UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2056 WESTINGS AVENUE, SUITE 400
NAPERVILLE, IL 60563-2657

June 9, 2025

Werner K. Paulhardt, Jr.
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 – TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" INSPECTION REPORT 05000282/2025012 AND 05000306/2025012

Dear Werner Paulhardt:

On May 15, 2025, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Prairie Island Nuclear Generating Plant and discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

One finding of very low safety significance (Green) is documented in this report. This finding involved a violation of NRC requirements. We are treating this violation as a non-cited violation (NCV) consistent with Section 2.3.2 of the Enforcement Policy.

Licensee-identified violations which were determined to be of very low safety significance are documented in this report. 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 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 06/09/25

Néstor J Féliz Adorno, Branch 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

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Letter to Werner K. Paulhardt, Jr. from Néstor J Féliz Adorno dated June 9, 2025.

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT – TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" INSPECTION REPORT 05000282/2025012 AND 05000306/2025012

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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:	05000282/2025012 and 05000306/2025012
Enterprise Identifier:	I-2025-012-0009
Licensee:	Northern States Power Company, Minnesota
Facility:	Prairie Island Nuclear Generating Plant
Location:	Welch, MN
Inspection Dates:	May 05, 2025 to May 15, 2025
Inspectors:	M. Abuhamdan, Reactor Inspector J. Corujo-Sandin, Senior Reactor Inspector M. Leech, Senior Reactor Analyst J. Park, Reactor Inspector
Approved By:	Néstor J Féliz Adorno, Branch 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 Title 10 of the *Code of Federal Regulations* 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" Inspection 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. Licensee-identified non-cited violations are documented in report section: 37060.

List of Findings and Violations

<table>
<tr><td colspan="4">Failure to Properly Categorize Structures, Systems, and Components (SSCs) Based on Non-Probabilistic Risk Assessment (PRA) Seismic Margins Analysis (SMA) for Seismic Hazards</td></tr>
<tr><td>Cornerstone</td><td>Significance</td><td>Cross-Cutting Aspect</td><td>Report Section</td></tr>
<tr><td>Mitigating Systems</td><td>Green
NCV 05000282,05000306/2025012-01
Open/Closed</td><td>[H.9] - Training</td><td>37060</td></tr>
<tr><td colspan="4">The inspectors identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of Renewed Facility Operating License Condition 2.C(9), "Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,"" for the licensee's failure to categorize SSCs based on the approved non-PRA SMA screening process when considering seismic hazards. Specifically, the licensee failed to categorize the SSCs, including Units 1 and 2 Safety Injection (SI) system accumulators and Unit 2 Chemical and Volume Control (VC) system letdown orifice isolation valves, as high safety-significant (HSS) Risk Informed Safety Class (RISC)-1, even though they were identified as credited equipment in safety shutdown equipment list (SSEL) in their SMA.</td></tr>
</table>

Additional Tracking Items

None.

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

37060 - 10 CFR 50.69 Risk-Informed Categorization and Treatment of Structures, Systems, and Components Inspection

The inspectors partially reviewed the licensee's program and implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," in accordance with the procedure sections listed below. The inspection was conducted as the first phase of a multi-phase review, consistent with the phased approach allowed by the IP. Because the licensee had not yet implemented the alternative treatment provisions and subsequent feedback and process adjustments required by 10 CFR 50.69(d) and (e), this phase focused solely on system categorizations, as outlined in IP Sections 02.01 through 02.03. As of the date of this inspection, two refueling outages have not occurred since system categorization was completed on at least three systems.

Review of the Licensee's Programs and Procedures (IP Section 02.01) (1 Sample)

(1) (Partial)
The team reviewed the licensee's programs and procedures to ensure that the procedures fully described the categorization and treatment process for systems, structures, and components (SSCs) as described in its Updated Final Safety Analysis Report (UFSAR) and as required by 10 CFR 50.69.

Review of the Licensee's 10 CFR 50.69 Program Implementation (IP Section 02.02) (1 Sample)

(1) (Partial)
The team reviewed the licensee's 10 CFR 50.69 categorization completed on the following systems:
1. Cooling Water (CL) System
2. Safety Injection (SI) System
3. Containment Ventilation and Shield Building Ventilation (ZC/ZS) System

Problem Identification and Resolution (IP Section 02.03) (1 Sample)

(1) The inspectors reviewed the licensee's past audits and self-assessments performed on the implementation of the 10 CFR 50.69 program related to the system categorization process to ensure that it took adequate corrective actions from these audits.

INSPECTION RESULTS

Failure to Properly Categorize SSCs Based on Non-Probabilistic Risk Assessment (PRA) Seismic Margins Analysis (SMA) for Seismic Hazards			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Mitigating Systems	Green NCV 05000282,05000306/2025012-01 Open/Closed	[H.9] - Training	37060
The inspectors identified a finding of very low safety significance (Green) and an associated non-cited violation (NCV) of Renewed Facility Operating License Condition 2.C(9), "Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,"" for the licensee's failure to categorize SSCs based on the approved non-PRA SMA screening process when considering seismic hazards. Specifically, the licensee failed to categorize the SSCs, including Units 1 and 2 Safety Injection (SI) system accumulators and Unit 2 Chemical and Volume Control (VC) system letdown orifice isolation valves, as high safety-significant (HSS) Risk Informed Safety Class (RISC)-1, even though they were identified as credited equipment in safety shutdown equipment list (SSEL) in their SMA.			

Description:

On November 12, 2019, the NRC approved Prairie Island Nuclear Generating Plant (PINGP) License Amendment 230/218 (ML19276F684), which allows the implementation of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors." In the UFSAR, PINGP stated they will use the methodology for SSC categorization outlined in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance." PINGP also stated in the submittal letter dated July 20, 2018, that they will use the SMA performed for the Individual Plant Examination of External Events (IPEEE) in response to Generic Letter (GL) 88-20, Supplement 4, for evaluation of safety significance related to seismic hazards during the categorization process.

NEI 00-04 adopted an approach that utilizes the SMA SSEL as a screening process. Once all system functions and associated SSCs involved in the seismic margin success path are identified, they are included in the SMA SSEL. The SSCs included in the SMA SSEL result in the HSS categorization according to the screening process. At the time of the submittal of their license amendment request, the licensee stated they had performed review of the as-built and as-operated plant conditions against the original SMA SSEL. Differences were reviewed to identify any potential impacts to the equipment credited on the SSEL, and appropriate changes were made and documented in their PRA document V.SPA.18.010, "Prairie Island Nuclear Generating Station – 10 CFR 50.69 Seismic IPEEE Equipment List Review," Revision 0. The SSCs identified in this document served as the basis for the licensee's SSC categorization evaluations related to seismic hazards.

On April 30, 2025, the inspectors performed a review of PRA document V.SPA.18.010. The document included Table A, IPEEE Seismic Essential Equipment List, synonymous to the SMA SSEL, that identified the SSCs credited as seismic safe shutdown success path. The SSCs included in the table were to be screened as HSS. Among the SSCs included in the table were Units 1 and 2 SI system accumulators 101-011, 101-012, 201-031, and 201-032.

The SI system was categorized by the licensee under the rule of 10 CFR 50.69 and documented in their system categorization document (SCD) PI-SCD-SI, "SCD for SI System," Revision 1. During the review of the SCD, the inspectors identified that all SI system accumulators had been improperly assigned a final categorization of low safety-significant (LSS) RISC-3, as indicated in the SCD Appendices J1.3 and J2.3, RISC-3 Components for Unit 1 and Unit 2, respectively. The inspectors also noted that the SCD Appendices K1 and K2, Basis for HSS Categorization for Unit 1 and Unit 2, respectively, included a section for the non-PRA-modeled seismic risk that did not identify the SI system accumulators as HSS. Further line of inspector inquiry revealed that not all SSCs included in the licensee's updated IPEEE Seismic Essential Equipment List, or the SMA SSEL, had been reflected as HSS in the seismic risk section in Appendices K1 and K2. The preliminary extent-of-condition review further identified that Unit 2 VC system letdown orifice isolation valves CV31347, CV31348, and CV31349, had been improperly assigned the final categorization of LSS RISC-3 when they should have been categorized as HSS RISC-1 according to the approved SMA SSEL screening process.

Corrective Actions: The licensee entered the issues into their corrective action program (CAP) and initiated actions to revise the affected SCDs to properly reflect the SMA SSEL screening of the SSCs.

Corrective Action References: CAP 501000098100

Performance Assessment:

Performance Deficiency: The licensee's failure to properly categorize SSCs (including the SI system accumulators 101-011, 101-012, 201-031 and 201-032, and VC system letdown orifice isolation valves CV31347, CV31348 and CV31349) as HSS RISC-1 based on the approved non-PRA SMA screening process related to seismic hazards was contrary to PINGP's Renewed Facility Operating License Condition 2.C(9) and was a performance deficiency.

Screening: The inspectors determined the performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, because the SSCs were improperly categorized as LSS RISC-3, the licensee's program and procedures would allow the removal of special treatment requirements that must remain in place for HSS RISC-1 SSCs. The removal of the special treatment requirements would not provide reasonable assurance that the SSCs would perform their intended safety functions.

Significance: The inspectors assessed the significance of the finding using IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings At-Power." The inspectors answered "Yes" to Question A.1, "If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or PRA functionality?" under Exhibit 2 – Mitigating Systems Screening Questions. Accordingly, the finding was determined to be of very low safety significance (Green).

Cross-Cutting Aspect: H.9 - Training: The organization provides training and ensures knowledge transfer to maintain a knowledgeable, technically competent workforce and instill nuclear safety values. Specifically, inspector interviews revealed that knowledge transfer and retention strategies had not been effectively implemented to capture the knowledge and skill of experienced individuals to advance those of less experienced individuals during the system categorization reviews for the affected SSCs.

Enforcement: Violation: Prairie Island Renewed Facility Operating License Condition 2.C(9) for Units 1 and 2 states, in part, that "NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using [...] the results of non-PRA evaluations that are based on the IPEEE Screening Assessments for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk" as specified in License Amendment 230/218 dated November 12, 2019. Contrary to the above, as of April 30, 2025, the licensee failed to implement 10 CFR 50.69 using the approaches for categorization of SSCs using the results of non-PRA evaluations that are based on the SMA. Specifically, the licensee failed to categorize SSCs (including Units 1 and 2 SI system accumulators 101-011, 101-012, 201-031 and 201-032, and Unit 2 VC system letdown orifice isolation valves CV31347, CV31348 and CV31349) as HSS RISC-1 based on the approved SMA screening process. Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.

Licensee-Identified Non-Cited Violation	37060
This violation of very low safety significance was identified by the licensee and has been entered into the licensee corrective action program and is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.	
Violation: Prairie Island Renewed Facility Operating License Condition 2.C(9) for Units 1 and 2 states, in part, that "NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events" as described in NSPM's submittal letter dated July 20, 2018, and all its subsequent supplements as specified in License Amendment 230/218 dated November 12, 2019. Section 3.1.1 of the submittal letter dated July 20, 2018, states that NSPM will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. NSPM established procedure FP-ENG-RIEP-02, "Risk-Informed Engineering Program System Categorization," Revision 1, to satisfy NEI 00-04. Section 5.5.1 of FP-ENG-RIEP-02 states, in part, "The Defense in Depth (DID) Evaluation is performed in accordance with Section 6 of NEI 00-04, 10 CFR 50.69 SSC Categorization Guideline […]. The purpose of this assessment is to determine the sufficiency of Safety-Related (SR) LSS SSCs by confirming that DID remains preserved. All LSS SSCs are subject to this DID evaluation." Section 5.5.3.1 of FP-ENG-RIEP-02 states, in part, "PERFORM a Core Damage DID evaluation for each LSS system function and SSC […]." Contrary to the above, the licensee failed to implement 10 CFR 50.69 using the approaches for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs using PRA models to evaluate risk associated with internal events as described in NSPM's submittal letter dated July 20, 2018. Specifically, the licensee did not implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. The licensee failed to perform a core damage DID evaluation for each LSS system function and SSC to determine	

the sufficiency of SR LSS SSCs by confirming that the DID remained preserved, as evidenced by the following examples: • As of February 24, 2025, the licensee incorrectly assumed that the core damage DID assessment was not required if component functions were not modeled in PRA. As a result, LSS system functions CL-2.3, MS-2.2b, SI-4.2, and SI-2.3 were identified as having been excluded within the scope of the core damage DID assessment during the initial categorization process. • As of April 17, 2025, the licensee failed to perform a core damage DID evaluation for the LSS active function of 26 SR valves in CL, Containment Spray (CS), Reactor Coolant (RC), Residual Heat Removal (RH), and VC systems to confirm the DID remained preserved. Significance/Severity: Green. The finding was screened using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors answered “Yes” to Question A.1, “If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or PRA functionality?” under Exhibit 2 – Mitigating Systems Screening Questions. Accordingly, the finding was determined to be of very low safety significance (Green). Corrective Action References: CAPs 501000095497 and 501000097345

Licensee-Identified Non-Cited Violation	37060
This violation of very low safety significance was identified by the licensee and has been entered into the licensee corrective action program and is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.	
Violation: 10 CFR 50.69(c)(1)(ii) requires, in part, the licensee to determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external) and SSCs. Contrary to the above, as of July 14, 2023, the licensee failed to determine SSC functional importance using an integrated, systematic process for addressing initiating events and SSCs. Specifically, the licensee did not determine the functional importance of Units 1 and 2 RH System Function 2.2 as HSS based on the PRA assessment. As a result, the functional importance of all SSCs mapped to System Function RH-2.2, which should have been determined as candidate HSS, were incorrectly determined as candidate LSS. Significance/Severity: Green. The finding was screened using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors answered “Yes” to Question A.1, “If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or PRA functionality?” under Exhibit 2 – Mitigating Systems Screening Questions. Accordingly, the finding was determined to be of very low safety significance (Green). Corrective Action References: CAP 501000075226	

Licensee-Identified Non-Cited Violation	37060
This violation of very low safety significance was identified by the licensee and has been entered into the licensee corrective action program and is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.	

Violation: Prairie Island Renewed Facility Operating License Condition 2.C(9) for Units 1 and 2 states, in part, “NSPM is approved to implement 10 CFR 50.69 using the approaches for categorization of Risk Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events” as described in NSPM’s submittal letter dated July 20, 2018, and all its subsequent supplements as specified in License Amendment 230/218 dated November 12, 2019. Section 3.1.1 of the submittal letter dated July 20, 2018, states that NSPM will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201.

NSPM established procedure FP-ENG-RIEP-03, “PRA Evaluations of Component Risk Significance for Risk-Informed Engineering Program,” Revision 2, to satisfy NEI 00-04. Section 5.5.1 of FP-ENG-RIEP-03 states, in part, “Upon completion of the system categorization, both prior to the initial IDP review and following the IDP review, PERFORM a system sensitivity for the categorized system’s components modeled in the PRA models.” Section 5.5.8 of FP-ENG-RIEP-03 states, “PERFORM a cumulative sensitivity following the SSC categorization of all systems.”

Contrary to the above, as of April 17, 2025, the licensee failed to implement 10 CFR 50.69 using the approaches for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs using PRA models to evaluate risk associated with internal events as described in NSPM’s submittal letter dated July 20, 2018. Specifically, the licensee did not implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. The licensee did not perform system sensitivity studies for 26 categorized valves in CL, CS, RC, RH, and VC systems modeled in the PRA models upon completion of their system categorization, nor did it perform a cumulative sensitivity study analysis after completing the SSC categorization of all systems. These studies were necessary to confirm that the categorization process resulted in acceptably small increases in core damage frequency (CDF) and large early release frequency (LERF) for the valves’ LSS active function.

Significance/Severity: Green. The finding was screened using IMC 0609 Appendix A, “The Significance Determination Process (SDP) for Findings At-Power.” The inspectors answered “Yes” to Question A.1, “If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or PRA functionality?” under Exhibit 2 – Mitigating Systems Screening Questions. Accordingly, the finding was determined to be of very low safety significance (Green).

Corrective Action References: CAP 501000097345

EXIT MEETINGS AND DEBRIEFS

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

- On May 15, 2025, the inspectors presented the Title 10 of the *Code of Federal Regulations* 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors” Inspection results to Werner Paulhardt, and other members of the licensee staff.

DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
37060	Calculations	PRA-CALC-PI-EPCS	50.69 Cumulative Sensitivity Studies for RIEP	1
37060	Corrective Action Documents	500000331988	OTHA: Review RC-02 Maintenance Rule Scope	10/09/2024
37060	Corrective Action Documents	501000061114	Potential 50.69 Program Risk Impact	03/08/2022
37060	Corrective Action Documents	501000063582	All LSS Captured for Sensitivity Studies	06/09/2022
37060	Corrective Action Documents	501000075226	PI 50.69 Categorization	07/14/2023
37060	Corrective Action Documents	501000088525	Trip Device is Not Working	08/05/2024
37060	Corrective Action Documents	501000095497	50.69 Categorization Process	02/24/2025
37060	Corrective Action Documents	501000096710	SI MOV Motor SR vs. NSR	03/28/2025
37060	Corrective Action Documents	501000097345	50.69 Scope of Sensitivity Studies	04/17/2025
37060	Corrective Action Documents	501000097717	50.69 System Categorization Doc Issues	04/24/2025
37060	Corrective Action Documents Resulting from Inspection	501000097716	Incorrect Safety Class in CL SCD	04/24/2025
37060	Corrective Action Documents Resulting from Inspection	501000098100	50.69 INSP: Seismic SSCs Are LSS	04/30/2025
37060	Corrective Action Documents Resulting from Inspection	501000098360	50.69 INSP: Scoping Components for a Given System	05/05/2025

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
37060	Corrective Action Documents Resulting from Inspection	501000098426	50.69 INSP: Procedural Discrepancy	05/07/2025
37060	Corrective Action Documents Resulting from Inspection	501000098461	50.69 INSP: RISC-2 Evaluation	05/07/2025
37060	Corrective Action Documents Resulting from Inspection	501000098496	50.69 INSP: SI Documentation Issues	05/08/2025
37060	Corrective Action Documents Resulting from Inspection	501000098702	50.69 INSP: RISC-2 MRule Requirement	05/13/2025
37060	Corrective Action Documents Resulting from Inspection	501000098705	50.69 INSP: Create MRule Function	05/13/2025
37060	Corrective Action Documents Resulting from Inspection	501000098769	50.69 INSP: RIEP-03 Editorial Correction	05/14/2025
37060	Drawings	NE-40008 Sheet 128	Prairie Island Nuclear Generating Plant	76
37060	Drawings	NE-40008 Sheet 64	Prairie Island Nuclear Generating Plant	78
37060	Drawings	NE-40406 Sheet 42	Prairie Island Nuclear Generating Plant	76
37060	Drawings	NE-40406 Sheet 92	Prairie Island Nuclear Generating Plant	76
37060	Drawings	NF-40211-1	Wiring Diagram Bus-1 Motor Control Center 1LA	76
37060	Drawings	NF-40211-2	Wiring Diagram Bus-2 Motor Control Center 1LA	Y

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		Sheet 2 of 2		
37060	Drawings	NF-40585-1	Wiring Diagram Bus-1 Motor Control Center 2LA	76
37060	Drawings	NF-40585-2	Wiring Diagram Bus 2 Motor Control Center 2LA	P
37060	Engineering Evaluations	PI-SCD-CL	Prairie Island Nuclear Generating Station 10 CFR 50.69 Categorization Document – Cooling Water System	1
37060	Engineering Evaluations	PI-SCD-RC	Prairie Island Nuclear Generating Station 10 CFR 50.69 Categorization Document – Reactor Coolant System	1
37060	Engineering Evaluations	PI-SCD-SI	Prairie Island Nuclear Generating Station 10 CFR 50.69 Categorization Document – SI System	1
37060	Engineering Evaluations	PI-SCD-ZC-ZS	Prairie Island Nuclear Generating Station 10 CFR 50.69 Categorization Document – Containment Ventilation & Shield Building Ventilation System	1
37060	Miscellaneous	CWO-EN-RIEPR-1-2-001	Create SQA Paperwork for Newly Developed Software - RIEPR	05/18/2023
37060	Miscellaneous	PINGP-PCDs-completed-after-5-4	PRA Change Database	5-4
37060	Miscellaneous	PRA-CALC-MT-23-003	RIEPR (Risk Informed Engineering Program) User's Manual	07/19/2023
37060	Miscellaneous	V.SPA.18.010	Prairie Island Nuclear Generating Station - 10 CFR 50.69 Seismic IPEEE Equipment List Review	0
37060	Procedures	FP-E-MR-01	Maintenance Rule Process	12
37060	Procedures	FP-E-RTC-02	Functional Location Classification	25
37060	Procedures	FP-ENG-RIEP-01	Risk-Informed Engineering Programs (RIEP)	1
37060	Procedures	FP-ENG-RIEP-02	Risk-Informed Engineering Program System Categorization	1
37060	Procedures	FP-ENG-RIEP-03	PRA Evaluations of Component Risk Significance for Risk-Informed Engineering Program	2
37060	Procedures	FP-ENG-RIEP-04	Risk-Informed Engineering Program Passive Component Categorization	2
37060	Procedures	FP-ENG-RIEP-05	Risk-Informed Engineering Programs Maintenance Requirements	1
37060	Procedures	FP-ENG-RIEP-06	Alternative Treatment Process Risk-Informed Engineering Program	0
37060	Procedures	FP-OP-IDP-02	Integrated Decision Making Panel - Risk Informed Engineering	2

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
			Program	
37060	Procedures	FP-PE-PRA-02	PRA Guideline for Model Update and Maintenance	21
37060	Self-Assessments	606000001603	SnapShot Report for PI 50.69 NRC Inspection 2022	02/18/2022
37060	Self-Assessments	606000002243 and 600001239199	AER SSA, 2025 Self-Assessment for 50.69 NRC Inspection	04/24/2025