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

June 7, 2017

Mr. Joel P. Gebbie
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: LICENSE AMENDMENT REQUEST REGARDING CONTAINMENT LEAKAGE RATE TESTING PROGRAM (CAC NOS. MF8483 AND MF8484)

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 336 to Renewed Facility Operating License No. DPR-58 and Amendment No. 318 to Renewed Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant (CNP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated October 18, 2016, as supplemented by letter dated February 27, 2017.

The amendments revise TS 5.5.14, "Containment Leakage Rate Testing Program," to clarify the containment leakage rate testing pressure criteria.

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

Sincerely,

Jennivine Rankin, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 336 to DPR-58
2. Amendment No. 318 to DPR-74
3. Safety Evaluation

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SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: LICENSE AMENDMENT REQUEST REGARDING CONTAINMENT LEAKAGE RATE TESTING PROGRAM (CAC NOS. MF8483 AND MF8484) DATED JUNE 7, 2017

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ADAMS Accession No. ML17131A277 ***via memo**

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 336
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 18, 2016, as supplemented by letter dated February 27, 2017, 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-58 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 336, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License No. DPR-58 and Technical Specifications

Date of Issuance: June 7, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 336

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Renewed Facility Operating License No. DPR-58 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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

REMOVE	INSERT
5.5-14	5.5-14

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, 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; and

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

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, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3304 megawatts thermal in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 336, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Less than Four Loop Operation</u>

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) <u>Fire Protection Program</u>

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013, 5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

b. The containment design pressure is 12 psig. For the Containment Leakage Rate Testing Program, P_a is 12.0 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 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 ≤ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

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

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 318
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 18, 2016, as supplemented by letter dated February 27, 2017, 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-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 318, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License No. DPR-74 and Technical Specifications

Date of Issuance: June 7, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 318

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Renewed Facility Operating License No. DPR-74 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

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

REMOVE	INSERT
5.5-14	5.5-14

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, 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; and

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

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, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3468 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to the renewed operating license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed operating license shall be completed. Attachment 1 is an integral part of this renewed operating license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 318, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) Leak Testing of Emergency Core Cooling System Valves

Indiana Michigan Power Company shall prior to completion of the first inservice testing interval leak test each of the two valves in series in the

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

b. The containment design pressure is 12 psig. For the Containment Leakage Rate Testing Program, P_a is 12.0 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

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

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

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

5.5.15 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 including 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. 336 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

AND

AMENDMENT NO. 318 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated October 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16294A257), as supplemented by letter dated February 27, 2017 (ADAMS Accession No. ML17061A025), Indiana Michigan Power Company (I&M, the licensee) requested license amendments for the Donald C. Cook Nuclear Plant (CNP), Unit Nos. 1 and 2. The licensee requested to revise technical specifications (TSs) 5.5.14, "Containment Leakage Rate Testing Program," to clarify the containment leakage rate testing pressure criteria.

The supplemental letter dated February 27, 2017, 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* (FR) on December 6, 2016 (81 FR 87972).

2.0 REGULATORY EVALUATION

2.1 Description of CNP Containment Leakage Testing Requirements

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(o) require that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR Part 50 includes two options, Option A – Prescriptive Requirements, and Option B – Performance-Based Requirements, either of which can be chosen for meeting the requirements of Appendix J. The testing requirements in Appendix J ensure that leakage through the primary reactor containment and related systems and components penetrating primary containment does not exceed allowable leakage rate values specified in the TSs or associated bases and that integrity of the containment structure is maintained during its service life.

The licensee has adopted and has been implementing Option B for meeting the requirements of Appendix J. Option B of Appendix J specifies the performance-based requirements and criteria for preoperational and subsequent leakage-rate testing.

2.2 Proposed TS Changes

The licensee requested to modify paragraph b in TS 5.5.14. TS 5.5.14 b currently states:

> b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig [pounds per square inch gauge].

The licensee requested to revise TS 5.5.14 b to state the following:

> b. The containment design pressure is 12 psig. For the Containment Leakage Rate Testing Program, P_a is 12.0 psig.

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements, guidance, and licensing information during its review of the proposed change.

The construction permits for CNP were issued and the majority of construction was completed prior to issuance of Appendix A, General Design Criteria (GDC), to 10 CFR Part 50, in 1971 by the Atomic Energy Commission. CNP was designed and constructed to comply with the GDC as proposed on July 11, 1967. Section 1.4 of the CNP updated final safety analysis report, "Plant Specific Design Criteria" (PSDC), defines the principal criteria and safety objectives for the design of CNP. The current Appendix A to 10 CFR Part 50 GDC differ both in numbering and content from the PSDC for CNP. The NRC staff considered the following CNP PSDCs in its review of this request.

PSDC-54, "Initial Leak Rate Testing for Containment" states:

> The containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.
>
> The containment was designed so that its maximum integrated leakage under accident conditions meets the site exposure criteria set forth in 10 CFR 100 guidelines.

PSDC-55, "Periodic Containment Leakage Rate Testing" states:

> The containment shall be designed so that an integrated leakage rate can be periodically determined by tests during the plant lifetime.
>
> The containment is designed to permit full-integrated leak rate tests.

Paragraph 50.36(c) of 10 CFR requires TSs to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements (SRs); (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. The regulation does not specify the particular requirements to be included in a plant's TSs.

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4.0 (ADAMS Accession No. ML12100A222), provides the improved standard technical specifications for Westinghouse plants. TS 5.5.16 b, Option B of NUREG-1431, states "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]. The containment design pressure is [50 psig]."

3.0 TECHNICAL EVALUATION

3.1 Background

The CNP containment design pressure, P_d, is 12 psig. Historically, the calculated peak containment pressure (CPCP) values have been relatively close to the design pressure value of 12 psig. Recently, the licensee performed a loss-of-coolant accident (LOCA) reanalysis using the methodology in WCAP-17721-P-A, "Westinghouse Containment Analysis Methodology – PWR [pressurized-water reactor] LOCA Mass and Energy Release Calculation Methodology," dated September 2015 (ADAMS Accession No. ML15272A050). This methodology was approved by the NRC by letter dated October 7, 2015 (ADAMS Accession No. ML15274A179). This reanalysis resulted in CPCP values of 10.37 psig for CNP, Unit No. 1, and 10.78 psig for CNP, Unit No. 2. The re-calculated LOCA pressure is solely due to a new analysis method and does not correspond to plant modifications that would impact the actual response during a design basis event. This has resulted in confusion regarding the language in TS 5.5.14 as the TS is not specific in describing the 12 psig as the P_a value for the containment leakage rate testing program purposes. Therefore, CNP requested to clarify that P_a is defined as 12.0 psig for the Containment Leakage Rate Testing Program.

3.2 Licensee's Submittal and NRC Staff's Technical Evaluation

TSs 5.5.14 for CNP, Unit Nos. 1 and 2, cite Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012 (ADAMS Accession No. ML12221A202), as the 10 CFR Part 50, Appendix J, Option B, program's implementation document. NEI 94-01, Revision 3-A, invokes as guidance the testing methodology of ANSI/ANS-[American National Standards Institute/American Nuclear Society] 56.8-2002, "Containment System Leakage Testing Requirements." ANSI/ANS-56.8-2002, Section 3.3.2, "Test pressure," states:

> Type B and Type C tests shall be conducted at a differential pressure of greater than or equal to P_a unless otherwise specified in the plant's licensing basis. When a higher differential pressure results in increased sealing, the differential pressure shall not exceed 1.1 P_a.

Using the re-calculated CPCP values and the criteria in ANSI/ANS-56.8-2002, the maximum test pressures for Type B and Type C local leak rate tests (LLRTs) shall not exceed 11.41 psig (i.e., 10.37 x 1.1) for Unit No. 1 and 11.86 psig (i.e., 10.78 x 1.1) for Unit No. 2. However, the licensee proposal to continue to use the design pressure of 12 psig as P_a for the containment leakage rate testing program purposes would exceed these values. The licensee stated in the license amendment request at Section 3.2, "Evaluation," that the purpose of the NEI 94-01, Revision 3-A, guidance:

> … is to prevent testing to be performed at significantly higher pressures than those expected to be observed during a design basis LOCA event. For example, when compared to CNP's 12 psig containment design pressure, testing a check valve at 55 psig would cause the check valve to seat tighter and therefore leak less. This limitation would not be a concern for a large majority of components tested under the Containment Leakage Rate Testing Program, which would have conservative results at higher pressures.
>
> Use of P_a at 12 psig for the Containment Leakage Rate Testing Program will not result in a significantly larger differential pressure to seal components whose characteristics result in improved sealing based on increased pressure. This allowance also results in having consistent LLRT pressures for each unit.

The NRC staff notes that all the previous CPCP values as reported in the LAR, prior to the current one that is derived via WCAP-17721-P-A, are such that using the design pressure of 12 psig as the test pressure for LLRTs would not result in exceeding the 1.1 P_a criteria. The NRC staff acknowledges that using test pressures slightly in excess (i.e., ≤ 6 percent) of the calculated peak containment internal pressure for the design basis LOCA for specific Type B and Type C components could result in non-conservative LLRT leakage rates for a limited number of components. NEI 94-01, Revision 3-A, does not directly address this specific issue. Section 8.0, "Testing Methodologies for Type A, B and C Tests," of NEI 94-01, Revision 3-A, reads in part:

> Type A, Type B and Type C tests should be performed using the technical methods and techniques specified in ANSI/ANS-56.8–2002, …
>
> It should be noted that the Type B or C tests performed on associated pathways must test all of its containment barriers. This includes bonnets, packings, flanged joints, threaded connections, and compression fittings. If the Type B or C test pressurizes any of the pathway's containment barriers in the reverse direction, **it must be shown that test results are not affected in a non-conservative manner** *(emphasis added)* by directionality.

By email dated January 26, 2017 (ADAMS Accession No. ML17027A019), the NRC staff requested that the licensee demonstrate how testing at a pressure of greater than 1.1 P_a is acceptable as an exception to ANSI/ANS-56.8–2002. More specifically, the staff requested that the licensee demonstrate that testing at a pressure in excess of the allowance in ANSI/ANS-56.8–2002 will not produce non-conservative overall aggregate Type B and Type C test results.

The licensee responded to the NRC staff's request for additional information (RAI) by letter dated February 27, 2017. The RAI response provided information on how the use of P_a as 12 psig could produce non-conservative results. The licensee stated the following:

> Acceptable containment leakage is maintained by finding and repairing valve seat and disc seat scratches or other very small leaks in Local Leak Rate Testing (LLRT) components. Given any containment boundary leakage pathway (hole, crack, scratch), using a P_a of 12 psig versus a lower LLRT test pressure results in increased leakage. The only way for LLRT results to be non-conservative using P_a of 12 psig (12 psig vs 10.37 or 10.78 psig) is if the slight increase in P_a results in reducing the area of the leak pathway (further closing of valve disc) on the

LLRT component.

The licensee also documented in the RAI response that it had reviewed station LLRT procedures to identify components where potential non-conservative results might be obtained. The CNP, Unit No. 1 and Unit No. 2, LLRT program contains, in total, approximately 1,130 Type B and Type C components. In all, between both units, the licensee identified 83 components with potentially non-conservative test results by testing at the slightly elevated P_a value of 12 psig in lieu of using the analytically derived CPCP values. A synopsis of the analyses performed for these components is as follows:

a. For all 42 check valve Containment Isolation Valves (CIVs), the licensee's analysis concluded that, "[s]ince the total combined containment leakage allowed is approximated by a 0.1 inch diameter hole, it is known that a disc not fully closed will grossly fail LLRT by leaking greater than the LLRT equipment can measure (~55,000 sccm [standard cubic centimeters per minute]). The test boundary will not pressurize without a closed disc and the desired test pressure is irrelevant."

b. For all 4 single wedge motor-operated valve (MOV) gate valves and all 6 double disc wedge MOV gate valves, the licensee's analysis concluded that the additional LLRT test forces created were either inconsequential or insignificant when compared to the forces generated by the motor operated valve actuators.

c. For all 4 globe MOVs where LLRT pressure is applied above the seat, the licensee's analysis concluded that "the larger test pressure will increase leakage results but will not shut the disc further than the actuator already closed the disc."

d. For all 13 identified air-operated valves (AOVs) tested with LLRT pressure applied above the seat, the licensee's analysis concluded that "the larger test pressure will increase the leakage results but will not shut the disc further than the actuator already closed the disc."

e. For all 5 manual globe valves where LLRT pressure is applied above the seat and the 1 manual single wedge gate valve, the licensee's analysis concluded that after seat to disc or wedge contact is reached by manual force, the additional LLRT force being applied with the slightly higher test pressure should neither drive the disc/wedge further into its seating surface nor produce non-conservative LLRT results.

f. For the 8 airlock door seals, the licensee's analysis concluded that performing these leakage rate tests at the slightly higher LLRT pressure may yield "potentially non-conservative leakage during the barrel test. However, the airlock door seals are all tested individually also during the barrel LLRT procedure and on a frequent basis (weekly) to verify their condition."

For the 75 LLRT components identified in "a." through "e." above, the NRC staff finds the licensee's analyses reasonable and concludes, based on the licensee's stated findings, that the cumulative effects of using an LLRT test pressure of 12.0 psig will, at worst, insignificantly affect the LLRT results.

For the 8 airlock door seals identified in "f." above, the licensee concludes that non-conservative LLRT values may result; however, the airlock doors are tested and monitored frequently. The NRC staff notes that the CNP, Unit No. 1 and Unit No. 2, TS 3.6.2, "Containment Air Locks"

SR 3.6.2.1 has an APPLICABILITY of Modes 1, 2, 3 and 4 and requires the performance of air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program and at a FREQUENCY that is in accordance with the Containment Leakage Rate Testing Program. The Containment Leakage Rate Testing Program identified in TS 5.5.14 lists NEI 94-01, Revision 3-A, as a 10 CFR Part 50, Appendix J, Option B, implementation document. NEI 94-01, Revision 3-A, Section 10.2.2.1, "Test Interval" reads in part:

> 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.

Therefore, the NRC staff concludes that any non-conservative leakage rate effect created by the testing of the eight containment air lock door seals at 12.0 psig, in lieu of the CPCP values, will be rendered inconsequential by SR 3.6.2.1, which verifies each air lock door seal's integrity during Modes 1, 2, 3, and 4 at a frequency based on each containment air lock's usage.

Based on the foregoing, the NRC staff concludes that a significant majority of the remaining approximately 1122 Type B and Type C components contained in the CNP Unit No. 1 and Unit No. 2 LLRT program may likely experience increased LLRT leakage rates, however small they might be, due to the use of 12.0 psig in lieu of the CPCP test values. Therefore, the NRC staff finds that, in the aggregate, the potential outcome of testing at 12.0 psig is conservative.

3.3 Conclusion

The number of components with potentially non-conservative results (e.g., 83) as compared to the total number of components in the CNP Unit No. 1 and Unit No. 2 LLRT program (e.g., 1130) is small. As such the NRC staff concludes that by testing at pressures slightly in excess of the allowance contained in Section 3.3.2 of ANSI 56.8-2002, non-conservatisms introduced into the aggregate Type B and Type C test results will be negated.

Therefore, the NRC staff concludes that defining the CNP Containment Leakage Rate Testing Program P_a value at 12.0 psig, which equals the containment design pressure P_d, is acceptable. This provides consistency with the testing history of the CNP program and clarification of the containment leakage rate testing pressure criteria.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments on May 11, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 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 and there has been no public comment on such finding as published in the *Federal Register* on December 6, 2016 (81 FR 87972). 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.

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