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

March 3, 2021

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. 2 - ISSUANCE OF AMENDMENT NO. 338 REGARDING ONE-TIME DEFERRAL OF THE STEAM GENERATOR TUBE INSPECTIONS (EPID L-2020-LLA-0271 [COVID-19])

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 338 to Renewed Facility Operating License No. DPR-74, for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the License and Technical Specifications (TSs) in response to your application dated December 14, 2020.

The amendment revises TS 5.5.7, "Steam Generator (SG) Program," to allow a one-time deferral of the SG tube inspections from the spring of 2021 to the fall of 2022 refueling outage. The proposed change was requested in response to social distancing recommendations provided by the Centers for Disease Control and Prevention, which have been issued as a defensive measure against the spread of the Coronavirus Disease 2019.

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 No. 50-316

Enclosure:
1. Amendment No. 338 to DPR-74
2. Safety Evaluation

cc: Listserv

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. 338
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 December 14, 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-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 338, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.03.03 15:34:26 -05'00'

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: March 3, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 338

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 page of the Renewed Facility Operating License, 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.

REMOVE	INSERT
5.5-6	5.5-6
5.5-7	5.5-7

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. 338, 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 (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 each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections), except for a one-time extension for the Unit 2 Cycle 26 inspection to be deferred to be performed during the Cycle 27 refueling outage in Fall 2022 and will be performed thereafter at the frequency specified above. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program (continued)

scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

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 not exceed 24 effective full power months or one 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.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 338 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By application dated December 14, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20352A221), Indiana Michigan Power Company (the licensee), requested one-time changes to the technical specifications (TSs) for the Donald C. Cook Nuclear Plant, Unit No. 2 (CNP-2).

The proposed change would allow a one-time deferral of the steam generator (SG) tube inspections required in TS 5.5.7, "Steam Generator (SG) Program," from the spring of 2021 to the fall of 2022 refueling outage. The proposed changes were requested in response to social distancing recommendations provided by the Centers for Disease Control and Prevention, which have been issued as a defensive measure against the spread of the Coronavirus Disease 2019 (COVID-19).

2.0 REGULATORY EVALUATION

2.1 System Description

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, SG tube integrity means that the tubes can perform this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to SG tube integrity are established in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provide regulatory requirements that state that the RCPB shall have "an extremely low probability of abnormal leakage … and of gross rupture" (GDC 14, "Reactor coolant pressure boundary"); "shall be designed with sufficient margin" (GDC 15, "Reactor coolant system design," and GDC 31, "Fracture prevention of reactor coolant pressure boundary"); shall be of "the highest quality standards practical" (GDC 30, "Quality of reactor coolant pressure boundary"); and shall be designed to permit "periodic inspection and testing … to assess … structural and leaktight integrity" (GDC 32, "Inspection of reactor coolant pressure boundary").

CNP-2 received a construction permit prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. Although the plant is exempt from compliance with the current GDC, the licensee states it is in compliance with Criterion 33 "Reactor Coolant Pressure Boundary Capability" of the Plant Specific Design Criteria that were in effect when CNP-2, was licensed, and discusses how CNP-2 meets this criterion in Section 1.4 of its Updated Final Safety Analysis Report (UFSAR).

Section 182(a) of the Atomic Energy Act requires nuclear power plant operating licenses to include TSs as part of any license. In 10 CFR 50.36, "Technical specifications," the U.S. Nuclear Regulatory Commission (NRC or Commission) regulatory requirements related to the content of the TSs are established. The TSs for all current pressurized-water reactor (PWR) licenses require that an SG program be established and implemented to ensure that SG tube integrity is maintained.

2.3 SG Tube Integrity Requirements in the CNP-2 TSs

At CNP-2, programs established by the licensee, including the SG program, are listed in the administrative controls section of the TSs. The requirements for performing SG tube inspections and plugging are described in TS 5.5.7, while the requirements for reporting the SG tube inspections and plugging are described in TS 5.6.7, "Steam Generator Tube Inspection Report."

For CNP-2, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.7.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS 5.5.7.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 5.5.7.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube plugging criteria. The applicable tube plugging criterion specified in TS 5.5.7.c is that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged.

The CNP-2 TS 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," includes a limit on operational primary-to-secondary leakage beyond which the plant must be promptly shut down. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases.

As part of the plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and a steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67 or 10 CFR 100.11 for offsite doses; GDC 19 of 10 CFR Part 50, Appendix A, for control room operator doses (or some fraction thereof, as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for CNP-2 are being changed because of the proposed amendment; thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 SG Design

The CNP-2 had four Westinghouse Model 54T replacement SGs installed in 1989 during the Unit 2 refueling outage 7 (U2C7). Each SG has 3,592 thermally treated Alloy 690 (Alloy 690TT) tubes with a nominal outside diameter of 0.875 inch and a nominal wall thickness of 0.050 inch. The tubes are supported by seven stainless steel tube support plates (TSPs) with quatrefoil-shaped holes and three sets of antivibration bars (AVBs) for the U-bend region of the tubes. The tubes installed in rows 1-8 were thermally treated after bending to reduce residual stress.

3.1.2 Operating Experience

Plugging History

Following the most recent inspection in 2016 (U2C23), a total of 19 tubes have been removed from service by plugging since the SGs were replaced in 1989, as shown below in Table 1.

Table 1: CNP-2 – Tube Plugging Summary

Date	SG 21	SG 22	SG 23	SG 24	Total
Pre-Service	0	1	0	0	1
1990	0	0	0	0	0
1992	0	0	0	0	0
1994	0	3	6	0	9
1997	1	0	0	4	5
2002	0	0	0	0	0
2004	0	1	0	0	1
2012	0	1	0	0	1
2016	0	2	0	0	2
Total Tubes Plugged	**1**	**8**	**6**	**4**	**19**
Total Percentage	0.028%	0.223%	0.167%	0.111%	0.132%

One tube was plugged prior to placing the SGs into operation. In 1994, nine tubes were plugged due to mechanical damage sustained during pressure pulse cleaning operations. Foreign object (FO) wear resulted in five tubes being plugged in 1997 and one tube being plugged in 2004. Due to repeated difficulty inspecting the U-bend region of one low row tube, that one tube was plugged in 2012. Two tubes were plugged in 2016 due to volumetric indications that resembled FO wear.

Recent Inspection Summaries

The last two SG inspections at CNP-2 were in spring 2012 (end-of-cycle (EOC) outage 20 (U2C20)) and fall 2016 (U2C23). More information regarding the SG inspections is available in the spring 2012 and fall 2016 tube inspection reports (ADAMS Accession Nos. ML12284A278 and ML17150A304, respectively).

U2C20 Summary

A single AVB wear indication with a depth of 11 percent through-wall was reported in SG 23. This was the only AVB wear indication found since the SGs commenced operation in 1989. The affected tube was left in service.

There were 40 TSP wear indications detected in the four SGs, of which, 33 were new and seven were repeat indications. The largest indication was 14 percent through-wall. The affected tubes were left in service.

The tube in row 2 column 21 of SG 22 was plugged due to inspectability issues in the small radius U-bend of this tube.

The secondary-side inspection activities included visual inspections of the divider lane and annulus at the top of the tubesheet and select inner bundle passes in all four SGs. FOSAR efforts were also conducted in SG 23. Application of Advanced Scale Conditioning Agent and water lancing was also applied to all four SGs. The combined efforts removed 5,835.5 pounds of material from the four SGs.

Channel head and plug visual inspections were also performed and no degradation was noted.

U2C23 Summary

Three AVB wear indications were reported. Two of the indications were in SG 22 and one was in SG 23. The indications in SG 22 were new and were sized at 8 and 11 percent through-wall. The indication in SG 23 was historical and had grown from 11 percent through-wall in the previous inspection to 15 percent through-wall in 2016. The affected tubes were left in service.

There were 79 TSP wear indications detected in the four SGs, of which, 39 were new and 40 were repeat indications. The largest indication was 13 percent through-wall. The affected tubes were left in service.

Two volumetric indications were detected in SG 22 just above the fifth hot-leg TSP in the outer periphery tubes. The two volumetric indications showed a change in the bobbin voltage signal from the inspection in 2012. Examination with a +Point™ probe confirmed that they were volumetric indications with no loose part present. The indications measured 38 and 39 percent through-wall. Both tubes were stabilized and plugged.

The secondary-side activities for the CNP-2 SGs in U2C23 included visual inspections of the swirl vanes and steam dryers in all four SGs, a general steam drum inspection in all four SGs, and a feedring / J-nozzle inspection in three of the four SGs. Sludge lancing and a post-sludge lancing inspection were performed, as well as a foreign object search and retrieval (FOSAR) in all four SGs. Sludge lancing removed 87 pounds of material from the four SGs.

Channel head and plug visual inspections were also performed and no degradation was noted.

3.2 Proposed TS Changes

3.2.1 Current TS Requirements

The SG program in CNP-2 TS 5.5.7 provides the SG tube inspection requirements. TS 5.5.7.d for CNP-2 requires periodic SG tube inspections to be performed and specifies provisions to be met for such inspections. TS 5.5.7.d.1 specifies the tube inspection scope required to be met during the first refueling outage following installation. TS 5.5.7.d.2 states, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full-power months or at least every third refueling outage (whichever results in more frequent inspections)."

Additionally, TS 5.5.7 states that 100 percent of the tubes are required to be inspected during each sequential period of 144 effective full-power months (EFPM), 120 EFPM, 96 EFPM, and 72 EFPM. The U2C26 outage in spring 2021 is the first outage in the third inspection period, which is 96 EFPM in duration.

3.2.2 Description of Proposed TS Changes

The license amendment request proposes to add language into TS 5.5.7.d.2 that indicates the SG tube inspections scheduled for refueling outage U2C26 are to be deferred to refueling outage U2C27 in the fall of 2022 and, thereafter, will be performed at the 72 EFPM frequency specified above

TS 5.5.7.d.2 would be changed by adding to the end of the first sentence language that states:

> , except for a one-time extension for Unit 2 Cycle 26 inspection to be deferred to be performed during the Cycle 27 refueling outage in Fall 2022 and will be performed thereafter at the frequency specified above

3.3 NRC Staff Evaluation of Proposed TS Changes

The NRC staff evaluation of the proposed one-time TS changes for Cycle 27 was performed within the context of the COVID-19 pandemic and the potential impacts of the COVID-19 virus on plant personnel. Therefore, this safety evaluation should not be considered precedent setting for future routine plant amendments or generic industry licensing actions related to SG inspection intervals.

The NRC staff evaluation of the proposed one-time TS changes focused on the potential for affecting SG tube integrity, since maintaining SG tube integrity ensures the plant will meet the SG Program related TS, thereby protecting public health and safety. In particular, the staff evaluation assessed whether the amendment request demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for Cycle 27. The SIPC and AILPC are defined in TS Section 5.5.7.b

The CNP-2 inspections have detected tube degradation from AVB and TSP wear, and the operational assessment (OA) provided by the licensee evaluates these as existing mechanisms using arithmetic deterministic analyses. These analyses use the worst-case single-tube analysis method of the Electric Power Research Institute (EPRI), "Steam Generator

Management Program: Steam Generator Integrity Assessment Guidelines," Revision 4 (EPRI IAG) (ADAMS Accession No. ML16208A273, non-publicly available), to provide a conservative estimate of the projected end-of-cycle (EOC) condition considering all uncertainties at a probability of 0.95 and at 50 percent confidence. The uncertainties used in the assessment are for the burst equation, the material strength, and the nondestructive examination (NDE) flaw sizing technique. The single tube methods are referred to as "worst-case degraded tube" methods because the most severely flawed tube is selected for evaluation. The worst-case degraded tube OA methods involve selecting the most severely flawed tube at the beginning-of-cycle and applying conservative flaw growth over the intended inspection interval, to arrive at a predicted EOC flaw size and then determine if the SIPC and AILPC will be met at the EOC.

The OA also included a probabilistic full-bundle analysis for TSP wear, which was created with Framatome's Mathcad SG Full-Bundle, Fully Probabilistic model. This analysis uses probabilistic models to determine the probability of burst and leakage. The projected EOC results were compared with the SIPC and AILPC acceptance criteria.

While FO wear has occurred in the CNP-2 SGs, FO wear was evaluated as a potential mechanism, since all known FOs causing the wear were either removed from the SGs or the tubes exhibiting the FO wear were plugged. Since there are no remaining FOs with the potential to cause wear or in service tubes with indications of FO wear, the OA provides a discussion of ongoing FO prevention and inspection.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

Wear at AVBs

Wear at AVBs has been detected in two of the four SGs at CNP-2 but has not resulted in any tubes being plugged. In the most recent U2C23 SG inspections, three AVB wear indications were detected in three tubes, two indications in SG 22 and one indication in SG 23. The U2C23 inspections for AVB wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. Sizing of the AVB wear indications was based on the bobbin probe results using an EPRI-qualified technique. The deepest indications returned to service were 11 and 15 percent through-wall in SG 22 and SG 23, respectively.

The licensee's OA for AVB wear was performed using the arithmetic approach described in the EPRI IAG. This approach adjusts the deepest AVB wear flaw returned to service to account for eddy current regression and sizing uncertainties, applies a 95th percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits. Due to the small number of AVB wear indications in the CNP-2 SGs, the licensee used the maximum measured growth rate at U2C23 instead of a 95th percentile growth rate.

The structural limit for AVB wear flaws is 41.7 percent through-wall, which includes burst strength and material property uncertainties. Since the EOC projected AVB wear flaw size of 31.8 percent through-wall is less than the structural limit for AVB wear flaws, the SIPC would be satisfied at the end of Cycle 27. Based on the detectability of wear flaws with the bobbin probe and the 100 percent scope of the in-service tube exams, the licensee assumed that the maximum size of a missed AVB wear flaw during the last inspection was equal to the maximum AVB wear size returned to service, thus, the AVB wear analysis is also bounding for any potentially missed indications during the U2C23 inspection. As noted in the EPRI guidance document, "Steam Generator Management Program: Steam Generator Degradation Specific

Management Flaw Handbook," Revision 2 (EPRI SG Degradation Handbook) (ADAMS Accession No. ML17103A395, non-publicly available), wear flaws from AVBs will leak and burst at essentially the same pressure, and since the differential pressure limit for accident-induced leakage integrity is much lower than the differential pressure limit for structural integrity, by meeting the SIPC, the NRC staff concludes that the AILPC will also be satisfied.

Wear at TSPs

Wear at TSPs has been detected in three of the four SGs at CNP-2 but has not resulted in any tubes being plugged. In the most recent U2C23 SG inspections, 79 TSP wear indications were detected in fifty-five tubes. There are no TSP indications in SG 22. The U2C23 inspections for TSP wear consisted of full-length bobbin probe examinations of 100 percent of the active tubes. Sizing of the TSP wear indications was based on the bobbin probe results using an EPRI-qualified technique. The deepest indications returned to service in each SG were 13 percent through-wall in SG 21 and 10 percent through-wall in SG 23 and SG 24.

The licensee's OA for TSP wear was performed using the arithmetic approach described in the EPRI IAG. This approach adjusts the deepest TSP wear flaw returned to service to account for eddy current regression and sizing uncertainties, applies a 95^{th} percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits. The licensee used the maximum measured growth rate over the last three inspections, instead of a 95^{th} percentile growth rate.

The structural limit for TSP wear flaws is 43.2 percent through-wall, which includes burst strength and material property uncertainties. Since the EOC projected TSP wear flaw size of 29.2 percent through-wall is less than the structural limit for TSP wear flaws, the SIPC would be satisfied at the end of Cycle 27. Based on the detectability of wear flaws with the bobbin probe and the 100 percent scope of the tube exams at CNP-2, the licensee assumed that the maximum size of a missed TSP wear flaw during the last inspection was equal to the maximum TSP wear size returned to service, thus, the TSP analysis is bounding for any potentially missed indications during the U2C23 inspection. As noted in the EPRI SG Degradation Handbook, wear flaws from TSPs will leak and burst at essentially the same pressure, and since the differential pressure limit for accident-induced leakage integrity is much lower than the differential pressure limit for structural integrity, by meeting the SIPC, the NRC staff concludes that the AILPC will also be satisfied.

The licensee also assessed TSP wear with a full-bundle probabilistic analysis, which was created with Framatome's Mathcad SG Full-Bundle, Fully Probabilistic model that included material property, NDE detection, and burst equation relationship uncertainties. This analysis is more responsive to extreme growth rates because it explicitly accounts for an increased probability that large growth rates and large EOC depths will occur, as more deep flaws are returned to service. To estimate future TSP wear growth rates from all SGs, the probabilistic model assumed the existing TSP wear indications from all four SGs existed in one SG bundle and the growth rates of the U2C23 TSP wear indications were then modeled over 7.5 effective full-power years. The quantity of new indications was projected by using a Weibull curve that was fit to the historical TSP wear indications at CNP-2. Both existing and new TSP wear scars were assumed to have a bounding length of 1.2 inches, since this length is greater than the TSP thickness, and a fixed structural to maximum depth ratio of 1.0, which represents a bounding flat wear profile. The full-bundle probabilistic results had a probability of survival of 0.981, which exceeded the required 0.95 value, and was therefore acceptable.

Evaluation Summary for Existing Mechanisms (Wear at AVBs and TSPs)

The NRC staff finds the licensee's evaluation of tube wear at AVBs and TSPs to be acceptable. Wear at these locations in the CNP-2 SGs has been effectively managed since SG installation in 1989, without challenging tube integrity. Wear at AVB and TSP locations in the CNP-2 SGs has been effectively managed since SG installation in 1989, without challenging tube integrity. Wear at support structures is readily detected with standard eddy current examination techniques and wear sizing errors are considered in the projection of existing flaws to U2C27.

The licensee provided deterministic wear analyses for operation until U2C27 and U2C28 at the AVBs and TSPs. For AVB wear growth, the analysis used the maximum growth rate observed in the U2C23 inspection. In a similar manner, the licensee evaluation of TSP wear predicts a conservative number of new TSP indications and applied the maximum growth rate observed in the last three inspections. The results of the AVB and TSP wear analyses, with projected conservative wear rates through U2C27, predict that tube integrity will be maintained. Therefore, the staff finds the licensee's evaluation of wear at support structures to be acceptable since the SIPC and AILPC will be satisfied.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

Foreign Object Wear

The CNP-2 SGs have reported FO wear at the top of the tubesheet only two times since SG installation in 1989, once in 1997 and again in 2004. These instances of FO wear resulted in the plugging of six SG tubes. The U2C23 inspections for FO wear in each SG consisted of full-length bobbin probe examination of 100 percent of the active tubes in all SGs and array probe inspections of all tubes from the first TSP to the tube end in SG 22. No FO wear was detected at the top of tubesheet during the U2C23 inspections. As noted previously, two volumetric indications less than the TS plugging limit were detected in SG 22, just above the fifth hot-leg TSP in the outer periphery tubes. Both tubes were stabilized and plugged. Sludge lancing and post-sludge lancing inspections were performed, as well as a FOSAR in all four SGs and 87 pounds of material were removed. Based on the experience at CNP-2, the licensee concluded it was unlikely that FO wear exceeding the structural limit of 43.2 percent through-wall would occur by the end of Cycle 27.

Evaluation Summary for Potential Mechanisms

The NRC staff finds the licensee's analysis of FO wear acceptable based on the few known detrimental FOs in the CNP-2 SGs since the replacement SGs were placed in service in 1989. The staff also acknowledges that predicting future FO generation is not possible, since past fleet-wide operating experience has shown that new FO generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of FOs by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with FOs. Plants in general, have demonstrated the ability to conservatively manage FOs once they are detected by eddy current examinations or by secondary-side FOSAR inspections. If unanticipated aggressive tube wear from new FOs should occur in a CNP-2 SG, operating experience has shown that a primary-to-secondary leak will probably occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the staff will interact with the licensee in accordance with established procedures in Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (ADAMS Accession

No. ML18093B067), to confirm the licensee's conservative decision making.

3.4 Primary-to-Secondary Leakage Actions

CNP-2 operated with no detectable primary-to-secondary leakage from the time of SG replacement in 1989 until 2016, when the air ejector radiation monitor detected a small leak in SG 22 that ranged from 0.04 to 0.08 gallons per day (gpd). During the U2C23 outage, extensive visual and eddy current examination in SG 22 did not reveal the source of the leakage. After replacement of two damaged fuel assemblies and restart of the reactor, the leakage became undetectable. TS 3.4.13, "RCS Operational LEAKAGE," has a requirement to "verify primary-to-secondary LEAKAGE is ≤150 gpd through any one SG." In addition to the TS requirements, CNP-2 has administrative limits for responding to primary-to-secondary leakage during operation. These limits require increased levels of monitoring starting at leakage of 5 gpd or more. The licensee proposed no changes to these existing TS and administrative limits. The NRC staff finds this acceptable since the administrative limits require prompt and controlled shut down at a significantly lower primary-to-secondary leakage level compared to the TS Operational Leakage limits.

3.5 Technical Evaluation Conclusion

Based on the above, the NRC staff finds that the licensee has demonstrated there is reasonable assurance that the structural and leakage integrity of the CNP-2 SG tubes will be maintained until the next SG tube inspections during Refueling Outage 27 in the fall of 2022. Therefore, the NRC staff concludes that the licensee may incorporate the proposed changes into TS 5.5.7.

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 February 9, 2021. 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, as published in the *Federal Register* on December 31, 2020 (85 FR 86969), 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 amendments 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: March 3, 2021

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT NO. 338 REGARDING ONE-TIME DEFERRAL OF THE STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2020-LLA-0271) DATED MARCH 3, 2021

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