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

January 23, 2020

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

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 349 AND 330 TO APPLY LEAK-BEFORE-BREAK METHODOLOGY TO REACTOR COOLANT SYSTEM BRANCH LINES AND DELETION OF CONTAINMENT HUMIDITY MONITOR REQUIREMENT (EPID L-2018-LLA-0726)

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 349 and 330 to Renewed Facility Operating License Nos. DPR-58 and DPR-74, for the Donald C. Cook Nuclear Plant (CNP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated November 20, 2018, as supplemented by letter dated August 22, 2019.

The amendments revise the CNP TSs to apply leak-before-break methodology to the piping associated with the CNP, Unit No. 2, accumulator, residual heat removal system, and safety injection systems and change CNP, Unit No. 2 TS 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," to change the value for unidentified leakage from 1 gallon per minute (gpm) to 0.8 gpm. The amendments also revise the CNP, Unit Nos. 1 and 2, TS 3.4.15, "RCS Leakage Detection Instrumentation," to delete the reference to the containment humidity monitor.

A copy of our related safety evaluation is also enclosed. A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert F. Kuntz, 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. 349 to DPR-58
2. Amendment No. 330 to DPR-74
3. Safety Evaluation

cc: Listserv

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 349
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 20, 2018, as supplemented by letter dated August 22, 2019, 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:

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 349, 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 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

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: January 23, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 349

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-315

Renewed Facility Operating License No. DPR-58

Replace the following pages of the Renewed Facility Operating License No. DPR-58 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change

INSERT	REMOVE
Page 3	Page 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 areas of change.

INSERT	REMOVE
3.4.15-1	3.4.15-1
3.4.15-2	3.4.15-2
3.4.15-3	3.4.15-3
3.4.15-4	3.4.15-4

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

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

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

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 349, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Less than Four Loop Operation

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

(4) Fire Protection Program

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump monitor in each sump;

b. One containment atmosphere particulate radioactivity monitor; and

c. One containment atmosphere gaseous radioactivity monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment sump monitor(s) inoperable.	A.1	-------------NOTE------------- Not required until 12 hours after establishment of steady state operation. ----------------------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>		
	A.2	Restore containment sump monitor(s) to OPERABLE status.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Containment atmosphere particulate radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 12 hours
	<u>OR</u>	
	B.1.2 -------------NOTE------------- Not required until 12 hours after establishment of steady state operation. ----------------------------------	
	Perform SR 3.4.13.1.	Once per 12 hours
	<u>AND</u>	
	B.2 Restore containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
C. Containment atmosphere gaseous radioactivity monitor inoperable.	C.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	<u>OR</u>	
	C.1.2 -------------NOTE------------- Not required until 12 hours after establishment of steady state operation. ----------------------------------	
	Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>	
	C.2 Restore containment atmosphere gaseous radioactivity monitor to OPERABLE status.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Containment sump monitor(s) inoperable. AND Containment atmosphere particulate radioactivity monitor inoperable.	D.1 Analyze grab samples of the containment atmosphere.	Once per 12 hours
	AND	
	D.2.1 Restore containment sump monitor(s) to OPERABLE status.	7 days
	OR	
	D.2.2 Restore containment atmosphere particulate radioactivity monitor to OPERABLE status.	7 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3.	6 hours
	AND	
	E.2 Be in MODE 5.	36 hours
F. All required monitors inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the containment sump monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitors.	In accordance with the Surveillance Frequency Control Program

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 330
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 20, 2018, as supplemented by letter dated August 22, 2019, 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:

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 330, 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 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

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: January 23, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 330

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 pages of the Renewed Facility Operating License No. DPR-74 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change

INSERT	REMOVE
Page 3	Page 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 areas of change.

INSERT	REMOVE
3.4.13-1	3.4.13-1
3.4.15-1	3.4.15-1
3.4.15-2	3.4.15-2
3.4.15-3	3.4.15-3
3.4.15-4	3.4.15-4

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. 330, are hereby incorporated into 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 0.8 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump monitor in each sump;

b. One containment atmosphere particulate radioactivity monitor; and

c. One containment atmosphere gaseous radioactivity monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment sump monitor(s) inoperable.	A.1	-------------NOTE------------- Not required until 12 hours after establishment of steady state operation. ----------------------------------- Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	A.2	Restore containment sump monitor(s) to OPERABLE status.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Containment atmosphere particulate radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere.	Once per 12 hours
	OR	
	B.1.2 -------------NOTE------------- Not required until 12 hours after establishment of steady state operation. -----------------------------------	
	Perform SR 3.4.13.1.	Once per 12 hours
	AND	
	B.2 Restore containment atmosphere particulate radioactivity monitor to OPERABLE status.	30 days
C. Containment atmosphere gaseous radioactivity monitor inoperable.	C.1.1 Analyze grab samples of the containment atmosphere.	Once per 24 hours
	OR	
	C.1.2 ------------NOTE------------ Not required until 12 hours after establishment of steady state operation. -----------------------------------	
	Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	C.2 Restore containment atmosphere gaseous radioactivity monitor to OPERABLE status.	30 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Containment sump monitor(s) inoperable. AND Containment atmosphere particulate radioactivity monitor inoperable.	D.1 Analyze grab samples of the containment atmosphere. AND D.2.1 Restore containment sump monitor(s) to OPERABLE status. OR D.2.2 Restore containment atmosphere particulate radioactivity monitor to OPERABLE status.	Once per 12 hours 7 days 7 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. All required monitors inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the containment sump monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 349 AND 330 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

1.0 INTRODUCTION

By application dated November 20, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18334A267) as supplemented by letter dated August 22, 2019 (ADAMS Accession No. ML19238A066), Indiana Michigan Power (the licensee) submitted a license amendment request (LAR) for the Donald C. Cook Nuclear Plant (CNP), Unit Nos. 1 and 2. The LAR proposed to apply leak-before-break (LBB) methodology to the piping associated with the CNP, Unit No. 2, accumulator, residual heat removal (RHR) system, and safety injection (SI) systems and change the value for unidentified leakage in CNP, Unit No. 2, Technical Specifications (TS) 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," from 1 gallon per minute (gpm) to 0.8 gpm. The LAR also requested to revise the CNP, Unit Nos. 1 and 2, TS 3.4.15, "RCS Leakage Detection Instrumentation," to delete to the requirements for the containment humidity monitors. The LAR stated that the request to apply LBB for CNP, Unit No. 2 is consistent with the LAR submitted for CNP, Unit No. 1 on March 7, 2018 (ADAMS Accession No. ML18072A012). Application of the LBB methodology to the CNP, Unit No. 1, was previously approved by amendment dated August 12, 2019 (ADAMS Accession No. ML19170A362).

The supplement dated August 22, 2019, provided additional information and clarified the application but did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 5, 2019 (84 FR 7937).

2.0 REGULATORY EVALUATION

2.1 System Descriptions

The LAR, as supplemented, provided a summary of the applicable systems related to the LBB methodology for CNP, Unit No. 2, including the accumulator lines, RHR lines, and safety injection (SI) line. The LAR described the leakage detection relied upon to support the deletion

of the containment humidity monitors. The leakage detection which will be credited are the containment atmosphere particulate radioactivity monitor, containment atmosphere gaseous radioactivity monitor, and an increase in the amount of coolant make-up water, which is required to maintain normal level in the pressurizer, or an increase in containment sump level.

2.1.1 Accumulator Lines

The LAR described that the CNP, Unit No. 2 accumulator lines are attached to each of the four primary reactor coolant loop (RCL) cold leg piping segments. The accumulator lines allow for injection from the accumulator tanks into the reactor vessel, via the cold leg piping, to provide emergency core cooling.

2.1.2 RHR Lines

The LAR described that the CNP, Unit No. 2, RHR lines evaluated for LBB are separated into the RHR suction line and RHR return lines. The RHR lines draw coolant from the RCL, pass it through a heat exchanger to remove residual heat, and return the coolant back to the RCL. The RHR suction line is attached to one of the four RCL hot leg piping segments. The RHR return lines are attached to the accumulator line piping for two of the four reactor coolant loops.

2.1.3 SI

The LAR described that the SI lines evaluated for LBB are associated with both the RCL hot leg piping and cold leg piping. Hot leg SI lines are directly attached to each of the four RCL hot leg piping segments.

2.1.4 Containment Atmosphere Particulate Radioactivity Monitors

The LAR stated that for CNP, Unit Nos. 1 and 2, the containment atmosphere particulate radioactivity monitors are the most sensitive instruments of those available for detection of RCS leakage into containment. The containment atmosphere particulate radioactivity monitor systems are not safety-related, have no safety or accident mitigation functions, and have no nonsafety related functions. The containment atmosphere particulate radioactivity monitors provide indication, record, and alarm functions in the control room, but no control function.

The LAR states that these instruments are capable of detecting particulate radioactivity in concentrations as low as 10^{-9} microcurie/cubic centimeter (μCi/cc) of containment air. The sensitivity of the particulate radioactivity monitors to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal baseline leakage into containment. The sensitivity is greatest where baseline leakage is low. Assuming a low background of particulate radioactivity, a reactor coolant corrosion product radioactivity of 0.2 μCi/cc (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment atmosphere, the particulate radioactivity monitor is capable of detecting leaks as small as approximately 0.0013 gpm (5 cc/minute) within 30 minutes after they occur. If only 10 percent of the particulate activity is actually dispersed in the air, the threshold of detectable leakage is raised to approximately 0.013 gpm (50 cc/minute). For cases where baseline reactor coolant leakage falls within the detectable limits of the particulate radioactivity monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the baseline value.

2.1.5 Containment Atmosphere Gaseous Radioactivity Monitors

The LAR stated that the CNP, Unit Nos. 1 and 2, containment atmosphere gaseous radioactivity monitors are less sensitive (threshold at 10^{-6} μCi/cc) than the containment atmosphere particulate radioactivity monitors and would function only in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects. The containment atmosphere gaseous radioactivity monitors provide indication, record, and alarm functions in the Control Room, but no control function. The system is not safety-related, has no safety or accident mitigation functions, or nonsafety-related/diverse actuation functions.

Assuming a reactor coolant gaseous activity of 0.3 μCi/cc, the occurrence of a leak of 2 to 4 gpm would double the background activity (predominantly argon-41) in less than 1 hour. In these circumstances this instrument is a useful backup to the particulate radioactivity monitor.

According to the existing TS Bases, the gaseous containment atmosphere radioactivity monitor is operable when it is capable of detecting a 1 gpm increase in unidentified leakage, within 4 hours given an RCS activity equivalent to that assumed in the design calculations for the monitors.

2.1.6 Containment Humidity Monitors

For CNP, Unit Nos. 1 and 2, the containment humidity monitor provides an indirect indication of leakage into containment. This instrumentation does not have the sensitivity of the particulate or gaseous radioactivity monitors, but detects vapor originating from all sources: the reactor coolant, the steam, and the feedwater systems. The containment humidity detection system components are not safety-related and provide indication and record functions only. No control or alarm functions are provided. The system has no safety or accident mitigation functions and no nonsafety-related/diverse actuation functions.

2.1.7 Containment Sump Monitors

The LAR stated that for CNP, Unit Nos. 1 and 2, an increase in the amount of coolant make-up water, which is required to maintain normal level in the pressurizer, will be indicated by an increase in charging flow or change in volume control tank level. Gross RCS leakage will be indicated by a rise in normal containment sump level and periodic operation of containment sump pumps. A run time meter is provided to monitor the frequency of operation and running time of each containment sump pump. The containment sumps, consisting of three sumps; the lower containment sump, the reactor cavity sump, and the pipe tunnel sump, are used to collect unidentified leakage.

The LAR supplement dated August 29, 2019, stated that the liquid inventory equipment (charging and letdown flow, as well as volume control tank level) provides control, indication, record, and alarm functions in the control rooms. The three sumps provide high sump level alarm function only in the control rooms. Each of the three sumps contain two sump pumps. The monitor for the containment sump detects the operating frequency of the installed sump pumps via pump run time meters located on the containment auxiliaries subpanel in the auxiliary building. Operators monitor the six-containment sump pump run time meters once per 12-hour shift.

2.2 Description of Licensees Proposed Change

2.2.1 LBB Analysis

LBB is used to demonstrate that a small leaking flaw grows slowly, and the limited leakage would be detected by the RCS leakage detection system allowing for operator response to shut down the plant to repair the degraded pipe before the pipe ruptures.

The LBB analysis provided in the LAR, as supplemented, pertains to the accumulator, RHR and SI lines at CNP, Unit No. 2, as further described in Section 3.0 of this safety evaluation (SE). Specific piping configurations are presented in Chapter 3 of:

- WCAP-18295, Revision 0, "Technical Justification for Eliminating Accumulator Line Rupture as the Structural Design Basis for D.C. Cook Units 1 and 2, Using Leak-Before-Break Methodology" (ADAMS Accession No. ML18334A268)

- WCAP-18302, Revision 0, "Technical Justification for Eliminating Residual Heat Removal Line Rupture as the Structural Design Basis for D.C. Cook Units 1 and 2, Using Leak-Before-Break Methodology" (ADAMS Accession No. ML18334A269)

- WCAP-18309, Revision 0, "Technical Justification for Eliminating Safety Injection Line Rupture as the Structural Design Basis for D.C. Cook Units 1 and 2, using Leak-Before-Break Methodology" (ADAMS Accession No. ML18334A270)

2.2.1 Deletion of Containment Humidity Monitor for TSs

The LAR proposed to change TS 3.4.15 "RCS Leakage Detection Instrumentation," to delete the containment humidity monitors from the limiting condition for operation (LCO) for CNP, Unit Nos. 1 and 2. The LAR also proposed to revise the CNP, Unit No. 2, TS 3.4.15 to be consistent with the CNP, Unit No. 1, TS 3.4.15. The specific TS changes proposed in the LAR are:

The changes for CNP, Unit No. 1, TS 3.4.15, Actions are as follows:

- LCO 3.4.15c. – Deleted "containment humidity or."

- Condition C – "Required containment humidity or" was deleted. Added an AND to the new Required Action C.2 to "Restore containment atmosphere gaseous radioactivity monitor to Operable status," within "30 days." Required Actions C.1 and C.2 are renumbered as C.1.1 and C.1.2, respectively.

- Action D - The Note was deleted from the Condition. Added "(s)" to monitor.

- sump monitor inoperable in the Condition.

- Action E – The Action was deleted in its entirety.

- Action F - Renumbered to reflect the deleted Action E and deleted "E" from the Condition.

- Action G - Renumbered to reflect the deleted Action E. Rephrased the Condition from "LCO 3.4.15.a, b, and c not met" to "All required monitors inoperable."

- Surveillance Requirement (SR) 3.4.15.5 – Deleted surveillance requirement.

The change for CNP, Unit No. 2, TS 3.4.13, LCO 3.4.13b, is to change "1" to "0.8."

The changes for CNP, Unit No. 2, TS 3.4.15, Actions are as follows:

- LCO 3.4.15b. – Added "particulate" and deleted "(gaseous or particulate)."

- LCO 3.4.15c. – Deleted "humidity" and added "atmosphere gaseous radioactivity."

- Action B – Deleted "Required" from the Condition and Required Action B.2, added "particulate" to the Condition, and "24 hours" was changed to "12 hours" in the Completion Time.

- Action C - Deleted "humidity" from the Condition and replaced it with "atmosphere gaseous radioactivity." Added an AND to the Required Action to restore containment atmosphere gaseous radioactivity monitor to OPERABLE status with a Completion Time of 30 days.

- Action D - Deleted the Note from the Condition. Deleted "humidity" from the Condition and replaced it with "atmosphere particulate radioactivity." Deleted "humidity" from the Required Action D.2.2 and replaced with "atmosphere particulate radioactivity."

- Action E - Action was deleted in its entirety.

- Action F - Renumbered to reflect the deleted Action E and deleted "E" from the Condition.

- Action G - Renumbered to reflect the deleted Action E. Rephrased the Condition from "LCO 3.4.15.a, b, and c not met" to "All required monitors inoperable."

- SR 3.4.15.5 – Deleted surveillance requirement.

2.3 <u>Description of Regulatory Requirements</u>

Regulation 10 CFR 50.36, "Technical specifications," states:

> (c) Technical specifications will include items in the following categories:
>
> ...
>
> (2) Limiting conditions for operation. (i) Limiting condition for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

The applicable regulatory requirement for submitting the LBB evaluation to exclude the dynamic effects associated with postulated pipe ruptures from the design basis is specified in 10 CFR 50, Appendix A, General Design Criterion (GDC 4 "Environmental and Dynamic Effects Design Bases").

Regulation 10 CFR 50, Appendix A, GDC 4:

> Structures, systems, and components [SCCs] important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents [LOCA]. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

2.4 Description of Plant-Specific Design Criteria

The construction permits for CNP, Unit Nos. 1 and 2, were issued and the majority of construction was completed prior to issuance of 10 CFR 50, Appendix A, in 1971 by the Atomic Energy Commission (AEC). CNP, Unit Nos. 1 and 2, were designed and constructed to comply with the AEC GDC as proposed on July 10, 1967. The application of the AEC proposed GDC to CNP, Unit Nos. 1 and 2, is contained in the Updated Final Safety Analysis Report (UFSAR), Section 1.4 "Plant Specific Design Criteria (PSDC)."

The CNP, Unit Nos. 1 and 2, UFSAR states the requirement for PSDC, Criterion 16, "Monitoring Reactor Coolant Leakage" is that "[m]eans shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary." The CNP, Unit Nos. 1 and 2, USFAR further states regarding PSDC 16 that "[t]he basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, noble gas activity, humidity and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump."

The CNP, Unit Nos. 1 and 2, UFSAR states the requirement for PSDC, Criterion 33, "Reactor Coolant Pressure Boundary Capability" is that:

> The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The CNP, Unit Nos. 1 and 2, USFAR further states regarding PSDC, Criterion 33, that "[t]he reactor coolant pressure boundary is designed to be capable of accommodating, without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection."

The CNP, Unit Nos. 1 and 2, UFSAR states the requirement for PSDC, Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure," is that:

> The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The CNP, Unit Nos. 1 and 2, USFAR further states regarding PSDC, Criterion 34, that

> Protection against non-ductile failure has been provided by conformance with Section III of the ASME Boiler and Pressure Vessel Code fracture toughness rules as implemented by Code Case 1514, wherever possible. Conservative estimates of the pertinent material toughness properties were made in cases where it was not possible to run the prescribed tests.
>
> Pressure containing components of the RCS are designed, fabricated, inspected and tested in conformance with the applicable codes.

2.5 Description of Guidance

Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (ADAMS Accession No. ML003740113), states, in part:

> At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be selected from the following:
>
> a. Monitoring of condensate flow rate from air coolers,
> b. Monitoring of airborne gaseous radioactivity
>
> Humidity, temperature or pressure monitoring of the containment atmosphere should be considered as alarms or indirect indication of leakage to the containment.

RG 1.45 also states that nuclear power plants may be operating at the time an earthquake occurs and may continue to operate after earthquakes, it is prudent to require the leakage detection systems to function under the same conditions. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it would be important for the operator to assess the

condition within the containment quickly. The proper functioning of at least one leakage detection system would assist in evaluating the seriousness of the condition within the containment in the event leakage has developed in the reactor coolant pressure boundary (RCPB). The airborne particulate radioactivity monitoring equipment has the desirable sensitivity to indicate RCPB leakage, and it should be included for all plants. Components for the airborne particulate radioactivity equipment should be qualified to function through the SSE. The LAR stated that while CNP is not committed to RG 1.45, the requirements of RG 1.45 were followed to the extent practical.

NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 3.6.3, "Leak-Before-Break Evaluation Procedures," Revision 1 (ADAMS Accession No. ML063600396), provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for LBB.

NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of potential for Pipe Breaks," (ADAMS Accession No. ML11343A034) provides guidance for LBB analyses.

SRP, Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection" (ADAMS Accession No. ML070610277), states, in part, that:

> The RCPB leakage detection system reliably monitors reactor coolant leakage from RCPB components by combinations of atmospheric particulate monitors, radio-gas monitors, and level, pressure, humidity, and temperature indicators.

NUREG-1431, "Standard Technical Specification (STS): Westinghouse Plants - Bases," Revision 4, Volume 2 (ADAMS Accession No. ML12100A228), B 3.4.15, "RCS Leakage Detection Instrumentation," states, in part, that

> Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.
>
> Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers][1]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.
>
> [An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.][1]

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1 [ ] Bracket text within STS is plant specific or shows differences in design/licensing basis.

NUREG-1061, "Report of the U.S Nuclear Regulatory Commission Piping, Review Committee, Evaluation of Potential for Pipe Breaks," Volume 3, (ADAMS Accession No. ML093170485) provides a methodology that the NRC accepts for LBB submittals. The LBB approach described applies the fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience double-ended ruptures or their equivalent in longitudinal or diagonal splits. NUREG-1061, Volume 3, also provides a step by step approach to performing LBB analysis. The LAR stated that the guidance of NUREG-1061, Volume 3, was used in performing the LBB analyses.

3.0 TECHNICAL EVALUATION

3.1 LBB Analysis

The NRC staff followed the guidance in SRP 3.6.3, Revision 1 to review the LBB analysis. The NRC staff reviewed the segments of piping lines for the LBB analysis (i.e., scope of LBB analysis), as documented in Section 3.1 below. The NRC staff also evaluated whether the subject piping lines satisfy the screening criteria for various degradation mechanisms, as documented in Section 3.2 of this SE. The NRC staff further reviewed the limit load analysis of the subject piping, as discussed in Section 3.3. In addition, the NRC staff evaluated the capability of the RCS leakage detection systems as addressed in Section 3.4.

3.1.1 Scope of the LBB Analysis

The LAR, as supplemented, stated that the LBB analysis pertains to the accumulator, RHR and SI lines at CNP, Unit No. 2, as further described below. Specific piping configurations are presented in Chapter 3 of WCAP-18295, WCAP-18302, and WCAP-18309, reports as supplemented by the letter dated August 22, 2019.

Accumulator Lines

The accumulator lines are attached to each of the four primary RCL cold leg pipes. The accumulator lines allow for injection from the accumulator tanks into the reactor vessel, via the cold leg piping, to provide emergency core cooling. For each cold leg, the scope of the LBB analysis is the accumulator piping starting at the cold leg up to, and including, the first check valve. The LAR, as supplemented, states that the material for fabrication of the subject piping is A376, TP316 stainless steel for the pipe and A403, WP316, stainless steel for the fittings.

RHR Lines

The RHR lines evaluated in the LBB analysis are certain portions of the RHR suction line (attached to the hot leg) and RHR return lines (attached to the cold leg). The scope of the LBB analysis is piping starting at the loop 2 hot leg up to, and including, the first isolation valve; piping starting at loop 2 cold leg up to, and including, the first check valve; and piping starting at loop 3 cold leg up to, and including, the first check valve. The LAR, as supplemented, states that the material for fabrication of the subject piping is A376, TP316 stainless steel for the pipe and A403, WP316 stainless steel for the fittings.

SI Lines

The SI lines in the LBB analysis are associated with both the RCL hot leg piping and cold leg piping. For each hot leg, the scope of the LBB analysis is the piping starting at the hot leg up to,

and including, the first check valve. For each cold leg, the scope of LBB analysis is the piping portion starting at the cold leg up to, and including, the first check valve. The LAR, as supplemented, states that the material for fabrication of the subject piping is A376, TP316 stainless steel for the pipe and A403, WP316 stainless steel for the fittings.

In addition, Table 1 in Enclosure 2 of the November 20, 2018, letter provides the various operating temperatures and pressures for the subject piping lines as input parameters for the LBB analysis. The NRC staff finds that the LAR, as supplemented, has clearly identified the analyzable portions of the accumulator, RHR and SI lines for the LBB analysis and, therefore, the scope of the LBB analysis is identified appropriately.

3.1.2 Screening Based on Applicable Degradation Mechanisms

SRP, Section 3.6.3.III, specifies that active degradation should not be a potential source to cause pipe rupture in the application of LBB to the subject piping (e.g., degradation due to stress corrosion cracking, fatigue, corrosion, wall thinning, creep or brittle cleavage-type failure) or unanticipated loading conditions (e.g., water hammer).

The NRC staff evaluated the LBB analysis in accordance with the degradation screening criteria of SRP, Section 3.6.3.III.

3.1.2.1 Stress Corrosion Cracking

The following elements of a reactor coolant environment are known to increase the susceptibility of austenitic stainless steel to stress corrosion cracking (SCC): oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides and sulfites). During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to SCC. The water chemistry control standards are also included in the plant operating procedures. Therefore, during plant operation, the likelihood of SCC is minimized.

The Westinghouse RCS primary loops and connected Class 1 piping have an operating history that demonstrates the inherent operating stability characteristics of the design. The operating experience also confirms that the RCS piping is resistant to intergranular stress corrosion cracking (IGSCC). In comparison, the operating experience has shown that primary water stress corrosion cracking (PWSCC) has occurred in nickel-based Alloy 82/182 dissimilar metal butt welds in pressurized water reactor (PWR) coolant environment. The LAR, as supplemented, confirmed that this material susceptible to PWSCC is not used in the subject portions of the accumulator, RHR and SI lines of CNP, Unit No. 2. Therefore, the NRC staff finds that SCC is not an active degradation mechanism for the subject piping.

3.1.2.2 Low Cycle and High Cycle Fatigue

The LAR, as supplemented, stated that the subject piping was constructed in accordance with the 1967 Edition of the American National Standards Institute (ANSI) B31.1 Code. The ANSI B31.1 Code does not specify an explicit low cycle fatigue analysis for the piping. Instead of an explicit fatigue analysis, the subject piping complies with the provision that an adequate stress range reduction factor be applied to the allowable stress to address fatigue effect from full temperature cycles for thermal expansion stress evaluation. The LAR, as supplemented, stated that the stress range reduction factor is 1.0 (i.e., no reduction) for equivalent full temperature

cycles less than 7000. For CNP, Unit No. 2, the equivalent full temperature cycles for the applicable design transients are less than 7000, so no reduction is required for the stress range.

The LAR, as supplemented, stated that pump vibrations during operation would result in high cycle fatigue loads in the piping system. Field measurements on typical PWR plant indicate vibration stress amplitudes less than 1 ksi (kilopounds per square inch). When translated to the accumulator, RHR and SI lines connected to the RCS primary loops, these stresses would be even lower, well below the fatigue endurance limit for the materials of the subject piping and would not result in fatigue crack growth.

The NRC staff finds that low cycle and high cycle fatigue is not a potential source of pipe rupture for the subject piping because the licensee has shown that the low cycle fatigue meets the allowable cycle limit in accordance with ANSI B31.1 Code and the high cycle fatigue due to pump vibration is insignificant.

The LAR, as supplemented, stated that thermal stratification occurs when conditions permit hot and cold layers of water to exist simultaneously in a horizontal pipe. This condition can result in significant thermal loadings due to the high fluid-temperature differentials. The LAR, as supplemented, indicated that changes in the stratification state result in thermal cycling, which can cause fatigue damage. The issue of RHR valve leakage described in NRC Bulletin 88-08, Supplement 3, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," April 11, 1989, identifies a scenario that could lead to stratification conditions which would challenge piping integrity. The LAR, as supplemented, also stated that WCAP-12143, "Report on Evaluation of Auxiliary Piping attached to the Reactor Coolant System per NRC Bulletin 88-08 for American Electric Power Service Corporation D. C. Cook Units 1 and 2," April 1989, identifies three auxiliary piping systems that are susceptible to the valve leakage and the potential stratification. The LAR, as supplemented, noted that the RHR lines are not identified as one of the three susceptible lines.

The NRC staff finds that the subject RHR piping is not susceptible to thermal stratification and, therefore, fatigue due to thermal stratification is not an active degradation mechanism for the RHR lines.

The licensee also documented a detailed fatigue crack growth (FCG) analysis for the subject piping in WCAP-18394, Revision 1. The LAR, as supplemented, stated that even though a FCG analysis is not an explicit requirement for LBB analyses that is specified in SRP 3.6.3, the FCG analysis for the subject piping was performed to confirm that degradation related to cyclic fatigue is insignificant and small postulated surface cracks will not become through-wall flaws during the entire operating period of the piping system (i.e., 60 years), and a through-wall crack is stable under the cyclic loadings of plant operation so that a leakage crack can be identified and addressed in a timely manner prior to growing to a critical flaw size.

The LBB analysis indicates that, the FCG during the 60 years of operation is very small compared to the initial crack sizes and the postulated surface cracks will not grow to though-wall cracks. Therefore, the NRC staff finds the FCG analysis results support the conclusion that potential FCG would not affect the structural integrity of the subject piping and would be insignificant between the time when leakage reaches 8 gpm and the time that the plant would be shut down to perform corrective actions.

The LAR, as supplemented, stated that the LBB analysis for the subject piping considers time dependencies since the FCG analysis involves a time-limited assumption. The licensee

evaluated this time dependency for the license renewal term (60 years of operation). Specifically, the LAR, as supplemented, stated that the FCG analysis for each subject piping system bounds the transient projections for the 60 years of operation. The NRC staff finds the evaluation acceptable because the LBB analysis has adequately performed the FCG analysis and confirmed the validity for the renewed license term.

3.1.2.3 Brittle Fracture and Cleavage-Type Failure

The LAR, as supplemented, stated that brittle fracture for stainless steel material occurs when the operating temperature is approximately minus 200 degree Fahrenheit (°F). The operating temperature of the accumulator, RHR and SI lines is higher than 120 °F and, therefore, brittle fracture is not a concern for potential failure of these lines. The licensee also stated that brittle cleavage-type failures are not a concern based on the operating temperatures and the stainless steel material used in the subject piping lines.

The NRC staff finds that brittle fracture is not relevant for the subject piping because the operating temperature of the subject piping is outside the temperature range that would cause brittle fracture. In addition, the NRC staff finds that the subject piping fabricated with stainless steel is not subject to cleavage-type brittle fracture at the operating temperature of the RCS and, therefore, cleavage failure is not relevant for the subject piping.

3.1.2.4 Wall Thinning

The LAR, as supplemented, stated that wall thinning by erosion and erosion-corrosion effects would not occur in the accumulator, RHR, and SI piping because of the low velocity of the flow, typically less than 1.0 ft/sec (feet/second) and the stainless steel material used for the piping, is highly resistant to these degradation mechanisms. The LAR, as supplemented, stated that the wall thinning is related to the high water velocity and, therefore, would not affect the subject piping at CNP, Unit No. 2.

The NRC staff finds that because of the low flow velocity in the subject piping and piping fabrication material, wall thinning is not an active degradation mechanism for the subject piping. The NRC staff also notes that industry operating experience has not shown wall thinning in the subject piping lines.

3.1.2.5 Creep Failure

The LAR, as supplemented, stated that the maximum normal operating temperature of the accumulator piping is about 549 °F. The maximum normal operating temperatures of the RHR and SI piping lines are approximately 617 °F and 618 °F, respectively. These temperatures are well below the temperature that would cause any creep damage in stainless steel piping.

The NRC staff recognizes that the operating temperature of the subject piping is well below the temperature (800 °F) that would cause creep damage in stainless steel material; therefore, creep damage is not an active degradation mechanism.

3.1.2.6 Water Hammer

The LAR, as supplemented, stated that the potential for water hammer in the accumulator, RHR, and SI lines is low because they are designed and operated to preclude the voiding condition in normally water-filled lines. The LAR, as supplemented, stated that to ensure

dynamic system stability, reactor coolant conditions are stringently controlled. The coolant temperature during normal operation is maintained within a narrow range by the control rod positions. The pressure is also controlled within a narrow range for steady-state conditions by the pressurizer heaters and pressurizer spray. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters (namely system resistance and the reactor coolant pump characteristics) are controlled in the design process.

Additionally, the LAR, as supplemented, stated that Westinghouse performed instrumentation and monitoring activities to verify the flow and vibration characteristics of the RCS and the connected auxiliary lines. Preoperational testing and operating experience have verified that the Westinghouse approach is effective.

The NRC staff finds that the subject lines are designed and operated to minimize the water hammer event. The water hammer event would not likely occur in a water solid pipe. Therefore, the NRC staff finds that water hammer is not an anticipated loading condition for the subject piping.

3.1.2.7 Screening Based on Applicable Degradation Mechanisms Conclusion

On the basis of the above screening evaluation, the NRC staff finds that accumulator, RHR and SI lines are not subject to any active degradation or unanticipated loading condition as specified in SRP 3.6.3. In addition, the LAR, as supplemented, confirmed that the inservice inspection (ISI) examination records for the subject piping show no recordable indications. Therefore, the NRC staff finds that, given the absence of active degradation or unanticipated loading conditions, the subject piping can be analyzed to determine the applicability of LBB in accordance with 10 CFR Part 50, GDC 4.

3.1.3 Limit Load Analyses

3.1.3.1 Material Properties

SRP, Section 3.6.3, Subsection III.11, specifies that the LBB analysis should identify the types of materials and materials specifications used for base metal and weldments. It also specifies that the licensee should provide the material properties including toughness, tensile data, and long-term effects such as thermal aging.

As previously discussed, the LAR, as supplemented, reported that the subject piping is fabricated with stainless steel materials. Specifically, the material for fabrication of the subject piping is A376 TP316 stainless steel for seamless pipes and A403 WP316 stainless steel for fittings. In addition, the welding processes used in the subject pipes are submerged arc welding (SAW) and shielded metal arc welding (SMAW) processes.

The LAR, as supplemented, also stated that certified material test reports (CMTRs) with mechanical properties were not readily available for the accumulator, RHR and SI piping. The LBB analysis used the code tensile properties at temperatures for the operating conditions and considered the material properties by performing linear interpolations. The LBB analysis interpolated material modulus of elasticity from ASME Code values for the operating temperatures considered. The Poisson's ratio was taken as 0.3. The yield strengths, ultimate strengths, and elastic moduli for the materials of each subject piping are provided in Table 4-1 of the corresponding WCAP report (referenced in Section 3.0 of this SE).

The NRC staff finds the LBB analysis approach acceptable because the analysis used the appropriate ASME Code material properties for the subject piping.

3.1.3.2 Load Combinations

SRP, Section 3.6.3, Subsection III.1, specifies that the LBB evaluation should use design basis loads which are based on the as-built piping configuration. As described in all three WCAP reports, the LBB analysis used the as-built piping configurations and the associated piping loads.

The LBB analysis used the absolute sum load combination (i.e., combining the individual absolute values of the loads) method to combine the deadweight, thermal expansion, pressure, SSE and seismic anchor motion (SAM) loads. The LBB analysis considered the bending and torsional moments to obtain the limiting total applied moment and calculated the applied moment based on the square root of the sum of squares of the bending and torsional moments, which is consistent with SRP, Section 3.6.3, Subsection III.11.C.v.

The NRC staff finds that the load combinations are acceptable because LBB analysis used the absolute sum load combination method; appropriately considered the maximum loads under the faulted conditions in the load combinations (including deadweight, thermal expansion, SSE, and SAM loads); the calculations of the total moment considered the bending and torsional moments; and these methods are consistent with the guidance in SRP Section 3.6.3, Subsection III.11.C.v.

3.1.3.3 Leakage Crack Size Calculation

SRP, Section 3.6.3, Subsection III.11.C.iii, specifies that the estimated leak rate from a leakage crack in the piping, during normal operation ,should be 10 times greater than the minimum leak rate that the RCS leakage detection systems can detect. Based on a detectable leak rate of 0.8 gpm, the LBB analysis calculated crack sizes that correspond to a leak rate of 8 gpm at the critical locations.

The NRC staff finds that the size of the leakage crack is large enough so that leakage from the flaw during normal operation would be 10 times greater than the minimum leakage that the detection system is capable of sensing. The NRC staff finds that the LAR, as supplemented, is consistent with the guidance in SRP, Section 3.6.3, Subsection III.11.C.iii.

3.1.3.4 Crack Stability Analysis

SRP, Section 3.6.3, Subsection III.11.C, specifies how the critical and leakage crack sizes should be calculated. SRP Section 3.6.3, Subsection III.11.C.ii specifies that the pipe location with the worst material property should be used. SRP, Section 3.6.3, Subsection III.11.C.iv, specifies that a crack stability analysis should be performed to demonstrate that the leakage crack size will not become unstable by comparing the leakage crack size to the critical crack size.

The LAR, as supplemented, stated that the analysis determined the highest stresses and identified the corresponding weld location node for each welding process type in the accumulator lines, RHR lines and SI lines. Therefore, the critical locations are established based on the loads and piping material properties.

In addition, the LAR, as supplemented, stated that the critical crack sizes at the critical locations were derived in accordance with the guidance in SRP, Section 3.6.3, as discussed below. If the size of a postulated crack is less than the critical crack size in the corresponding pipe segment, the crack is stable and is not subject to rupture under the bounding load conditions with the highest faulted stresses. The critical crack sizes were determined in a conservative manner (e.g., using the absolute sum method in the load combinations and the lower-bound material strengths). The critical crack size is also at least twice the leakage crack size at the critical locations, thereby, demonstrating a margin of 2 between the leakage crack size and critical crack size.

The LAR, as supplemented, stated that the crack stability analysis used the limit load method to predict the critical crack size for the critical locations in the subject piping. The failure criterion was obtained by requiring equilibrium of the section containing a through-wall circumferential crack. The applied loads are calculated in consideration of internal pressure, axial force, and imposed moments. The limiting moment for the analyzed pipe is calculated based on the flow stress, axial force, pipe dimensions, and crack size and configuration. For the limit load method, the licensee also multiplied the pipe loads by the Z factor because the shop and field welds were fabricated with SMAW and SAW processes. The licensee stated that it derived Z factors for SMAW and SAW in accordance with SRP, Section 3.6.3. The crack stability analysis using the limit load method confirms the stability of the critical cracks in the critical locations.

In the evaluation of the crack stability analysis provided in the LAR, as supplemented, the NRC staff noted that the limit load method was used with relevant Z factors for the SMAW and SAW locations; a safety margin of 2 is demonstrated between the leakage crack size and the critical crack size; and that these approaches are consistent with the guidance in SRP, Section 3.6.3. The NRC staff finds that if a crack exists in the subject piping and the crack size is smaller than the critical crack size, it will be stable as discussed above. The NRC staff finds that the LAR, as supplemented, has demonstrated crack stability in the subject piping by performing the appropriate limit load analyses.

3.1.4 LBB Analysis Conclusion

On the basis of its review of the LAR, the NRC staff finds that, for the subject accumulator, RHR, and SI piping, the LBB analysis has demonstrated the following:, (1) the screening criteria of SRP, Section 3.6.3, are satisfied in the evaluation of applicable degradation mechanisms, (2) a margin of 10 exists between the calculated leak rate from the postulated leakage crack sizes and the RCS leakage detection system capability; (3) a margin of 2 exists between the critical crack sizes and the leakage crack sizes; (4) the leakage crack sizes were estimated based on the reasonable technical basis considering the relevant characteristics of the leakage flow; (5) the critical cracks were calculated conservatively in consideration of the material properties and bounding load conditions in the limit load analyses; (6) the LBB methods are consistent with the guidance in SRP, Section 3.6.3; and (7) the leakage and critical cracks are stable. The NRC staff finds that the LBB analysis has demonstrated that the subject accumulator, RHR, and SI, piping lines have a low probability of rupture.

Pursuant to GDC 4 of Appendix A to 10 CFR Part 50, the NRC staff concludes that the dynamic effects associated with the postulated rupture of the subject accumulator, RHR and SI piping can be excluded from the design basis of CNP, Unit No. 2.

3.2 RCS Leakage Detection System Capability

Currently CNP, Unit Nos. 1 and 2, RCS leakage into containment is detected by the following methods:

- The containment atmosphere particulate radioactivity monitor.
- The containment atmosphere gaseous radioactivity monitor.
- The containment humidity monitor.
- An increase in the amount of coolant make-up water, which is required to maintain normal level in the pressurizer, or an increase in containment sump level.

The LAR has proposed that the containment humidity monitors for CNP, Unit Nos. 1 and 2, be deleted from TSs 3.4.15.

RCS leakage detection system capability are further discussed in detail below for the proposed changes to TS 3.4.13 and TS 3.4.15.

3.2.1 NRC Staff Evaluation of Proposed Changes CNP, Unit No. 2, TS 3.4.13

The LAR proposed for CNP, Unit No. 2, to change the TS 3.4.13 requirement for unidentified leakage from 1 gpm to 0.8 gpm.

The NRC staff's evaluation of the LAR, as supplemented, request to adopt LBB for CNP, Unit No. 2, is provided in Section 3.1 of this SE. The analysis provided in the LAR, as supplemented, selected worst-case locations in each piping system and derived the critical crack size and calculated the leakage flaw size at these locations which would result in an 8.0 gpm leak (10 times the leak detection capability of 0.8 gpm). The LBB included a stability analysis to demonstrate that the calculated leakage crack size yielding 8.0 gpm are stable with a margin of at least 2.0 between the calculated crack size and the critical crack size. The LAR, as supplemented, stated that the RCS containment air particulate monitor can detect a 0.8 gpm RCS leak in 1 hour.

The LAR, as supplemented, stated that the containment monitors can detect particulate radioactivity in concentrations as low as 10^{-9} micro Curies/cc (cubic centimeter). The containment gas monitor has a threshold of 10^{-6} micro Curies/cc.

The SRP, Section 3.6.3, Revision 1, states that plant-specific leakage detection systems inside containment should be equivalent to those in RG 1.45, Revision 1, which states that plants should use leakage detection systems with a response time of no greater than 1 hour for a leakage rate of 1 gpm. The LAR, as supplemented, stated that calculations show that the containment monitors can detect a 0.8 gpm leak within 1 hour.

During the NRC's review of the CNP, Unit No. 1, LAR, the NRC staff requested additional information on if the calculation used to determine that the containment particulate monitor could detect a 0.8 gpm leak within 1 hour, used realistic reactor coolant activity, holdup time caused by piping insulation, and delay transport time of the activity to the containment monitor (response time). The NRC staff's request was to determine whether the activity released from a

postulated RCS leak was isotope activity representative of the average degassed activity from no known fuel leaks with typical noble gas activity during normal operations.

The response to the NRC staff's request was provided by letter dated May 6, 2019 (ADAMS Accession No. ML19129A126), and as stated in the LAR is consistent with the proposed TS change to CNP, Unit No. 2. The response stated that the leak detection system capability calculation considers both the effects of RCS piping insulation holdup time and monitor delay and transport time. The calculation assumed a 5-minute holdup time for the insulation since the steam leak would be expected to easily cut through the insulation. The calculation assumed a 2-minute transport and delay time since the ventilation turnover in CNP, Unit No. 1, lower primary containment would be less than 2 minutes. The response also stated that the RCS radioactivity concentrations used are from a composite of actual operating data using the lowest measured activity for each isotope. Activation products were minimized, and activity used was from the beginning of a fuel cycle. The NRC staff determined the calculations that qualified the containment monitors for the LBB analysis used realistic reactor coolant activity levels and considered the attenuating factors of piping insulation and monitor delay and transport time.

The May 6, 2019, response stated that the leakage calculation demonstrates that the alarm setpoint would be exceeded to indicate a 0.8 gpm leak to main control room operators within 1 hour, since the alarm annunciates in the CNP, Unit No. 1, main control room. Therefore, the NRC staff concludes that the CNP, Unit Nos. 1 and 2, containment monitors can detect a 0.8 gpm RCS leak within 1 hour based on the information provided in the LARs, as supplemented, thereby, meeting the requirements of RG 1.45, Revision 1.

RG 1.45, Revision 1, recommends that plants use multiple, diverse and redundant detectors at various locations in the containment, as necessary, to ensure that the transport delay of the leakage from the source to the detector will yield an acceptable overall response time. SRP, Section 3.6.3, Revision 1, states the leakage detection systems are evaluated to determine whether they're sufficiently reliable, redundant, and sensitive so that the margin on the detection of unidentified leakage exists for through-wall flaws to support the deterministic fracture mechanics evaluation.

The LAR, as supplemented, requested to modify the CNP, Unit Nos. 1 and 2, TS 3.4.15, "RCS Leakage Detection Instrumentation," to eliminate the containment humidity monitor. With the proposed change, TS 3.4.15 would continue to require one containment sump monitor in each sump, one containment atmosphere containment air particulate radioactivity monitor, and one containment atmosphere gaseous radioactivity monitor to be operable when the unit is in Modes 1, 2, 3, and 4.

The May 6, 2019, response stated that the primary containment sump has two sump pumps. The backup pump would only operate if the first pump was unable to function. The identification of an increase in unidentified leakage would be delayed by the time required for the unidentified leakage to travel to the containment sump. With a 0.8 gpm leak to the sump, the first pump would start pumping approximately 6 hours after leakage started entering the sump. If the first pump was inoperable, the second pump would start within 11.4 hours. The operators monitor pump run time instrumentation every 12 hours. The response stated that it would take no more than 36 hours for operations to realize that there is significant leakage into the sump using this backup method.

The May 6, 2019, response stated that RCS inventory balance, as required by SR 3.4.13.1, is an alternate method of detecting changes in RCS leakage. The data from the RCS inventory

balance is used to establish the unidentified leakage rate from the RCS. SR 3.4.13.1 is required to be performed every 72 hours but is performed daily by procedure after the plant reaches steady state operation. RCS inventory mass balance may also be used if the gaseous or containment monitors are inoperable instead of the backup grab samples identified in TS 3.4.13. The TSs requires performance of either SR 3.4.13.1 or grab samples of the containment atmosphere if the containment monitors are inoperable. The procedure for performing SR 3.4.13.1 specifies increased monitoring activities if a 0.1 gpm unidentified leakage is calculated over a 7-day rolling average or any calculated unidentified leakage over 9 consecutive days. Therefore, performance of TS SR 3.4.13.1 ensure that the mass balance can be used to back up the containment monitors for LBB detection.

The response also stated that a 0.8 gpm leak is significantly greater than normal operational leakage of less than 0.05 gpm. During normal plant operation the volume control tank level would decrease at greater than 2 percent per hour with a 0.8 gpm leak.

The NRC staff concluded that the likelihood of a crack growth resulting in a LOCA in the RCS piping segments associated with this LAR, prior to detection by one of the RCS detection systems discussed above, is very low. The NRC staff's conclusion is based on the diversity of the leak detection systems discussed above, the very slow rate of stable crack growth from a crack yielding 0.8 gpm, and the very low probability of an added load (e.g., earthquake), beyond the capability of the existing piping support systems.

3.2.2 NRC Staff Evaluation of Proposed Change to CNP, Unit Nos. 1 and 2, TS 3.4.15

The LAR, as supplemented, proposed deletion of the containment humidity monitor, from LCO 3.4.15c for both units. The LAR, as supplemented, also proposed to remove the parenthetical (gaseous or particulate) from TS LCO 3.4.15b for CNP, Unit No. 2, to make it consistent with the

CNP, Unit No. 1, TS requirement is one containment atmosphere particulate radioactivity monitor shall be operable.

The NRC staff guidance provided RG 1.45, Revision 0, recommends that at least three separate detection methods should be employed and two of these methods should be; sump level and flow monitoring; and airborne particulate radioactivity monitoring. RG 1.45 recommends that the third method could be either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous radioactivity. RG 1.45 further states that humidity, temperature or pressure monitoring of the containment atmosphere should be considered as alarms or indirect indication of leakage to the containment.

The guidance in NUREG-1431 does not specifically identify containment humidity monitors in LCO 3.4.15 to provide monitoring consistent with the guidance in RG 1.45. Further, NUREG-1431, STS Bases does specifically describe containment humidity monitors in B 3.4.15-1, and states:

> Other indications may be used to detect an increase in unidentified LEAKAGE; however, they are not required to be OPERABLE by this LCO. An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers][2]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

[An increase in humidity of the containment atmosphere could indicate the release of water vapor to the containment. Condensate flow from air coolers is instrumented to detect when there is an increase above the normal value by 1 gpm. The time required to detect a 1 gpm increase above the normal value varies based on environmental and system conditions and may take longer than 1 hour. This sensitivity is acceptable for containment air cooler condensate flow rate monitor OPERABILITY.][2]

Enclosure 14 of the LAR provided a historical review related to containment humidity monitoring. Actual plant data for the currently installed containment humidity monitoring instrumentation's ability to detect less than TS allowable RCS leakage was provided. Two instances were identified in which RCS leakage into containment was identified by other monitored parameters. There were no instances where humidity monitors were the only means to detect an RCS leak. The two examples provided were:

<u>2004</u>

The Unit 1 pressurizer manway experienced leakage which was identified by the containment atmosphere particulate radioactivity monitor. The recorded leakage was a maximum of 0.27 gpm. There was not a corresponding increase indicated on the containment humidity monitor.

<u>2015</u>

The Unit 1 reactor coolant pump seal experienced elevated leakage. This leakage was indicated on both Unit 1 containment atmosphere particulate radioactivity monitors. There was not a corresponding increase indicated on the containment humidity monitor.

The LAR, as supplemented, stated that the CNP, Unit Nos. 1 and 2, containment humidity monitors are obsolete and significant resources are needed to maintain this equipment, and each instance of maintenance causes dose to be received because the instruments are located in lower containments.

Based on the discussion above, the NRC staff finds that the proposed change to CNP, TS 3.4.15 meets the intent of RG 1.45 and will require three independent means for RCS leakage detection that includes; one containment sump monitor, one containment atmospheric particulate monitor, and one containment atmospheric gaseous monitor. These three means are the most direct instrumentation for RCS leakage detection, as described below.

1. The containment atmosphere particulate radioactivity monitor is the most sensitive instrument of those available for detection of RCS leakage into containment. This instrument is capable of detecting particulate radioactivity in concentrations as low as 10^{-9} microcurie/cubic centimeter (μCi/cc) of containment air. In addition, the

containment particulate radioactivity monitor is capable of detecting leaks as small as approximately 0.0013 gpm (5 cc/minute) within thirty [30] minutes after they occur. The MCR [main control room] has associated instruments, recorders, and alarms for the monitoring for system leakage.

2. The containment atmosphere gaseous radioactivity monitor is less sensitive (threshold at 10^{-6} μCi/cc) than the containment atmosphere particulate radioactivity monitor and would function only in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects. The MCR has associated instruments, recorders, and alarms for the monitoring for system leakage.

3. The containment sump monitor detects gross RCS leakage by a rise in normal containment sump level and periodic operation of containment sump pumps. A run time meter is provided to monitor the frequency of operation and running time of each containment sump pump. The MCR has associated instrument alarms for sump monitoring related to leak detection.

The CNP, Unit No. 1 and 2, containment humidity monitors are an indirect indication of leakage to containment and have been shown, based on historical data at CNP, to have limited value in identifying RCS leakage. With the deletion of the containment humidity monitor the NRC staff finds that the proposed TS requirements are consistent with the RG 1.45 because the TS will require three containment RCP leakage detection instrumentation. The three methods are containment sump monitoring, containment airborne particulate radioactivity monitoring, and containment airborne gaseous radioactivity monitoring. As described in RG 1.45, monitoring of humidity in the containment atmosphere is considered an indirect indication of leakage into containment. With the removal of the containment humidity monitor, the control room operators would continue to have containment temperature and pressure monitoring instrumentation for use as indirect indication of leakage into containment. These instances demonstrate that the remaining RCS leakage detection instrumentation is sufficient to detect significant leakage from the RCPB.

The NRC staff finds that this proposed change is consistent with NUREG-1431. The STS does not describe usage of containment humidity monitor and utilizes other means of RCS monitoring such as containment sump (level or discharge flow), containment atmospheric radioactivity (gaseous or particulate), and (if available) containment air cooler condensate flow rate monitor.

Per 10 CFR 50.36(c)(2)(ii), a TS LCO of a nuclear reactor must be established for each item meeting one or more of the four criteria in 10 CFR 50.36(c)(2)(ii)(A)-(D). However, per 10 CFR 50.36(c)(2)(iii), a licensee is not required to propose to modify TS that are included in any license issued before August 18, 1995, to satisfy the four criteria. Criterion 1 in 10 CFR 50.36(c)(2)(ii)(A) is "Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." The containment humidity monitor is not used to detect, and indicate in the control room, a significant degradation of the RCS boundary, and therefore does not meet Criterion 1. In addition, 10 CFR 50.36(c)(2)(ii)(B)-(D) (i.e. Criteria 2, 3, and 4), are not applicable to the containment humidity monitors. Criterion 2 is related to process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Criterion 3 is related to SSC that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Criterion 4 is

related to structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Therefore, the NRC staff finds that an LCO is not needed for the containment humidity monitors.

3.2.2.1 NRC Staff Evaluation of CNP, Unit No. 1, TS 3.4.15, Proposed ACTIONS Changes

For CNP, Unit No. 1, the LAR, as supplemented, proposed deletion of "Required containment humidity" from TS 3.4.15 Condition C, leaving containment atmosphere gaseous monitor. The LAR, as supplemented, added an 'AND' to the Required Action to restore containment atmosphere gaseous radioactivity monitor to OPERABLE status with a Completion Time of 30 days (proposed Required Action C.2).

Removing the containment humidity monitor from Condition C is consistent with the removal of the containment humidity monitor from the LCO statement. Adding a 30-day required restoration time brings Action C into alignment with existing Actions A and B and ensures a timely return to maintaining three separate detection methods. Editorial changes are made to renumbering existing Required Actions C.1, and C.2 to C.1.1 and C.1.2, respectively.

For TS 3.4.15 Condition D, the Note was deleted from the Condition. The Note presently states:

> Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor.

The Note is redundant once the containment humidity monitor is removed from the LCO statement.

For TS 3.4.15, Condition D and Required Action D.2.1, editorial change is made to change sump monitor to sump monitor(s).

TS 3.4.15, Condition E, is deleted in its entirety. The Required Actions contained in Action E are redundant to existing Required Action B.2 (particulate radioactivity monitor) and the proposed Required Action C.2 (gaseous radioactivity monitor).

TS 3.4.15, Condition F, is renumbered as "E" and reference to E is deleted from the Condition to reflect the deletion of Condition E.

TS 3.4.15, Condition G, is renumbered to F to reflect the deleted Condition E. The LAR also proposed to edit the Condition consistent with the wording in NUREG 1431. The revised wording does not change the intent of the Condition and eliminates the need to refer back to the LCO statement.

3.2.2.2 NRC Staff Evaluation of CNP, Unit No. 2, TS 3.4 15, Proposed ACTIONS Changes

The LAR, as supplemented, proposed to modify TS 3.4.15, Condition B, to remove "Required" and add "particulate" to the Condition and change "24 hours" to "12 hours" in the Completion Time (two places). Condition B and Required Action B.2 are modified to address only the particulate radioactivity monitor, consistent with the changes to CNP, Unit No. 2, LCO 3.4.15 b. discussed above. The change in Completion Time is due to the LBB methodology for CNP, Unit No. 2, as discussed in Enclosure 2 of the LAR, and is consistent with the information that the

NRC evaluated and found sufficient as documented in the letter dated November 8, 2000 (ADAMS Accession No. ML003767675).

The LAR, as supplemented, proposed to replace "humidity" with "atmosphere gaseous radioactivity" from TS 3.4.15, Condition C, and added "30 days" to Completion Time for Required Action C.2.

The modification to the scope of Condition C is consistent with the changes to CNP, Unit No. 2, LCO 3.4.15 c, discussed above. Adding a 30-day-required restoration time brings this Condition in alignment with Conditions A and B and ensures a timely return to maintaining three separate containment detection methods. Editorial changes are made to renumber existing Required Actions C.1, and C.2 to C.1.1 and C.1.2, respectively.

The LAR, as supplemented, proposed to remove the Note from TS 3.4.15 Condition D. The Note presently states "Only applicable when the containment atmosphere gaseous radiation monitor is the only OPERABLE monitor." The LAR, as supplemented also proposed to replace "humidity" from the Condition and replace it with "atmosphere particulate radioactivity" from TS 3.4.15 Condition D. The proposed change also Deleted "humidity" from the Required Action D.2.2 and replaced with "atmosphere particulate radioactivity." The Note is made redundant by the other changes to Condition D and by the removal of the containment humidity monitor from the LCO 3.4.15 statement. Replacing the humidity monitor with the particulate radioactivity monitor is consistent with the intent of the Condition C which is that the gaseous radioactivity monitor is the only OPERABLE monitor, with the Required Action to restore one other method of detection within the Completion Time of 7 days.

The LAR, as supplemented, proposed to delete TS 3.4.15 Condition E in its entirety. Condition E and the Required Actions contained in Action E are redundant to existing Required Action B.2 (particulate radioactivity monitor) and Required Action C.2 (gaseous radioactivity monitor), which is added by this amendment request.

TS 3.4.15, Condition F, is renumbered as "E" and reference to E is deleted from the Condition to reflect the deletion of Condition E.

The LAR, as supplemented, proposed renumbering TS 3.4.15, Condition G to Condition F, to reflect the deleted Condition E. The LAR, as supplemented, also proposed to modify this Condition by changing "LCO 3.4.15.a, b, and c not met" to "All required monitors inoperable." The Condition was rephrased to be consistent with the wording used in NUREG 1431. The revised wording does not change the intent of the Condition and eliminates the need to refer back to the LCO statement.

3.2.2.3 NRC Staff Evaluation of CNP, TS SR 3.4.15.5

The LAR, as supplemented, proposed removal of SR 3.4.15.5 which requires channel calibration of the containment humidity monitor. The deletion of this SR is to reflect removal of the containment humidity monitor from CNP, Unit No. 1 and 2, TS LCO 3.4.15 c and is, therefore, acceptable.

3.2.3 NRC Staff Technical Summary of Changes to TS 3.4.15

The proposed change will modify CNP, Unit Nos. 1 and 2, TS 3.4.15, "RCS Leakage Detection Instrumentation." The proposed change allows for removal of an instrument that is not required

in order to indicate a significant abnormal degradation of the RCPB. The containment humidity monitors do not meet the criteria for inclusion in TS found at 10 CFR 50.36(c)(2)(ii)(A)-(D).

Specifically, the NRC staff evaluated the proposed CNP, Unit Nos. 1 and 2, changes against each of the design requirements discussed in Section 2 of this SE. The NRC staff finds that the proposed changes remain consistent with PSDC 16 Monitoring Reactor Coolant Leakage.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. Since the CNP, Unit Nos. 1 and 2, containment humidity monitors are deleted, associated testing of the containment humidity monitor is no longer required in accordance with 10 CFR 50.36(c)(3).

The NRC staff finds that the occurrence of RCS leakage will continue to be monitored by the remaining required instrumentation, the atmosphere radioactive particulate and gaseous monitors and containment sump monitors. Therefore, the NRC staff finds the proposed changes to CNP, Unit Nos. 1 and 2, TS 3.4.15, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Michigan official was notified of the proposed issuance of the amendment on December 10, 2019. 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 revises SRs. 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 on March 5, 2019 (84 FR 7937), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be

conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date of issuance: January 23, 2020

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NUMBERS 349 AND 330 TO APPLY LEAK BEFORE-BREAK METHODOLOGY TO REACTOR COOLANT SYSTEM BRANCH LINES AND DELETION OF CONTAINMENT HUMIDITY MONITOR (EPID L-2018-LLA-0726) DATED JANUARY 23, 2020

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