacy of the OLS steel girder. The AISC specification required the allowable stress to be based on the specified minimum yield strength of the material. However, the licensee used certified material test report strength, which is reflective of actual material yield strength of the girder material for the evaluation of OLS girder. The use of actual material yield strength did not ensure that the appropriate AISC factor of safety was established for the OLS girder. The AISC factor of safety ensures the OLS girder would maintain structural integrity (no permanent deformation or structural failure) when subjected to the applied loads (lifted load, wind load, design basis earthquake load). The load handling reliability of the OLS girder inherently decreased when the AISC requirements were not met.

The failure to comply with the AISC specification on safety factor and demonstrate structural integrity of the OLS reduced the confidence that a failure of the OLS would not occur. A failure of the OLS could result in a heavy load drop accident that challenges plant stability. This issue was entered into the licensee's Corrective Action Program as CAP 1404203, "OLS calculation used actual material strength rather than ASTM," dated October 30, 2013. As part of the licensee's corrective actions, the licensee performed a functionality assessment of the OLS girder for the noncompliance with AISC requirements. The licensee functionality assessment concluded that there was reasonable assurance that the OLS girder remained capable of performing its intended design functions under the loads applied during the steam generator lifts. The inspectors reviewed the licensee's functionality assessment and concluded that there was sufficient rationale to support the final conclusion.

Analysis: The inspectors determined the licensee's failure to meet AISC requirements for the OLS main support girder was a performance deficiency. In accordance with IMC 0612, "Issue Screening", Appendix B, the inspectors determined the performance deficiency was more than minor, and a finding, because the performance deficiency was associated with the Initiating Events Cornerstone attribute of design control and adversely affected the cornerstone objective to limit the likelihood of those events that upset the plant stability and challenge critical safety functions during shutdown, as well as power operations.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase I -- Initial Screening and Characterization of Findings," Table 3. Since the finding was associated with shutdown (defueled) conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined that none of the conditions constituting a loss of control were met as described in Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRS and BWRS," for this finding and no Phase II or Phase III analysis was required. Therefore, the inspectors determined that this finding was of very low safety significance. Specifically, the OLS girder remained capable of performing its intended design functions under the loads applied during the steam generator lifts. In addition, the OLS was load tested before its use during the steam generator replacement.

The inspectors identified a Human Performance, Design Margin (H.6) cross-cutting aspect associated with this finding. Specifically, the licensee failed to ensure the OLS girder reflected the intended design margins.

Enforcement: No violation of regulatory requirements is associated with this finding (FIN 05000306/2013011-02).

.3. Radiation Protection Program Controls, Planning, and Preparation

a. Inspection Scope

The inspectors reviewed radiation protection program controls, planning, and preparation in the following areas utilizing applicable portions of Baseline Inspection Procedures (IP) 71124.01, 71124.02, 71124.03, 71124.04, and 71124.06 as guidance:

- As-Low-As-Is-Reasonably- Achievable (ALARA) planning;
- Dose estimates and dose tracking;
- Exposure controls including temporary shielding;
- Contamination controls;
- Radioactive material management;
- Radiological work plans and controls;
- Emergency contingencies;
- Project staffing and training plans; and
- Airborne radioactivity effluent controls

b. Findings

No findings were identified.

.4 Security Considerations Associated with Vital and Protected Area Barriers

a. Inspection Scope

The inspectors walked down areas associated with vital and protected area barriers that could be affected during replacement activities and concluded that they were operational and that the licensee was in compliance with security requirements.

b. Findings

No findings were identified.

.5 Controls and Plans to Minimize Adverse Impacts on Operating Unit and Common Systems

a. Inspection Scope

The inspectors also reviewed the licensee's plan to minimize impacts to the operating unit and any common systems during SGR activities. The inspectors walked down plant areas that had the potential to be impacted by steam generator replacement (SGR) activities.

These walk downs included the area surrounding the condensate storage tanks since these tanks were at increased risk of being punctured if a heavy load was dropped during lifting/rigging activities. The inspectors verified that the licensee had appropriately managed the configuration of the operating unit, including maintaining defense-in-depth strategies for safety-related equipment. The inspectors also verified that the licensee implemented the compensatory measures discussed in their plan for the specific areas reviewed. Lastly, the inspectors ensured that any increased risk to the operating unit or common systems was appropriately reflected in the daily operating unit risk-assessments as required by 10 CFR 50.65(a)(4).

.6 Welding and Non-Destructive Examination (NDE) Activities

a. Inspection scope

The inspectors reviewed the following welding and NDE activities associated with the steam generator replacement project to evaluate compliance with the American Society of Mechanical Engineers (ASME) Code Section XI and Section V requirements. These activities were inspected in accordance with IP 71111.08, "Inservice Inspection Activities."

- Special Procedures for welding and NDE on the replacement steam generators (RSGs) and connecting reactor coolant system (RCS) piping;
- Training and qualifications for personnel performing welding and NDE on RSGs and connecting RCS piping;
- NDE including radiography results and work packages for welds fabricated on RSGs and connecting RCS piping;
- Completion of pre-service NDE requirements for welds fabricated on RSGs and connecting RCS piping; and
- Completion of baseline eddy current examination (ECT) of new SG tubes.

b. Findings

No findings were identified.

.7 Activities Associated with Lifting and Rigging

a. Inspection Scope

The inspectors examined the SGRP lifting equipment necessary to perform steam generator rigging and transport; design evaluation and use of the OLS; and load drop protection. The inspectors performed direct observation of some of the heavy lifts performed both inside and outside containment to remove the old steam generators and install the new steam generators. The inspectors also verified that these activities were bound by the analyses and evaluations the licensee performed to support these activities. In addition, the inspectors reviewed crane and OLS personnel training certifications.

b. Findings

No findings were identified.

.8 Old and New Steam Generator Cutting, Movement and Reconnection

a. Inspection Scope

The inspectors observed various portions of the process of the old steam generators being cut and lifted from the steam generator vaults through the penetrations in the steel containment vessel and shield building to the hydraulic trailer transporter. The inspectors also observed various portions of the sequence of the replacement steam generators being transferred from the hydraulic trailer transporter, upended, lifted, and positioned into their respective steam generator vaults, and reconnected to the reactor coolant system piping. During these observations, the inspectors performed visual inspections of the OLS and the hydraulic trailer transporter.

b. Findings

No findings were identified.

.9 Steam Generator Hold-Down Bolts and Major Structural Modifications

a. Inspection Scope

There were no major structural modifications to the plant to facilitate the replacement of the steam generators for the inspectors to inspect or review. In addition, the old steam generator hold-down bolts were replaced with new hold-down bolts. The inspectors reviewed the evaluation associated with the replacement of the steam generator hold-down bolts.

b. Findings

No findings were identified.

.10 Operating Conditions throughout the Steam Generator Replacement Process

a. Inspection Scope

The inspectors routinely inspected the following activities as they occurred throughout this inspection period:

- Establishment of operating conditions including defueling, reactor coolant system drain-down, and system isolation and safety tagging/blocking;

- Implementation of radiation protection controls including: implementation of ALARA, radiological exposure, contamination, and airborne contamination controls planned for cutting, welding, and other activities including contaminated interference removal. Also, implementation of any special controls for contaminated tools and waste were reviewed;

- Implementation of controls for excluding foreign materials in the primary and secondary side of the SGs and in the reactor coolant system openings; and

- Installation, use and removal of temporary services directly related to steam generator replacement activities.

b. Findings

No findings were identified.

.11 Radiological Safety Plans for Disposal of Old Steam Generators

a. Inspection Scope

The inspectors reviewed the licensee's plans for disposal of old steam generators offsite and evaluated 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 for the following radioactive shipments:

- Shipping Record 13-047; Radioactive Waste Shipment Old Steam Generators Upper Section; November 16, 2013;
- Shipping Record 13-048; Radioactive Waste Shipment Old Steam Generators Upper Section; November 16, 2013;
- Shipping Record 13-052; Radioactive Waste Shipment Old Steam Generators Lower Section; December 26, 2013; and
- Shipping Record 13-053; Radioactive Waste Shipment Old Steam Generators Lower Section; December 26, 2013.

Additionally, the inspectors assessed whether the shipment placarding was consistent with the information in the shipping documentation.

b. Findings

No findings were identified.

.12. Steam Generators Post-Installation Verification and Testing

a. Inspection scope

The inspectors performed selective reviews and inspections, consistent with the safety significance, of the following areas: containment leak testing; post-installation inspections and verification; reactor coolant system leakage testing; calibration and testing of instrumentation for both the primary and secondary side systems affected by the SG replacement; and procedures for equipment performance testing required to confirm the design and to establish baseline measurements, during post-installation and power ascension. Specific activities observed included:

- SP 2070 – Reactor Coolant System Integrity Test;
- SP 2072.EH – Local Leak Rate Test of Equipment Hatch Containment System;
- SP 2132 – Unit 2 Personnel and Maintenance Airlock Door Leakage Test;
- SP 2750 – Unit 2 Containment Closeout Verification;

- ST 2RSG-HYDRO21 – 21 Replacement Steam Generator Hydrostatic Test; and
- ST 2RSG-HYDRO22 – 22 Replacement Steam Generator Hydrostatic Test.

b. Findings

No findings were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On January 16, 2014, the inspectors presented the inspection results to Mr. Kevin Davison and other members of the licensee staff. The licensee personnel acknowledged the inspection results presented and did not identify any proprietary content. The inspectors confirmed that all proprietary material reviewed during the inspection was either returned to the licensee staff or will be properly disposed of when no longer needed.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee

K. Davison, Site Vice President
J. Hallenbeck, Site Engineering Director
S. Sharp, Plant Manager
M. Murphy, Regulatory Affairs Director
S. Marty, SG Replacement Project Director
J. Wren, Program Engineering NDE Level III
B. Boyer, Radiation Protection Manager
S. Martin, Nuclear Oversight Manager
T. Downing, Site ISI Program
J. Hamilton, Security Manager
J. Ruttar, Operations Manager
D. Vincent, Regulatory Assurance

Nuclear Regulatory Commission

K. Stoedter, Senior Resident Inspector
E. Sanchez, Acting Resident Inspector
T. Wengert, Senior Project Manager, Office of Nuclear Reactor Regulation

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

| | | |
|---|---|---|
| 05000306/2013011-01 | NCV | Steam Generator Blowdown (SGBD) Pipe Support Anchorages Failure to Meet Design Requirements (Section 4OA5.1) |
| 05000306/2013011-02 | FIN | Outside Lift System (OIS) Girder Failure to Meet American Institute of Steel Construction (AISC) Requirements (Section 4OA5.2) |

Closed

| | | |
|---|---|---|
| 05000306/2013011-01 | NCV | Steam Generator Blowdown (SGBD) Pipe Support Anchorages Failure to Meet Design Requirements (Section 4OA5.1) |
| 05000306/2013011-02 | FIN | Outside Lift System (OLS) Girder Failure to Meet American Institute of Steel Construction (AISC) Requirements (Section 4OA5.2) |

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.

4OA5 Other Activities

- Evaluation of Outside Lifting System (OLS) Impacts on Plant Operation Prairie Island Nuclear Generating Plant, Revision 0, dated August 2013
- Procedure No. PINGP 1389, “Crane Daily-Frequent and Functional Inspection,” Revision 9
- Work Package C-07-16804-004, “OLS Inspection Checklist,” Attachment 13
- Work Package C-07-16804-006, “Rigging International Outside Lift System (OLS) Checklist,” Attachments 7A to 7D
- Work Package C-07-16804-007, “Rigging International Outside Lift System (OLS) Checklist,” Attachments 7A to 7D
- Prairie Island Work Order 455951-08, B&W Work Package C-07-16803-006, Step 270 and Attachment 6, Revision 0
- Calculation No. C-3218-14, “Outside Lift System (OLS) SGUP Rigging Analysis,” Revision 2
- Calculation No. C-3218-15, “Outside Lift System (OLS) SGLP Rigging Analysis,” Revision 2
- Calculation No. C09916.30, “230/20 Ton Polar Crane (S/N 9916) Evaluation Report For Planned Engineered Lifts," Revision 2
- EC 13181, SGR N-1 Unit 2 Polar Crane Part 21 Restoration, Revision 0
- EC 16803, “Steam Generator Rigging Inside Containment,” Revision 2
- EC16804, "Steam Generator Rigging and Transport Outside Containment," Revision 5
- 50.59 Screening No. 3623, “Unit 2 Polar Crane Part 21 Restoration,” Revision 1
- Calculation No. 020781- 02 - PISGR- 45DK- 0019, “Load Qualification of OSG Upper Part Lifting Lugs,” Revision 2
- Calculation No. 020781- 02 - PISGR- 45DK- 0003, “OSG Lower Part Cover Plate and Lifting Lugs," Revision 3
- Calculation No. 020781-02-PISGR-41DK-0015, “Evaluation of Polar Crane Ring Girder for Engineering Lift During Unit 2 SGR Prairie Island Unit 2 Steam Generator Replacement," Revision 1
- Report No. 020781-02-PISGR-41RA-0006, “Geotechnical Recommendations Technical Report, Unit 2 SGR, Prairie Island Nuclear Generating Plant," Revision 5.
- Drawing No. 3218-110, “Outside Lift System and Runway General Arrangement,” Sheet 3 of 4, Revision 5
- Drawing No. 3218-110, “Outside Lift System and Runway General Arrangement," Sheet 2 of 4, Revision 5
- Drawing No. 3218-110, “Outside Lift System and Runway General Arrangement,” Sheet 4 of 4, Revision 4
- Calculation No. C-3218-26, “Runway Towing Analysis” Revision 0
- Calculation No. C-3218-12, “Outside Lift System Structural Analysis and Interface Loads," Revision 2
- Drawing No. 3218-050, “Steam Generator Upper Parts," Revision 5, Sheet 1 and 2 of 2
- Drawing No. 3218-056, “Steam Generator Lower Parts," Revision 4, Sheet 1 and 2 of 2

- Crane Inspection Report No. C09916.31, “Crane Inspection Requirement Report for 230 Ton Polar Crane," dated December 13, 2012
- Calculation No. 020781-02-PISGR-52PO-0001-0376_R00, Structural Calculations Prairie Island Unit 2 – Steam Generator Replacement Runway Towing Analysis (C-3218-26 R0)
- Calculation No. 020781-02-PISGR-52PO-0001-0034, “Runway System Interface Loads (NTE)," Revision 2
- Calculation No. 020781-02-PISGR-52PO-0001-0461, “Runway System Structural Qualifications and Interface Loads (NTE)," Revision 2
- -Calculation No. 020781-02-PISGR-52PO-0001-0461, “Runway System Structural Qualifications and Interface Loads (NTE)," Revision 2, Addendum 1)
- Calculation No. PI-996-111-007-S01, “Analysis of Outside Lift System for DBE Loads," Revision 1
- ANSI/ASME NQA-1-2004, “Quality Assurance Requirements for Nuclear Facility Applications," Subpart 2.15
- Calculation No. C09915.71 and C09916.51, “Design Evaluation of the Main Hoist Drum Reductions (Third Reductions) Gear Sets for Class ‘A’ Service per Specifications” EOCI No. 61 (1961), CMAA No. 70 (1975) and CMAA No. 70 (2010) When Handling a 230 Ton Lifted Load," July 12, 2013
- CAP 1393691, “De-Rate Capacity of Polar Cranes," August 16, 2013
- CAP 1408752, “Unanalyzed Cribbing Configuration During HTS Removal," December 1, 2013
- CAP 1404203, “OLS Calculation Used Actual Material Strength Rather Than ASTM," October 30, 2013
- CAP 1398969, “Added Outside Lift Systems Inspections," September 27, 2013
- EC 13131, “Replacement of the Prairie Island Unit 2 Steam Generators with AREVA Model 56/19 Steam Generators," Revision 0
- EC 16807, “Steam Generator-Secondary Side Piping," Revision 4
- EC 16805, “Reactor Coolant System Piping and Steam Generator Supports," Revision 6
- Drawing No. 020781-02-PISGR-41DD-0047, “Reactor Building-Unit No.2 Steam Generator Steel Column Supports-Sections and Details," Revision 0
- Calculation No. PI-221-01.2 AND 01.3, "Unit 2 Main Steam Parts 1, 2 And 3 Stress Analysis," Revision 0B
- Calculation No. 09Q4836-CAL·006," “Unit 2 Main Steam Line Pipe Stress Analysis," Revision 0
- Calculation No. 09Q4836-CAL·006," “Unit 2 Main Steam Line Pipe Stress Analysis," Minor Revision 0A
- Calculation No. 020781-02-PISGR-45DK-0011, “Piping Stress Analysis for Blowdown System-Unit 2," Revision 1
- Calculation No. 020781-02-PISGR-45DK-0011, “Piping Stress Analysis for Blowdown System-Unit 2," Revision 3
- Calculation No. 020781-02-PISGR-45DK-0031, “Blowdown Support Analysis- SGR U2," Revision 1
- Calculation No. 020781-02-PISGR-45DK-0031, “Blowdown Support Analysis- SGR U2," Revision 4
- Calculation No. Book No. 225, “Unit 2 Reactor Building Steam Generator Blowdown System Pipe Rupture Restraints," Revision 0
- Calculation No. PI-221-05, “Stress Analysis of Steam Generator Blowdown System Piping – Parts 8C and 8D," Revision 2
- Calculation No. 321.4200-AD "Design of Pipe Hangers at Fuel Handling Floor and Operating Floor," Revision 00

- Calculation No. ENG-ME-114, "Reanalysis of 1-MSDH-24 Support," Revision 00
- Drawing No. XH-1106-3889, “Main Steam Pipe Support 2-MSH-53," Revision 0
- 50.59 Evaluation No. 1100, “Unit 2 Replacement Steam Generators – Stress and Fatigue Analysis Report," Revision 0
- 50.59 Evaluation No. 1105, “Unit 2 MSLB Containment Response with RSGs," Revision 0
- 50.59 Evaluation No. 1101, “Evaluation of Updated Loads due to Revised Reactor Vessel Support Stiffness," Revision 0
- 50.59 Screening No. 3643, “EC-16805, Reactor Coolant System Piping and Steam Generator Supports," Revision 0
- 50.59 Screening No. 3645, “EC 16807, Steam Generator Secondary Side Piping," Revision 2
- Calculation No. BUCRPR/NGV 2580, “Prairie Island Unit 2 Replacement Steam Generator Design Report," Revision 0
- ASME Certified Design Specification No. M530-0001-007-05, “Certified Design Specification for Replacement Steam Generators," dated March 25, 2013
- Calculation No. 020781-02-PISGR-45DK-0014, “Elimination of Main Steam Whip Restraints MSH-47 and MSH-48," Revision 0
- Calculation No. 020781-02-PISGR-45DK-0013, “External Recirculation Piping and Isolation Valve Analysis," Revision 0
- Calculation No. 020781-02-PISGR-45DK-0030, “Feedwater Gamma Plug Addition - SGR U2," Revision 0
- ND-92412-16, “Rigid Base Support Main Steam," Revision 76
- ND-92412-17, “Pipe Support Drawing 2-MSH-81," Revision B
- Calculation No. ENG-CS-392, “As-Built Main Steam Pipe Supports," Revision 0
- Technical Specification for Containment Vessel Units 1 and 2 Prairie Island, Revision 0 and Addendum No. 4
- Calculation No. 09Q4836-CAL-019, “Pipe Stress Analysis of Unit 2 Main Steam Lines Between Steam Generators and Anchors Inside Containment," Revision 0
- Calculation No. 020781-02-PISGR-41DK-0012, “Steam Generator Lower Lateral Support & Support Column Permanent Modifications for Unit 2 SGR," Revision 4
- CAP 1399214, “SLN Calculation 020781-02-PISGR-45DK-0031 Contains Errors," September 30, 2013
- CAP 1399725, “Attachment Missing From Legacy Calculation PI-P-099," October 2, 2013
- CAP 1403558, “Main Steam Piping Re-Analysis Inappropriately Conservative,” October 25, 2013
- CAP 1405404, “Legacy Calculation Has Missing Pages,” November 7, 2013
- CAP 1406242, “NRC Question SGR2-132," November 13, 2013
- CAP 1408246, “Lack of Clarity On Response to SGR2-142," November 26, 2013
- CAP 1410077, “Drawing XH-1106-1548 does not Match As-Built," December 10, 2013
- CAP 1410115, “Containment Design Spec Addendum not Located with Main Spec,” December 10, 2013
- CAP 1411135, “Documentation of Support Calculations Not Being Retrievable," December 16, 2013
- CAP 1412225, “Deficiency in Analysis of U2 Blowdown Supports," December 22, 2013
- CAP 1412692, “SLN Blowdown Support 2-RBDH-688 Analysis Issue,” December 29, 2013
- CAP 1411807, “Calculation Discussion Needs Clarification," December 19, 2013
- CAP 1411495, “Calculation Book 223 has Illegible Portions,” December 18, 2013

- CAP 1411462, “Design Load on Support 2-MSH-81 Increased W/O Full Analysis," December 18, 2013
- CAP 1412106, “NRC Question SGR2 198," December 20, 2013
- CAP 1413743, “Variation in Bolt Area Used In Calculation ENG-CS-392," January 8, 2013
- CAP 1411370, “Issue Identified By The NRC Not Entered into CAP Promptly," December 17, 2013
- CAP 1403816, “Wrong Calculation Updated, But Not Issued, for EC 16807," October 28, 2013
- CAP 1411275, “Difficulty Locating Information On Microfilm," December 17, 2013
- CAP 1396313, “SLN Calculation 45DK-0011 Contains Coordinate System Error”, September 10, 2013
- CAP 1405381; Unacceptable Film Radiography on RSG 21 Cold Leg Weld; November 7, 2013
- Document No. 020781-02-PISGR-40IM-0001; Metrology, Machining, and Welding Plan; Revision 03
- Eddy Current Report No. 51-9193620-000; Technical Summary of Steam Generators Eddy Current Examination for Prairie Island Nuclear Generating Plant; Unit 2; September/October 2012
- NDE Report No. 2013V058; VT-3 Examination Report for MS Double Snubber/Support, Component ID H-8; October 3, 2013
- NDE Report No. 2013V059; VT-3 Examination Report for MS Support, Component ID
- NDE Report No. MT-NP8-419; Magnetic Particle Examination of RSG 21 Girth Weld; November 15, 2013
- NDE Report No. MT-NP8-426; Magnetic Particle Examination of RSG 21 Girth Weld; November 16, 2013
- NDE Report No. RT-NP8-023; Radiographic Examination of RSG 22 Girth Weld; November 15, 2013
- NDE Report No. RT-NP8-024; Radiographic Examination of RSG 21 Girth Weld; November 17, 2013
- NDE Report PT-NP8-089; Liquid Penetrant Examination of RSG 21 Cold Leg ID; November 2, 2013
- NDE Report PT-NP8-090; Liquid Penetrant Examination of RSG 21 Hot Leg ID; November 2, 2013
- NDE Report PT-NP8-093; Liquid Penetrant Examination of RSG 22 Cold Leg ID; November 2, 2013
- NDE Report PT-NP8-094; Liquid Penetrant Examination of RSG 22 Hot Leg ID; November 2, 2013
- NDE Report PT-NP8-098; Liquid Penetrant Examination of RSG 22 Hot Leg OD; November 4, 2013
- NDE Report PT-NP8-099; Liquid Penetrant Examination of RSG 22 Cold Leg OD; November 4, 2013
- NDE Report PT-NP8-103; Liquid Penetrant Examination of RSG 21 Cold Leg OD; November 5, 2013
- NDE Report PT-NP8-104; Liquid Penetrant Examination of RSG 21 Hot Leg OD; November 5, 2013
- NDE Report RT-NP8-018; Radiographic Examination of RCS 21 Cold Leg Nozzle Safe End-to-Elbow Weld; November 13, 2013
- NDE Report RT-NP8-019; Radiographic Examination of RCS 21 Hot Leg Nozzle Safe End-to-Elbow Weld; November 13, 2013

- NDE Report RT-NP8-027; Radiographic Examination of RCS 22 Hot Leg Nozzle Safe End-to-Elbow Weld; November 22, 2013
- NDE Report RT-NP8-028; Radiographic Examination of RCS 22 Cold Leg Nozzle Safe End-to-Elbow Weld; November 22, 2013
- Procedure 8-QPP-710; General Procedure for Magnetic Particle Examination (DRY); Revision 003
- Procedure 8-QPP-720; General Procedure for Liquid Penetrant Examination (Visible, Solvent Removable); Revision 004
- Procedure FP-PE-NDE 530; Visual Examination VT-3; Revision 6
- Procedure No. H2; Boric Acid Corrosion Control Program; Revision 23
- Procedure No. SP 1403 [2403]; Reactor Vessel Closure Head Bare Metal Visual Examination; Revision 4
- Procedure SWI NDE-PT-1; Solvent Removable, Visible Dye Penetrant Examination; Revision 1
- Procedure SWI NDE-UT-16; Ultrasonic Examination of Vessels Not Greater Than 2” In Thickness; Revision 1
- Procedure SWI NDE-VT-6.0; Visual Examinations for Leakage on Reactor Penetrations (VT-2); Revision 1
- Weld Procedure Specification (WPS) 8-MGTSM-33-III-1; WPS for Girth Welding on SGs 21 and 22; Revision 000
- Weld Procedure Specification (WPS) 8-MGTSM-88-IX-1; WPS for RCS Safe End-to-Cast Austenitic Piping Welding for SGs 21 and 22; Revision 002
- WO 00455953; RCS Safe end-to-Cast Austenitic Piping Welding on SGs 21 and 22
- WO 00455954; Girth Welding on SGs 21 and 22
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Containment Access Facility (CAF) RP Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; FOSAR and New Steam Generator Bowl Closeout RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; HEPA Ventilation and Vacuum Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; OSG Removal and Transport RP Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Pipe End Decontamination and Internal Shielding Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Radioactive Material Handling Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RCS Cutting and Laser Metrology of Pipe Ends RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RCS Machining and Welding RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Audio/Visual/Remote Monitoring Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Instrument Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Radiography Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Staffing Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Temporary Shielding Plan; Revision 00

- 01393501; Snap Shot Self-Assessment; In-Plant Radioactivity Control and Mitigation, Occupational Dose Assessment; dated August 8, 2013
- 01396643; Snap Shot Self-Assessment; Radiological Hazard Assessment and Exposure Controls; dated September 24, 2013
- 020781-02-PISGR-68HS-0002; Radiation Protection and ALARA Plan; Revision 05
- 1C19.2 Containment System Ventilation Unit 1; Revision 25
- 1R28 Radiation Protection Outage; dated January 25, 2013
- 2C19.2 Containment System Ventilation Unit 2; Revision 2
- 2R28 Radiation Protection Department Outage Manual; dated September 12, 2013
- 5AWI 5.3.0; Key and Seal Control; Revision 14
- 5AWI 10.1.2; Respirator Qualification Program; Revision 05
- CR 01401995; PCE No. 11, 2R28, SGRP; While Performing Pipe End Decon; dated October 16, 2013
- CR 01404460; Equipment Hatch Effluent Release Monitor Loss of Power; dated October 31, 2013
- CY-ENVR-401; Liquid Waste Tank Release Report; Revision 00
- CY-ENVR-502; Containment Release Instruction; Revision 00
- CY-ENVR-503; Outage Control of Containment Openings; Revision 01
- CY-ENVR-504; Satellite RCA Effluent Control; Revision 00
- CY-ENVR-512; Gas Decay Tank Release Instruction; Revision 01
- CY-ENVR-513; Effluent Surveillance Sample Collection; Revision 00
- CY-ENVR-623; Effluent Release Offsite Dose Report; Revision 00
- D11.12; Radioactive Materials Shipment – Old Steam Generator, Upper and Lower Parts; Revision 3
- DOT Special Permit 14455; June 3, 2013
- FG-RP-BSR-01; Bioassay Sample Report; Revision 01
- FP-RP-AM-01; Alpha Monitoring Program; Revision 03
- FP-RP-BP-01; Bioassay Program; Revision 05
- FP-RP-JJP-01; RP Job Planning; Revision 14
- FP-RP-RW-02; Radioactive Shipping Procedure; Revision 8
- FP-RP-RWP-01; Radiation Work Permit; Revision 13
- FP-RP-SD-01; Special Dosimetry; Revision 08
- FP-RP-WBC-01; Whole Body Counter Use and Functional Check; Revision 01
- FP-WM-W01-01; Work Identification, Screening, Validation and Cancellation; Revision 17
- GG-FO-WI-001; OSGLP Preparation for Offsite Transport; Advance Copy; Undated
- GG-FO-WI-002; OSGUP Preparation for Offsite Transport; dated May 6, 2013
- H4; Offsite Dose Calculation Manual (ODCM); Revision 27
- H4.2; Offsite Dose Calculation Manual (ODCM) Supporting Data; Revision 01
- H 26; Respiratory Protection Program; Revision 09
- NVLAP Scope of Accreditation; effective dates July 2012 through June 2013
- NVLAP Scope of Accreditation; effective dates July 2013 through June 2014
- NOS Observation Report 2013-01-013; Radiological Protection; Respiratory Protection; dated February 2013
- Old Steam Generator (OSG) Removal Schedule; dated September 17, 2013
- Personnel Contamination Logs; dated 2012 and 2013
- Positive Whole Body Count Documentation; dated 2011 and 2012
- PING 1027; Breathing Air System Air Quality Tests; Selected Records; various dates 2012 and 2013
- PING 1028; PSI Visual Cylinder Inspection Evaluations; Selected Records; various dates 2012 and 2013

- PING 1037; Quarterly Update of Emergency Response Organization; Revision 30
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Containment Access Facility (CAF) RP Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; FOSAR and New Steam Generator Bowl Closeout RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; HEPA Ventilation and Vacuum Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; OSG Removal and Transport RP Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Pipe End Decon and Internal Shielding Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Radioactive Material Handling Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RCS Cutting and Laser Metrology of Pipe Ends RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RCS Machining and Welding RP Plan; Revision 01
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Audio/Visual/Remote Monitoring Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Instrument Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Radiography Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; RP Staffing Plan; Revision 00
- Prairie Island Nuclear Generating Plant; Unit-2 Steam Generator Replacement Project; Temporary Shielding Plan; Revision 00
- Prairie Island 2R28 Radiation Protection Department Manual; Revision 00
- Radiation Occurrence Report; Selected Records; Selected Individuals; dated 2013
- Radioactive Release Permit PILB2013-181; dated September 4, 2013
- Radioactive Release Permit PIGC2013-079; dated March 28,, 2013
- Radioactive Release Permit PILC2013-142; dated August 15, 2013
- Radioactive Release Permit PILC2013-179; dated September 10, 2013
- Radiological Work Assessment Form CRPC Instructions for WO 455951; dated September 17, 2013
- Radiological Work Control Planning Packages for Reactor Cavity Decontamination and Work Inside the RSG Channel Head; Various dates 2013
- Radiation Work Permit 1747-01; Outage Satellite RCA Work; dated September 17, 2013
- Radiation Work Permit 1758-00; Work in OSGLA on the Lower Part of OSG; dated September 17, 2013
- Respiratory Protection Equipment Maintenance Records; Selected Records; Various dates 2012 and 2013
- Respiratory Protection Equipment NIOSH Certifications; Selected Records; Various dates 2013
- Respiratory Protection Qualification Records; Selected Records; Various dates 2013
- RPIP 1008; Radiation Protection Key Control; Revision 17
- RPIP 1106; Access Control Procedures; Revision 18
- RPIP 1118; Conducting Radiological Surveys; Revision 24
- RPIP 1120; Posting of Restricted Areas; Revision 38
- RPIP 1121; RWP Issue; Revision 24
- RPIP-1123; Alpha Characterization Smears; Revision 01

- RPIP 1124; Evaluation of Isotopic Mix; Revision 01
- RPIP 1126; Contamination Monitor Alarm Response and Personnel Decontamination; Revision 24
- RPIP 1131; Control of Radiography; Revision 18
- RPIP 1135; RWP Coverage; Revision 32
- RPIP-1204; Evaluation of Airborne Radioactivity; Revision 18
- RPIP 1210; Charging SCBA Air Cylinders; Revision 13
- RPIP-1214; Respiratory Protection Equipment Testing; Revision 18
- RPIP-1215; Respiratory Equipment Control; Revision 07
- RPIP-1219; Respirator Use; Revision 19
- RPIP-1226; Control Room Breathing Air System Testing; Revision 04
- RPIP 1300; Control and Tagging of Radioactive Material; Revision 22
- RPIP 1302; Unconditional Release of Materials; Revision 25
- RPIP 1304; Conditional Release of Equipment to Outside the Radiological Controlled Area; Revision 11
- RPIP 1309; Tracking Radwaste Shipments; Revision 6
- RPIP 1330; Satellite RCA Process; Revision 09
- RPIP 1331; Radioactive Material Control; Revision 01
- RPIP-1404; CAM Alarm Response; Revision 10
- RPIP 1645; Shutdown Radiation Surveys; Revision 06
- RPIP 1677; SAM-11 Small Articles Monitor Operation and Calibration; Revision 06
- RPIP 1686; SAM-12 Small Articles Monitor Operation and Calibration; Revision 00
- RPIP 1729; Initial Containment Entry; Revision 14
- RPIP-2016; Max Air 2000-800 Respiratory Protection; Revision 00
- RWP and WOs 407277 and 407278; Perform Eddy Current Testing 21 and 22 RHR Heat Exchangers; dated June 11, 2013
- RWP and WO 455878; SFP Transfer Canal Repairs and Inspections; dated July 24, 2013
- RWP and WO 455953; Cutting and Welding Activities Outside of RCS Piping; dated July 02, 2013
- RWP and WO 476259 and 476571; Replace 21 and 22 RCP Seal with FLOWSERVE N-9000 Seal; dated September 14, 2013
- Shipping Record 13-047; Radioactive Waste Shipment Old Steam Generators Upper Section; November 16, 2013
- Shipping Record 13-048; Radioactive Waste Shipment Old Steam Generators Upper Section; November 16, 2013
- Shipping Record 13-052; Radioactive Waste Shipment Old Steam Generators Lower Section; December 26, 2013
- Shipping Record 13-053; Radioactive Waste Shipment Old Steam Generators Lower Section; December 26, 2013
- Unit-2 Steam Generator Replacement Project Removal and Transport Plan; dated September 15, 2013

## LIST OF ACRONYMS

| | |
|---|---|
| ADAMS | Agencywide Document Access Management System |
| AISC | American Institute of Steel Construction |
| ALARA | As-Low-As-Is-Reasonably-Achievable |
| ASME | American Society of Mechanical Engineers |
| CAP | Corrective Action Program |
| CC | Code Case |
| CFR | Code of Federal Regulations |
| CR | Condition Report |
| CST | Condensate Storage Tank |
| DRP | Division of Reactor Projects |
| EC | Engineering Change |
| EPRI | Electric Power Research Institute |
| ET | Eddy Current Testing |
| HP | Health Physics |
| HRA | High Radiation Areas |
| ID | Inner Diameter |
| IMC | Inspection Manual Chapter |
| IP | Inspection Procedure |
| IPEEE | Individual Plant Examination of External Events |
| IR | Inspection Report |
| ISI | Inservice Inspection |
| LER | Licensee Event Report |
| MS | Main Steam |
| NCV | Non-Cited Violation |
| NDE | Non-destructive Examinations |
| NEI | Nuclear Energy Institute |
| NOS | Nuclear Oversight Department |
| NRC | U.S. Nuclear Regulatory Commission |
| OD | Outer Diameter |
| OPR | Operability Recommendation |
| OLS | Outside Lift System |
| PARS | Publicly Available Records System |
| PWR | Pressurized Water Reactor |
| RCA | Radiologically Controlled Area |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RHR | Residual Heat Removal |
| RFO | Refueling Outage |
| RPM | Radiation Protection Manager |
| RSG | Replacement Steam Generator |
| RWP | Radiation Work Permit |
| SCBA | Self-Contained Breathing Apparatus |
| SDP | Significance Determination Process |
| SG | Steam Generator |
| SGBD | Steam Generator Blowdown |
| SGRP | Steam Generator Replacement Project |
| SG | Steam Generator |
| SLOCA | Small Loss of Coolant Accident |
| SRA | Senior Reactor Analyst |

## LIST OF ACRONYMS

| | |
|---|---|
| SSC | Systems, Structures, and Components |
| TS | Technical Specification |
| USAR | Updated Safety Analysis Report |
| VHRA | Very High Radiation Areas |
| WO | Work Order |
| WR | Work Request |

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

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-306
License No. DPR-60

Enclosure:
Inspection Report 05000306/2013011
w/Attachment: Supplemental Information

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