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

May 2, 2022

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

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 359 AND 340 REGARDING ADOPTION OF TSTF-577, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0205)

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment Nos. 359 and 340 to Renewed Facility Operating License Nos. DPR-58 and DPR-74, for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the license and technical specifications (TSs) in response to your application dated November 8, 2021, as supplemented by letter dated February 1, 2022.

The amendments would revise the TSs to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections."

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's monthly *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 Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 359 to DPR-58
2. Amendment No. 340 to DPR-74
3. Safety Evaluation
4. Notice and Environmental Finding

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. 359
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company dated November 8, 2021, as supplemented by letter dated February 1, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 359, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2022.05.02 09:19:18 -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: May 2, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 359

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

REMOVE	INSERT
-3-	-3-

Technical Specifications

Replace the following pages of the Renewed Facility Operating License, Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	---
5.6-4	5.6-4
---	5.6-5

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

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

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

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

(1) <u>Maximum Power Level</u>

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

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

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 359, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

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

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

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

 1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm for an individual SG, for a total leakage of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.”

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.8 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.9 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.9.a and 5.5.9.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

Tests described in Specification 5.5.9.c shall be performed once per 24 months; after 720 hours of adsorber operation; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the charcoal adsorber capability.

Tests described in Specification 5.5.9.d shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a removal efficiency of > 99% of the dioctyl phthalate (DOP) when tested in accordance with the standard and at the system flowrate specified below:

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	≥ 5,400 and ≤ 6,600
ESF Ventilation System	N510-1980	≥ 22,500 and ≤ 27,500
FHAEV System	N510-1980	≥ 27,000 and ≤ 33,000

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a removal efficiency of ≥ 99% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with the standard and at the system flowrate specified below:

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	≥ 5,400 and ≤ 6,600
ESF Ventilation System	N510-1980	≥ 22,500 and ≤ 27,500
FHAEV System	N510-1980	≥ 27,000 and ≤ 33,000

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers, shows the methyl iodide penetration less than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity (RH) specified below:

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor temporary liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a Surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A Surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A Surveillance program to ensure that the quantity of radioactivity contained in all outdoor temporary liquid storage tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.11 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity, an absolute specific gravity, or a specific gravity within limits;

5.5 Programs and Manuals

5.5.11 Diesel Fuel Oil Testing Program (continued)

2. A flash point within limits and, if the gravity was not determined by comparison with the supplier's certification, a kinematic or saybolt viscosity within limits; and

3. A clear and bright appearance with proper color;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in Specification 5.5.11.a above, are within limits; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with required UFSAR updates submitted pursuant to 10 CFR 50.71.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2. Provisions for ensuring the unit is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable;

2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.13.b.1 and 5.5.13.b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

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

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

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

d. Leakage rate acceptance criteria are:

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

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

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

5.5 Programs and Manuals

5.5.15 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed, with certain regulatory positions, in Regulatory Guide 1.129, Revision 3, or of the battery manufacturer including the following:

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

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

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following is an exception to Section C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

> The appropriate application of ASTM E741-00 required by C.1.1 may include minor exceptions to the test methodology. These exceptions shall be documented in the test report.

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREV System, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by testing described in Paragraph C. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Paragraphs C and D, respectively.

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions of Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, “Risk-Informed Method of Control of Surveillance Frequencies,” Revision 1.

c. The provisions of Surveillance Requirement 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” July 2006 (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.7, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the application performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 340
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company dated November 8, 2021, as supplemented by letter dated February 1, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-74 is hereby amended to read as follows:

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

    The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 340, 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 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2022.05.02 09:20:34 -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: May 2, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 340

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-316

Renewed Facility Operating License No. DPR-74

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

REMOVE	INSERT
-3-	-3-

Technical Specifications

Replace the following pages of the Renewed Facility Operating License, Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	---
5.6-4	5.6-4
5.6-5	5.6-5

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 3468 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to the renewed operating license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed operating license shall be completed. Attachment 1 is an integral part of this renewed operating license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 340, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) Leak Testing of Emergency Core Cooling System Valves

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

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm for an individual SG, for a total leakage of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.”

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.8 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.9 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specifications 5.5.9.a and 5.5.9.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

Tests described in Specification 5.5.9.c shall be performed once per 24 months; after 720 hours of adsorber operation; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the charcoal adsorber capability.

Tests described in Specification 5.5.9.d shall be performed once per 24 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test Frequencies.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a removal efficiency of > 99% of the dioctyl phthalate (DOP) when tested in accordance with the standard and at the system flowrate specified below:

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	≥ 5,400 and ≤ 6,600
ESF Ventilation System	N510-1980	≥ 22,500 and ≤ 27,500
FHAEV System	N510-1980	≥ 27,000 and ≤ 33,000

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a removal efficiency of ≥ 99% of a halogenated hydrocarbon refrigerant test gas when tested in accordance with the standard and at the system flowrate specified below:

ESF Ventilation System	ANSI Standard	Flowrate (cfm)
CREV System	N510-1975	≥ 5,400 and ≤ 6,600
ESF Ventilation System	N510-1980	≥ 22,500 and ≤ 27,500
FHAEV System	N510-1980	≥ 27,000 and ≤ 33,000

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers, shows the methyl iodide penetration less than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity (RH) specified below:

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor temporary liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a Surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A Surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A Surveillance program to ensure that the quantity of radioactivity contained in all outdoor temporary liquid storage tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program Surveillance Frequencies.

5.5.11 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity, an absolute specific gravity, or a specific gravity within limits;

5.5 Programs and Manuals

5.5.11 Diesel Fuel Oil Testing Program (continued)

2. A flash point within limits and, if the gravity was not determined by comparison with the supplier's certification, a kinematic or saybolt viscosity within limits; and

3. A clear and bright appearance with proper color;

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in Specification 5.5.11.a above, are within limits; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test Frequencies.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.12.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with required UFSAR updates submitted pursuant to 10 CFR 50.71.

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

   2. Provisions for ensuring the unit is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable;

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to the support system(s) for the supported systems described in Specifications 5.5.13.b.1 and 5.5.13.b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

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 April 22, 2006 Type A test shall be performed no later than plant startup after the fall 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.

5.5.15 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-2010, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed, with certain regulatory positions, in Regulatory Guide 1.129, Revision 3, or of the battery manufacturer including the following:

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

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

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

   The following is an exception to Section C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

   > The appropriate application of ASTM E741-00 required by C.1.1 may include minor exceptions to the test methodology. These exceptions shall be documented in the test report.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREV System, operating at the flow rate required by the VFTP, at a Frequency of 24 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in Paragraph C. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

5.5 Programs and Manuals

5.5.16 Control Room Envelope Habitability Program (continued)

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by Paragraphs C and D, respectively.

5.5.17 Surveillance Frequency Control Program

This program provides controls for Surveillance frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method of Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirement 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," (Westinghouse Proprietary); and

8. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (Westinghouse Proprietary)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.7, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

e. The number and percentages of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 359 AND 340 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKET NOS. 50-315 AND 50-316

Application (i.e., initial and supplements)	Safety Evaluation Date
• November 8, 2021, ADAMS Accession No. ML21312A518 • February 1, 2022, ADAMS Accession No. ML22032A317	• **May 2, 2022** Principal Contributors to Safety Evaluation • Clinton Ashley • Leslie Terry

1.0 PROPOSED CHANGES

Indiana Michigan Power Company (I&M, the licensee) requested changes to the technical specifications (TSs) for Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (CNP), by license amendment request (application). In its application, as supplemented, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendments under the Consolidated Line Item Improvement Process (CLIIP). The proposed changes would revise TS 5.5.7, “Steam Generator (SG) Program,” and TS 5.6.7, “Steam Generator Tube Inspection Report,” based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, “Revised Frequencies for Steam Generator Tube Inspections” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation (SE) of TSTF-577 (ADAMS Accession No. ML21098A188).

The tubes within a SG function as an integral part of the reactor coolant pressure boundary and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

CNP SGs have thermally treated Alloy 690 (Alloy 690TT) tubes.

1.1 Proposed TS Changes to Adopt TSTF-577

In accordance with NRC staff-approved TSTF-577, the licensee proposed changes that would revise CNP TS 5.5.7 and TS 5.6.7.

1.1.1 TS 5.5.7, “Steam Generator (SG) Program”:

- The introductory paragraph to TS 5.5.7 would be revised by replacing “steam generator” with “SG” in a couple instances.
- TS 5.5.7.b.1 would be revised by replacing “steam generator” with “SG” in one instance.
- TS 5.5.7.d.2 would be revised by deleting the requirement to base inspection frequency on the more restrictive metric between either the effective full power months (EFPM) or refueling outage and to use just the EFPM metric.
- TS 5.5.7.d.2 would be revised by deleting the requirement to inspect 100 percent of the tubes during each period in paragraphs d.2.a, d.2.b, d.2.c, and d.2.d (144, 120, 96, and 72 EFPM, respectively) and by adding the requirement to inspect 100 percent of the tubes every 96 EFPM.
- TS 5.5.7.d.2 would be revised by deleting the allowance to extend the inspection period by 3 EFPM months and by deleting the discussion of prorating inspections.
- TS 5.5.7.d.3 would be revised by replacing “shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)” with “shall be at the next refueling outage.”

1.1.2 TS 5.6.7, “Steam Generator Tube Inspection Report”:

- Existing reporting requirement b. would be renumbered as c. and be revised by editorial and punctuation changes.
- New reporting requirement b. would be added to require the nondestructive examination (NDE) techniques utilized for tubes with increased degradation susceptibility be reported.
- Existing reporting requirement c. would be renumbered as c.1. and be revised by editorial and punctuation changes.
- Existing reporting requirement d. would be renumbered as c.2. and be revised to note that the location, orientation (if linear), measured size (if available), and voltage response do not need to be reported for tube wear indications at support structures that are less than 20 percent through-wall. However, the total number of tube wear indications at support structures that are less than 20 percent through-wall would be reported.
- New reporting requirement d. would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis methodology, inputs, and results.

- Existing reporting requirement e. would be renumbered as c.4. and be revised by editorial and punctuation changes.

- Existing reporting requirement f. would be renumbered as e. and be revised by editorial and punctuation changes.

- New reporting requirement f. would be added to require the results of any SG secondary side inspections be reported.

- Existing reporting requirement g. would be renumbered as c.3. and be revised to add the requirements to report a description of the condition monitoring assessment, the margin to the tube integrity performance criteria, and a comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. In addition, the requirement to report the results of tube pulls and in-situ testing would be deleted.

1.2 Additional Proposed TS Changes

In addition to the changes proposed consistent with the traveler discussed in Section 1.1, the licensee proposed the following variations.

1.2.1 Editorial Variations

The licensee noted that CNP TSs have different numbering than Standard Technical Specifications (STSs) on which TSTF-577 was based. Specifically, the "Steam Generator (SG) Program" is numbered 5.5.7 in CNP, Unit 1 and Unit 2, TSs rather than 5.5.9 as stated in the TSTF. In addition, the licensee noted that Unit 2 TS 5.5.7.b.2 is being changed to be consistent with the Unit 1 TS 5.5.7.b.2. Specifically, the wording change is: *"Leakage is not to exceed 0.25 gpm [gallons per minute] for ~~in~~ an individual SG, for a total leakage ~~rate~~ of 1 gpm for all SGs,"* where "for" replaces "in" and "rate" is deleted. The difference between the Unit 1 TS and the Unit 2 TS is being corrected for clarity and to make the two TSs consistent.

1.2.2 Other Variation

The licensee noted that CNP Unit 2 TS 5.5.7.d.2 contains a one-time extension of SG tube inspections from the Unit 2 Cycle 26 refueling outage to the Unit 2 Cycle 27 refueling outage (Fall of 2022). The licensee proposed to delete the one-time extension.

2.0 REGULATORY EVALUATION

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.36(c)(5), "Administrative controls," state that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4." TS Section 5.0, "Administrative Controls," requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. The programs established by the licensee, including the SG program, are listed in the administrative controls section of the TSs to operate the facility in a safe manner.

The NRC staff's guidance for the review of TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STSs for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with NUREG-1431[1], as modified by NRC-approved travelers.

TSTF-577 revised the STSs related to SG tube inspections and SG tube inspection reporting requirements. The NRC approved TSTF-577, under the CLIIP on April 14, 2021 (ADAMS Package Accession No. ML21099A086).

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes to Adopt TSTF-577

The NRC staff compared the licensee's proposed TS changes in Section 1.1 of this SE against the changes approved in TSTF-577. In accordance with SRP Chapter 16.0, the NRC staff determined that the STS changes approved in TSTF-577 are applicable because CNP is a pressurized-water reactor (PWR) design plant and the NRC staff approved the TSTF-577 changes for PWR designs. The NRC staff finds that the licensee's proposed changes to the CNP, Units 1 and 2, TSs in Section 1.1 of this SE are consistent with those found acceptable in TSTF-577.

In the SE of TSTF-577, the NRC staff concluded that the changes to STS 5.5.9, "Steam Generator (SG) Program," and STS 5.6.7, "Steam Generator Tube Inspection Report," were acceptable because, as discussed in Section 3.0 of that SE, they continued to ensure SG tube integrity and, therefore, protected the public health and safety. In particular, the structural integrity performance criterion and accident-induced leakage performance criterion (explained in STS 5.5.9.b, items 1 and 2, respectively) will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). Additionally, the proposed changes to the reporting requirements will provide more detailed and consistent information to the NRC. Therefore, the NRC staff found that the proposed changes to the SG program and inspection reporting requirements were acceptable because they continued to meet the requirements of 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner. For these same reasons, the NRC staff concludes that the corresponding proposed changes to the CNP TSs in Section 1.1 of this SE continue to meet the requirements of 10 CFR 50.36(c)(5).

3.2 Additional Proposed TS Changes

3.2.1 Editorial Variations

The licensee noted that CNP TSs have different numbering than STS. Specifically, the "Steam Generator (SG) Program" is numbered 5.5.7 in CNP, Unit 1 and Unit 2, TSs rather than 5.5.9 as

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[1] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Volume 1, "Specifications," and Volume 2, "Bases," Revision 5, September 2021 (ADAMS Accession Nos. ML21259A155 and ML21259A159, respectively).

stated in the TSTF. The NRC staff finds that the different TS numbering is acceptable because it does not substantively alter TS requirements.

The licensee noted that CNP, Unit 2, TS 5.5.7.b.2 is being changed to be consistent with the CNP, Unit 1, TS 5.5.7.b.2. Specifically, the wording change is to replace "in" with "for" and delete "rate" in the following: *"Leakage is not to exceed 0.25 gpm for ~~in~~ an individual SG, for a total leakage ~~rate~~ of 1 gpm for all SGs,"* where "for" replaces "in" and "rate" is deleted. The difference between the CNP, Unit 1, TS and the CNP, Unit 2, TS is being corrected for clarity and to make the two TSs consistent. The NRC staff finds that CNP, Unit 2, TS 5.5.7.b.2 wording change is acceptable because it does not substantively alter TS requirements.

3.2.2 Other Variation

The licensee noted that CNP, Unit 2, TS 5.5.7.d.2 contains a one-time extension of SG tube inspections from the CNP, Unit 2, Cycle 26 refueling outage to the CNP, Unit 2, Cycle 27 refueling outage (Fall of 2022). The licensee proposed to delete the one-time extension because upon implementation of TSTF-577 prior to the CNP, Unit 2, Cycle 27 refueling outage the one-time extension is no longer needed. Therefore, because the one-time extension in no longer needed, the NRC staff finds that deletion of the CNP, Unit 2, TS 5.5.7.d.2 one-time extension is acceptable.

3.3 TS Change Consistency

The NRC staff reviewed the proposed TS changes for technical clarity and consistency with the existing requirements for customary terminology and formatting. The NRC staff finds that the proposed changes are consistent with Chapter 16.0 of the SRP and are therefore acceptable.

4.0 CONCLUSION

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

NOTICES AND ENVIRONMENTAL FINDINGS

RELATED TO

RELATED TO AMENDMENT NOS. 359 AND 340 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74

INDIANA MICHIGAN POWER COMPANY

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

DOCKET NOS. 50-315 AND 50-316

Application (i.e., initial and supplements)	Safety Evaluation Date
• November 8, 2021, ADAMS Accession No. ML21312A518 • February 1, 2022, ADAMS Accession No. ML22032A317	**May 2, 2022** [redacted]

1.0 INTRODUCTION

Indiana Michigan Power Company (I&M, the licensee) requested changes to the technical specifications (TSs) for Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (CNP), by license amendment request (application). In its application, as supplemented, the licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendments under the Consolidated Line Item Improvement Process. The proposed changes would revise the CNP "Steam Generator (SG) Program" and the "Steam Generator Tube Inspection Report" TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21060B434), and the associated NRC staff safety evaluation of TSTF-577 (ADAMS Accession No. ML21098A188).

2.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Michigan official was notified of the proposed issuance of the amendment on April 11, 2022. The State official had no comments.

3.0 ENVIRONMENTAL CONSIDERATION

The amendments relate, in part, to changes in recordkeeping, reporting, or administrative procedures or requirements. The amendments also relate, in part, to changing requirements with respect to the installation or use of facility components located within the restricted area as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 22, 2022 (87 FR 9651). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 359 AND 340 REGARDING ADOPTION OF TSTF-577, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0205) DATED MAY 2, 2022

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ADAMS Accession No.: ML22102A012 *** via memo**

OFFICE	NRR/DORL/LPL3/PM	NRR/DORL/LPL3/LA	NRR/DSS/STSB/BC*
NAME	SWall	SRohrer	VCusumano
DATE	04/11/2022	04/12/2022	02/08/2022
OFFICE	NRR/DNRL/NCSG	NRR/DORL/LPL3/BC	NRR/DORL/LPL3/PM
NAME	SBloom	NSalgado	SWall
DATE	04/12/2022	05/02/2022	05/02/2022

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