INDIANA MICHIGAN POWER

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June 8, 2020

AEP-NRC-2020-40
10 CFR 50.90

Docket No.: 50-315

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

Donald C. Cook Nuclear Plant, Unit 1
LICENSE AMENDMENT REQUEST FOR ONE-TIME EXTENSION OF THE CONTAINMENT TYPE A LEAK RATE TESTING FREQUENCY

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 1. The proposed license amendment request (LAR) would allow for a one-time extension to the 15-year frequency of the CNP Unit 1 containment leakage rate test (i.e. Integrated Leak Rate Test (ILRT) or Type A test). This test is required by TS 5.5.14, Containment Leakage Rate Testing Program. The proposed one-time change would permit the current ILRT interval of 15 years to be extended by approximately five months to no later than the plant startup after the spring 2022 refueling outage.

The most recently completed Type A test at CNP Unit 1 was on November 1, 2006, therefore the next Type A test is due to start no later than November 30, 2021. In order to perform the test during a regularly scheduled refueling outage, the next Type A test at CNP Unit 1 would need to be completed no later than startup after the fall 2020 refueling outage.

On January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency for the United States to aid the nation's healthcare community in responding to the Novel Coronavirus and its associated disease, COVID-19. On March 10, 2020, Michigan Governor Gretchen Whitmer declared a state of emergency. The COVID-19 outbreak was subsequently characterized as a pandemic by the World Health Organization on March 11, 2020, and on March 13, 2020, President Donald Trump declared the COVID-19 pandemic a national emergency.

While there is no way to effectively predict the course of the COVID-19 pandemic, multiple medical authorities, including the Director of the National Institute of Allergy and Infectious Diseases, Dr. Anthony Fauci, and researchers at Johns Hopkins University, have expressed major concerns over a likely resurgence of COVID-19 this fall. Performing the Type A test would require 25 vendor personnel from across the United States working alongside plant personnel in close proximity for extended periods of time. Including the ILRT in the fall outage scope would also increase the overall

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outage duration by approximately two days, increasing the amount of time that supplemental workforce would remain on-site.

In response to concerns of a Fall resurgence of COVID-19, in the interest of personnel safety, and to preclude the potential for transmittal and spread of COVID-19, I&M requests a one-time extension of the CNP Unit 1 Type A test interval. This request is part of an overall effort by I&M to reduce the number of outside personnel required on-site, and the overall outage scope, in response to the developing COVID-19 pandemic situation while maintaining the safety and reliability of the plant for the next operating cycle. Current efforts in this regard have reduced the number of scheduled man-hours from an estimate of 240,000 to approximately 155,000 man-hours, and reduced the number of outside personnel required from an estimate of 1,250 to approximately 800 personnel. This reduction in scope and required outside personnel will allow CNP Unit 1 outage personnel to more effectively follow guidelines for social distancing established by the Centers for Disease Control and Prevention, and mandated by State Order, during the fall outage.

Enclosure 1 provides an affirmation statement pertaining to the information contained herein. Enclosure 2 provides a description and assessment of the proposed changes. Enclosure 3 provides an assessment of risk associated with the one-time extension. Enclosure 4 provides Unit 1 TS pages, marked to show the proposed changes. New clean TS pages with proposed changes incorporated will be provided to the Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan state officials.

I&M requests approval of the proposed amendment by August 11, 2020, in order to facilitate preparation and planning for the next Unit 1 refueling outage, currently scheduled to occur during fall 2020. Once approved, the amendment shall be implemented within 30 days.

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

Sincerely,

Quinton A. Lies

Q. Shane Lies
Site Vice President
Indiana Michigan Power Company

BMC/mll

Enclosures:

1. Affirmation
2. Description and Assessment of the Technical Specification Changes
3. Evaluation of Risk Significance of Short-Term ILRT Extension
4. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked to Show Proposed Changes

c: R. J. Ancona – MPSC
EGLE – RMD/RPS
J. B. Giessner – NRC Region III
NRC Resident Inspector
S. P. Wall – NRC Washington, D.C.
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2020-40

AFFIRMATION

I, Q. Shane 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

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 8th DAY OF June 2020

Micah L. Lange
Notary Public

My Commission Expires 02-20-2025

MICAH L LANGE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 02-20-2025
Acting in the County of Berrien

Enclosure 2 to AEP-NRC-2020-40

Description and Assessment of Technical Specification Changes

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 1. The proposed license amendment request (LAR) would allow for a one-time extension to the 15-year frequency of the CNP Unit 1 containment leakage rate test (i.e. Integrated Leak Rate Test (ILRT) or Type A test). This test is required by TS 5.5.14 "Containment Leakage Rate Testing Program." The proposed one-time change would permit the current ILRT interval of 15 years to be extended by approximately five months to no later than the plant startup after the spring 2022 refueling outage. For conservatism, an extension of six months was assumed in the analysis.

The most recently completed Type A test at CNP Unit 1 was on November 1, 2006, therefore the next Type A test is due to start no later than November 30, 2021. In order to perform the test during a regularly scheduled refueling outage, the next Type A test at CNP Unit 1 would need to be completed no later than startup after the fall 2020 refueling outage.

On January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency for the United States to aid the nation's healthcare community in responding to the Novel Coronavirus and its associated disease, COVID-19. On March 10, 2020, Michigan Governor Gretchen Whitmer declared a state of emergency. The COVID-19 outbreak was subsequently characterized as a pandemic by the World Health Organization on March 11, 2020, and on March 13, 2020, President Donald Trump declared the COVID-19 pandemic a national emergency.

While there is no way to effectively predict the course of the COVID-19 pandemic, multiple medical authorities, including the Director of the National Institute of Allergy and Infectious Diseases, Dr. Anthony Fauci, and researchers at Johns Hopkins University, have expressed major concerns over a likely resurgence of COVID-19 this fall. Performing the Type A test would require 25 vendor personnel from across the United States working alongside plant personnel in close proximity for extended periods of time. Including the ILRT in the fall outage scope would also increase the overall outage duration by approximately two days, increasing the amount of time that supplemental workforce would remain on site.

In response to concerns of a Fall resurgence of COVID-19, in the interest of personnel safety, and to preclude the potential for transmittal and spread of COVID-19, I&M requests a one-time extension of the CNP Unit 1 Type A test interval. This request is part of an overall effort by I&M to reduce the number of outside personnel required on-site, and the overall outage scope, in response to the developing COVID-19 pandemic situation while maintaining the safety and reliability of the plant for the next operating cycle. Current efforts in this regard have reduced the number of scheduled man-hours from an estimate of 240,000 to approximately 155,000 man-hours, and reduced the number of outside personnel required from an estimate of 1,250 to approximately 800 personnel. This reduction in scope and required outside personnel will allow CNP Unit 1 outage

more effectively follow guidelines for social distancing established by the Centers for Disease Control and Prevention, and mandated by State Order, during the fall outage.

2.0 PROPOSED CHANGE

CNP Unit 1 TS 5.5.14, Containment Leakage Rate Testing Program, currently states the following:

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.

The proposed change to CNP Unit 1 TS 5.5.14, Containment Leakage Rate Testing Program, will add an exception to allow for the performance of the next Type A test no later than the spring 2022 refueling outage for Unit 1, as follows (added text in bold italic):

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***, except that the next Type A test performed after the November 1, 2006 Type A test shall be performed no later than the plant startup after the spring 2022 refueling outage***.

3.0 BACKGROUND

The CNP Unit 1 containment is a steel-lined, reinforced concrete structure. The containment structure, including all its penetrations, includes a low-leakage steel liner designed to contain the radioactive material that may be released from the reactor core following a design basis loss of coolant accident. Additionally, the containment structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

As required by 10 CFR 50.54(o), the CNP Unit 1 containment is subject to the requirements set forth in 10 CFR 50, Appendix J. The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from containment, including systems and components that penetrate containment, does not exceed the allowable leakage values specified in TS 5.5.14. The testing requirements assure that periodic surveillance of containment penetrations and isolation valves is performed so that proper maintenance and repairs are performed on the systems and components penetrating containment. The limitation on containment leakage provides assurance that containment would perform its design function following an accident up to and including the plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for containment

penetrations; and (3) Type C tests, intended to measure containment isolation (CI) valve leakage. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing. This request modifies the existing Appendix J Type A testing interval but does not change the Appendix J Type A, Type B, or Type C test methods.

Chronology of 10 CFR 50 Appendix J Testing Requirements

In 1995, 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was amended to provide a performance-based Option B for containment leakage testing requirements. Performance-based test intervals are based on consideration of the operating history of the component and resulting risk from its failure. The use of the term "performance-based" in 10 CFR 50, Appendix J refers to both the performance history necessary to extend test intervals, as well as to the criteria necessary to meet the requirements of Option B. Also in 1995, Regulatory Guide (RG) 1.163 (Reference 1) was issued. The RG endorsed NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 2), with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allowed licensees with a satisfactory ILRT performance history (i.e., two consecutive successful type A tests) to reduce the frequency of the ILRT from three tests in ten years to one test in ten years. This relaxation was based on an Nuclear Regulatory Commission (NRC) risk program and Electric Power Research Institute (EPRI) Topical Report (TR)-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (Reference 3), which illustrated that the risk increase associated with extending the ILRT surveillance interval was very small.

By letter dated August 31, 2007, NEI submitted NEI 94-01, Revision 2 (Reference 4), to the NRC Staff for review. NEI 94-01, Revision 2, describes an approach for implementing the performance-based requirements of Option B, which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163 (Reference 1). It delineates a performance-based approach for determining containment leakage rate surveillance testing frequencies using industry performance data, plant-specific performance data, and risk insights. The NRC final Safety Evaluation (SE) issued by letter dated June 25, 2008 (Reference 5), documents the evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 was subsequently issued as Revision 2-A, dated October 2008 (Reference 6).

On December 8, 2008, the NRC issued Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50" (Reference 7). The RIS clarifies the NRC position concerning licensee requests to extend Type A test intervals beyond 15 years, stating that a licensee can commence the test no later than the last day of the month in which it becomes due, without seeking NRC approval through a license amendment. The RIS also endorses the statement made in NEI 94-01, Revision 2, that if the test interval ends while primary containment integrity is not required, or is required solely for shutdown activities, the test interval may be extended indefinitely, but a Type A test shall be completed prior to entering the operating mode requiring primary containment integrity.

By letter dated June 9, 2011, NEI submitted NEI 94-01 Revision 3 (Reference 8) to the NRC Staff for review. NEI 94-01, Revision 3, added guidance for extending Type C Local Leak Rate Test (LLRT) surveillance intervals beyond sixty months. The NRC final SE issued by letter dated June 8, 2012 (Reference 9), documents the NRC's evaluation and acceptance of NEI 94-01, Revision 3, subject to the specific limitations and conditions listed in Section 4.0 of the SE. The accepted version of NEI 94-01 has subsequently been issued as Revision 3-A dated July 2012 (Reference 10).

Current CNP Unit 1 Testing Requirements Under 10 CFR 50 Appendix J Option B

By letter dated March 7, 2014, as supplemented by letters dated September 30, 2014, December 16, 2014, January 15, 2015, and February 20, 2015, I&M submitted an amendment request to allow a permanent extension of the Type A primary containment integrated leak rate test frequency from once every 10 years to once every 15 years. On March 30, 2015, the NRC approved Amendment No. 326 for CNP Unit 1, authorizing the adoption of NEI 94-01 Rev 3-A as the implementation document to develop the performance-based primary containment leakage testing program at CNP Unit 1, in accordance with 10 CFR Part 50, Appendix J, Option B, and allowing I&M to extend the containment Type A test interval for CNP Unit 1 from 10 years to 15 years.

With the approval of CNP Unit 1 Amendment 326 (Reference 11), the due date for the Unit 1 Type A test moved from November 1, 2016, to November 1, 2021. The proposed change would defer the Type A test for CNP Unit 1 until no later than the startup after the spring 2022 refueling outage, which is currently scheduled to begin on April 9, 2022. This represents an extension of approximately five months. For conservatism, an extension of six months was assumed in the analysis. The intervals for the Type B and Type C tests at CNP Unit 1 would remain unchanged at 120 months and 75 months, respectively.

4.0 TECHNICAL ANALYSIS

As required by 10 CFR 50.54(o), the CNP Unit 1 containment is subject to the requirements set forth in 10 CFR 50, Appendix J. Option B of Appendix J requires that containment leakage test intervals be determined using a performance-based approach. In a letter dated March 30, 2015 (Reference 11), the NRC approved a license amendment submitted by I&M, extending the Type A testing interval to 15 years, and establishing November 1, 2021, as the date by which I&M is to conduct the next CNP Unit 1 Type A test.

The NRC clarified its position concerning licensee requests to extend Type A tests beyond the currently approved 15 years in RIS 2008-27 (Reference 7), stating that any extension beyond the end of the month in which the test is due would require a license amendment request.

Per RIS 2008-27, the license amendment request should demonstrate:

- A sound technical justification and/or undue hardship or unusual difficulty
- The requested amendment poses minimal safety risk
- Acceptable plant-specific containment performance, including a plant-specific risk-informed analysis
- That containment does not have a history of significant degradation issues

4.1 Description of Containment

The CNP Unit 1 containment is a steel-lined, reinforced concrete structure. The containment structure, including all its penetrations, includes a low-leakage steel liner designed to contain the radioactive material that may be released from the reactor core following a design basis loss of coolant accident. Additionally, the containment structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions. The steel-lined, reinforced concrete containment structure, including foundations, access hatches, and penetrations is designed and constructed to maintain full containment integrity when subject to accident temperatures and pressure, and the postulated earthquake conditions.

The structure consists of side walls measuring 113 feet (ft.) (nominal) in height from the liner on the base to the spring line of the dome and has a nominal inside diameter of 115 ft. The cylinder is 4 ft. – 6 inch (in.) thick at the base, tapering to 3 ft. – 6 in. seven feet above the base. The thickness of the cylinder remains at 3 ft. – 6 in. to the dome spring line. The thickness of the dome is 3 ft. – 6 in. at the spring line tapering uniformly to 2 ft. - 6 in. at the peak of the dome. The base mat consists of a 10 ft. thick structural concrete slab, increasing to 20 ft. adjacent to the recirculation sump area.

The basic structural elements considered in the design of the containment structure are the base slab, the vertical cylinder and the hemispherical dome, all acting as one structure. The vertical cylindrical wall and the dome of the steel liner are anchored to the concrete by means of horizontal and vertical stiffener angles. In addition, Nelson studs welded to the stiffener angles extend into the concrete and are anchored behind the first layer of reinforcing, thereby preventing pull-out in case of local concrete cracking. The steel base liner is anchored to the concrete by welding it to continuous steel tee bars which in turn are welded to structural members anchored into the base mat. The base liner is covered by a 2 ft. - 0 in. concrete mat. The underground portion of the containment vessel is waterproofed in order to prevent possible corrosion of the reinforcing steel and liner plate due to seepage of ground water. The waterproofing consists of a continuous impervious membrane, which is placed under the mat, and on the outside of the walls. The membrane placed under the mat extends up and around the walls and is taped to the membrane placed on the outside of the walls, thus providing a continuous waterproof surface.

The reinforced concrete structure was designed in accordance with the applicable portions of the American Concrete Institute (ACI) codes ACI-318-63 and ACI-301-66. The structural steel components were designed in accordance with the American Institute of Steel Construction, AISC-69 specifications.

The containment is divided into three main compartments. These are:
a) The lower compartment.
b) The upper compartment.
c) The ice condenser compartment.

The lower compartment encloses the reactor system and associated auxiliary systems equipment. The upper compartment contains the refueling cavity, refueling equipment and polar crane used during refueling and maintenance operations. The upper and lower compartments are separated by a divider barrier. The ice condenser, which contains borated ice provided to absorb the loss of-coolant accident (LOCA) energy, is in the form of an enclosed and refrigerated annular compartment,

located circumferentially between the crane wall and the outer wall of the containment and extends from below to above the operating deck and divider barrier.

The reactor containment structure is a reinforced concrete vertical right cylinder with a slab base and a hemispherical dome. A welded steel liner with a nominal thickness of 3/8 in. at the dome and wall, and 1/4 in. at the bottom is attached to the inside face of the concrete shell, to insure a high degree of leak tightness. The containment structure is designed to contain the radioactive material, which might be released, following a LOCA. The structure serves as both a biological shield and a pressure container.

The ice condenser is a completely enclosed annular compartment located around, approximately 300 degrees, of the perimeter of the upper compartment of the containment, but penetrating the operating deck so that a portion extends into the containment lower compartment. The lower portion has a series of hinged doors that are exposed to the atmosphere of the lower containment compartment which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors that are exposed to the atmosphere of the upper compartment; these also remain closed during normal plant operation. Intermediate deck doors are located below the top deck doors. These doors form the floor of a plenum at the upper part at the Ice condenser and remain closed during normal plant operation. In the ice condenser, ice is held in baskets arranged to promote heat transfer to the ice. A refrigeration system maintains the ice in the solid state. Suitable insulation surrounding both the ice condenser volume and the refrigeration ducts serves to minimize the heat transfer to the ice condenser boundaries.

In the event of a LOCA or steam line break in the containment, the pressure rises in the lower compartment and the door panels located below the operating deck (a portion of the divider barrier) open. This allows the air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the door panels at the top of the ice condenser to open, allowing the air to flow out of the ice condenser into the upper compartment. Steam entering the ice condenser compartment is condensed by the ice, thus limiting the peak pressure and temperature buildup in the containment. Condensation of steam within the ice condenser results in a continual flow of steam from the lower compartment to the condensing surface of the ice, thus reducing the lower compartment pressure. The divider barrier separates the upper and lower compartments and ensures that the steam is directed into the bottom of the ice condenser. Only a limited amount of steam can bypass the ice condenser through the divider barrier.

The containment liner is enclosed within the containment and thus is not directly exposed to the temperature of the environs. The containment ambient temperature during operation is between 60 and 120 Degrees Fahrenheit (°F) in lower containment, and between 60 and 100°F in upper containment.

4.2 Integrated Leak Rate History

Previous CNP Unit 1 ILRT results have confirmed that the containment is acceptable, with considerable margin, with respect to the TS acceptance criterion of 0.25% leakage of containment air weight per day at the design basis loss of coolant accident pressure. Since the last three Type A test results meet the performance leakage rate criteria from NEI 94-01, Revision 3-A (Reference 10), a test frequency of 15 years would be acceptable.

It should be noted that Amendment 332 to CNP Unit 1 TS, issued in October 2016 (Reference 12), changed the value of the allowable leakage rate (La) from 0.25% of containment air weight per day to 0.18% of containment air weight per day. When implementing the TS change into the ILRT procedure and the calculation of La, a more conservative value for containment free volume was also used, resulting in a change of La from 110,219 standard cubic centimeters per minute (sccm) to 68,559 sccm in March of 2017. However, even comparing the past ILRT leakage to the newer, more stringent value of La shows significant margin.

Unit 1 ILRT Results (Type A Test)			
Test Date	Performance Criterion		Acceptance Limit*
June 1989	Mass point Upper Confidence Limit (UCL) leakage with penalties:	0.419 of La 0.582 of new La**	1.0 La
October 1992	Mass point UCL leakage with penalties:	0.044 of La 0.061 of new La**	1.0 La
November 2006	Mass point UCL leakage with penalties:	0.336 of La 0.467 of new La**	1.0 La

* The total allowable "as-left" leakage is 0.75 La

** The new La, established in March 2017, is 68,559 sccm, compared to the La in use at the time of the test, which was 110,219 sccm

No modifications that require a Type A test are planned at CNP Unit 1 prior to the spring 2022 refueling outage, when the next Type A test will be performed in accordance with this proposed change. Any unplanned modifications to containment prior to the next scheduled Type A test would be subject to the special testing requirements of Section IV.A of 10 CFR 50, Appendix J. There have been no pressure or temperature excursions in Unit 1 containment which could have adversely affected containment integrity. There is no anticipated addition or removal of plant hardware within Unit 1 containment which could affect leak-tightness.

4.3 Type B and Type C Testing Programs

CNP Unit 1 Appendix J, Type B and Type C leakage rate test program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B and TS 5.5.14. The Type B and Type C testing program consists of local leak rate tests (LLRT) of penetrations with a resilient seal, double-gasketed manways, hatches and flanges, and CI valves that serve as a barrier to the release of the post-accident containment atmosphere. As discussed in NUREG-1493 (Reference 13), Type B and Type C tests can identify the vast majority (greater than 95%) of all potential containment leakage paths. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life.

A review of the most recent Type B and Type C test results and a comparison with the allowable leakage rate was performed. The combined Type B and Type C leakage for CNP Unit 1 has consistently remained well below allowable leakage (0.6 La). The as-found minimum and as-left maximum pathway leak rate summary totals for the last several refueling outages are shown below.

Unit 1 LLRT Results* (Type B and Type C Tests)						
	La (sccm)	0.6La (sccm)	As-Found Minimum (sccm)	As-Found Minimum as % of 1.0La	As-Left Maximum (sccm)	As-Left Maximum as % of 1.0La
Spring 2010	110,219	66,131	41,616	37.76%	28,481	25.84%
Fall 2011	110,219	66,131	46,107	41.83%	18,561	16.84%
Spring 2013	110,219	66,131	9,838	8.93%	20,883	18.95%
Fall 2014	110,219	66,131	9,836	8.92%	30,252	27.45%
Spring 2016	110,219	66,131	15,725	14.27%	18,612	16.89%
Fall 2017	68,559	41,135	14,001	20.42%	18,151	26.48%
Spring 2019	68,559	41,135	12,847	18.74%	15,663	22.85%

*the total allowable as-found minimum pathway leakage, or as-left maximum pathway leakage is 0.6La

4.4 Supplemental Inspection Requirements

In the SER for NEI 94-01, Revision 2 (Reference 5), the NRC stated the following requirement for the performance of Supplemental Visual Inspections in the SE Section 3.1.1.3, "Adequacy of Pre-Test Inspections (Visual Inspections):"

> Subsections IWE and IWL of the ASME Code, Section XI, as incorporated by reference in 10 CFR 50.55a, require general visual examinations two times within a 10-year interval for concrete components (Subsection IWL), and three times within a 10-year interval for steel components (Subsection IWE). To avoid duplication or deletion of examinations, licensees using NEI TR 94-01, Revision 2, have to develop a schedule for containment inspections that satisfy the provisions of Section 9.2.3.2 of this TR and ASME Code, Section XI, Subsection IWE and IWL requirements.

The second ten-year containment inservice inspection (CISI) interval began March 1, 2010, and concluded February 29, 2020. The CISI Program Plan for the second interval was developed in accordance with the requirements of the 2004 Edition, No Addenda, of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (BPV) Code, Section XI, Subsections IWE and IWL for Inspection Program B, as modified by 10 CFR 50.55a. Identification and evaluation of inaccessible areas during the second ten-year CISI interval are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A) for IWE and 10 CFR 50.55a(b)(2)(viii)(E) for IWL. Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation during the second ten-year CISI interval are performed in accordance with the requirements of 10 CFR 50.55a(b)(ix)(G) and 10 CFR 50.55a(b)(ix)(H).

The third ten-year CISI interval began March 1, 2020 and will conclude on February 28, 2030. The CISI Program Plan for the third interval is based on the rules of ASME Section XI, Subsections IWE and IWL, 2013 Edition, as modified by 10 CFR 50.55a. Identification and evaluation of inaccessible areas during the third ten-year CISI interval are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A) for IWE and 10 CFR 50.55a(b)(2)(viii)(H) and (I) for IWL. Requirements for examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation located in 10 CFR 50.55a(b)(ix)(G) and (H) have been incorporated into the 2013 Edition of Section XI applicable to the third ten-year CISI interval.

Each ten-year CISI interval is divided into three approximately equal-duration inspection periods. A minimum of one inspection during each inspection period of the ISI interval is required by the IWE program. Visual examinations of accessible concrete containment components in accordance with ASME Code, Section XI, Subsection IWL are performed every five years, resulting in at least two IWL examinations being performed during a 15-year Type A test interval. The examinations performed in accordance with the IWE/IWL program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B.

In addition to the IWL examinations, I&M performs a visual inspection of the accessible interior and exterior of the CNP Unit 1 Containment Building prior to each Type A test. This examination is performed in sufficient detail to identify any evidence of deterioration which may affect the reactor building's structural integrity or leak tightness. The examination is conducted in accordance with approved plant procedures to satisfy the requirements of the 10 CFR 50 Appendix J Testing Program. The activity is coordinated with the IWE/IWL examinations to the extent possible.

The table below provide dates of completed and scheduled ILRTs, completed containment surface examinations, along with an approximate schedule for future containment surface examinations.

Unit 1			
Calendar Year	Type A Test (ILRT)	General Visual Examination of Accessible Exterior Surface	General Visual Examination of Accessible Interior (Liner) Surface
2006	November 2006	August 2007 (IWL)	October 2006 (IWE)
2007			
2008			
2009		August 2009 (Appendix J)	October 2009 (IWE)
2010			
2011		August 2011 (IWL)	October 2011 (IWE)
2012			
2013			
2014			October 2014 (IWE)
2015			
2016			
2017		July 2017 (IWL)	
2018			
2019			March 2019 (IWE)
2020			
2021		August 2021 (IWL) or April 2022 (Appendix J)	
2022	April 2022		April 2022 (IWE)

4.4.1 IWE Examinations

A review was conducted for CNP Unit 1 per IWE-1241, Examination Surface Areas (1992 Edition with 1992 Addenda of ASME Section XI) for the initial 10-year Category E-C examination requirements. No areas were deemed susceptible to accelerated degradation and aging; therefore, augmented examinations per Category E-C were not required. This information is documented in the first 10-year CISI Plan for CNP Unit 1. The examinations performed during the first ten-year interval identified the following:

- In 1998, a visual examination of the Unit 1 containment liner in the floor slab to liner area was performed after seal material removal to address the issue of liner corrosion as described in NRC Information Notice 97-10. Corrosion and pitting of the steel liner plate were identified along the moisture barrier seal near the containment cylinder base at elevation 598 ft., 9-3/8 in. Based on a detailed engineering analysis, I&M concluded that the structural integrity of containment to withstand normal operating, design basis accident, and severe accident loads was not affected by the as-found condition of the liner. The leak tight integrity of containment was not impaired, and the liner, as-found, would have fulfilled its function as an effective leak-tight barrier. This condition was documented in an I&M letter to the NRC dated March 8, 2000, (Reference 14). Modifications were made to the floor-liner seal to prevent further degradation of the liner. This issue was included in a special inspection pursuant to NRC Manual Chapter 0350, "Staff Guidelines for Restart Approval." Closure of the issue was documented in an NRC Inspection Report dated January 19, 2000, (Reference 15).

  The area behind the moisture barrier seal is normally inaccessible. In accordance with IWE-1220(b), the area is exempt from the examination requirements of IWE-2000, including the additional examination requirements of IWE-2430. The corrective action for the liner corrosion included modifying the seal design to prevent moisture intrusion and re-coating the affected area. Therefore, augmented examination per IWE-1240 is not required because the area is no longer likely to experience accelerated degradation or aging. Additionally, IWE-1240 does not apply to inaccessible areas. Although not required by IWE-2430 or IWE-1240, I&M has performed supplemental inspections of portions of the area behind the moisture barrier seal to verify the effectiveness of the new design. Sections of the redesigned moisture barrier seal were removed approximately three years after installation, and a visual examination was performed on the liner area where corrosion was previously identified. The visual examination found no moisture intrusion and no active corrosion. Continued monitoring of the moisture barrier seal was performed with the scheduled VT-3 visual examination of the moisture barrier seal area each inspection period (3 inspections are required in a 10-year interval) until the third inspection period of the second CISI interval, when the exam requirement for the moisture barrier was changed back to a general visual examination.

- In 1999, a visual examination of the containment liner identified an apparent weld repair of the liner plate. Surface preparation to allow further inspection dislodged repair material exposing a hole through the liner plate. The hole was circular in appearance with a diameter of approximately 3/16 in. on the exterior surface and 3/4 in. on the interior surface. It appeared that the liner hole resulted from an inadequate repair of a hole drilled in error during plant construction. After the damaged liner plate section was cut out, a piece of wood, determined to be the handle of a wire brush, was found embedded in the concrete. Some minor corrosion was noted on the concrete side of the liner plate in the area of the embedded wire brush. This condition was also determined to be construction-related. The affected area of containment was restored to an acceptable design configuration. The repair was vacuum box tested and subjected to an LLRT. This condition was initially reported pursuant to 10 CFR 50.73 in Licensee Event Report (LER) 2000-001-00 (Reference 16). Additional evaluation of the condition determined that the containment structure would have performed its safety-related function during a design basis accident in the as-found condition, and the LER was retracted by LER 2000-001-01 (Reference 17).

The through-wall hole in the liner plate did not invoke the requirements for augmented examination per IWE-1240 since the hole was not the result of degradation or aging. The corrective action for the through-wall hole included removal of the wire brush and wooden handle to the maximum extent practical without cutting any stiffener steel, performing a repair of the concrete, and replacing the liner section that contained the hole. Therefore, augmented examination per IWE-1240 is not required because the area is no longer likely to experience accelerated degradation or aging since there is no wood in contact with the liner and the inadequate repair/hole has been eliminated.

The examinations performed during the second ten-year interval identified the following:

In the October 2011 IWE inspection, one examination area was identified as being susceptible to accelerated degradation per IWE-1241 (2004 Edition of ASME Section XI). A glycol pipe penetration in Unit 1 had a large amount of wet discoloration due to condensation below the insulated piping. After insulation removal, a VT-1 (detailed) visual examination was performed which revealed minor pitting that did not impact the leak tightness or structural integrity of the containment boundary. This area was classified as an Augmented Examination per Category E-C to get insulation removed for continued monitoring with a VT-1 visual examination during successive inspection periods. The requirement for augmented examination was determined to be no longer required per Paragraph IWE-2420(c) after the results of the fall 2014 inspection found essentially no change from the exam performed in the fall of 2011.

The most recent IWE examination occurred in March 2019. During this examination nine containment penetrations were identified as having rust and/or discoloration caused by condensation, with chipped and/or peeling paint. No structural degradation or wastage was noted, and the condition of all nine penetrations was similar to that identified in the October 2014 and October 2011 IWE examinations.

Since the last ILRT a containment interior surface coating inspection was performed each outage that included the liner plate as part of the Safety-Related Coatings program. Additionally, four IWE inspections have been completed on CNP Unit 1 since the last ILRT. Either the IWE inspections or the Safety-Related Coatings program inspections would satisfy the Appendix J interior inspection requirements and neither has indicated any significant degradation in the containment liner that would prevent it from fulfilling its leak-tight integrity purpose for 10 CFR Part 50 Appendix J. An additional IWE inspection will be performed between now and the requested ILRT performance.

4.4.2 IWL Examinations

Since the last ILRT, in November 2006, there have been three ASME Section XI, Subsection IWL examinations completed, with the most recent examination taking place in July of 2017. These examinations on the concrete exterior were conducted under the direction of the Responsible Engineer using the General and Detailed Visual Exam methods. The first IWL examination to be completed after the 2006 Type A test was performed in 2007, in accordance with the requirements of the 1992 Edition with the 1992 Addenda of ASME Section XI, as part of the first 10-year interval of the CISI Program. The actual IWL examination took place over the course of several months, between February 2006 and July 2007, overlapping the timeframe of the most recent Type A test.

The IWL examination completed in July of 2007 did not reveal any significant observations that could potentially affect the structural integrity of the Unit 1 containment or the calculated design safety margins. During the 2007 inspection, a section of concrete in the Unit 1 containment dome, located 15 feet above the east main steam enclosure (4 feet in length, 15 inches wide, and about 3.5 inches thick) was discovered to have come loose. Exposed rebar and a popout at elevation 700' (azimuth 155°) was also discovered. The other recordable observations consisted of two surface cracks between 1/32" and 1/16", popout and spalling of maximum 1" depth, and efflorescence and loosening of some previously installed patches on the Unit 1 dome. The conditions observed in the 2007 inspection were evaluated and determined to not pose a threat to the design basis margin for the concrete containment structure. The concrete above the east main steam enclosure and the exposed rebar at elevation 700' were subsequently repaired in April of 2008 and April of 2010, respectively.

The second IWL examination performed after the November 2006 ILRT was completed in August 2011 in accordance with the requirements of the 2004 Edition of ASME Section XI in the second 10-year interval. The conditions observed in the 2011 inspection are only associated with Group 7 elements, which are those elements located above elevation 710'-6" (springline) that form the shape of the dome. The maximum depth of the spalling and popout identified in 2011 was only 1 inch. The conditions identified in the 2011 inspection are still bounded by the evaluation performed for the conditions identified in the 2001 and 2006 inspections, which included evaluating Group 7 elements for a loss of concrete cover up to 3 inches.

The requirements of the second 10-year interval of the CISI Program have been met for CNP Unit 1. The most recent IWL examinations for Unit 1 were completed in July 2017, in accordance with the requirements of the 2004 Edition of ASME Section XI in the second 10-year interval. When compared with the previous IWL inspections, the new conditions observed in the 2017 inspection are only associated with Group 7 elements (spalling and popout of maximum 1" depth). The examination of the Unit 1 containment structure did not reveal any significant observations that could potentially affect the structural integrity or the calculated design safety margins. The subject conditions are within the bounds of the conditions that were previously identified and evaluated in the 2001, 2006, and 2011 CISI Program inspections, and the condition of the containment concrete is being tracked under the CISI program. The conditions that were observed in the previous IWL inspections have either been repaired or determined to be structurally acceptable.

An additional IWL examination or Appendix J inspection will be completed prior to the requested ILRT performance date.

4.5 Deficiencies Identified

Consistent with the guidance provided in NEI 94-01, Revision 3-A (Reference 10), Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE/IWL program examinations or at any other times is entered into the corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions.

4.6 Plant-Specific Confirmatory Analysis

4.6.1 Methodology

An evaluation has been performed assessing the risk impact of a one-time extension of the CNP Unit 1 ILRT surveillance interval by six months from the currently approved value of 15 years to 15.5 years. The evaluation is included as Enclosure 3 to this letter. The plant-specific risk assessment followed the guidance in NEI 94-01, Revision 3-A (Reference 10), the NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals" from November 2001 (Reference 18), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in RG 1.200 (Reference 19) as applied to ILRT interval extensions, risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174 (Reference 20), the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval (Reference 21), and the methodology used in Electric Power Research Institute (EPRI) Topical Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (Reference 22).

4.6.2 Summary of Plant-Specific Risk Assessment Results

The risk assessment associated with a one-time extension to the CNP Unit 1 ILRT surveillance interval from 15 years to 15.5 years is considered to be small since it represents a small change to the CNP Unit 1 risk profile. Details of the CNP Unit 1 risk assessment are contained in Enclosure 3 to this letter. The plant-specific results for a one-time extension of the CNP Unit 1 ILRT surveillance interval from the current 15 years to 15.5 years are summarized below.

- RG 1.174 (Reference 20) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small" changes in risk as resulting in increases in Core Damage Frequency (CDF) less than 1.0E-06/year and increases in Large Early Release Frequency (LERF) less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The one-time increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 15 years to 1 in 15.5 years is estimated as 8.35E-9/year for Unit 1 using the EPRI guidance; this value increases negligibly if the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval is included. Therefore, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of RG 1.174.

- When external event risk is included, the one-time increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 15 years to 1 in 15.5 years is estimated as 3.88E-8/year for Unit 1 using the EPRI guidance. As such, the estimated change in LERF is determined to be "very small" using the acceptance of RG 1.174.

- The effect resulting from temporarily changing the Type A test frequency to 1 in 15.5 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing is 0.0014 person-rem/year for Unit 1. NEI 94-01, Revision 3-A (Reference 10), states that a "small" population dose is defined as an increase of ≤ 1.0 person-rem per year, or ≤ 1% of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The results of this calculation meet these

criteria. Moreover, the risk impact for the ILRT extension when compared to other severe accident risks is negligible.

- The one-time increase in the conditional containment failure probability (CCFP) from the 1 in 15 year interval to a 1 in 15.5 year interval is 0.035% for Unit 1. NEI 94-01, Revision 3-A, states that increases in CCFP of ≤ 1.5% is "small." Therefore, this increase is judged to be "small."

5.0 REGULATORY ASSESSMENT

5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

NUREG-1831, Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2, documents the NRC's technical review of I&M's license renewal application for CNP Unit 1 and Unit 2, including Aging Management Programs (AMPs) in place at CNP Unit 1. Section 3.5.2.3.1 of NUREG-1831 lists the AMPs intended to manage the aging effects of containment, and the containment components. Of those listed, three AMPs are relevant to managing containment leakage- the Containment Leakage Rate Testing Program, Inservice Inspection - ASME Section XI, Subsection IWE Program, and Inservice Inspection - ASME Section XI, Subsection IWL Program. The requirements of the Containment Leakage Rate Testing Program, which includes Type A testing, continue to be met, as the Type A test interval continues to meet the guidelines contained in NEI 94-01. The Inservice Inspection - ASME Section XI, Subsection IWE Program, which is discussed in Section 4.4.1 of this enclosure, is unaffected by this proposed amendment request. The Inservice Inspection - ASME Section XI, Subsection IWL Program, which is discussed in Section 4.4.2 of this enclosure, is unaffected by this proposed amendment request.

10 CFR 50.36(c)(3), "Surveillance requirements," states, in part, that TS shall include the "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." This proposed change revises TS 5.5.12 to add the date-related information for the next Type A test performance. Therefore, this 10 CFR 50.36 requirement continues to be met by this change.

10 CFR 50.36(c)(5), "Administrative controls," requires that "provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner" will be included in the TS. 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," requires that the implementation document used to develop a performance-based leakage testing program be included by general reference in the TS. The Appendix J Testing Program is included in the Administrative Controls section of the CNP Unit 1 TS, as TS 5.5.14, "Containment Leakage Rate Testing Program." This proposed change does not remove this administrative control requirement, but simply revises TS 5.5.14, to extend the interval for performing the next Type A Test by approximately five months. Therefore, this 10 CFR 50.36 requirement continues to be met by this proposed change.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of 10 CFR Part 50, Appendix J. Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B, and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed, however, it did alter the frequency at which Type A, Type B, and Type C containment leakage tests must be performed. Under the performance-based option of 10 CFR Part 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The one-time extension of the frequency of the containment Type A test will not affect the design, fabrication, or construction of the containment structure, and the design will continue to account for the effects of natural phenomena. The containment Type A test will continue to be done in accordance with 10 CFR 50 Appendix J using 10 CFR 50 Appendix B quality standards. The frequency of the containment Type A test is being changed in accordance with standards reviewed and approved as compliant with Appendix J. Therefore, there will be no instances where the applicable regulatory criteria are not met.

Based on the considerations 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 continue to be conducted in accordance with the site licensing basis, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, CNP has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements/criteria.

5.2 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP), is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 1. The proposed license amendment request (LAR) would allow for a one-time extension to the 15-year frequency of the CNP Unit 1 containment leakage rate test (i.e. Integrated Leak Rate Test (ILRT) or Type A test). This test is required by TS 5.5.14 "Containment Leakage Rate Testing Program." The proposed one-time change would permit the current ILRT interval of 15 years to be extended by approximately five months to no later than the plant startup after the spring 2022 refueling outage.

I&M has evaluated whether or not 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 amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment involves changes to the CNP Unit 1 containment leakage rate testing program. The proposed amendment does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself, and the testing requirements to periodically demonstrate the integrity of containment, exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment.

The proposed amendment modifies TS 5.5.14, "Containment Leakage Rate Testing Program" to allow for a one-time extension to the containment Type A test interval. The potential consequences of extending the containment Type A test interval one-time by up to six months have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in the Nuclear Regulatory Commission (NRC) Final Safety Evaluation for NEI Topical Report (TR) 94-01, Revision 3-A. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. I&M has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment modifies TS 5.5.14, "Containment Leakage Rate Testing Program" to allow for a one-time extension to the containment Type A test interval. Containment, and the testing requirements to periodically demonstrate the integrity of containment, exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is operated or controlled. 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 amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed amendment modifies TS 5.5.14, "Containment Leakage Rate Testing Program" to allow for a one-time extension to the containment Type A test interval. This amendment does not

alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A. Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that containment would not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current CNP Unit 1 PRA model concluded that extending the ILRT test interval one-time by up to six months results in a very small change to the risk profile. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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 NRC'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. 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.

5.3 Environmental Consideration

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 CONCLUSION

In RIS 2008-27 (Reference 7), the NRC has established its position on the extension of the containment Type A test interval beyond 15 years under 10 CFR Part 50, Appendix J, Option B. The NRC will consider such extensions only under compelling circumstances. The licensee should demonstrate a sound technical justification and/or undue hardship or unusual difficulty, that the requested amendment poses minimal safety risk, acceptable plant-specific containment performance, including a plant-specific risk-informed analysis, and that the containment does not have a history of significant degradation issues.

Based on the results of previous Type A, Type B, and Type C tests performed at CNP Unit 1, along with recent IWE and IWL examinations, CNP Unit 1 containment does not have a history of significant degradation issues. The plant-specific risk analysis demonstrates that the increased risk due to an extension of approximately five months to the containment Type A test is minimal. The probability of a resurgence of COVID-19 in the fall represent an unforeseen emergent condition, and the benefits

of reducing the scope of the fall 2020 outage to mitigate the risk of transmittal and spread of COVID-19 provide a compelling basis to grant an extension.

7.0 PRECEDENT

The proposed amendment incorporates into the CNP Unit 1 TS a change that is similar (i.e., an ILRT interval greater than 15 years), to the following license amendments previously approved by the NRC to extend the Type A test frequency:

- December 20, 2018 (ML18337A422), for Indian Point Nuclear Generating Unit 3
- June 29, 2007 (ML071800319), for Three Mile Island Nuclear Station, Unit 1
- March 24, 2006 (ML060520032), for Seabrook Station, Unit 1
- February 9, 2006 (ML060410310), for River Bend Station, Unit 1
- December 23, 2005 (ML053190343), for St. Lucie Plant, Unit 2

The proposed amendment is also similar in nature to the amendment approved on April 15, 2020 (ML20101G054), for Grand Gulf Nuclear Station, Unit 1, which provided a one-cycle extension to the Type A test frequency due to the need to minimize exposure of essential and non-essential personnel to the COVID-19 virus.

5.0 REFERENCES

1. Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," Revision 0, dated September 1995.
2. NEI document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, dated July 21, 1995.
3. EPRI report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994.
4. NEI document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2, dated August 31, 2007 (ADAMS Accession No. ML072970206).
5. Letter from M. J. Maxin, NRC, to J. C. Butler, NEI, "Final Safety Evaluation For Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, "Industry Guideline For Implementing Performance-Based Option of 10 CFR PART 50, Appendix J" and Electronic Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" (TAC NO. MC9663), dated June 25, 2008 (ADAMS Accession No. ML081140105).
6. NEI document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2-A, dated November 19, 2008 (ADAMS Accession No. ML100620847).
7. NRC Regulatory Issue Summary 2008-27, Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50, dated December 8, 2008 (ADAMS Accession No. ML080020394).

8. NEI document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3, dated June 9, 2011 (ADAMS Accession No. ML112920567).

9. Letter from Sher Bahadur, NRC, to Mr. Biff Bradley, NEI, "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, "Industry Guideline For Implementing Performance-Based Option of 10 CFR PART 50, Appendix J" (TAC NO. ME2164), dated June 08, 2012 (ADAMS Accession No. ML121030286).

10. NEI document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 31, 2012 (ADAMS Accession No. ML12221A202).

11. Letter from A. W. Dietrich, NRC, to L. J. Weber, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Containment Leakage Rate Testing Program (TAC Nos. MF3568 and MF3569), dated March 30, 2015 (ADAMS Accession No. ML15072A264).

12. Letter from A. W. Dietrich, NRC, to J. P. Gebbie, Indiana Michigan Power Company, Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Adoption of TSTF-490, Rev. 0, "Deletion of E-Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification" and Implementation of Full-Scope Alternative Source Term (CAC Nos. MF5184 and MF5185), dated October 20, 2016 (ADAMS Accession No. ML16242A111).

13. NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995.

14. Letter from M. W. Rencheck, I&M, to NRC Document Control Manager, transmitting LER 315/1998-011-03, "LER Retraction, Containment Liner Pitting," dated March 8, 2000 (ADAMS Accession No. ML003695085).

15. Letter from J. A. Grobe, NRC, to R. P. Powers, Indiana Michigan Power Company, "NRC Inspection Report 50-315/99026(DRS); 50-316/99026(DRS)," dated January 19, 2000 (ADAMS Accession No. ML003677524).

16. Letter from M. W. Rencheck, Indiana Michigan Power Company, to NRC Document Control Manager, transmitting LER 316/2000-001-00: "Through-Liner Hole Discovered in Containment Liner," dated February 16, 2000 (ADAMS Accession No. ML003687066).

17. Letter from J. E. Pollock, Indiana Michigan Power Company, to NRC Document Control Manager, transmitting LER 316/2000-001-01: "Through-Liner Hole Discovered in Containment Liner," dated March 16, 2001 (ADAMS Accession No. ML010810216).

18. Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.

19. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009.

20. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, dated January 2018.

21. Letter from C. H. Cruse, Calvert Cliffs Nuclear Power Plant, to NRC Document Control Desk, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, March 27, 2002 (ADAMS Accession No. ML020920100).

22. EPRI report TR-1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325," dated October 2008.

Enclosure 4 to AEP-NRC-2020-40

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, except that the next Type A test performed after the November 1, 2006 Type A test shall be performed no later than the plant startup after the spring 2022 refueling outage.

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.