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October 18, 2016

AEP-NRC-2016-64
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1 and Unit 2
Docket Nos.: 50-315 and 50-316
License Amendment Request Regarding Containment Leakage Rate Testing Program

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Unit 1 and Unit 2, proposes to amend the Appendix A Technical Specifications (TS) to Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to change TS 5.5.14, Containment Leakage Rate Testing Program, to clarify the containment leak rate testing pressure criteria. I&M has evaluated the proposed changes in accordance with 10 CFR 50.92 and concluded that they involve no significant hazards consideration.

Enclosure 1 to this letter provides an affirmation statement pertaining to the information contained herein. Enclosure 2 provides I&M's evaluation of the proposed TS change. Enclosures 3 and 4 provide Unit 1 and Unit 2 TS pages, respectively, marked to show the proposed changes. New clean Unit 1 and Unit 2 TS pages with proposed changes incorporated will be provided to the Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested.

I&M requests approval of the proposed change in accordance with the NRC's normal review and approval schedule. The proposed change will be implemented within 90 days of NRC approval.

Copies of this letter and its enclosures are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Quinton S. Lies
Site Vice President

DB/kmh

A001
NRR

Enclosures:

1. Affirmation
2. Proposed License Amendment Request Regarding Containment Leakage Rate Testing Program.
3. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked To Show Proposed Changes
4. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked To Show Proposed Changes

c: R. J. Ancona, MPSC
A. W. Dietrich, NRC Washington DC
MDEQ- RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson - AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2016-64

AFFIRMATION

I, Quinton S. Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Quinton S. Lies

Quinton S. Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 18 DAY OF October 2016

Danielle Burgoyne
Notary Public

My Commission Expires 04-04-2018

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2018
Acting in the County of Berrien

Enclosure 2 to AEP-NRC-2016-64

Proposed License Amendment Request Regarding Containment Leakage Rate Testing Program

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, proposes to amend the Appendix A Technical Specifications (TS) to Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to change TS 5.5.14, Containment Leakage Rate Testing Program, to clarify the containment leak rate testing pressure criteria.

2.0 DETAILED DESCRIPTION

TS 5.5.14 b. currently states, "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig." I&M proposes to change this statement to read, "The containment design pressure is 12 psig. For the containment Leakage Rate Testing Program, P_a is defined as 12.0 psig."

Enclosures 3 and 4 to this letter provide the Unit 1 and Unit 2 TS pages, respectively, marked to show proposed changes. New text on these pages is marked with a single-line border. New clean Unit 1 and Unit 2 TS pages with proposed changes incorporated will be provided to the Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested.

3.0 TECHNICAL EVALUATION

3.1 Background

By letter dated April 6, 2004 (Reference 2), I&M requested to convert from Current TS (CTS) to Improved TS (ITS). Amendment 287 for Unit 1 and Amendment 269 for Unit 2 were issued June 1, 2005. The calculated peak containment pressure values have historically been relatively close to the design pressure value. Recent containment integrity re-analyses using new analysis methods have resulted in reduced calculated peak containment pressures. I&M's conversion from CTS to ITS combined with the new containment integrity analyses have resulted in confusion regarding the language in TS 5.5.14. The language is not specific in describing the 12 psig value as the P_a value for containment leak rate testing purposes. The 12 psig value was previously established in the licensing basis as the test value and is not the same as the calculated peak containment pressure.

The containment design pressure is 12 psig for both CNP units. Current calculated peak pressure from a loss-of-coolant accident (LOCA) is 10.37 psig for Unit 1 and 10.78 psig for Unit 2. The Updated Final Safety Analysis Report (UFSAR), Section 5.7.3, states that periodic leak rate testing is performed at the design pressure of 12 psig. The UFSAR testing criteria has remained consistent since the initial issue of the FSAR.

NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" endorses American National Standard ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements" for testing details. NEI 94-01, Revision 3-A, Section 8, "Testing Methodologies for Type A, B and C Tests" states that Type A, Type B and Type C tests should be performed using the technical methods and techniques specified in ANSI/ANS-56.8-2002. Section 3.3.2, Test Pressure, of this ANSI Standard states:

"Type B and 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".

3.2 Evaluation

CNP desires to clarify that P_a is defined as 12.0 psig for the Containment Leakage Rate Testing Program. The definition for the term P_a in ANSI/ANS-56.8-2002 and 10 CFR Part 50 Appendix J is the peak containment internal pressure related to the design basis LOCA. Containment design pressure, P_d, has remained at 12.0 psig for both CNP units. Containment leakage rate testing has also consistently been conducted using a value of P_a equal to 12.0 psig, as defined in the CNP licensing basis. Maximum peak containment internal pressure related to LOCA event P_a, has been revised three times at CNP for the values of 11.85 psig, 11.75 psig, and 11.43 psig for both units (using a single bounding analysis for both units). For the majority of CNP's operation, the calculated LOCA pressure has been relatively close to the licensing basis value of 12 psig. A recent LOCA reanalysis resulted in calculated peak pressures of 10.37 psig for Unit 1 and 10.78 psig for Unit 2. This reanalysis was implemented under a 10 CFR 50.59 review and uses the methodology in WCAP-17721-P-A (Reference 4) which was approved for use by the NRC in Reference 5. Note that the reduced calculated LOCA pressure is due to a new analysis method and does not correspond to plant modifications that would impact the actual response during a design basis event. As a comparative reference point, using the current calculated peak pressure, the difference in test pressure for local leak rate test (LLRT) is approximately 1.63 psig for Unit 1 and 1.22 psig for Unit 2. Using a P_a of calculated peak pressure would result in test pressures slightly lower than the licensing basis value P_a of 12 psig for CNP. Therefore, CNP's licensing basis value of 12 psig for containment leakage rate testing program is acceptable.

As discussed, the NEI 94-01, Revision 3-A, guidance for Appendix J limits the differential pressure used during testing to 1.1 P_a. The purpose of this 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.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Title 10 Code of Federal Regulations (CFR) 50.36, "Technical specifications" states:

(c) Technical specifications will include items in the following categories:

4) *Design features.* Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

The proposed change will modify TS 5.5.14 b., for the Containment Leakage Rate Testing Program. This change does not modify the testing requirements for Leak Rate Testing. Therefore, the requirements of Title 10 CFR 50.36 continue to be met with the changes proposed in this license amendment request for TS 5.5.14 b.

General Design Criteria

The construction permits for CNP were issued and the majority of construction was completed prior to issuance of 10 CFR 50, Appendix A, General Design Criteria, in 1971 by the Atomic Energy Commission (AEC). CNP was designed and constructed to comply with the AEC General Design Criteria (GDC) as proposed on July 10, 1967. The application of the AEC proposed General Design Criteria to the CNP is contained in the CNP UFSAR as the Plant Specific Design Criteria (PSDC). Appendix A of 10 CFR Part 50 GDC differs both in numbering and content from the PSDC for CNP.

The impact of the Surveillance Requirment changes proposed in this submittal on the PSDC applicable to this license amendment request are discussed below:

PSDC 54 Initial Leak Rate Testing for Containment

The CNP UFSAR 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

The CNP UFSAR 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."

With the changes proposed in this license amendment request, the requirements of PSDC 54 and 55 continue to be met and the plant TS will continue to provide the basis for safe plant operation.

4.2 Precedent

By letter dated September 30, 2015, the NRC issued amendments to Sequoyah Nuclear Plant, Units 1 and 2 for the Conversion to the Improved Technical Specifications with beyond Scope Issues. This amendment approved the use of wording for the Containment Leakage Rate Testing Program which is consistent with the proposed wording in this License Amendment Request (LAR).

4.3 No Significant Hazards Consideration Determination

Pursuant to 10 CFR 50.90, I&M, the licensee for CNP Unit 1 and Unit 2, proposes to amend the Appendix A TS to Facility Operating Licenses DPR-58 and DPR-74. I&M proposes a change to TS 5.5.14, Containment Leakage Rate Testing Program, to clarify the calculated peak containment internal pressure related to the design basis accident. I&M has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Response: No

   The proposed changes do not involve changes to the installed structures, systems or components of the facility. The proposed change is consistent with Westinghouse Owners Group Standard Technical Specification language for the Containment Leak Rate Program.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change does not introduce new accident initiators or impact assumptions made in the safety analysis. Testing requirements continue to demonstrate that the Limiting Conditions for Operation are met and the system components are functional.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not exceed or alter a design basis or safety limit, so there is no significant reduction in the margin of safety.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

5.0 ENVIRONMENTAL CONSIDERATION

I&M has evaluated this LAR against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this LAR meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR Part 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.
As demonstrated in Section 4.3, the proposed TS change does not involve a significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.
This LAR will not change the types or amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.
This LAR will not increase the individual or cumulative occupational radiation exposure.

6.0 REFERENCES

1. Donald C Cook Nuclear Plant Updated Final Safety Analysis Report

2. Letter from M. K. Nazar, Indiana Michigan Power Company, to Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1 and Unit 2, License Amendment Request - Conversion of Current Technical Specifications (CTS) to Improved Technical Specifications (ITS)," AEP:NRC:4901, dated April 6, 2004.

3. NUREG 1431, Revision 2, Standard Technical Specification Westinghouse Plants, dated June 2001.

4. WCAP-17721-P-A, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology," Revision 0, dated September 2015.

5. Letter from Mirela Gavrilas, NRC, James Gresham, Westinghouse, "Verification Letter of the Approval Version of Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-17721-P, Revision), and WCAP-17721-NP, Revision 0, 'Westinghouse Containmnet Analysis Methodology – PWR [Pressurized Water Reactor] LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation Methodology,' " dated October 7, 2015

Enclosure 3 to AEP-NRC-2016-64

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW PROPOSED CHANGES

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 defined as 12.0 psig. ~~The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.~~

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% 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.

Enclosure 4 to AEP-NRC-2016-64

DONALD C. COOK NUCLEAR PLANT UNIT 2 TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW PROPOSED CHANGES

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 defined as 12.0 psig. ~~The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.~~

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% 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.