



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

September 11, 2013

Mr. James E. Lynch
Site Vice President
Northern States Power Company - Minnesota
Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089-9642

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2 - ISSUANCE OF
AMENDMENT RE: EXCEPTION TO TECHNICAL SPECIFICATION 5.5.14
TESTING REQUIREMENTS ASSOCIATED WITH STEAM GENERATOR
REPLACEMENT (TAC NO. ME9141)

Dear Mr. Lynch:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Renewed Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant (PINGP), Unit 2. The amendment consists of changes to the technical specifications (TSs) in response to your application dated July 25, 2012, as supplemented by letter dated July 25, 2013.

The amendment revises TS 5.5.14, "Containment Leakage Rate Testing Program," to except the licensee from the requirement to perform an Appendix J, Type A, containment integrated leakage rate test, following modifications to the containment pressure boundary resulting from the replacement of the PINGP, Unit 2, steam generators, scheduled for fall 2013.

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

Sincerely,

A handwritten signature in black ink that reads "Thomas J. Wengert".

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-306

Enclosures:

1. Amendment No. 197 to DPR-60
2. Safety Evaluation

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 197
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 July 25, 2012, as supplemented by letter dated July 25, 2013, 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:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 11, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 197

RENEWED FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Replace the following page of the Renewed Facility Operating License No. DPR-60 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE

Page 3

INSERT

Page 3

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

REMOVE

5.0-28

INSERT

5.0-28

- (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. 197, 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

5.5 Programs and Manuals (continued)

5.5.14 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
1. Unit 1 and Unit 2 (steam generator (SG) replacement commencing Fall 2013) are excepted from post-modification integrated leakage rate testing requirements associated with SG replacement.
 2. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J", Section 9.2.3, to allow the following:
 - (i). The first Unit 1 Type A test performed after December 1, 1997 shall be performed by December 1, 2012.
 - (ii). The first Unit 2 Type A test performed after March 7, 1997 shall be performed by March 7, 2012.
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident is less than the containment internal design pressure, P_a , of 46 psig.
- c. The maximum allowable primary containment leakage rate, L_a , at P_a , shall be 0.15% of primary containment air weight per day. For pipes connected to systems that are in the auxiliary building special ventilation zone, the total leakage shall be less than 0.06% of primary containment air weight per day at pressure P_a . For pipes connected to systems that are exterior to both the shield building and the auxiliary building special ventilation zone, the total leakage past isolation valves shall be less than 0.006% of primary containment air weight per day at pressure P_a .



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-60

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

DOCKET NO. 50-306

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated July 25, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12207A523), as supplemented by letter dated July 25, 2013 (ADAMS Accession No. ML13206A313), Northern States Power Company, a Minnesota Corporation (NSPM, the licensee), doing business as Xcel Energy, requested a change to the technical specifications (TSs) for the Prairie Island Nuclear Generating Plant (PINGP), Unit 2. The proposed change concerns primary containment leakage testing, and would revise TS 5.5.14 to permit the licensee to operate without performance of a Type A Test following modifications to the containment pressure boundary resulting from the replacement of the steam generators (SGs), scheduled for fall 2013.

The licensee will move new SG assemblies, and the old SG assemblies, through the existing primary containment equipment hatch. Thus, the physical integrity of the reactor containment vessel and shield building are unaffected.

Per TS 5.5.14, the Containment Leakage Rate Testing Program shall be in accordance with the guidelines contained in NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by two enumerated exceptions. The proposed change would revise the first exception to eliminate the requirement to perform an overall containment integrated leak rate test (IRLT) following installation of the Unit 2 replacement SG.

The updated TS Page 5.0-28 also includes the changes associated with the January 22, 2013, issuance of the Alternative Source Term (AST) License Amendment Nos. 206 and 193, for PINGP Units 1 and Unit 2, respectively (ADAMS Accession No. ML112521289). These changes were not incorporated on the TS page submitted with the licensee's July 25, 2012, application due to the timing of the issuance of the AST amendments. The supplement dated July 25, 2013, provided additional information that did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant

hazards consideration determination as published in the *Federal Register* on September 14, 2012 (77 FR 56880).

2.0 REGULATORY EVALUATION

The licensee will replace the existing Unit 2 SGs during the fall 2013 refueling outage. The SG replacement affects only the closed piping inside containment and, as previously mentioned, does not affect the physical integrity of the reactor containment vessel and shield building. The inside-containment of this closed system consists of the outer shell of the SGs, the main steam lines, the main and Auxiliary Feedwater (AFW) lines, SG blowdown lines, the recirculation lines, sampling lines, and instrument lines. During a design basis loss-of-coolant accident (LOCA), these elements inside containment form a barrier against the uncontrolled release of radioactivity to the environment and, thus, are considered part of the primary containment system boundary.

The containment system is the principal barrier, after the reactor coolant pressure boundary, to prevent the release of quantities of radioactive material that would have a significant radiological effect on the health and safety of the public. As specified in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(o), primary containments for water cooled power reactors shall be subject to the leakage rate test requirements of Appendix J to Part 50.

To meet 10 CFR 50.54(o), the licensee must meet certain performance-based leakage test requirements. Among other things, the licensee must perform Type A tests to measure the containment system overall ILRT. The Type A test must be conducted under conditions representing design basis loss-of-coolant accident (LOCA) peak pressure within containment. A Type A test must be conducted (1) after the containment system has been completed and is ready for operation, and (2) at periodic intervals based on the historical performance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents. The leakage rate must not exceed the allowable leakage rate (L_a) with margin, as specified in the TSs. The test results must be compared with previous results to examine performance history of the overall containment system to limit leakage.

TS 5.5.14, "Containment Leakage Rate Testing Program," requires that a program be established to implement the leakage testing of the containment as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. Option B of Appendix J identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing. Option B performance-based test intervals are based on consideration of operating history of the component and resulting risk from its failure. The intent of Option B requirements is to reduce the testing burden commensurate with the safety significance while providing a reduction, in part, in occupational radiation exposure and cost. It continues to specify the maximum allowable primary containment leakage rate.

RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, states that Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50 Appendix J," provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J, subject to certain modifications.

In Section 9.2.4 of NEI 94-01, Revision 0 (ADAMS Accession No. ML11327A025), it states that repairs and modifications that affect the containment leakage integrity require Type A or local leakage rate testing prior to returning the primary containment system to operation.

For pre-service and in-service inspection requirements, the affected area of the primary containment system boundary is subject to the American Society of Mechanical Engineers (ASME) Class 2 per Section XI. The pressure boundary of the replacement SGs is constructed in accordance with ASME Section III Class 1. As such, the replacement of the SGs is subject to the requirements of ASME Sections III and XI. The ASME Section III/XI system pressure testing acceptance criteria for the base metal and welds is "no leakage." Furthermore, the acceptance criterion for the specified field hydrostatic tests is "no leakage" (no exceptions for manways and handholes).

The ASME Section III/XI pressure test does not require the leakage rate to be quantified. The acceptance criterion for the proposed test is no visual through-wall leakage; therefore, there is no need to quantify the leakage rate. This acceptance criterion is more conservative than the Type A test which allows some leakage.

ASME Section III/XI requires non-destructive examination (NDE) and visual examination of welds and system leakage testing. If any through-wall leakage is detected from the welds, the leakage is required to be repaired before plant service continues. The licensee proposes to use the ASME Section III/XI pressure test requirements to satisfy the intent of the Appendix J requirements, rather than performing a Type A test.

The Commission's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls. This amendment request concerns 10 CFR 50.36(c)(2), 10 CFR 50.36(c)(3), and 10 CFR 50.36(c)(5).

The SG replacement project is considered by the licensee to be a modification that affects containment leakage integrity. The NRC staff has approved similar requests for several licensees, including an amendment to PINGP Unit 1 that was issued on August 20, 2004 (ADAMS Accession No. ML042090525).

3.0 TECHNICAL EVALUATION

The Unit 2 SG replacement will consist of the following activities, as described in the licensee's application.

- Cutting and removing the main steam lines, main and AFW lines, SG blowdown lines, and associated instrument and sampling lines
- Cutting and removing the upper part of the SGs
- Cutting the reactor coolant piping and removing the SG lower assemblies
- Installing the new SG lower subassemblies and re-welding the reactor coolant piping
- Installing the new SG upper subassemblies on the new lower assemblies

- Re-installing and re-welding the main steam lines, main and AFW lines, SG blowdown lines, recirculation lines and associated instrument lines

The Unit 2 reactor containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom which houses the reactor pressure vessel, the steam generators, reactor coolant pumps, the reactor coolant loops, the accumulators of the safety injection system, the primary coolant pressurizer, the pressurizer relief tank, and other branch connections of the reactor coolant system. The reactor containment vessel is, in turn, housed completely within the shield building. The SG replacement affects only the closed piping inside the reactor containment vessel. In other words, the integrity of the reactor containment vessel structure and shield building are unaffected by the SG replacement project. The new SG assemblies and the old SG assemblies will transit through the containment equipment hatch. However, the outer shell of the SGs, the inside-containment portions of the main steam lines, the main and AFW lines, SG blowdown lines, the recirculation lines, sampling lines, and instrument lines are part of the primary reactor containment, as discussed in the PINGP updated safety analysis report, Section 5.2.2.1.1 and Table 5.2-1.

For the PINGP Unit 2 SG replacement project, the performance of local leakage rate testing (Type B or Type C) is not practical. The intent of performing a Type A test is to assure the leak-tight integrity of the area affected by the modification (i.e., the closed system inside the reactor containment vessel formed by the outer shell of the steam generators and the main steam, feedwater, steam generator blowdown, and feedwater piping) does not alter the overall leakage rate of the primary containment. An exception is requested to avoid performing an unnecessary ILRT. The ILRT is unnecessary because the ASME Section III/XI pressure test requirements for the replacement steam generators will satisfy the intent of RG 1.163 and NEI 94-01. Additionally, the next scheduled Type A test for Unit 2 is scheduled to occur in 2022. Unless the licensee obtains an exception to the requirement, it would need to perform a Type A test following the SG replacement project.

Since the activities associated with an SG replacement are "repairs or modifications that affect containment leakage integrity," the licensee must, pursuant to SR 3.6.1.1, as implemented in TS 5.5.14, RG 1.163, NEI 94-01 and 10 CFR 50, Appendix J, Option B, perform a Type A leakage rate test before entering Mode 4. Through its amendment request to TS 5.5.14, the licensee is requesting not to be required to perform the Type A test following the Unit 2 steam generator replacement.

The licensee proposes to eliminate the requirement to perform post-modification integrated leakage rate (Type A) testing following the replacement, which is not consistent with the current TS 5.5.14. The licensee has proposed revising the following exception to section "a." of TS 5.5.14:

... as modified by the following exceptions:

1. **Unit 1 and Unit 2 (steam generator (SG) replacement commencing Fall 2013) are** excepted from post-modification integrated leakage rate testing requirements associated with **SG** replacement."

Based on the quality of the repairs/modifications made to the containment, and the fact that the containment shell is not being affected, the NRC staff has determined that performing the Type A test after the SG replacement is not necessary to assure that LCO 3.6.1 will be met, and that containment will be operable in Modes 1 through 4.

In addition, the licensee stated that, because the affected area of the primary containment system boundary is subject to ASME Class 2 per Section XI, and the pressure boundary of the replacement SGs is constructed in accordance with ASME Section III Class 1, replacement of the SGs is subject to the requirements of ASME Section III/XI. The ASME Section III/XI testing provides a method that allows testing of only the modified portions of the containment barrier (SG shell and associated closed piping and components) instead of the more comprehensive Type A testing which would be performed on the entire containment barrier.

ASME Section III/XI also requires NDE and visual examination of welds and system leakage testing. This NDE provides confidence to pressurize the secondary side of the SGs and demonstrate leak-tight integrity with the unit in Mode 3 under no-load conditions. NDE of the welds (ultrasonic or radiographic testing) provides additional assurance that the joints are free from defects that could result in significant leakage.

The NRC staff has reviewed the ASME Section III/XI requirements and determined that the ASME Section III/XI surface examination, volumetric examination, and system pressure testing requirements accomplish the intent of Type A testing requirements of Appendix J (which are currently required by TS 5.5.14). The objective of the Type A test is to ensure the leak-tight integrity of the containment area affected by the modification. The ASME Section III/XI inspection and testing requirements fulfill the intent of the requirements of Appendix J and the provisions of NEI 94-01, Section 9.2.4. In addition, the hydrostatic test pressure for the SG vessel and the in-service leak test pressure for the main steam line weld will be at least 20 times that of a Type A test. Although the pressure is applied in the reverse direction to accident pressure, any potential for accident direction leakage would be revealed by the high pressure used.

Summary

The proposed amendment would eliminate the post-modification containment leakage rate (Type A) testing required for the modifications to the primary containment system boundary specifically associated with replacement of the steam generators. Based on the above, the NRC staff concludes that the exception from performing a post-modification Type A test following the Unit 2 SG replacement, and the proposed change to TS 5.5.14, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on September 14, 2012 (77 FR 56880). Accordingly, the amendment meets 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 amendment.

6.0 CONCLUSION

The NRC staff 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: B. Lee, NRR/DSS/SCVB

Date of issuance: -September 11, 2013

Mr. James E. Lynch
 Site Vice President
 Northern States Power Company - Minnesota
 Prairie Island Nuclear Generating Plant
 1717 Wakonade Drive East
 Welch, MN 55089-9642

September 11, 2013

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2 - ISSUANCE OF AMENDMENT RE: EXCEPTION TO TECHNICAL SPECIFICATION 5.5.14 TESTING REQUIREMENTS ASSOCIATED WITH STEAM GENERATOR REPLACEMENT (TAC NO. ME9141)

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

Sincerely,
/RA/
 Thomas J. Wengert, Senior Project Manager
 Plant Licensing Branch III-1
 Division of Operating Reactor Licensing
 Office of Nuclear Reactor Regulation

Docket No. 50-306

Enclosures:

1. Amendment No. 197 to DPR-60
2. Safety Evaluation

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