UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, ILLINOIS 60532-4352

September 26, 2022

Mr. Christopher P. Domingos
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–PHASE 4 POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL REPORT 05000282/2022011 AND 05000306/2022011

Dear Mr. Domingos:

On August 29, 2022, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Prairie Island Nuclear Generating Plant and discussed the results of this inspection with Mr. H. Hanson, Plant Manager and other members of your staff. The results of this inspection are documented in the enclosed report.

Three findings of very low safety significance (Green) are documented in this report. Two of these findings involved violations of NRC requirements and one was determined to be Severity Level IV. We are treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance or severity of the violations documented in this inspection report, 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 Prairie Island Nuclear Generating Plant.

If you disagree with a cross-cutting aspect assignment or a finding not associated with a regulatory requirement 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; and the NRC Resident Inspector at 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 Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

Signed by Feliz-Adorno, Nestor
on 09/26/22

Néstor J. Féliz Adorno, Chief
Engineering and Reactor Projects Branch
Division of Operating Reactor Safety

Docket Nos. 05000282 and 05000306
License Nos. DPR-42 and DPR-60

Enclosure:
As stated

cc w/ encl: Distribution via LISTSERV

Letter to Christopher P. Domingos from Néstor J. Féliz Adorno dated September 26, 2022.

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT–PHASE 4 POST-APPROVAL SITE INSPECTION FOR LICENSE RENEWAL REPORT 05000282/2022011 AND 05000306/2022011

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U.S. NUCLEAR REGULATORY COMMISSION
Inspection Report

Docket Numbers:	05000282 and 05000306
License Numbers:	DPR-42 and DPR-60
Report Numbers:	05000282/2022011 and 05000306/2022011
Enterprise Identifier:	I-2022-011-0031
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant
Location:	Red Wing, MN
Inspection Dates:	July 17, 2022 to July 23, 2022
Inspectors:	M. Domke, Reactor Inspector E. Fernandez, Reactor Inspector B. Jose, Senior Reactor Inspector G. Pick, Senior Reactor Inspector
Approved By:	Néstor J. Féliz Adorno, Chief Engineering and Reactor Projects Branch Division of Operating Reactor Safety

Enclosure

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a Phase 4 Post-Approval Site Inspection for License Renewal at Prairie Island Nuclear Generating Plant, in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to https://www.nrc.gov/reactors/operating/oversight.html for more information.

List of Findings and Violations

Failure to Monitor In-Scope Compressed Air Piping Subject to Aging Effects			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Initiating Events	Green FIN 05000282,05000306/2022011-01 Open/Closed	[H.3] - Change Management	71003
The team identified a finding of very low safety significance (Green) for the licensee's failure to monitor for the effects of aging of compressed air piping. Specifically, the licensee changed a preventive maintenance activity from periodic to on-demand that prevented them from managing the effects of aging on in-scope piping of the compressed air system as required by their compressed air aging management program.			

Failure to Determine the Reactor Coolant System (RCS) Pressure and Temperature Limits Using the Analytical Method Required by Technical Specification (TS)			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000282,05000306/2022011-02 Open/Closed	None (NPP)	71003
The team identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of TS 5.6.6.b for the licensee failure to determine the RCS pressure and temperature limits using WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."			

Incorrect Fluence Design Values Result in Non-Conservative Reactor Coolant System (RCS) Pressure and Temperature Limits and Inaccurate Information Provided to the NRC			
Cornerstone	Significance/Severity	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green Severity Level IV NCV 05000282,05000306/2022011-03 Open/Closed	None (NPP)	71003
The team identified a finding of very low safety significance (Green) and an associated Severity Level IV non-cited violation (NCV) of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion III, "Design Control," and 10 CFR 50.9, "Completeness and Accuracy of Information," for the licensee failure to verify the adequacy of the reactor vessel design fluence values and submit related information to the Commission that was accurate in all material respects. Specifically, a design error resulted in the licensee using incorrect			

beltline fluence values to calculate upper shelf energy, adjusted reference temperature, and pressurized thermal shock. This resulted in inaccurate information submitted to the NRC and a non-conservative Technical Specification (TS) limiting condition for operation (LCO).

Additional Tracking Items

Type	Issue Number	Title	Report Section	Status
URI	05000282,05000306/ 2021004-01	Unresolved Issue Associated with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.9, "Completeness and Accuracy of Information"	71003	Closed

INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

OTHER ACTIVITIES–TEMPORARY INSTRUCTIONS, INFREQUENT AND ABNORMAL

71003 - Post-Approval Site Inspection for License Renewal

The NRC conducted this Phase 4 Post-Approval Site Inspection for License Renewal in accordance with the License Renewal Inspection Program (LRIP). Per Inspection Manual Chapter 2516, the LRIP is the process used by NRC staff to verify the adequacy of Aging Management Programs (AMPs), Time Limited Aging Analyses (TLAAs), and other activities associated with an applicant's request to renew an operating license of a commercial nuclear power plant beyond the initial licensing period under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54, "Requirements for the Renewal of Operating Licenses for Nuclear Power Plants."

The inspection evaluated the licensee implementation of aging management activities by performing detailed reviews of the aging management programs listed below. The team reviewed license renewal program assessments, commitments, engineering evaluations, procedures, program documents, completed tests, completed maintenance activities, nondestructive examination reports, technical reports, correspondence, and drawings. This inspection took place 8 to 9 years after the licensee entered the period of extended operation (PEO) for Unit 2 and Unit 1, respectively. The PEO included the additional 20 years beyond the original 40-year licensed term. Unit 1 will end the period of extended operation at midnight on August 9, 2033. Unit 2 will end the period of extend operation at midnight on October 29, 2034.

Post-Approval Site Inspection for License Renewal (16 Samples)

(1) L.2.2 - Aboveground Steel Tanks Program
(2) L.2.8 - Buried Piping and Tanks Inspection Program
(3) L.2.10 - Compressed Air Monitoring Program
(4) L.2.13 - Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Sensitive Instrumentation Circuits Program
(5) L.2.14 - External Surfaces Program
(6) L.2.16 - Fire Water System Program (only sprinkler heads)
(7) L.2.21 - Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program
(8) L.2.25 - Masonry Wall Program
(9) L.2.26 - Metal-Enclosed Bus Program

(10) L.2.31 - Open-Cycle Cooling Water System Program (only microbiologically influenced corrosion monitoring)
(11) L.2.32 - PWR Vessel Internals Program
(12) L.2.33 - Reactor Head Closure Studs Program
(13) L.2.34 - Reactor Vessel Surveillance Program
(14) L.2.35 - RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program
(15) L.2.36 - Selective Leaching of Materials Program
(16) L.2.38 - Structures Monitoring Program

INSPECTION RESULTS

<table>
<tr><td colspan="4">Failure to Monitor In-Scope Compressed Air Piping Subject to Aging Effects</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Initiating Events</td><td>Green
FIN 05000282,05000306/2022011-01
Open/Closed</td><td>[H.3] - Change Management</td><td>71003</td></tr>
<tr><td colspan="4">The team identified a finding of very low safety significance (Green) for the licensee’s failure to monitor for the effects of aging of compressed air piping. Specifically, the licensee changed a preventive maintenance activity from periodic to on-demand that prevented them from managing the effects of aging on in-scope piping of the compressed air system as required by their compressed air aging management program.</td></tr>
<tr><td colspan="4"><u>Description</u>:

On July 18, 2022, the team performed a walkdown and observed grid-marks used for mapping ultrasonic examinations on carbon steel piping and fittings between the instrument air compressors and air dryers. Interviews of licensee staff revealed the licensee initiated these ultrasonic examinations after internal operating experience showed wall thickness corroded below the minimum acceptance criteria in 2007. Ultrasonic examination reports from 2009 to 2020 of this wet-air piping showed three areas eroded to 25 percent of nominal wall thickness. Some license renewal components had 8-11 percent thickness remaining before achieving their established minimum wall thickness acceptance criteria. In June 2022, the licensee changed the maintenance plan from a 2-year periodicity to “on-demand.” The licensee believed the air dryers installed in 2005 resolved the corrosion effects. However, the team determined that the piping of concern was located upstream of the air dryers and continued to experience a wet air environment that was conducive to carbon steel corrosion. The team also noted that the licensee had not performed any “on-demand” maintenance to this piping section.

The licensee implemented Procedure H65.2.10, “Compressed Air Monitoring Aging Management Program,” Revision 4, to manage the effects of corrosion on the interior of components that consisted of the station and instrument air systems. The licensee included the piping and components from the instrument air compressors to the air dryers within the scope of the program as described in Table 1, “Managed Aging Effects.”
Procedure H65.2.10, Section 4.5.1 specified the effects of corrosion to be monitored by performing periodic inspections and tests. Section 4.6.5 stated that test and inspection results must be evaluated to ensure that the station and instrument air system could perform its intended function. However, the team identified carbon steel components in a wet-air</td></tr>
</table>

environment between the instrument air compressors and dryers where corrosion effects were no longer monitored or evaluated.

Corrective Actions: The licensee initiated actions to develop a periodic maintenance plan.

Corrective Action References: Quality Issue 501000065702

Performance Assessment:

Performance Deficiency: The team determined that the failure of the licensee to monitor aging effects for in-scope compressed air piping was contrary to Revision 4 of Procedure H65.2.10 and was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the failure to manage the aging effect of wall thinning of carbon steel in a wet gas environment does not limit the likelihood of events that upset plant stability because it increased the likelihood of a loss of instrument air that would require a plant shutdown.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” Specifically, the team determined the finding was of very low safety significance (Green) because it did not result in an actual or partial loss of instrument air or in an increased likelihood of a complete loss instrument air that would result in a plant trip.

Cross-Cutting Aspect: H.3 - Change Management: Leaders use a systematic process for evaluating and implementing change so that nuclear safety remains the overriding priority. Specifically, the licensee changed the periodicity of the aging management activity to the most vulnerable part of an in-scope system without considering all the impacts of this change.

Enforcement:

Inspectors did not identify a violation of regulatory requirements associated with this finding.

Failure to Determine the Reactor Coolant System (RCS) Pressure and Temperature Limits Using the Analytical Method Required by Technical Specification (TS)			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green NCV 05000282,05000306/2022011-02 Open/Closed	None (NPP)	71003
The team identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of TS 5.6.6.b for the licensee failure to determine the RCS pressure and temperature limits using WCAP-14040-NP-A, Revision 2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.”			

Description:

TS 5.6.6.b required the licensee to, in part, determine the RCS pressure and temperature limits using Revision 2 of WCAP-14040-NP-A. However, in the fourth quarter of 2021, the resident inspectors noted the licensee was using Revision 4 of WCAP-14040 and opened Unresolved Item 05000282; 05000306/2022004-01, in part, to review this issue further. The licensee issued Quality Issue 501000060322 to document this concern.

During the Phase 4 Inspection, the team noted the licensee received NRC approval to increase the reactor vessel fluence operating conditions from 20 to 35 effective full power years (EFPY) using Revision 2 of WCAP-14040 via Safety Evaluation Report titled, "Prairie Island Nuclear Generating Plant, Units1 and 2 - Issuance of Amendments Re: Use of a Pressure and Temperature Limits Report (TAC Nos. MA1121 and MA1122)," dated May 4, 1998 [ML20247F916]. The team also noted the licensee revised their calculations to adopt WCAP-14040, Revision 4 in 2010 to follow the recommendations of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Revision 0. In addition, as part of their measurement uncertainty recapture amendment, the licensee requested the NRC to change TS 5.6.6.b to use WCAP-14040, Revision 4. However, on April 23, 2010, the licensee issued Letter L-PI-10-035, "Supplement to License Amendment Request for Measurement Uncertainty Recapture - Power Uprate, Withdrawal of Proposed Change to Analysis Methodology for Pressure Temperature Limits Report (TAC Nos. ME3015 and ME30162)," [ML101130449] withdrawing this request.

The licensee continued to use Revision 4 of WCAP-14040 after withdrawing the TS change request. The licensee believed that the requirements of WCAP-14040, Revision 2, could be separated into three distinct portions–fluence calculations, pressure temperature limits determination, and cold overpressure mitigation systems evaluation. Consequently, the licensee revised their fluence methodology described in their updated safety analysis report (USAR) to WCAP-14040, Revision 4. The licensee performed 10 CFR 50.59 Screening 4305, "Adjusted Reference Temperature for Unit 1 and Unit 2 Reactor Vessel Materials at 54 EFPY," and determined a 50.59 evaluation and a license amendment were not needed to implement the USAR change.

The team noted the licensee failed to recognize that the proposed analytical change would result in a non-compliance with TS when performing the 10 CFR 50.59 screening. As a result, the licensee continued to adopt Revision 4 of WCAP-14040 in their calculations and did not re-attempt to request NRC approval to change TS 5.6.6.b. The team determined that the pressure-temperature curves required by TS 5.6.6 remained valid (bounding) until each unit exceeded 35 EFPY based on the licensee analysis that used Revision 2 of WCAP-14040. Unit 1 exceeded 35 EFPY in May 2015 and Unit 2 exceeded 35 EFPY in July 2015.

The licensee determined that sufficient margin remained from the revised pressure-temperature curves and the actual heatup and cooldown operating bands, even when calculated using Revision 4 of WCAP-14040, to reasonably assure safety until compliance was restored. Because using Revision 4 of WCAP-14040 was not in accordance with TS, the team consulted with the Office of Nuclear Reactor Regulations (NRR) and determined it had been generically reviewed and approved by the NRC and, thus, its use in this licensee assessment was reasonable.

Corrective Actions: The licensee initiated actions to submit a change to TS 5.6.6.b that would refer to WCAP-14040, Revision 4. As interim actions the licensee implemented administrative controls that specified actions to take immediately if operators had exceeded the limits of the revised pressure-temperature curves. Corrective Action References: Quality Issue 501000060322
Performance Assessment: Performance Deficiency: The team determine the failure to determine the RCS pressure and temperature limits using WCAP-14040-NP-A, Revision 2, was contrary to TS 5.6.6.b and was a performance deficiency. Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to follow technical specifications when determining the RCS pressure and temperature limits did not reasonably assure the reactor was operated within its design limits throughout its expected end-of-life. Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The team determined the finding was of very low safety significance (Green) because the licensee reasonably determined that they maintained sufficient margin to the pressure-temperature curve limits when calculated using the current EFPY and the end-of-life EFPY using an alternative NRC-approved method. Cross-Cutting Aspect: Not Present Performance. No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.
Enforcement: Violation: TS 5.6.6.b required the licensee to, in part, determine the RCS pressure and temperature limits using Revision 2 of WCAP-14040-NP-A (including any exemption granted by NRC to American Society of Mechanical Engineers (ASME) Code Case N-514). Contrary to the above, since May 2015 for Unit 1 and July 2015 for Unit 2, the licensee failed to determine the RCS pressure and temperature limits using WCAP-14040-NP-A, Revision 2 (including any exemption granted by NRC to ASME Code Case N-514). Specifically, the licensee used Revision 4 of WCAP-14040-NP-A to increase the reactor vessel fluence operating conditions from 35 to 54 EFPY. Unit 1 exceeded 35 EFPY in May 2015 and Unit 2 exceeded 35 EFPY in July 2015. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy. The disposition of this finding and associated violation closes URI: 05000282,05000306/2021004-01

Incorrect Fluence Design Values Result in Non-Conservative Reactor Coolant System (RCS) Pressure and Temperature Limits and Inaccurate Information Provided to the NRC			
Cornerstone	Significance/Severity	Cross-Cutting Aspect	Report Section
Barrier Integrity	Green Severity Level IV NCV 05000282,05000306/2022011-03 Open/Closed	None (NPP)	71003

The team identified a finding of very low safety significance (Green) and an associated Severity Level IV non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion III, "Design Control," and 10 CFR 50.9, "Completeness and Accuracy of Information," for the licensee failure to verify the adequacy of the reactor vessel design fluence values and submit related information to the Commission that was accurate in all material respects. Specifically, a design error resulted in the licensee using incorrect beltline fluence values to calculate upper shelf energy, adjusted reference temperature, and pressurized thermal shock. This resulted in inaccurate information submitted to the NRC and a non-conservative Technical Specification (TS) limiting condition for operation (LCO).

<u>Description</u>:

In September 2021, during review of the draft version of WCAP-18660-NP, "Analysis of Capsule N from the Xcel Energy Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program," the licensee recognized that the fluence reported for the nozzle shell to intermediate shell weld (i.e., W2 weld) at 54 effective full power years (EFPY) measured more than two times the predicted values from their analysis of record. The vendor subsequently confirmed the WCAP-18660-NP report fluence had the correct value. The vendor determined they made an error in 2007 when they performed Calculation CN-REA-07-28, "Reactor Vessel Fluence Calculations for Prairie Island MUR Uprate Program," Revision 0. Specifically, they calculated the fluence with the W2 weld at 179 cm above the core midplane when the actual location for estimating the fluence should have been 163 cm.

The licensee accepted the calculation on August 24, 2007. The licensee used Calculation CN-REA-07-28 to extend the heatup and cooldown curves from 35 to 54 EFPY for their end-of-life extension (EOLE). This location error similarly affected the Unit 2 W2 weld and the Unit 1 intermediate shell to lower shell weld (i.e., W3 weld) fluence values. However, the licensee determined the change in fluence impact for these additional welds was not as pronounced. The licensee documented this deficiency in Quality Issue 501000056729.

The WCAP-18660-NP report also revealed that the Unit 1 W3 weld chemistry factor had increased beyond that originally reported. Because the Unit 2 W2 weld had the same heat of weld wire as Unit 1 W3 weld, the licensee applied the chemistry factor error to the Unit 2 W2 weld. The licensee documented this deficiency in Quality Issue 501000056798.

In the fourth quarter of 2021, the resident inspectors noted that the licensee had not initiated an extent of condition review. After questioning, the licensee identified prior submittals that had used the inaccurate information. The resident inspectors opened Unresolved Item 05000282; 05000306/2022004-01 to review this issue further.

Inaccurate Submittals to the NRC

The fluence values from Calculation CN-REA-07-28 provided input into Calculation CN-MRCDA-07-59, “Prairie Island Units 1 and 2 Measurement Uncertainty Recapture Reactor Vessel Integrity Evaluation,” Revision 1. Calculation CN-MRCDA-07-59 determined the upper shelf energy, adjusted reference temperature, and pressurized thermal shock, which the licensee had reported in submittals to NRC since 2008. The licensee reviewed the historical impact of this error on these three key parameters and determined that the impact would not have resulted in exceeding the parameter limits derived using WCAP-14040, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” Revision 4. However, the team noted that these evaluations could not be adjusted to remain within the limits derived using WCAP-14040, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” Revision 2, which was the methodology required by TS 5.6.6.b. This additional performance deficiency is discussed separately in the Results section of this inspection report.

The team noted the licensee failed to provide information to the Commission that was complete and accurate in all material respects for “Application for Renewed Operating Licenses for Prairie Island Nuclear Generating Plant, Units 1 and 2,” dated April 11, 2008 [ML081130666]; and Letter L-PI-09-133, “License Amendment Request for Measurement Uncertainty Recapture - Power Uprate,” dated December 28, 2009 [ML093650045]. Specifically, the fluence values provided in these submittals for the W2 and W3 welds were lower than the actual fluence. However, the values for the related parameters (i.e., upper shelf energy, adjusted reference temperature, and pressurized thermal shock) at 54 EFPY remained lower than the limits. In consultation with the Office of Nuclear Reactor Regulations (NRR), the team determined this information was material to the NRC because NRR had relied on the information to assess compliance with NRC regulations (i.e., 10 CFR 50.60 and/or 10 CFR 50.61).

The licensee determined that the W2 weld location error may have impacted relief requests related to the W2 weld on both units. Letter L-PI-19-002, “10 CFR 50.55a Requests Nos. 1-RR-5-10 and 2-RR-5-10, Proposed Alternative to Reactor Vessel Inservice Inspection (ISI) Intervals for Prairie Island Unit 1 and Unit 2,” dated June 13, 2019 [ML19164A166], requested extending nondestructive examination of the W2 and W3 seam welds on each unit from the fifth to sixth 10-year intervals. The team noted that the licensee had not identified the full impact on these calculations of the pending Unit 2 Capsule N results. These results may cause additional adjustments. However, the licensee had reasonable time to address this since these results were expected in the fourth quarter of 2022 and the end of the fifth 10-year interval was on December 24, 2024.

Non-Conservative Technical Specifications

Technical Specification 5.6.6.a required the licensee to establish pressure temperature limits for heatup and cooldown for several TS LCOs. Particularly, TS LCO 3.4.3 required that operators control plant heatup and cooldown within the limits of the curves contained in the pressure-temperature limits report. The W2 weld error increased the adjusted reference temperature and shifted the actual pressure temperature heatup and cooldown curves further to the right than that shown in the pressure-temperature limits report. Thus, the team determined that TS LCO 3.4.3 was non-conservative since 2015.

Corrective Actions: The licensee initiated actions to submit a change to TS 5.6.6.b that would refer to WCAP-14040, Revision 4. In addition, the licensee determined that sufficient margin remained from the revised pressure-temperature curves and the actual heatup and cooldown operating bands, even when calculated using Revision 4 of WCAP-14040, to reasonably assure safety until compliance was restored. Because using Revision 4 of WCAP-14040 was not in accordance with TS, the team consulted with NRR and determined it had been generically reviewed and approved by the NRC and, thus, its use in this licensee assessment was reasonable. As interim actions the licensee implemented administrative controls to ensure they continued to meet their design limits and specified actions to take immediately if operators find they are operating outside of the revised pressure-temperature curves.

Corrective Action References: Quality Issues 501000055874, 501000056729, 501000056798, and 501000059447

Performance Assessment:

Performance Deficiency: The team determined the licensee failure to verify the adequacy of the reactor vessel design fluence values was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. This performance deficiency resulted in inaccurate information submitted to the NRC, which was contrary to 10 CFR 50.9, "Completeness and Accuracy of Information," and a non-conservative TS LCO.

Screening: The inspectors determined the performance deficiency was more than minor because it was associated with the Design Control attribute of the Barrier Integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to verify the adequacy of the reactor vessel design fluence values resulted in operating limits that did not reasonably assure the reactor vessel was operated within its actual pressure boundary design limits and necessitated the licensee to use a different analytical approach to demonstrate reasonable assurance the reactor vessel would protect the public from radionuclide releases caused by accidents or events.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The team determined the finding was of very low safety significance (Green) because the licensee reasonably determined the reactor was never operated outside the correct operating limits and that the corrected limits provided margin to brittle failure using an alternative NRC-approved method.

Cross-Cutting Aspect: Not Present Performance. No cross-cutting aspect was assigned to this finding because the inspectors determined the finding did not reflect present licensee performance.

Enforcement:

The reactor oversight process's (ROP's) significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC's ability to regulate using traditional enforcement to adequately deter non-compliance.

Severity: This violation was less serious than the Severity Level III violation example described by Section 6.9.c.1 of the NRC Enforcement Policy, which stated "Incomplete or inaccurate information is provided or maintained. If this information had been completely and accurately provided or maintained, it would likely have caused the NRC to reconsider a regulatory position or undertake a substantial further inquiry." Specifically, the team consulted with NRR and determined that, if the information had been accurate, it would likely have not caused the NRC to reconsider a regulatory position or undertake a substantial further inquiry because sufficient margin continued to exist after the licensee reperformed their calculations using methods the NRC had generically approved. The Policy did not contain an applicable Severity Level IV violation example. However, the team determined the violation met the Policy definition of Severity Level IV in that it created the potential of more than minor safety consequences because it resulted in the NRC approval of operating limits that did not reflect the actual pressure boundary design limits of the reactor vessel.

Violation: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," specifies, in part, that measures shall be established 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. In addition, 10 CFR 50.9, "Completeness and Accuracy of Information," states, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Contrary to the above, from August 24, 2007 until August 29, 2022, the licensee failed to verify or check the adequacy of the reactor vessel design by the performance of design reviews, the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Specifically, the licensee evaluated the reactor vessel fluence in Calculation CN-REA-07-28, "Reactor Vessel Fluence Calculations for Prairie Island MUR Uprate Program," Revision 1. However, the calculation used incorrect design inputs resulting in non-conservative fluence values that subsequently caused the operating limits of TS LCO 3.4.3 to be non-conservative. In addition, on April 11, 2008, and December 28, 2009, the licensee failed to provide accurate information to the Commission in all material respects. Specifically, the licensee provided the incorrect design information to the Commission in the following submittals:

- "Application for Renewed Operating Licenses for Prairie Island Nuclear Generating Plant, Units 1 and 2," dated April 11, 2008 [ML081130666]
- Letter L-PI-09-133, "License Amendment Request for Measurement Uncertainty Recapture - Power Uprate," dated December 28, 2009 [ML093650045]

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

The disposition of this finding and associated violation closes URI: 05000282,05000306/2021004-01

URI	Unresolved Issue Associated with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.9, "Completeness and Accuracy of Information" URI 05000282,05000306/2021004-01	71003
Description: This URI was closed to the two Green findings associated with non-cited violations (NCVs) described above.		

Minor Performance Deficiency	71003
Minor Performance Deficiency: On May 26, 2022, the licensee identified a through-wall leak at the bottom of condensate storage tank 11 when inspecting it per work order 700096533-0010, “Inspection of condensate storage tank 11 exterior.” The licensee stopped the leak via temporary modification 601000003804. Licensee Procedure H65.2.2, “Aboveground Steel Tanks Aging Management Program,” Revision 2, stated, in part, that any degradation of carbon steel tank external surfaces will be recorded and evaluated to ensure the component intended function will be maintained. It also stated that inspection results will be evaluated to determine if additional inspections are needed to assure that the extent of corrosion is adequately determined. Contrary to this, the team determined the licensee failed to evaluate the external surface degradation of condensate storage tank 11 to ensure its intended function would be maintained and to determine if additional inspections were needed to assure that the extent of corrosion was adequately determined. Screening: The inspectors determined the performance deficiency was minor. Specifically, it could not reasonably be viewed as a precursor to a significant event, would not have the potential to lead to a more significant safety concern, and does not adversely affect the cornerstone objective of any cornerstones listed in IMC 0612, Appendix B.	

Minor Performance Deficiency	71003
Minor Performance Deficiency: In 2018, the licensee created Department Action Request 600000360016 to evaluate GALL-SLR, Chapter M.33, “Selective Leaching.” The licensee performed this as an annual review of operating experience required by Procedure H65, “License Renewal Implementation and Aging Management Programs,” Revision 8. However, the team determined that the licensee did not evaluate new environments and materials identified as causing or experiencing selective leaching described in the Scope of Program subsection of GALL-SLR, Chapter M.33. Procedure H65, Step 4.6.4.C, instructed the licensee to perform a directed search for applicable operating experience and incorporate applicable lessons learned into the aging management program. Similarly, procedure FP-PE-RLP-01, “License Renewal Implementation,” Revision 8, Step 5.2.1 instructed licensee personnel to review relevant operating experience for impact and incorporate relevant lessons learned. Contrary to this, the licensee did not review relevant operating experience contained in GALL-SLR, Chapter M.33, Scope of Program regarding ductile iron components and components exposed to a wastewater environment for impact and incorporate any applicable lessons learned into the Selective Leaching of Materials Program. Specifically, during inspection interviews, the licensee described that they had ductile iron components and had wastewater as an environment. However, the team noted that the gap analysis did not address this material or environment, which was identified by GALL-SLR as causing or experiencing selective leaching. Screening: The team determined the performance deficiency was minor. Specifically, it could not reasonably be viewed as a precursor to a significant event, would not have the potential to lead to a more significant safety concern, and does not adversely affect the cornerstone objective of any cornerstones listed in IMC 0612, Appendix B.	

EXIT MEETINGS AND DEBRIEFS

The inspectors verified no proprietary information was retained or documented in this report.

- On August 29, 2022, the inspectors presented the Phase 4 Post-Approval Site Inspection for License Renewal results to Mr. H. Hanson, Plant Manager and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71003	Calculations	CN-MRCDA-07-59	Prairie Island Units I and 2 Measurement Uncertainty Recapture: Reactor Vessel Integrity Evaluation	1
		CN-REA-07-28	Reactor Vessel Fluence Calculations for Prairie Island MUR Uprate Program	1
		ENG-ME-819	Adjusted Reference Temperatures for Unit 1 and Unit 2 Reactor Vessel Materials at 54 EFPY	0
	Corrective Action Documents	Action Request 01114595	Low Thickness on 125 STA Air Receiver	10/11/2007
		Action Request 01408630	Preliminary Indications in Baffle to Former Bolting	11/29/2013
		Action Request 01452005	Preliminary Flaw Indications in U1 Baffle Former Bolts	10/20/2014
		Action Request 01466818	Decision to defer CRGT Guide Card Examination	02/08/2015
		Action Request 501000059447	Relief Request Rx Vessel Fluence Error	01/04/2022
		Quality Issue 501000055874	Pressure Temperature Limits Report 54 EFPY Fluence Not Bounding	09/08/2021
		Quality Issue 501000056729	Fluence Analysis Unexpected High W2 Weld	10/05/2021
		Quality Issue 501000056798	U1 Capsule Report Fluence Impacts on U2	10/06/2021
		Quality Issue 501000060322	NRC Questions Fluence Methodology Change	02/03/2022
		Quality Issue 501000063397	11 Condensate Storage Tank Leaking	05/26/2022
	Corrective Action Documents Resulting from Inspection	Quality Issue 501000063580	NRC Information Request Omission	06/02/2022
		Quality Issue 501000063745	2020 11 CST External Inspection	06/10/2022

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Quality Issue 501000064858	License Renewal 22 Potential 11 CST Performance Deficiency	07/20/2022
		Quality Issue 501000064879	License Renewal 22 50.59 Applicability Determination for H71 Revision Change	07/20/2022
		Quality Issue 501000065166	Code Compliance Evaluation Inconsistencies	08/01/2022
		Quality Issue 501000065702	License Renewal 2022 - Justification for PM Change	08/18/2022
	Drawings	NF-39216-3	Flow Diagram Cooling Water - Aux Bldg, Unit 1	81
		NF-39255-1	Flow Diagram Diesel Generators D1 & D2, Unit 1 & 2	82
		NF-39264-1	Circulating Water Piping Plan, Unit 1	76
	Engineering Evaluations	WCAP-14040-A	Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves	4
		WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves	2
		WCAP-18660-NP	Analysis of Capsule N from the Xcel Energy Prairie Island Unit 1 Reactor Vessel Radiation Surveillance Program	0
	Miscellaneous		Pressure and Temperature Limits Report	4 & 8
			SW/MIC Asset Management Plan	2
		50.59 Applicability Determination and Prescreening 602000005419	PINGP Selective Leaching of Materials Program Procedure	0
		50.59 Screening 4232	USAR Appendix L Change for License Renewal Commitment #29	0
		501000000796	2016 Xcel Prairie Island CP Survey	09/06/2016
		501000012830	2018 Xcel Prairie Island CP Survey	05/07/2018
		Form QF0935	Underground Pipe or Tank Inspection	1
		Letter L-PI-09-133	License Amendment Request for Measurement Uncertainty Recapture - Power Uprate	12/28/2009
		Letter L-PI-10-035	Supplement to License Amendment Request for Measurement Uncertainty Recapture - Power Uprate,	04/23/2010

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Withdrawal of Proposed Change to Analysis Methodology for Pressure Temperature Limits Report (TAC Nos. ME3015 and ME30162)	
		Letter L-PI-19-002	10 CFR 50.55a Requests Nos. 1-RR-5-10 and 2-RR-5-10, Proposed Alternative to Reactor Vessel Inservice Inspection (ISI) Intervals for Prairie Island Unit 1 and Unit 2	06/13/2019
		Letter L-PI-22-005	Prairie Island Unit 1 Reactor Vessel Material Surveillance Program Report	03/07/2022
		PM 3108-2	Cooling Water Emergency Intake Structure 5 Year Inspection	6
		PMCR 613000003517	UT: 3-SA-13 IA WET AIR PIPING PM Reduction	05/06/2022
		Relief Request	Prairie Island Nuclear Generating Plant, Units 1 And 2 – Proposed Alternative to The Requirements of The ASME Code (EPID: L-2019-LLR-0055)	11/05/2019
		XCE-45274	Laboratory Inspection of Valves for Selective Leaching	02/10/2022
	NDE Reports	180-9231999-000	Prairie Island Unit-I, 1R29 Refueling Outage Reactor Vessel and Internals 10 Year ISI Visual Examination	01/06/2015
		180-9248410-000	Prairie Island Unit-2, 2R29 Refueling Outage Reactor Vessel and Internals 10 Year ISI Visual Examination	01/16/2016
		2R30-MIC-013	XH-106-95-A	09/26/2017
		BOP-UT-12-034	3-FO-44 for Buried Pipe Program	08/14/2012
		BOP-UT-18-006	Buried Fire Protection 10"	06/19/2018
		BOP-VT-11-040	Interior of 121 DSL Fire Pump Fuel Oil Storage Tank	06/21/2011
		BOP-VT-11-058	Interior of 121 DSL CLP FOST	11/01/2011
		BOP-VT-12-131	Interior of 121 DSL Generator Fuel Oil Storage Tank	01/28/2021
		BOP-VT-18-019	Gray Cast Iron Housing for Selective Leaching.	06/13/2018
		BOP-VT-21-016	21 Circulating Water Pump Casing Vent	10/12/2021
		MIC-08-014	XH-106-95-B	02/21/2008
		MIC-09-069	NF-39256-1 P1	10/03/2009
		NRP-GTFWI-15-001	Prairie Island Unit 2 Upper Internals MRP-227A Guide Tube Flange Welds Inspection Field Service Report	11/01/2015
		NSP-GTFWI-16-001	Prairie Island Unit 1 Upper Internals MRP-227 Guide Tube Flange Welds Inspection Field Service Report	10/27/2016
		WDI-PJF-131-	Prairie Island Unit 1 1R29 MRP-227A Visual Examination	11/19/2014

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		2682-FSR-001	Field Service Report	
		WDI-PJF-1314501-FSR-001 -NP	Prairie Island Nuclear Generating Plant 2R29 Lower Internals MRP-227A Visual Inspections	12/10/2015
		XCE-71600	Laboratory Inspection of Valves for Selective Leaching	02/03/2021
	Procedures	1D7	Unit 1 Reactor Vessel Closure	17
		2D7	Unit 2 Reactor Vessel Closure	17
		D92	Excavation	13
		FP-E-SE-03	10 CFR 50.59 and 72.48 Processes	18
		FP-PE-RLP-01	License Renewal Implementation	9
		FP-PE-SW-1	SW/MIC Program	15
		FP-PE-UPT-01	Underground Piping and Tanks Integrity Program	4
		H44	Reactor Vessel Integrity Program	33
		H58.1	PINGP Underground Piping and Tank Integrity Inspection Plan	1
		H65	License Renewal Implementation and Aging Management Programs	12
		H65.2.10	Compressed Air Monitoring Aging Management Program	4
		H65.2.12	Electrical Cables and Connections not Subject to Environmental Qualification Requirements Aging Management Program	6
		H65.2.13	Electrical Cables and Connections Not Subject to Environmental Qualification Requirements Used in Instrumentation Circuits Aging Management Program	5
		H65.2.2	Aboveground Steel Tanks Aging Management Program	3
		H65.2.21	Inaccessible Medium and Low Voltage Cables not Subject to Environmental Qualification Requirements Aging Management Program	4
		H65.2.25	Masonry Wall Aging Management Program	1
		H65.2.26	Metal Enclosed Bus Aging Management Program	4
		H65.2.31	Open-Cycle Cooling Water System Aging Management Program	12
		H65.2.35	Water-Control Structures Aging Management Program	1
		H65.2.36	Selective Leaching of Materials Aging Management Program	5

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		H65.2.8	Buried Piping & Tanks Inspection Aging Management Program	4
		H71	PINGP Selective Leaching of Materials Program Procedure	5
		PE 0005-TC	4.16 Kv Bus and Duct Inspection	14
		PM 3586-10	Periodic Structures Inspection	11
		PM 3586-11	Shield Building Caulking Inspection	6
		SP 2123	General Visual Examination of the Containment Moisture Barrier for ASME Subsection IWE	14
		TP 1626	Cathodic Protection Monthly Inspection	17
		TP1805	Instrument Air System Joint Integrity Test	8
		WDI-STD-088	Underwater Remote Visual Examination of Reactor Vessel Internals	11
	Work Orders	00403421	Inspect 11 CST for License Renewal	04/30/2013
		00416459	Perform NDE Ultrasonic Testing of 3-SA-13 IA Wet Air Piping	05/27/2011
		00478305	Perform NDE Ultrasonic Testing of 3-SA-13 IA Wet Air Piping	04/21/2014
		00521532	Perform NDE Ultrasonic Testing of 3-SA-13 IA Wet Air Piping	06/12/2016
		700022735	PM 3108-2 CL EMERG INTAKE STRUCTURE	04/10/2019
		700023185	3-SA-13 NDE UT of Air Piping	05/23/2018
		700028870	Visual Inspection of 22 CST Exterior	02/13/2019
		700060647	3-SA-13 NDE UT of Air Piping	05/21/2020
		700060840	Visual Inspection of 22 CST Exterior	11/22/2020
		700071661	Buried CL System Pipe License Renewal Inspection	08/20/2021
		700073194	PM 3586-10 Periodic Structures Inspection	05/21/2021
		700089254	PM 3586-10 Periodic Structures Inspection	12/13/2021
		700094404	Visual Inspection of 22 CST Exterior	04/26/2022