UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 28, 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

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS 239 AND 227 RE: 24-MONTH OPERATING CYCLE (EPID L-2021-LLA-0146)

Dear Mr. Domingos:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 239 to Renewed Facility Operating License No. DPR-42 and Amendment No. 227 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications in response to your application dated August 6, 2021, as supplemented by letters dated December 7, 2021, March 7, 2022, and June 10, 2022.

The amendments revise the technical specifications to accommodate a 24-month operating cycle with a 25 percent grace period to permit up to 30 months to complete surveillance requirements.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures:
1. Amendment No. 239 to DPR-42
2. Amendment No. 227 to DPR-60
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 239
Renewed License No. DPR-42

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated August 6, 2021, as supplemented by letters dated December 7, 2021, March 7, 2022, and June 10, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-42 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 239, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2022.07.28 11:41:49 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 28, 2022

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
Renewed License No. DPR-60

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), dated August 6, 2021, as supplemented by letters dated December 7, 2021, March 7, 2022, and June 10, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2022.07.28 11:42:23 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 28, 2022

ATTACHMENT TO LICENSE AMENDMENT NOS. 239 AND 227

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Renewed Facility Operating License Nos. DPR-42 and DPR-60 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicate the areas of change.

Renewed Facility Operating License No. DPR-42

REMOVE	INSERT
Page 3	Page 3

Renewed Facility Operating License No. DPR-60

REMOVE	INSERT
Page 3	Page 3

Technical Specifications

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3.2-10	3.3.2-10
5.5-2	5.5-2
5.5-3	5.5-3
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6
5.5-19	5.5-19
5.5-20	5.5-20
5.5-21	5.5-21

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purpose of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 239, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission -approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, NSPM to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility;

(6) Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to transfer byproduct materials from other job sites owned by NSPM for the purposes of volume reduction and decontamination.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 1677 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, are hereby incorporated in the renewed operating license. NSPM shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

NSPM shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.5	NA
b. Automatic Actuation Relay Logic	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.8	NA
c. High Containment Pressure	1, 2, 3	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≤ 4.0 psig
d. Pressurizer Low Pressure	1, 2, 3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 1760 psig
e. Steam Line Low Pressure	1, 2, 3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.3 SR 3.3.2.6	≥ 505[(b)] psig
2. Containment Spray					
a. Manual Initiation	1, 2, 3, 4	2	B	SR 3.3.2.4	NA
b. Automatic Actuation Relay Logic	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.8	NA

(a) Pressurizer Pressure ≥ 2000 psig.
(b) Time constants used in the lead/lag controller are $t_1 \geq 12$ seconds and $t_2 \leq 2$ seconds.

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed. The date (i.e., month and year) the change was implemented shall be indicated.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The systems include portions of the Residual Heat Removal and Safety Injection Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

5.5 Programs and Manuals

5.5.3 Post Accident Sampling (continued)

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable.

This program shall allocate releases equally to each unit. The liquid radwaste treatment system, waste gas treatment system, containment purge release vent, and spent fuel pool vent are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. Therefore, all releases will be allocated equally in determining conformance to the design objectives of 10 CFR 50, Appendix I.

The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days from the liquid effluent releases would exceed 0.12 mrem to the total body or 0.4 mrem to any organ; or from the gaseous effluent releases would exceed 0.4 mrad for gamma air dose, 0.8 mrad for beta air dose, or 0.6 mrem organ dose;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. for noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 4.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT or PT) of exposed surfaces of the removed flywheels may be conducted at 20 intervals.

5.5 Programs and Manual (continued)

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operations and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Semiquarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified 2 years or less in the Inservice Testing Program for performing inservice testing activities.

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered in-leakage measured by the testing described in paragraph c. The unfiltered air in-leakage limit for radiological challenges is the in-leakage flow rate assumed in the licensing basis analysis of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions of the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability and determining CRE unfiltered in-leakage as required by paragraph c.

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The 24-Month Fuel Cycle related Surveillance Requirement Frequency changes approved by the NRC in Units 1 and 2 License Amendments 239/227 were not subject to provision b. Subsequent changes are subject to the Surveillance Frequency Control Program.

5.5 Programs and Manuals

5.5.17 Surveillance Frequency Control Program (continued)

d. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.5.18 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODES 1 and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

    1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

    2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

    3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

5.5 Programs and Manuals

5.5.18 Risk Informed Completion Time Program (continued)

1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 239 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 227 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated August 6, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21218A093), as supplemented by letters dated December 7, 2021 (ML21341B375), March 7, 2022 (ML22066B341), and June 10, 2022 (ML22161A943), Northern States Power Company, a Minnesota Corporation, doing business as Xcel Energy (NSPM, the licensee), requested changes to the technical specifications (TSs) for Prairie Island Nuclear Generating Plant, Units 1 and 2 (Prairie Island).

The proposed changes would revise the Prairie Island licensing basis and make corresponding changes to the TSs to implement a 24-month operating cycle with a 25 percent grace period to permit up to 30 months to complete surveillance requirements. The corresponding TS changes in support of the change in the maximum surveillance intervals from 24 months to 30 months includes a change to one TS Allowable Value (AV) and changes to TS 5.5.2 and TS 5.5.17. The change to TS 5.5.2 implements TSTF-299 “Administrative Controls Program 5.5.2.b Test Interval and Exception” (ML040620202), which clarifies the intent of refueling cycle intervals with respect to the system leak test requirements. The change to TS 5.5.17 documents the use of NRC Generic Letter 91-04 (GL 91-04) “Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle” (ML013100215), to increase the surveillance requirement (SR) intervals in lieu of the TS program.

The supplemental letters dated December 7, 2021, March 7, 2022, and June 10, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the Federal Register on October 5, 2021 (86 FR 55013).

2.0 REGULATORY EVALUATION

2.1 Background

Improved reactor fuels allow licensees to consider an increase in the duration of the fuel operating for their facilities. The NRC staff has reviewed requests for individual plants to modify TS surveillance intervals to be compatible with a 24-month operating cycle. The NRC staff issued GL 91-04 to provide guidance to licensees for preparing such license amendment requests (LARs).

On March 15, 2018, NSPM requested an amendment to the Prairie Island TSs to implement TS Task Force (TSTF) Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b" (ML18074A308). By letter dated April 16, 2019, the NRC issued Amendment Nos. 226 and 214 which approved the requested TS changes (ML19045A480). Amendment Nos. 226 and 214 incorporated TS 5.5.15, "Surveillance Frequency Control Program (SFCP)," into the Prairie Island TSs.

With the adoption of TSTF-425, TS SR 3.0.2 was updated to align with the NUREG-1431 "Standard Technical Specifications – Westinghouse Plants: Specifications" (ML21259A155), TS SR 3.0.2 and removed the restriction from TS SR 3.0.2. The restriction was relocated with the specific 24-month frequencies under SFCP defined by TSs. The limitation on grace was relocated from TS SR 3.0.2 to the surveillance test interval (STI) document. The LAR noted that the Prairie Island STI states, "Unless noted otherwise, TS SR 3.0.2 does not apply to the SRs in Table 2.1 that have a frequency of 24 months."

2.2 Proposed Changes

2.2.1 Changes to SRs in the Prairie Island STI

The LAR proposed to modify the following SRs with 24-month intervals in the Prairie Island STI which have limitations on the application of grace and would be revised to allow a grace period of up to 6 months or 30 months total:

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

- SR 3.3.1.10, Perform CHANNEL CALIBRATION
- SR 3.3.1.11, Perform CHANNEL CALIBRATION
- SR 3.3.1.12, Perform CHANNEL CALIBRATION
- SR 3.3.1.13, Perform Channel Operational Test (COT)
- SR 3.3.1.14, Perform Trip Actuating Device Operational Test (TADOT)

TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation

- SR 3.3.2.6, Perform CHANNEL CALIBRATION
- SR 3.3.2.7, Perform MASTER RELAY TEST

TS 3.3.3, Event Monitoring (EM) Instrumentation

- SR 3.3.3.2, Perform CHANNEL CALIBRATION

TS 3.3.6, Control Room Special Ventilation System (CRSVS) Actuation Instrumentation

- SR 3.3.6.3, Perform TADOT
- SR 3.3.6.4, Perform CHANNEL CALIBRATION

TS 3.4.1, RCS [Reactor Coolant System] Pressure, Temperature, and Flow – Departure from Nucleate Boiling (DNB) Limit

- SR 3.4.1.3, Verify RCS total flow rate is within the limit specified in the Core Operating Limits Report (COLR)

TS 3.4.9, Pressurizer

- SR 3.4.9.2, Verify capacity of each required group of pressurizer heaters is ≥ 100 kW
- SR 3.4.9.3, Verify required pressurizer heaters are capable of being powered from an emergency power supply

TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)

- SR 3.4.11.2, Perform a complete cycle of each PORV

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSPT) > Safety Injection (SI) Pump Disable Temperature

- SR 3.4.12.5, Perform CHANNEL CALIBRATION for each Over Pressure Protection System (OPPS) actuation channel

TS 3.4.13, LTOP – Reactor Coolant System Cold Leg Temperature (RCSPT) ≤ Safety Injection (SI) Pump Disable Temperature

- SR 3.4.13.6, Perform CHANNEL CALIBRATION for each OPPS actuation Channel

TS 3.4.16, RCS Leakage Detection Instrumentation

- SR 3.4.16.3, Perform CHANNEL CALIBRATION of the required containment sump monitor
- SR 3.4.16.4, Perform CHANNEL CALIBRATION of the required containment radionuclide monitor

TS 3.5.2, Emergency Core Cooling System (ECCS) - Operating

- SR 3.5.2.9, Verify each ECCS throttle valve listed in SR 3.5.2.9 is in the correct position
- SR 3.5.2.10, Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion

TS 3.6.5, Containment Spray and Cooling Systems

- SR 3.6.5.4, Verify cooling water flow rate to each containment fan coil unit is ≥ 900 gpm

TS 3.6.8, Vacuum Breaker System

- SR 3.6.8.2, Perform CHANNEL CALIBRATION

TS 3.7.3, Main Feedwater Regulation Valves (MFRVs) and MFRV Bypass Valves

- SR 3.7.3.2, Verify each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal

TS 3.7.4, Steam Generator (SG) PORVs

- SR 3.7.4.2, Verify one complete manual cycle of each SG PORV block valve

<u>TS 3.7.10, Control Room Special Ventilation System (CRSVS)</u>

- SR 3.7.10.4, Verify each CRSVS train in the Emergency Mode delivers 3600 to 4400 cfm through the associated CRSVS filters

<u>TS 3.8.1, AC [alternating current] Sources - Operating</u>

- SR 3.8.1.11, Verify on an actual or simulated loss of offsite power that the diesel generator (DG) auto-starts from standby condition

<u>TS 3.9.3, Nuclear Instrumentation</u>

- SR 3.9.3.2, Perform CHANNEL CALIBRATION of required channels

2.2.2 TS Changes

<u>TS Allowable Values</u>

The LAR proposed a change to TS Table 3.3.2-1, Function 1e, Safety Injection – Steam Line Low Pressure, Allowable Value, which would be revised from ≥ 500 pounds per square inch gauge (psig) to ≥ 505 psig.

<u>TS 5.5 Programs and Manuals</u>

The LAR proposed to revise TS 5.5.2 to modify the frequency requirements from "at refueling cycle intervals or less" to "at least once per 24 months" and to add that the provisions of SR 3.0.2 are applicable.

The LAR proposed to revise TS 5.5.17 to include a new provision "c." and modified existing provision "c." to provision "d." to account for the new provision. The proposed new provision "c." would be included to indicate that the changes proposed in the LAR were not subject to the SFCP and that subsequent changes would be subject to the SFCP.

2.3 <u>Regulatory Requirements and Guidance</u>

2.3.1 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," details the content and information that must be included in a facility's TS. Specifically, 10 CFR 50.36 states, in part:

> Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section.

The regulation at 10 CFR 50.36(c)(3) states:

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Additionally, 10 CFR 50.36(c)(5) states:

> Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The regulation at 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," requires that preventative maintenance activities must not reduce the overall availability of the systems, structures, and components.

2.3.2 Regulatory Guidance

GL 91-04, Enclosure 1, indicates that SRs with an 18-month frequency requirement that are not instrument calibration related (i.e., non-calibration changes) should be evaluated for the effect on safety associated with an extension to a 24-month required interval. This evaluation should address the following three criteria:

1. The licensee should analyze the effect on plant safety from the change in surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.
2. The licensee should confirm that historical plant maintenance and surveillance data support the conclusion that the effect on safety is small.
3. The licensee should confirm that the performance of surveillance at the bounding surveillance interval limit would not invalidate any assumption in the plant licensing basis.

For those SRs where the evaluation accomplishes these goals, the licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components. No change in the existence, testability, or availability of plant systems and components is being requested in this LAR, only an extension in the frequency of the tests or inspections.

GL 91-04, Enclosure 2, identifies the following seven steps for the evaluation of instrumentation calibration related (i.e., Calibration Changes) frequency changes.

1. Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval.

2. Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

3. Confirm the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model, number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

4. Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.

5. Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to affect a safe shutdown with the associated instrumentation.

6. Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant surveillance procedures for channel checks, channel functional tests, and channel calibrations.

7. Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety.

Regulatory Guide (RG) 1.105, Revision 4, “Setpoints for Safety-Related Instrumentation,” dated February 2021, (ADAMS Accession No. ML20330A329) describes a method acceptable to the NRC staff for complying with the NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits. RG 1.105 Revision 4 endorses American National Standards Institute / International Society of Automation (ANSI/ISA) Standard 67.04.01-2018, “Setpoints for Nuclear Safety-Related Instrumentation.” The NRC staff used this guide to establish the adequacy of the licensee’s setpoint calculation methodologies and the related plant surveillance procedures.

Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical specifications,’ Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,” dated August 24, 2006 (ML051810077), addresses requirements on limiting safety system settings that are assessed during the periodic testing and calibration of instrumentation. RIS 2006-17 discusses issues that could occur during the testing of limiting safety system settings and that, therefore, may have an adverse effect on equipment operability.

3.0 TECHNICAL EVALUATION

3.1 Method Applied to Surveillance Failure Analysis

The license amendment request (LAR) stated that surveillance failure analysis (SFA) was applied to those devices with non-calibration, calibration SRs with setpoints (TS Allowable Values) and calibration SRs without setpoints (no TS Allowable Values) interval changes. A failure history for each 24-month SR and its components was evaluated. The primary focus of the SFA dealt with those failures that could only be detected by the performance of the SR, and that by its extension to 30 months, may result in a decrease in availability (operability) of that function.

As these types of failures are not related to potential unavailability due to testing interval extension, the failures evaluated in this submittal exclude failures that:

1. Did not impact a TS safety function or TS operability.
2. Are detectable by required testing performed more frequently than the SR being extended; or
3. The cause can be attributed to an associated event such as a preventative maintenance task, human error, previous modification, or previously existing design deficiency; or that were subsequently re-performed successfully with no intervening corrective maintenance (e.g., plant conditions or malfunctioning measurement and test equipment may have caused aborting the test performance).

3.2 Non-Calibration Changes

The non-calibration SRs subject to the extension from 24 to 30 months are:

TS 3.3.1, Reactor Trip System (RTS) Instrumentation

- SR 3.3.1.13, Perform Channel Operational Test (COT)
- SR 3.3.1.14, Perform Trip Actuating Device Operational Test (TADOT)

TS 3.3.6, Control Room Special Ventilation System (CRSVS) Actuation Instrumentation

- SR 3.3.6.3, Perform TADOT

TS 3.4.1, RCS Pressure, Temperature, and Flow – Departure from Nucleate Boiling (DNB) Limit

- SR 3.4.1.3, Verify RCS total flow rate is within the limit specified in the COLR

TS 3.4.9, Pressurizer

- SR 3.4.9.2, Verify capacity of each required group of pressurizer heaters is ≥ 100 kW
- SR 3.4.9.3, Verify required pressurizer heaters are capable of being powered from an emergency power supply

TS 3.4.11, Pressurizer Power Operated Relief Valves (PORVs)

- SR 3.4.11.2, Perform a complete cycle of each PORV

TS 3.5.2, Emergency Core Cooling System (ECCS) - Operating

- SR 3.5.2.9, Verify each ECCS throttle valve listed in SR 3.5.2.9 is in the correct position
- SR 3.5.2.10, Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet strainers show no evidence of structural distress or abnormal corrosion

TS 3.6.5, Containment Spray and Cooling Systems

- SR 3.6.5.4, Verify cooling water flow rate to each containment fan coil unit is ≥ 900 gpm

TS 3.7.3, Main Feedwater Regulation Valves (MFRVs) and MFRV Bypass Valves

- SR 3.7.3.2, Verify each MFRV and MFRV bypass valve actuates to the isolation position on an actual or simulated actuation signal

TS 3.7.4, Steam Generator (SG) PORVs

- SR 3.7.4.2, Verify one complete manual cycle of each SG PORV block valve

TS 3.7.10, Control Room Special Ventilation System (CRSVS)

- SR 3.7.10.4, Verify each CRSVS train in the Emergency Mode delivers 3600 to 4400 cfm through the associated CRSVS filters

TS 3.8.1, AC Sources - Operating

- SR 3.8.1.11, Verify on an actual or simulated loss of offsite power that the diesel generator (DG) auto-starts from standby condition

3.2.1 Licensee Evaluation of Non-Calibration Changes

GL 91-04 identifies three steps to evaluate non-calibration changes. LAR section 4.3 provided the licensee's general evaluations of these three steps.

STEP 1: Licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.

Licensee Evaluation of STEP 1

The LAR stated each proposed non-calibration SR interval change has been evaluated with respect to the effect on plant safety. The methodology used was based on whether the function or feature is tested on a more frequent basis during the operating cycle by other plant programs, is designed to have a redundant counterpart(s) or be single failure proof, or is highly reliable.

A summary of the evaluation of the effect on safety for each proposed non-calibration SR interval change was provided in LAR in Enclosure 2 and the supplement dated June 10, 2022.

STEP 2: Licensees should confirm that historical plant maintenance and surveillance data support this conclusion.

Licensee Evaluation of STEP 2

The LAR stated that test history has been evaluation on SR test results and associated maintenance records back to at least 2010 and as far back as 1999 in some cases. The LAR stated that potential common cause failures of similar components tested by different SRs were also evaluated. In addition, potential common failures of similar components tested by other SRs were evaluated to determine whether evidence of repetitive failures among similar plant components existed. The LAR stated that the evaluation, as documented in Enclosure 2, determined that current plan programs are adequate to ensure system reliability.

The LAR stated that the review of SR test history justifies that applying a 6-month grace period to SRs with a 24-month interval will have minimal impact, if any, on system availability. Specific SR test failures and justification for this conclusion are discussed in Enclosure 2 and the supplement dated June 10, 2022.

STEP 3: Licensees should confirm that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding SR interval limit provided to accommodate a 24-month fuel cycle.

Licensee Evaluation of STEP 3

The LAR stated that the impact of the proposed SR changes was reviewed, including a review of the Prairie Island Updated Safety Analysis Report (USAR) and commitments and no assumptions in the plant licensing basis would be invalidated by the proposed SR interval changes.

3.2.2 NRC Staff Evaluation of Non-Calibration SR Interval Changes

The NRC staff reviewed the evaluation methods applied by the licensee related to non-calibration changes in the LAR. The LAR stated that in the event of degraded performance of a component within the surveillance program, the issue would be addressed via the Maintenance Rule program.

The NRC staff noted during its review of the LAR that Enclosure 2 did not include evaluations for non-calibration SRs 3.3.1.13 and 3.3.6.3 which were included in LAR Section 3.1 as SRs that would be revised by the proposed amendment. The supplement dated June 10, 2022, provided the evaluation for the proposed changes for SRs 3.3.1.13 and 3.3.6.3.

Based upon the NRC staff's review of the LAR, as supplemented, the changes meet the guidance of GL 91-04. The NRC staff finds the proposed interval extensions for the non-calibration SRs acceptable because (1) the effect on safety would be small, (2) historical data do not contradict this conclusion, and (3) no assumptions in the plant licensing basis would be invalidated as a result of the proposed change.

3.3 Calibration Changes

The calibration SRs subject to the extension from 24 to 30 months are:

TS 3.3.1, Reactor Trip System (RTS) Instrumentation
- SR 3.3.1.10, Perform CHANNEL CALIBRATION
- SR 3.3.1.11, Perform CHANNEL CALIBRATION
- SR 3.3.1.12, Perform CHANNEL CALIBRATION

TS 3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation
- SR 3.3.2.6, Perform CHANNEL CALIBRATION

TS 3.3.3, Event Monitoring (EM) Instrumentation
- SR 3.3.3.2, Perform CHANNEL CALIBRATION

TS 3.3.6, Control Room Special Ventilation System (CRSVS) Actuation Instrumentation
- SR 3.3.6.4, Perform CHANNEL CALIBRATION

TS 3.4.12, Low Temperature Overpressure Protection (LTOP) – Reactor Coolant System Cold Leg Temperature (RCSPT) > Safety Injection (SI) Pump Disable Temperature
- SR 3.4.12.5, Perform CHANNEL CALIBRATION for each Over Pressure Protection System (OPPS) actuation channel

TS 3.4.13, LTOP – Reactor Coolant System Cold Leg Temperature (RCSPT) ≤ Safety Injection (SI) Pump Disable Temperature

- SR 3.4.13.6, Perform CHANNEL CALIBRATION for each OPPS actuation Channel

TS 3.4.16, RCS Leakage Detection Instrumentation

- SR 3.4.16.3, Perform CHANNEL CALIBRATION of the required containment sump monitor
- SR 3.4.16.4, Perform CHANNEL CALIBRATION of the required containment radionuclide monitor

TS 3.6.8, Vacuum Breaker System

- SR 3.6.8.2, Perform CHANNEL CALIBRATION

TS 3.9.3, Nuclear Instrumentation

- SR 3.9.3.2, Perform CHANNEL CALIBRATION of required channels

3.3.1 Licensee Evaluation of Calibration Changes

GL 91-04 identifies seven steps for the evaluation of instrumentation calibration changes. LAR section 4.4 provided a general overview of the licensee's evaluation of these seven steps.

STEP 1: Confirm that instrument drift as determined by as-found and as-left calibration data from surveillance and maintenance records has not, except on rare occasions, exceeded acceptable limits for a calibration interval.

Licensee Evaluation of STEP 1

The LAR states that a statistical analysis was performed for all instruments which perform a SR function on the as-found and as-left data from surveillance procedures to determine a statistical drift value that is representative of data collected since 1999 or since an instrument was replaced. The statistically determined drift was extrapolated for interval of 30 months. This extrapolation resulted in a single allowable value that needed to be revised to support a 30-month surveillance interval.

STEP 2: Confirm that the values of drift for each instrument type (make, model, and range) and application have been determined with a high probability and a high degree of confidence. Provide a summary of the methodology and assumptions used to determine the rate of instrument drift with time based upon historical plant calibration data.

Licensee Evaluation of STEP 2

The effect of longer calibration intervals on TS instrumentation was evaluated by performing an instrument drift study from records going back to at least 2010 and in some cases back to 1999. The methodology used to perform the drift analysis is consistent with other utilities requesting a 24-month operating cycle and is based on Electric Power Research Institute (EPRI) technical report (TR)-103335, Revision 2, "Guidelines for Instrument Calibration Extension/Reduction Programs."

STEP 3: Confirm that the magnitude of instrument drift has been determined with a high probability and a high degree of confidence for a bounding calibration interval of 30 months for each instrument type (make, model number, and range) and application that performs a safety function. Provide a list of the channels by TS section that identifies these instrument applications.

Licensee Evaluation of STEP 3

LAR Enclosure 3 provides the methodology for determining drift probability. The LAR stated that the analysis demonstrates with a high degree of confidence for a bounding calibration interval of 30 months for each instrument make, model, and range. The basis for extension of instruments not in service long enough to establish a projected drift value is obtained from the methods presented in Enclosure 3. LAR table 1 in Enclosure 2 of the LAR provides a list of affected channels by TS section, including instrument make, model, and range.

STEP 4: Confirm that a comparison of the projected instrument drift errors has been made with the values of drift used in the setpoint analysis. If this results in revised setpoints to accommodate larger drift errors, provide proposed TS changes to update trip setpoints. If the drift errors result in revised safety analysis to support existing setpoints, provide a summary of the updated analysis conclusions to confirm that safety limits and safety analysis assumptions are not exceeded.

Licensee Evaluation of STEP 4

The LAR stated that the projected 30-month drift values were compared to the design allowances as calculated in the associated instrument setpoint analysis and one allowable value will need to be revised to support an extension to 30-months.

STEP 5: Confirm that the projected instrument errors caused by drift are acceptable for control of plant parameters to effect a safe shutdown with the associated instrumentation.

Licensee Evaluation of STEP 5

The analysis in LAR Enclosure 2 discusses the impact of drift on instrument setpoint and uncertainty calculations including instrumentation used for safe shutdown. The LAR stated that the revised setpoint and uncertainty calculations change calibration information if needed which provides assurance the instrumentation will perform with the required accuracy to effect a safe shutdown.

STEP 6: Confirm that all conditions and assumptions of the setpoint and safety analyses have been checked and are appropriately reflected in the acceptance criteria of plant SR procedures for channel checks, channel functional tests, and channel calibrations.

Licensee Evaluation of STEP 6

The LAR stated that the revised setpoint and uncertainty calculations will result in changes to calibration information which are implemented through plant procedures.

STEP 7: Provide a summary description of the program for monitoring and assessing the effects of increased calibration surveillance intervals on instrument drift and its effect on safety.

Licensee Evaluation of STEP 7

The LAR stated that the TS calibration SR intervals will be monitored and trended in accordance with station procedures including recording as-found and as-left calibration data and out of tolerance conditions are entered into the corrective actions program and evaluated and trended. The LAR states this approach will identify occurrences of instruments found outside their allowable value and instruments whose performance is not as assumed in the drift or setpoint analysis.

3.3.2 NRC Staff Evaluation of Calibration SR Interval Changes

The LAR stated that the evaluation of calibration SR interval changes is based on the guidance of EPRI TR-103335, Revision 2. Although not formally endorsed by the NRC, the NRC staff has previously accepted the use of drift evaluation methods based on EPRI TR-103335, Revision 2, as it has been used because it addresses regulatory issues that have surfaced since issuance of EPRI TR-103335, Revision 1. The loop and/or component drift data is characterized using probability analysis to understand the expected behavior of the loop and/or the components. It is confirmed that the data is normally distributed. The data is also checked for drift bias. Normally the mean of the data is zero or a very small number. Any significant value of mean is addressed as drift bias.

Based on the NRC staff's review of the method applied to the data within the Summary Technical Specification Trip Setpoint Calculation in Enclosure 2, "Summary Technical Specification Trip Setpoint Calculation" of the Response to Request for Additional Information (RAI) Letter dated December 7, 2021, the NRC staff found that the square root sum of the squares (SRSS) methodology used to calculate the proposed AVs was consistent with ANSI/ISA S67.04.01-2018. In particular, the method applied in the licensee's Summary Technical Specification Trip Setpoint Calculation was in agreement with Section 4.5.1 "Combination of uncertainties (SRSS method)," of Part I of ANSI/ISA S67.04.01-2018 and RG 1.105, Revision 4. It therefore provides reasonable assurance that the impacted setpoints were evaluated adequately. Equations used in the calculations within the summary report were therefore consistent with the guidance in RG 1.105.

The NRC staff concludes the method used by the licensee adequately addresses the associated drift term for the functions described in the license amendment.

Additionally, except for the function described in Section 3.4.1 below, the NRC staff's review of the calculations presented in the Summary Technical Specification Trip Setpoint Calculation, demonstrate that no alteration to any of the other affected setpoints' allowable values was required.

Therefore, the NRC staff finds the explanation provided in the Summary Technical Specification Trip Setpoint Calculation document and Enclosure 2 of the LAR describing the seven steps acceptable per the guidance of GL 91-04.

3.4 TS Changes

3.4.1 TS Allowable Values

The LAR stated that a comparison of the projected drift errors over the extended calibration interval identified one TS allowable value that required revision. The LAR proposed to revise TS table 3.3.2-1, Function 1e, Safety Injection – Steam Line Low Pressure from ≥ 500 psig to ≥ 505 psig.

3.4.1.1 NRC Staff Evaluation of TS Allowable Value Change

The licensee performed an analysis that demonstrated that the current TS setpoint, ≥ 500 psig, did not meet the acceptance criteria with new surveillance frequencies that may be extended up to 30 months. Therefore, the license amendment proposed to revise the TS setpoint from ≥ 500 psig to ≥ 505 psig. The NRC staff reviewed the SRSS method applied by the licensee in its analysis related to the Safety Injection - Steam Line Low Pressure function, which is described in Enclosure 2 of the LAR and the information contained in Attachment AA, 3.3.2-1, Function 1e: Safety Injection - Steam Line Low Pressure, within Enclosure 2 "Summary Technical Specification Trip Setpoint Calculation" of the letter dated December 7, 2021. The NRC staff determined that the analysis provided by the licensee supports the revised setpoint and is therefore acceptable.

3.4.2 TS 5.5 Programs and Manuals

The LAR proposed to revise TS 5.5.2 consistent with TSTF-299. The proposed change to TS 5.5.2b would modify the frequency of the integrated leak test requirements from "at refueling cycle intervals or less" to "at least once per 24 months" and to add that the provisions of SR 3.0.2 are applicable.

The LAR proposed to revise TS 5.5.17 to indicate that the SRs that are in the proposed amendment and that are in the Prairie Island STI are proposed to be revised by GL 91-04 as an exception to the use of the SFCP.

3.4.2.1 NRC Staff Evaluation of TS 5.5 Changes

The revised TS 5.5.2.b will require integrated system leak testing at least once per 24 months. This fixed testing frequency is more precise than the current frequency of "at refueling cycle interval or less." The provisions of SR 3.0.2 are appropriate because a refueling cycle interval may be longer than 24 months in order to achieve the required fuel burn-up. Providing a 25 percent frequency extension, as afforded by SR 3.0.2, to the integrated system leak test requirements allows flexibility in scheduling equivalent to that in the current requirement of "at refueling cycle interval or less" and avoids an unnecessary shutdown.

The NRC staff concludes that the licensee's proposed changes to TS 5.5.2b are acceptable because they clarify the frequency of testing and provide the necessary scheduling flexibility to avoid an unnecessary shutdown.

3.5 <u>Technical Evaluation Conclusion</u>

The LAR, as supplemented, provided information supporting non-calibration and calibration related SR changes. The information provided in the LAR, as supplemented, was reviewed and found acceptable by NRC staff. The justification for previously documented failures would have no effect on safety with the proposed SR interval was found to be consistent with the guidance of GL 91-04. The information provided related to past failures was reviewed by the staff and the failure data were found to be acceptable by the NRC staff per the guidance of GL 91-04. The NRC staff finds the extension of surveillance intervals from 24 months to up to 30 months is acceptable and the changes proposed in the LAR continue to meet 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments on June 13, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and/or change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration as published in the Federal Register on October 5, 2021 (86 FR 55013), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: William Roggenbrodt, NRR
Hari Kodali, NRR
Nan Chien, NRR
Caroline Tilton, NRR

Date of Issuance: July 28, 2022

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS 239 AND 227 RE: 24-MONTH OPERATING CYCLE (EPID L-2021-LLA-0146) DATED JULY 28, 2022

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