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

September 3, 2020

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

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 – ISSUANCE OF AMENDMENT NO. 353 RE: ONE CYCLE EXTENSION OF APPENDIX J, TYPE A, INTEGRATED LEAKAGE RATE TEST INTERVAL (EPID L-2020-LLA-0126 [COVID-19])

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 353 to Renewed Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1 (CNP, Unit 1). The amendment consists of changes to the Technical Specifications in response to your application dated June 8, 2020, as supplemented by letter dated July 9, 2020. The amendment revises Technical Specification 5.5.14, “Containment Leakage Rate Testing Program,” to extend the primary containment integrated leak rate test, or Type A test, interval at CNP, Unit 1. Specifically, the amendment allows for a one-time extension of the the current 15 year integrated leak rate test interval by approximately 5 months to 15.5 years and no later than the plant startup after the spring 2022 Refueling Outage.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission’s next biweekly *Federal Register* notice.

Sincerely,

/RA/

Scott P. Wall, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Amendment No.353 to DPR-58
2. Safety Evaluation

cc: Listserv

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 353
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company dated June 8, 2020, as supplemented by letter dated July 9, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 353, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2020.09.03 14:35:22 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: September 3, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 353

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-315

Renewed Facility Operating License No. DPR-58

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

REMOVE	INSERT
-3-	-3-

Technical Specifications

Replace the following page of the Renewed Facility Operating License No. DPR-58, Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

INSERT	REMOVE
5.5-14	5.5-14

and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components; and

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 353, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Less than Four Loop Operation

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

(4) Fire Protection Program

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

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J,” dated July 2012, and Section 4.1, “Limitations and Conditions for NEI TR 94-01, Revision 2,” of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008, 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 ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 353 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application dated June 8, 2020 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20164A044), as supplemented by letter dated July 9, 2020 (ADAMS Accession No. ML20198M423), Indiana Michigan Power Company (I&M, the licensee) submitted a license amendment request (LAR) to revise the technical specifications (TSs) for the Donald C. Cook Nuclear Plant, Unit No. 1 (CNP, Unit 1).

The amendment revises TS 5.5.14, "Containment Leakage Rate Testing Program," to extend the interval of the primary containment integrated leak rate test (ILRT), or Type A test, at CNP, Unit 1. Specifically, the amendment allows for a one-time extension of the the current 15 year integrated leak rate test interval by approximately 5 months to 15.5 years and no later than the plant startup after the CNP, Unit 1, spring 2022 Refueling Outage (RFO).

The licensee stated that the request to defer the ILRT until the plant startup after the spring 2022 RFO is based on the site's performance history, historical plant-specific containment leakage testing program results, containment inservice inspection (ISI) program results, and a supporting plant-specific risk assessment. The licensee further stated that the extension would minimize exposure of essential and non-essential personnel to COVID-19, and expeditiously return CNP, Unit 1, to service in support of the national emergency declaration due to the COVID-19 pandemic by allowing for the timely and efficient release of contracted outage support staff and the transition of non-essential staff personnel to remote working arrangements as soon as possible.

The supplement dated July 9, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 30, 2020 (85 FR 39219).

2.0 REGULATORY EVALUATION,

The CNP, Unit 1, Updated Final Safety Analysis Report (UFSAR), Subsection 5.1, “Application of Design Criteria” (ADAMS Accession No. ML19317D003), states, in part:

> The reinforced concrete structure was designed in accordance with the applicable portions of [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 CNP, Unit 1, UFSAR, Subsection 5.2, “Application of Design Criteria to the Containment Structure,” states, in part:

> 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°F. …

The CNP, Unit 1, UFSAR, Subsection 5.2.2, “Containment System Structure Design,” states, in part:

> The containment is divided into three main compartments. These are:
>
> - The lower compartment.
> - The upper compartment.
> - 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 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.
>
> 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” at the dome and wall, and 1/4” 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 loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.
>
> The structure … consists of side walls measuring 113 ft [feet] (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 thickness of the cylinder is 3 ft - 6 in [inches] and 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 containment structure is inherently safe with regards to common hazards such as fire, flood and electrical storm. The concrete walls are invulnerable to fire, a minimum amount of combustible material, such as lubricating oil in the pump and motor bearings, is present in the containment. A system of lightning rods is installed on the containment dome as protection against electrical storm damage. The dead weight of the structure is a minimum of ten times the buoyancy force that may be exerted on the structure when the ground water table is two feet below grade

The CNP, Unit 1, UFSAR, Subsection 5.3.1, “General Description,” states, in part:

> The Ice Condenser is a completely enclosed annular compartment located around approximately 300° 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 loss-of-coolant accident 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 CNP, Unit 1, UFSAR, Subsection 5.5.2, “Design Bases,” states, in part:

> The Containment Ventilation System is designed to the following parameters:
>
> …
>
> c. Maintain a maximum temperature of 100°F in the containment upper compartment during plant operation and a minimum of 60°F during plant shutdown to permit personnel access as required.
>
> d. Maintain a maximum temperature of 120°F in the lower compartment (135°F inside the primary concrete shield) during plant operation and a minimum of 60°F during an outage.

2.1 Licensee’s Proposed Changes

The proposed change would allow a one-time extension to the next performance of the required Type A containment ILRT required by the TSs.

The CNP, Unit 1, TS 5.5.14, “Containment Leakage Rate Testing Program,” currently states:

> 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.

In the proposed modification to TS 5.5.14, the performance of the next Type A test is changed to no later than the startup after the spring 2022 RFO. This is accomplished by adding “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" to the end of the last sentence in TS 5.5.14 excerpt.

2.2 Regulatory Requirements and Guidance

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the licensee requested a change to the Technical Specifications for CNP, Unit 1.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(o) require that the primary reactor containments for water-cooled power reactors shall be subject to the requirements set forth in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR Part 50 includes two options, "Option A − Prescriptive Requirements" and "Option B − Performance-Based Requirements," either of which can be chosen for meeting the requirements of Appendix J.

The testing requirements in 10 CFR Part 50, Appendix J, ensure that: (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TSs and (b) integrity of the containment structure is maintained during the service life of the containment.

Option B of 10 CFR Part 50, Appendix J, specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by: (1) Type A tests to measure the containment system overall integrated leakage rate; (2) Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries; and (3) Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the performance history of the overall containment system (for Type A tests), and based on the safety significance and performance history of each penetration boundary and isolation valve (for Types B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release.

The overall integrity (structural and leaktight integrity) of the primary containment is verified by a Type A ILRT, and the integrity of the penetrations and isolation valves is verified by Type B and Type C local leak rate tests (LLRT), as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the essential leaktight characteristics of the containment structure at the design-basis accident pressure.

The leakage rate test results must not exceed the maximum allowable leakage rate (L_a) with margin, as specified in the TSs. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration that may affect the containment leaktight integrity must be conducted prior to each Type A test and at a periodic interval between tests.

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires the TSs to include, by general reference, the regulatory guide (RG) or other implementation document used by a licensee to develop a performance-based leakage testing program. Further, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the NRC and endorsed in RG 1.163"Performance-Based Containment Leak-Test Program," dated September 1995 (ADAMS Accession No. ML003740058).

The NRC staff's final safety evaluation (SE) (ADAMS Accession No. ML081140105), dated June 25, 2008, 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” (ADAMS Accession No. ML072970206), and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” dated August 2007 (ADAMS Accession No. ML072970208), was incorporated into NEI 94-01, Revision 2-A, and issued on November 19, 2008 (ADAMS Accession No ML100620847). NEI 94-01, Revision 2-A, describes an NRC-approved approach for implementing the optional performance-based requirements of Option B described in 10 CFR Part 50, Appendix J, and incorporates the regulatory positions stated in RG 1.163. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate testing frequencies, and includes provisions for extending Type A ILRT intervals to up to 15 years. This approach uses industry performance, plant-specific data, and risk insights in determining the appropriate testing frequency, and also discusses the performance factors that licensees must consider in determining test intervals.

The NRC staff’s final SE dated June 8, 2012 (ADAMS Accession No. ML121030286), for NEI 94-01, Revision 3, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” was incorporated into NEI 94-01, Revision 3-A, and issued on July 2012 (ADAMS Accession No. ML12221A202).

EPRI Report No. 1009325, Revision 2-A, provides a generic assessment of the risks associated with a permanent extension of the ILRT surveillance interval to 15 years, and provides a risk-informed methodology to be used to confirm the risk impact of the ILRT extension on a plant-specific basis. Probabilistic risk assessment (PRA) methods are used in combination with ILRT performance data and other considerations to justify the extension of the ILRT surveillance interval. This is consistent with the guidance provided in RG 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 (ADAMS Accession No. ML17317A256), and RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” Revision 1, dated May 2011 (ADAMS Accession No. ML100910008) to support changes to surveillance test intervals.

On March 30, 2015 (ADAMS Accession No. ML15072A264), the NRC issued license Amendment No. 326 for CNP, Unit 1, which revised TS 5.5.14 to require the containment leakage rate testing program to be in accordance with the guidelines contained in NEI 94-01, Revsion 3-A, and the conditions and limitations in Section 4.1 of the NRC SE in NEI 94-01, Revision 2-A.

The NRC staff has previously issued a significant number of license amendments for licensees of reactor units that have requested to extend their Type A test intervals to 15 years on a permanent basis, based primarily on PRA criteria (e.g., see ADAMS Accession Nos. ML19022A324, ML18337A422, ML15028A308, ML17103A235). The NRC 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,” dated December 8, 2008 (ADAMS Accession No. ML080020394), provides guidance on justifications the NRC staff would consider for extending ILRT intervals beyond 15 years.

RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, dated March 2009 (ADAMS Accession No. ML090410014), describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors.

The regulations in 10 CFR 50.55a, "Codes and standards," contain the containment inservice inspection (ISI) program requirements that, in conjunction with the requirements of 10 CFR Part 50, Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

The regulations in 10 CFR 50.65(a)(1) state, in part, that the licensee:

> … shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of [10 CFR 50.65], are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience.

The regulations in 10 CFR 50.36(c) state that the TSs will include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations; (3) surveillance requirements; (4) design features; and (5) administrative controls. Section 50.36(c)(5), "Administrative controls," specifies, in part, that administrative controls are "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The LAR requested a change to the "Administrative Controls" section of the TSs for CNP, Unit 1.

3.0 TECHNICAL EVALUATION

3.1 Background

TS 5.5.14 for CNP, Unit 1, was revised by license Amendment No. 326 to allow the Type A test to be conducted at 15-year intervals based on acceptable performance history, as defined in NEI 94-01, Revision 3-A, and subject to the conditions and limitations in NEI 94-01, Revision 2-A. The last Type A test at CNP, Unit 1, was completed on November 1, 2006; therefore, the next Type A test would need to be performed on or before November 2021. The licensee's proposed change would defer the Type A test until the spring 2022 RFO.

The CNP, Unit 1, license Amendment No. 326 allowed for a maximum ILRTinterval of 15 years with provision for a grace period of up to 9 months beyond 15-year interval provided that an unforeseen emergent condition exists. NRC SE Section 3.1.1.2, "Deferral of Tests Beyond the 15-year interval," for NEI 94-01, Revision 2, states, in part:

> As noted above, Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore,

> if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

3.2 Historical Leakage Rate Test Results

The licensee provided the CNP, Unit 1, historical results of ILRT tests and the combined trend summary of Type B and C LLRTs. In addition, the LAR included a summary of the CNP, Unit 1, IWE examination results of the containment metal liner and the results of the IWL containment concrete visual inspections completed during the first and second 10-year containment ISI intervals.

3.2.1 Integrated Leakage Rate Testing History

By license Amendment No. 332, dated October 20, 2016 (ADAMS Accession No. ML16242A111), the TS acceptance criterion for maximum allowable containment leakage rate, L_a, at P_a, changed from 0.25 to 0.18 percent by weight of the containment air per 24 hours. Additionally, a more conservative value for containment free volume was also used when implementing Amendment No 332, the ILRT procedure, and the calculation of L_a, which resulted in a change of L_a from 110,219 standard cubic centimeters per minute (sccm) to 68,559 sccm. The last two consecutive, successful tests at CNP, Unit 1, were performed in October 1992 and November 2006. The licensee stated that the value of P_a for CNP, Unit 1, is 12.0 pounds per square inch gauge (psig) per TS 5.5.14.

In Section 4.2, "Integrated Leak Rate History" of Enclosure 2 to the LAR, the licensee provided a summary of Type A ILRT results, which demonstrated that the last two Type A tests had containment performance leakage rates less than the L_a (1.0 L_a at P_a) of 0.18 percent containment air weight per day. The licensee also stated:

> 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.

As shown in the LAR supplement dated July 9, 2020, the last two tests for the CNP, Unit 1, primary containment have shown a leakage rate much less than the acceptance criterion, L_a, of 0.18 percent of primary containment air weight per day at a test pressure in excess of P_a. These results permit the ILRT at CNP, Unit 1, to continue to be performed on a 15-year interval in accordance with the guidance of NEI 94-01, Revision 2-A and 3-A. For the CNP, Unit 1, primary containment, the margin of the test results relative to the acceptance criterion support a conclusion that exceeding the performance criterion of L_a would be unlikely with implementation of the proposed one-time interval extension.

The NRC staff reviewed the information related to the licensee's proposal to extend 10 CFR Part 50, Appendix J, ILRT Type A test intervals, including historical leakage test results. In Section 4, "Technical Analysis," of its LAR, the licensee provided test results for the two most recent CNP, Unit 1, ILRT Type A tests of 1992 and 2006. These test results indicate containment performance leakage rates are much less than the maximum allowable containment leakage rate (L_a at P_a) of 0.18 percent primary containment air weight per day. Therefore, the staff concludes that the performance history of the Type A tests supports extending the current CNP, Unit 1, ILRT interval by approximately 5 months to 15.5 years and no later than the plant startup after the CNP, Unit 1, spring 2022 RFO.

3.2.2 Local Leak Rate Testing (Types B and C) History

The CNP, Unit 1, TS 5.5.14 states, in part, that: "Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests." The containment performance is demonstrated by the as-found minimum pathway summations, whereas the as-left maximum pathway summations signify the acceptance criteria for restart. In Section 4.3, "Type B and Type C Testing Programs" of Enclosure 2 to the LAR, the licensee provided trend summaries for CNP, Unit 1, from 2010 to 2019 (total of seven RFOs). The results show that there have been no as-found aggregate Type B and C LLRT failures that resulted in exceeding the acceptance criterion of 0.6 L_a. The results indicate a good margin between the combined Type B and C test totals and the acceptance criterion (0.6 L_a), and suggest that acceptance criteria are unlikely to be exceeded during the proposed extension to conduct the next ILRT test.

The data contained in Section 4.3 of Enclosure 2 of the LAR indicates that the as-found minimum pathway summations represent a high quality of maintenance of Type B and C tested components, while the as-left maximum pathway summations represent an effective management of the 10 CFR 50, Appendix J, testing program" by the program owner. As discussed in NUREG-1493, "Performance-Based Leak-Test Program," dated September 1995 (ADAMS Accession No. ML20098D498), Types B and C tests can identify the vast majority of all potential leakage paths in an ILRT. The licensee is not proposing any changes to the Types B and C test intervals and, thus, the Types B and C testing during the interval between ILRT tests will continue to provide a high degree of assurance that containment integrity is maintained. Based on the above, the NRC staff concludes that continued testing of scheduled Types B and C components during the Fall 2020 RFO and beyond up to the start of the spring 2022 RFO will provide a measure of assurance of the leaktightness of the containment.

3.3 Containment Inservice Inspection Program

The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Type A ILRT), and the containment ISI requirements mandated by 10 CFR 50.55a, together, help ensure the continued leaktight and structural integrity of the containment during its service life. As required by TS 5.5.14, CNP, Unit 1, is subject to the requirements set forth in 10 CFR Part 50, Appendix J, Option B, which requires that test intervals for Type A ILRT be determined by using a performance-based approach. The CNP, Unit 1, ILRT program is based on implementation of the guidance in NEI 94-01, Revision 3-A, and the conditions and limitations in Section 4.1 of the NRC SE in NEI 94-01, Revision 2-A.

3.3.1 Description of Containment

In Section 4.1, "Description of Containment," of the LAR, I&M described the CNP, Unit 1, containment as a steel-lined, reinforced concrete structure that 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 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 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, and the structural steel components were designed in accordance with the American Institute of Steel Construction (AISC) specifications in AISC-69.

The reactor containment structure is a reinforced concrete vertical right cylinder with a slab base and a hemispherical dome. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The ice condenser is a completely enclosed annular compartment located around, approximately 300 degrees, the perimeter of the upper compartment of the containment but penetrating the operating deck so that a portion extends into the containment lower compartment.

3.3.2 ASME Code, Section XI, Subsection IWE Examinations

In Section 4.4.1, "IWE Examinations," of the LAR, the licensee stated that, 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. 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 IWE examinations performed in October 2014 and October 2011.

The LAR stated that 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. The LAR also stated that either the IWE inspections or the safety-related coatings program inspections would satisfy the 10 CFR Part 50, 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.

3.3.3 ASME Code, Section XI, Subsection IWL Examinations

In Section 4.4.2, “IWL Examinations,” of the LAR, the licensee stated that since the last ILRT in November 2006 there have been three ASME Code, Section XI, Subsection IWL, examinations completed with the most recent examination taking place in July 2017. These examinations of the concrete exterior were conducted under the direction of the responsible engineer using the general and detailed visual examination methods.

General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is conducted in accordance with the CNP IWE/IWL program, which implement the requirements of ASME, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g). The IWE/IWL inspections and supplemental inspections, in accordance with other approved CNP plant procedures, are used to satisfy the general visual examination requirements of 10 CFR Part 50, Appendix J, Option B, and to monitor and manage the age-related degradations of the primary containment to ensure that containment structural and leaktight integrity is maintained through its service life.

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 Part 50, Appendix J, testing program and is coordinated with the IWE/IWL examinations to the extent possible. Since the last Type A ILRT test performed for CNP, Unit 1, in November 2006, containment general visual examinations (IWL) of accessible exterior surfaces were performed in August 2007, August 2009 (Appendix J), August 2011, and July 2017. Containment general visual examinations (IWE) of accessible interior (liner) surfaces were performed in October 2006, 2009, 2011 and 2014, and March 2019.

The second 10-year 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 of the ASME Code Section XI, Subsection IWL, as modified by 10 CFR 50.55a. Identification and evaluation of inaccessible areas during the second 10-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 10-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 requirements of the second 10-year interval of the CNP CISI program have been met for CNP, Unit 1. When compared with the previous IWL inspections performed in August 2007 and August 2011, the new conditions observed in the 2017 inspection are only associated with Group 7 elements (spalling and pop out of maximum 1-inch depth). The examination of the CNP, 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 by I&M 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.

The LAR stated that 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 CNP corrective action program for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions.

Based on the results of the most recent ASME Code, Section XI, Subsection IWE and IWL inspections discussed above, the NRC staff concludes that there has not been evidence to date of significant degradation of the CNP, Unit 1, containment structure, and that the degradations noted have been entered into the CNP corrective action program and appropriately managed and corrected. The staff evaluation concludes that there is reasonable assurance that the licensee is adequately implementing the CNP CISI program to monitor and manage age-related degradation of the CNP, Unit 1, containment.

3.4 Risk Evaluation

Enclosure 3 of the LAR provides a plant-specific risk assessment for a one-time extension of the current 15 year ILRT interval by approximately 5 months to 15.5 years and no later than the plant startup after the CNP, Unit 1, spring 2022 RFO.

The licensee states that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A, as endorsed by the NRC. In addition, the proposed LAR follows RG 1.200 on the use of PRA as applied to ILRT interval extensions, the risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174, the methodology used for Calvert Cliffs Nuclear Power Plant to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval, and the methodology used in EPRI Report No. 1009325, Revision 2-A.

The licensee addressed each of the four conditions for the use of EPRI Report No. 1009325, Revision 2-A, contained in NEI 94-01, Revision 2-A, which are listed in Section 4.2 of the NRC SER for the EPRI report. A summary of how each condition is met is provided in Sections 3.4.1 through 3.4.4 below.

3.4.1 Technical Adequacy of the PRA

The first condition stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application. This RG describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors.

Consistent with the information provided in RIS 2007-06, "Regulatory Guide 1.200 Implementation," the NRC staff will use Revision 2 of RG 1.200 to assess technical adequacy of the PRA used to support risk-informed applications received after March 2010. RG 1.200 describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. In Section 3.2.4.1 of the NRC SE for EPRI Report No. 1009325, the NRC staff states that Capability Category (CC) I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, since approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

The licensee addresses CNP's PRA technical adequacy in Section A, "PRA Technical Adequacy for ILRT," of Enclosure 3 to the LAR.

The CNP internal events PRA (including internal flooding) received a full scope peer review in 2015, followed by several focused-scope reviews on various portions of the model, such as pre-initiator human reliability analysis and containment hydrogen analysis. For the purposes of this ILRT extension evaluation, which only requires an assessment of CC-I, only those supporting requirements (SRs) that are currently “Not Met” are evaluated. Of the remaining Internal Events SRs, no impacts were identified on the ILRT extension evaluation.

The CNP fire PRA was subject to a full scope peer review during initial model development in 2010, with follow-on focused-scope reviews occurring in 2015 and 2017. For the purposes of ILRT extension evaluation, which only requires an assessment of CC-I, only those SRs that are currently not met are evaluated. Of the remaining fire PRA SRs, open items related to SR IGN-A 1 are identified as a potential impact on the ILRT extension evaluation. This SR is evaluated as “Not Met” due to the use of fire ignition frequencies based on NUREG/CR-6850 Supplement 1 (ADAMS Accession No. ML15167A550) instead of NUREG-2169 (ADAMS Accession No. ML15016A069). For the purposes of the ILRT extension evaluation, the fire PRA model is quantified with each set of fire ignition frequencies and the more limiting results are used.

The CNP seismic PRA peer review was conducted in 2018, with a formal findings and observations closure review in 2019. For the purposes of ILRT, which only requires an assessment of CC-I, only those SRs that are “Not Met” are listed and evaluated. However, no seismic PRA-related SRs are currently “Not Met,” so no evaluation is provided for the purposes of the ILRT extension evaluation.

Based on review of the above information, the NRC staff concludes the licensee has addressed the relevant findings and gaps from the peer reviews and that they have no impact on the results of this LAR. Therefore, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of the request to take the interval from 15 to 15.5 years. Accordingly, the first condition for the use of EPRI Report No. 1009325, Revision 2-A, contained in NEI 94-01, Revision 2-A, is met.

3.4.2 Estimated Risk Increase

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval is “small,” and consistent with the guidance in RG 1.174. Specifically, a “small” increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a “small” increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment over-pressure for net positive suction head for Emergency Core Cooling System (ECCS) injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed in Section 3.4.4 of this SER, CNP, Unit 1, does not rely on containment over-pressure for ECCS performance. Thus, the associated risk metrics include LERF, population dose, and CCFP.

The licensee provided the results of the plant-specific risk assessment in Section 4.6.2 of the LAR. Details of the CNP, Unit 1, risk assessment are contained in Enclosure 3 to the LAR. The plant-specific results for a one-time extension of the CNP, Unit 1, ILRT interval from the current 15 years to 15.5 years are summarized below.

- RG 1.174 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 CDF less than 1.0E-06/year and increases in 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 frequency from 1 in 15 years to 1 in 15.5 years is estimated as 8.35E-9/year for CNP, 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 frequency from 1 in 15 years to 1 in 15.5 years is estimated as 3.88E-8/year for CNP, Unit 1, using the EPRI guidance. As such, the estimated change in LERF is also determined to be “very small” using the acceptance guidance 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 CNP, Unit 1. NEI 94-01, Revision 3-A, states that a “small” population dose is defined as an increase of 1.0 person-rem per year, or 1 percent 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 CCFP from changing the ILRT interval from 15 years to 15.5 years is 0.035% for CNP, Unit 1. NEI 94-01, Revision 3-A, states that increases in CCFP of less than 1.5% is “small.” Therefore, this increase is determined to be “small.”

Based on the risk assessment results, the NRC staff concludes that the increase in LERF is “very small” and within the acceptance guidelines of RG 1.174. Also, the increase in the total integrated plant risk and the magnitude of the change in the CCFP for the proposed change are “small” and supportive of the proposed change. The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded as a result of the requested change, and the use of the three quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition for the use of EPRI Report No. 1009325, Revision 2-A contained in NEI 94-01, Revision 2-A is met.

3.4.3 Leak Rate for the Large Pre-Existing Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI Report No. 1009325 acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in footnote 2 in Table 6-4 of Enclosure 3, the representative containment leakage for Class 3b sequences in the CNP, Unit 1, plant risk assessment is 100 L_a.

Accordingly, the third condition for the use of EPRI Report No. 1009325, Revision 2-A, contained in NEI 94-01, Revision 2-A, is met.

3.4.4 Applicability if Containment Over-Pressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 6.4 of Enclosure 3 of the LAR, the licensee stated that for CNP, Unit 1, containment over-pressure is not required in support of ECCS performance to mitigate design basis accidents. Accordingly, the fourth condition for the use of EPRI Report No. 1009325, Revision 2-A, contained in NEI 94-01, Revision 2-A, is not applicable.

3.5 Technical Evaluation Summary

The NRC staff concludes that the licensee is satisfactorily monitoring and managing the CNP, Unit 1, containment and performing supplemental inspections to periodically examine and monitor aging degradation, thereby providing reasonable assurance that the containment structural and leaktight integrity will continue to be maintained. The licensee justified the proposed change to extend the performance-based Type A ILRT interval by demonstrating adequate performance of the containment based on plant-specific Type A ILRT program results, consistent with the guidance in NEI 94-01, Revision 3-A, and the conditions and limitations in Section 4.1 of the NRC SE in NEI 94-01, Revision 2-A.

The licensee also demonstrated satisfactory containment inspection results consistent with the ISI program requirements of ASME Code, Section XI, Subsections IWE and IWL. Based on its review, the NRC staff concludes the requested one-time extension of the Type A ILRT interval from 15 years to 15.5 years acceptable.

The LAR provided justification of the proposed one-time change in ILRT interval from a maximum of 15 years to 15.5 years. Primary containment leakage testing intervals have been maintained calendar-based due to variability of refueling cycle lengths, with the expectation that an ILRT would be scheduled for performance during the RFO before the interval would be exceeded.

As noted in Section 3.1 of this SE, NEI 94-01 Revision 3-A, Section 3.1.1.2, includes a grace period not to exceed 9-months, as constrained by the requirements of the TR. As noted before, to enter into or beyond this grace period "… the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists." The guidance assumes a 9-month extension to the 15-year ILRT interval to be generally justifiable on the basis of an unforeseen emergent condition existing. The staff determined that the historical performance of CNP, Unit 1, containment regarding leakage potential suggests that the additional risk associated with a one-time nominal 6-month extension would be low and avoids risk associated with an unforeseen emergent condition—a potential fall 2020 resurgence of COVID-19, including a potential for transmittal and spread of COVID-19 associated with additional personnel that would be necessary to perform a Type A test during the upcoming fall refueling outage. Accordingly, the staff concludes that a Type A test interval extension from a maximum of 15 years to approximately 15.5 years is justified.

In summary, based on the preceding regulatory and technical evaluations, the NRC staff concludes that the licensee adequately implemented its performance-based containment leakage rate testing program. The results of past ILRT and LLRT providedin the LAR demonstrate acceptable performance, and further demonstrate that the structural and leaktight integrity of the containment is being adequately maintained. The staff concludes that, by granting a one-time extension of the current Type A test interval requiring completion of an ILRT prior to the plant startup after the spring 2022 RFO, there is reasonable assurance that the structural and leaktight integrity for the CNP, Unit 1, containment will continue to be maintained without undue risk to public health and safety, and the administrative controls requirement of 10 CFR 50.36(c)(5) will continue to be met. Therefore, the staff concludes that the proposed change to TS 5.5.14 to allow the requested one-time extension of the ILRT interval for CNP, Unit 1, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Michigan official was notified of the proposed issuance of the amendment on August 4, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration in the *Federal Register* on June 30, 2020 (85 FR 39219), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date of Issuance: September 3, 2020

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 – ISSUANCE OF AMENDMENT NO. 353 RE: ONE CYCLE EXTENSION OF APPENDIX J, TYPE A, INTEGRATED LEAKAGE RATE TEST INTERVAL (EPID L-2020-LLA-0126) DATED SEPTEMBER 3, 2020

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