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

January 8, 2015

Karen D. Fili
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATION 5.5.11, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (TAC NO. MF3161)

Dear Ms. Fili:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 187 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). The amendment consists of a change to MNGP technical specification (TS) 5.5.11, "Primary Containment Leakage Rate Testing Program," in response to your application dated November 14, 2013.

The amendment removes the requirement for reduced pressure drywell airlock door seal testing, since this capability is not required and does not reflect the current testing practice at MNGP. As such, the acceptance criteria currently specified in TS 5.5.11.d.2.b is removed, and the drywell airlock door seals will be tested solely by performance of an overall airlock leakage test as currently specified in TS 5.5.11.d.2.a.

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,

Terry A. Beltz

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 187 to DPR-22
2. Safety Evaluation

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

NORTHERN STATES POWER COMPANY - MINNESOTA

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 187
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company - Minnesota (NSPM, the licensee), dated November 14, 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-22 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 187, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

Enclosure 1

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-22 and
Technical Specifications

Date of Issuance: January 8, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 187

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License No. DPR-22 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of 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
5.5-11	5.5-11

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

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 calibration or associated with radioactive apparatus or components; and

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and 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 2004 megawatts (thermal).

2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 187, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

   NSPM shall 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

5.5 Programs and Manuals

5.5.11 Primary Containment Leakage Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criterion is an overall air lock leakage rate of $\leq 0.05\ L_a$ when tested at $\geq P_a$.

e. The resilient seals of each 18 inch primary containment purge and vent valve shall be replaced at least once every 9 years. The provisions of SR 3.0.2 are applicable to this requirement. If a common mode failure attributable to the resilient seals is identified based on the results of SR 3.6.1.3.11, the resilient seals of all 18 inch primary containment purge and vent valves shall be replaced.

f. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.5.12 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer of the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 187 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY - MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated November 14, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13322A446), Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, Inc., requested changes to the technical specifications (TSs) for the Monticello Nuclear Generating Plant (MNGP). Specifically, the licensee requested to revise TS 5.5.11, "Primary Containment Leakage Rate Testing Program," by removing the requirement for reduced pressure drywell airlock door seal testing, as the MNGP drywell airlock door seals are not configured to support being individually tested.

2.0 REGULATORY EVALUATION

MNGP was designed largely before publication of the 70 General Design Criteria (GDC) for nuclear power plant construction permits proposed by the Atomic Energy Commission (AEC) for public comment in July 1967, and constructed prior to 1971 publication of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50. As such, MNGP was not licensed to the GDCs of Appendix A. The MNGP Updated Safety Analysis Report (USAR), Section 1.2, lists the principal design criteria for the design, construction, and operation of MNGP. Appendix E of the USAR provides a plant comparative evaluation to the 70 proposed AEC design criteria. The licensee conforms to the intent of the 70 proposed GDCs.

The U.S. Nuclear Regulatory Commission (NRC) staff identified specific 10 CFR Part 50, Appendix A, GDCs that pertain to this license amendment request, which are discussed below:

- GDC-16, "Containment design," states that the reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Enclosure 2

- GDC-50, "Containment design basis," states that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

- GDC-52, "Capability for containment leakage rate testing," states that the reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

- GDC-53, "Provisions for containment testing and inspection," states that the reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

The NRC staff identified additional regulations that pertain to this license amendment request, which are discussed below:

10 CFR 50.54, "Conditions of licenses", Paragraph (o), states that primary reactor containments for water cooled power reactors, other than facilities for which the certifications required under paragraphs 50.82(a)(1) or 52.110(a)(1) of this chapter have been submitted, shall be subject to the requirements set forth in Appendix J to this part.

10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," states, in part, that primary reactor containments shall meet the containment leakage test requirements set for in this appendix, and the periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for these tests.

10 CFR 50.36 (c)(3), "Surveillance requirements," states that 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.

10 CFR 50.36 (c)(5), "Administrative controls," states that 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. Each licensee shall submit any reports to the Commission pursuant to approved technical

specifications as specified in § 50.4.

3.0 TECHNICAL EVALUATION

The MNGP is a General Electric (GE) boiling-water reactor (BWR) with a Mark I type pressure suppression containment. As described in the MNGP USAR, the primary containment includes a steel pressure boundary drywell with an upper cylindrical and lower spherical section connected by large vent pipes to a suppression chamber (or wetwell). The drywell houses the reactor pressure vessel (RPV) and its associated primary system. The primary function of the drywell is to contain the mass and radiation released from a design-basis accident (DBA) loss-of-coolant accident (LOCA), and to direct the steam released from the primary system into the wetwell and limit the total increase in containment pressure by condensation of the steam.

3.1 Background

The licensee proposes a change to the acceptance criteria for the drywell personnel airlock door seal testing as specified in TS 5.5.11.d.2. The licensee proposes to delete current TS 5.5.11.d.2.b, which states the following:

> b) For each door, leakage rate is $\leq 0.007\ L_a$ when pressurized to ≥ 10 psig.

The licensee provides justification for removal of this testing methodology based on the airlock door seals not being "double gasketed," such that the seals on each (i.e., inner and outer) airlock door cannot be tested individually and can only be tested by performing an overall airlock leakage test. The licensee describes the history of this TS requirement as originating with recognition of the single seal on each door. Testing the airlock at the containment DBA LOCA maximum accident pressure, P_a, requires bracing (strong-backs) to hold the inner airlock door closed, as pressurizing the airlock space tends to force open the inner airlock door towards the drywell. Accident pressure contained within the drywell actually tends to push both the inner and outer airlock doors to the closed position, thus ensuring good seal contact.

As an alternative, the airlock door seals were to be tested by pressurizing the airlock to a lower pressure of ≥ 10 psig. The more common configuration for drywell personnel airlock doors is for each door to have a double-seal arrangement that allows for just the seals to be tested by pressurizing between the seals of each door, although commonly at a pressure less than P_a to avoid the need for restraints.

The licensee states that plant practice has been to perform an overall airlock leakage test at P_a. Therefore, the licensee proposed that the testing methodology for the drywell personnel airlock door would be that currently specified in TS 5.5.11.d.2.a, such that TS 5.5.11.d.2 would be revised as follows:

> 2. Air lock testing acceptance criterion is an overall air lock leakage rate of $\leq 0.05\ L_a$ when tested at $\geq P_a$.

3.2 Requirements and Guidance for Primary Containment Leakage Rate Testing Programs

The GDCs establish requirements at a high level and do not specify leak testing methods and frequencies. The regulation at 10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors be subject to the requirements set forth in Appendix J to 10 CFR Part 50. Appendix J allows for licensees to choose from two methodologies, either an Option A - Prescriptive Requirements, or Option B - Performance-Based Requirements. The licensee adopted Option B for MNGP. Option B states that:

> The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under part 52 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications.

The MNGP TS 5.5.11 references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as the guidance document used. RG 1.163 endorses, with limitations and conditions, the industry guidance document NEI 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50 Appendix J," Revision 0, dated July 26, 1995. Regarding drywell air lock testing, NEI 94-01 states that:

> Subsequent periodic tests shall be performed at a frequency of at least once per 30 months. Containment airlock tests should be performed in accordance with ANSI/ANS 56.8-1994. In addition, equalizing valves, door seals, and penetrations with resilient seals (i.e., shaft seals, electrical penetrations, view port seals and other similar penetrations) which are testable, shall be tested at a frequency of once per 30 months. Airlock door seals should be tested prior to a preoperational Type A test. When containment integrity is required, airlock door seals should be tested within 7 days after each containment access.
>
> For periods of multiple containment entries where the airlock doors are routinely used for access more frequently than once every 7 days (e.g., shift or daily inspection tours of the containment), door seals may be tested once per 30 days during this time period.
>
> Door seals are not required to be tested when containment integrity is not required, however they must be tested prior to reestablishing containment integrity. Door seals shall be tested at P_a, or at a pressure stated in the plant Technical Specifications.

NEI 94-01 also describes the airlock as being a complex composite penetration, including equalized valves, door seals, and penetrations with resilient seals (e.g., operating shaft seals, electrical penetrations, view port seals and other similar penetrations) which are testable either individually or by performing an overall airlock leakage test. These components would be tested at a frequency not exceeding 30 months. These components would also be subject to as-found testing prior to any maintenance, repair, modification, or adjustment activity that could affect leak tightness, in addition to as-left testing similar to other Type B and Type C tested penetrations.

3.3 NRC Evaluation of Licensee Request

The guidance in NEI 94-01 was developed for the common configuration of airlock door seals, in that each door had an inner and outer seal with provision for testing the seals by pressurizing between the seals on each door. Since MNGP does not have this seal configuration, the door seals are only testable by pressurizing the entire primary containment (i.e., integrated leakage rate test) or by performing a local leakage rate test by pressurizing the airlock volume with restraints on the inner air lock door to hold the sealing surfaces together as they would be during accident conditions. The MNGP primary containment is a GE Mark I design and is inerted during plant operation at power which, along with personnel radiation dose considerations, results in very infrequent ingress and egress through the airlock between refueling outages when primary containment integrity is required. With limited personnel and equipment traffic through the airlock that might lead to seal gasket and mating surface damage or relative position changes from one closure to the next, the door seals generally remain undisturbed. As such, the potential for developing door seal leakage remains low between plant outages. The MRC staff finds that the 7-day and 30-day test provisions in NEI 94-01 are unnecessary for MNGP. Consistent with the guidance in NEI 94-01, the door seal test verifies seal (gasket and mating surface) condition, as the overall airlock (Type B) test measures the actual contribution of the penetration to the combined Type B and Type C test total, which is determined on an as-found minimum pathway basis each refueling outage for comparison to the performance criterion identified in the MNGP TS 5.5.11.

The NRC finds that performing the drywell air lock door testing at the frequency identified in the RG 1.163 endorsed guidance of NEI 94-01 (excepting the 7-day and 30-day provisions) with the criterion identified in the requested change is acceptable.

3.4 Conclusions

The NRC staff concludes that the proposed change to the MNGP TSs continues to meet the requirements of the applicable regulations and guidance for primary containment leakage testing. Therefore, the staff finds the requested change to the drywell personnel airlock door seal testing specified in MNGP TS 5.5.11.d.2 to be 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 requirements with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 as published in the *Federal Register* on August 5, 2014 (79 FR 45478).

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

Principal Contributor: Jerome Bettle, NRR

Date of issuance: January 8, 2015

January 8, 2015

Karen D. Fili
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATION 5.5.11, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (TAC NO. MF3161)

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Sincerely,
/RA/
Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 187 to DPR-22
2. Safety Evaluation

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