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

August 1, 2019

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

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 346 RE: "APPROVAL OF APPLICATION OF PROPRIETARY LEAK-BEFORE-BREAK METHODOLOGY FOR REACTOR COOLANT SYSTEM SMALL DIAMETER PIPING" (EPID L-2018-LLA-0054)

Dear Mr. Gebbie:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed License Amendment No. 346 to Renewed Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant (CNP), Unit 1. The amendment consists of changes to the license in response to your application dated March 7, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18072A012), as supplemented by letters dated September 18, 2018, September 27, 2018, November 27, 2018, and May 6, 2019 (ADAMS Accession Nos. ML18274A094, ML18274A093, ML18333A032, and ML19129A126, respectively).

The amendment approves the use of a leak-before-break (LBB) methodology on designated reactor coolant system (RCS) piping segments associated with the Unit 1 accumulator, residual heat removal (RHR), and safety injection (SI) systems. Use of an approved LBB methodology, as evaluated within this Safety Evaluation (Enclosure 2), provides CNP, Unit 1, with additional design margin for future RCS piping analysis. Additionally, to support the amendment, a modification is made to RCS technical specification Section 3.4.13, "RCS Operational LEAKAGE," TS Required Action including adding requirements to meet the operational leakage limits of limiting conditions for operations 3.4.13.

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,

Russell S. Haskell, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Amendment No. 346 to DPR-58
2. Safety Evaluation

cc: ListServ

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 346
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 (the licensee) dated March 7, 2018, as supplemented by letters dated September 18, 2018, September 27, 2018, November 27, 2018, and May 6, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 346, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Lisa M. Regner, Acting Branch Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License and
Technical Specifications

Date of Issuance: August 1, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 346

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following Renewed Facility Operating License DPR-58 page with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
-3-	-3-

Replace the following Technical Specification pages with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2
3.4.13-3	-

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. 346, 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)

3.4.13 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----NOTES----- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	-----NOTES----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RE: REQUEST FOR APPROVAL OF APPLICATION OF PROPRIETARY LEAK-BEFORE-BREAK METHODOLOGY FOR REACTOR COOLANT SYSTEM SMALL DIAMETER PIPING

FOR

AMENDMENT NO. 346 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application dated March 7, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18072A012), with supplements dated September 18, 2018, September 27, 2018, November 27, 2018, and May 6, 2019 (ADAMS Accession Nos. ML18274A094, ML18274A093, ML18333A032, and ML19129A126, respectively), Indiana Michigan Power Company (the licensee) submitted for the U. S. Nuclear Regulatory Commission (NRC or Commission) review and approval a license amendment request (LAR) to apply leak-before-break (LBB) methodology to portions of the accumulator, safety injection (SI), and residual heat removal (RHR) piping at Donald C. Cook Nuclear Plant (CNP), Unit 1. The enclosure to the application contains sensitive unclassified non-safeguards information and, per the licensee's request, has been withheld from public disclosure pursuant to Section 2.390, "Public inspections, exemptions, requests for withholding," of Title 10 of the *Code of Federal Regulations* (10 CFR).

The supplemental letters provided additional information that clarified the application and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 8, 2018 (83 FR 20862).

2.0 REGULATORY EVALUATION

2.1 Description of System

The LBB concept is based on calculations and experimental data demonstrating that certain pipe material has sufficient fracture toughness (ductility) to prevent a small through-wall flaw from propagating rapidly and uncontrollably to catastrophic pipe rupture and to ensure that the probability of a pipe rupture is extremely low. LBB is used to demonstrate that a small leaking flaw grows slowly, and the limited leakage would be detected by the unit's reactor coolant system (RCS) leakage detection system early on allowing for operator response to shut down the plant to repair the degraded pipe long before the pipe ruptures.

2.2 Description of Licensees Proposed Change

The proposed amendment requests NRC approval to use a leak-before-break (LBB) methodology on designated reactor coolant system (RCS) piping segments associated with the CNP, Unit 1, accumulator, residual heat removal (RHR), and safety injection (SI) systems. The licensee proposed in the LAR that removing the requirement to design against a double guillotine break would provide additional margin in future RCS piping analysis. The additional margin would be utilized as a risk mitigation technique for an upcoming CNP, Unit 1, reactor system modification should there be a piping reanalysis required to evaluate the impact of revised loss-of-coolant accident (LOCA) loads. In the LAR, the licensee stated that the margin associated with the approval of this request could eliminate future needs for additional pipe restraints or supports whose installation and maintenance could require personnel radiation exposure as well as resource expenditures to engineer, install, and maintain. Additionally, to support the use of the proposed LBB methodology, the amendment would modify technical specification (TS) 3.4.13, "RCS Operational LEAKAGE," TS Required Action including adding requirements to meet the operational leakage limits of limiting conditions for operations (LCO) 3.4.13.

2.3 Description of Regulatory Requirements

The regulation at 10 CFR, Section 50.36(a)(1), requires an applicant for an operating license to include proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must also include a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls." However, per 10 CFR 50.36(a)(1), these TS bases "shall not become part of the technical specifications."

The regulation at 10 CFR 50.36(b) requires TSs to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

As required by 10 CFR 50.36(c)(2)(i), the TSs will include limiting conditions for operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TSs to include items in the category of surveillance requirements (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.

Part 50 of 10 CFR, Appendix A, General Design Criteria (GDC) Criterion 4, "Environmental and dynamic effects design bases," requires, in part, that structures, systems, and components 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. 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.

The CNP updated final safety analysis report (UFSAR), Section 1.4, the Plant Specific Design Criteria define the principal criteria and safety objectives for the CNP design. As stated in the LAR, the following apply to the proposed amendment:

> Criterion 33, "Reactor Coolant Pressure Boundary Capability," states, 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.
>
> Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," states, 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.

2.4 Description of Regulatory Guidance

Regulatory Guide (RG) 1.45, Revision 1, "Reactor Coolant Pressure Boundary Leakage Detection Systems,"[1] provides acceptance criteria for the RCS leakage detection systems. The RCS leakage detection systems should be demonstrated to detect a certain leak rate with margins when compared to the leak rate from the leakage flaw size of the subject piping.

The NRC Standard Review Plan (SRP), NUREG-0800, Section 3.6.3, Revision 1, "Leak-Before-Break Evaluation Procedures,"[2] provides guidance on screening criteria, safety margins, and analytical methods for the piping systems to be qualified for LBB. The subject piping should satisfy the screening criteria of SRP, Section 3.6.3, Revision 1, by demonstrating that it experiences no active degradation. The fracture mechanics analysis of the subject piping should satisfy the safety margins in SRP, Section 3.6.3, Revision 1.

NUREG-1061, Volume 4, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of potential for Pipe Breaks,"[3] provides the technical basis for the LBB analyses.

3.0 TECHNICAL EVALUATION

As referenced above, the NRC staff uses the guidance in SRP, Section 3.6.3, Revision 1, to review the licensee's LBB evaluation. First, the NRC staff reviews the exact segments of piping that will be considered for the LBB application (i.e., scope of LBB application), as discussed in Section 3.1 below. Second, the NRC staff determines whether the subject piping satisfies the screening criteria with respect to various degradation mechanisms, as discussed in Section 3.2 below.

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1 ADAMS Accession No. ML073200271, dated May 2008.

2 ADAMS Accession No. ML063600396, dated March 2007.

3 ADAMS Accession No. ML11343A034, dated December 31, 1984.

Third, the NRC staff reviews the fracture mechanics analysis of the subject piping, as discussed in Section 3.3 below. Fourth, the NRC staff evaluates the RCS leakage detection system capability, as discussed in Section 3.4 below.

The licensee's LBB application is based on the following Westinghouse plant-specific topical reports:

- WCAP-18295-P/NP, 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."[4]
- WCAP-18302-P/NP, 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."[5]
- WCAP-18309-P/NP, 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."[6]

3.1 Scope of LBB Application

By licensee response to requests for additional information in letter dated May 6, 2019, the scope of this LBB application applies to the designated portions of the accumulator, RHR, and SI lines. These analyzed piping segments allow the licensee to credit the proposed LBB methodology at CNP, Unit 1. The following systems and associated piping segments were evaluated for NRC-approval as part of this LBB analysis.

3.1.1 Accumulator Lines

The accumulator lines attached to each of the four-primary reactor coolant loop (RCL) cold-leg pipes, as shown in Figures 3-1 to 3-5 of WCAP-18295-P/NP. The accumulator lines allow for injection from the accumulator tanks into the reactor vessel via the cold-leg piping to provide emergency core cooling. Each of the accumulator lines are nominal pipe size (NPS) 10-inch, Schedule 140, and only a portion of the piping evaluated for LBB extends from the cold-leg pipe through two check valves and ends at an isolation valve near the accumulator tanks.

The licensee indicated that for the accumulator lines, the LBB analysis does not address the locations after the isolation valve near the accumulator tank. The licensee further stated that any break after the isolation valve will not have any effect on the primary loop piping system because there are two check valves, and the one isolation valve will prevent the break propagation to the primary loop piping system.

The licensee stated that, per WCAP-18295-P/NP, the scope of the LBB analysis is from the cold-legs for each loop to the accumulator isolation valves 1-IMO-110, 1-IMO-120, 1-IMO-130, and 1-IMO-140. The licensee stated that per the current licensing basis, the piping attached to the RCS up to and including the first isolation valve (i.e., check valve or normally closed block valve) for systems not normally operating are considered high energy. From the licensee response dated May 6, 2019, the licensee clarified that accumulator piping downstream of the first isolation valves up to the second isolation valves and to the piping downstream of the second isolation valves will not be considered for this LBB application since the water in these

[4] (non-proprietary version) ADAMS Accession No. ML18072A013, dated January 31, 2018.

[5] (non-proprietary version) ADAMS Accession No. ML18072A015, dated January 31, 2018.

[6] (non-proprietary version) ADAMS Accession No. ML18072A018, dated January 31, 2018.

piping segments is not representative of normal operating RCS temperature and pressure conditions (not characterized as high-energy fluid systems). Therefore, the licensee revised the scope of the LBB application for the accumulator lines as follows:

(1) SI piping starting at the Loop 1 Cold-leg up to, and including, SI check valve 1-SI-170-L1.
(2) SI piping starting at the Loop 2 Cold-leg up to, and including, SI check valve 1-SI-170-L2.
(3) SI piping starting at the Loop 3 Cold-leg up to, and including, SI check valve 1-SI-170-L3.
(4) SI piping starting at the Loop 4 Cold-leg up to, and including, SI check valve 1-SI-170-L4.

3.1.2 RHR Lines

The RHR lines evaluated for this LBB application are certain portions of the suction and return lines, as shown in Figures 5-1 and 5-2 of WCAP-18302-P/NP. The RHR lines draw the reactor 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 lines. The RHR suction lines evaluated for LBB are NPS 14-inch, Schedule 160, piping segments extending from the hot-leg pipe through an isolation valve and ending at a second isolation valve. The RHR return lines are attached to the accumulator line for two of the four RCLs. The RHR return lines evaluated for LBB are NPS 8-inch, Schedule 140, piping segments starting at the connection to the accumulator line and ending at the first check valve upstream of the connection.

The licensee stated that for the RHR suction line attached to the Loop 2 hot-leg, the LBB analysis will not be performed at the locations beyond the second isolation valve away from the hot-leg. Two isolation valves will prevent the propagation of any piping breaks in the subsequent RHR piping from affecting the primary loop piping system. For the RHR return lines attached to the Loop 2 and Loop 3 accumulator lines, the LBB analysis will not be performed at the locations beyond the first check valve away from the accumulator lines. Two check valves, one on the RHR return line and one on the accumulator line, will prevent the propagation of any piping breaks in the subsequent RHR return line from affecting the primary loop piping system.

The licensee stated that, per WCAP-18302-P/NP, the scope of the LBB analysis is from the cold-legs on Loops 2 and 3 to the second RHR check valves 1-RH-133 and 1-RH-134 and from the hot-leg on Loop 2 to valve 1-ICM-129. However, per the current licensing basis, the piping attached to the RCS up to and including the first isolation valve (i.e., check valve or normally closed block valve) for systems not normally operating are considered high energy. From the licensee's response dated May 6, 2019, the licensee clarified that the RHR piping downstream of the first isolation valves up to the second isolation valves and to the piping downstream of the second isolation valves will not be considered for the LBB application since the water in these piping segments is not representative of normal operating RCS temperature and pressure conditions. Therefore, the licensee revised the scope of the LBB application for the RHR lines as follows:

(1) RHR piping starting at the Loop 2 Cold-leg up to, and including, RHR check valve 1-SI-170-L2.
(2) RHR piping starting at the Loop 3 Cold-leg up to, and including, RHR check valve 1-SI-170-L3.
(3) RHR piping starting at the Loop 2 Hot-leg up to and including isolation valve 1-IMO-128.

3.1.3 SI Lines

The SI lines evaluated tor LBB are associated with both the RCL hot-leg piping and cold-leg piping, as shown in Figures 3-1 to 3-4 of WCAP-18309-P/NP. Hot-leg SI lines are directly attached to each of the four RCL hot-leg piping segments. The hot-leg SI lines evaluated for LBB are NPS 6-inch, Schedule 160, piping segments extending from the hot-leg pipe through a check valve before transitioning to NPS 8-inch, Schedule 140, piping and ending at an isolation valve. Cold-leg SI lines are attached to the accumulator line piping and provide injection to the RCL cold-leg through the accumulator line. The cold-leg SI line evaluated for LBB is NPS 10-inch, Schedule 140, piping starting at the connection to the accumulator line transitioning to 6-inch, Schedule 160, piping and ending at the first check valve.

The licensee stated that for the cold-leg SI lines, the LBB analysis does not address the locations beyond the first check valve. The cold-leg SI line check valve, in conjunction with the 10-inch check valve on the accumulator line, provides protection against break propagation. Any break beyond the second check valve will not have any effect on the primary loop piping system. The licensee noted that similar justification is considered for the hot-leg SI lines, in that, LBB analysis will not be performed at the locations beyond the isolation valve. The check valve and isolation valve, in series on the hot-leg SI lines, provides protection against break propagation. Any break beyond the isolation valve will not have any effect on the primary loop piping system.

The licensee stated that, per WCAP-18309-P/NP, the scope of the LBB analysis is from the cold-legs for each loop to the second SI check valves 1-SI-161-L1, 1-SI-161-L2, 1-SI-161-L3, and 1-SI-161-L4. Also, per WCAP-18309-P/NP, the scope of the LBB analysis is from the hot-legs for each loop to the second SI isolation valves 1-IMO-315 and 1-IMO-325. However, per the current licensing basis, the piping attached to the RCS up to and including the first isolation valve (i.e., check valve or normally closed block valve) for systems not normally operating are considered high energy. From the licensee's response dated May 6, 2019, the licensee clarified that the piping downstream of the first isolation valves up to the second isolation valves and to the piping downstream of the second isolation valves is not considered for this LBB application since the water in this pipe segment is not representative of normal operating RCS temperature and pressure conditions. Therefore, the licensee revised the scope of the LBB application for the SI lines as follows:

(1) SI piping starting at the Loop 1 Cold-leg up to, and including, SI check valve 1-SI-170-L1.
(2) SI piping starting at the Loop 2 Cold-leg up to, and including, SI check valve 1-SI-170-L2.
(3) SI piping starting at the Loop 3 Cold-leg up to, and including, SI check valve 1-SI-170-L3.
(4) SI piping starting at the Loop 4 Cold-leg up to, and including, SI check valve 1-SI-170-L4.
(5) SI piping starting at the Loop 1 Hot-leg up to, and including, SI check valve 1-SI-158-L1.
(6) SI piping starting at the Loop 2 Hot-leg up to, and including, SI check valve 1-SI-158-L2.
(7) SI piping starting at the Loop 3 Hot-leg up to, and including, SI check valve 1-SI-158-L3.
(8) SI piping starting at the Loop 4 Hot-leg up to, and including, SI check valve 1-SI-158-L4.

In addition, the various normal operating RCS temperature and pressure parameters for the subject lines are provided in Enclosure 2, Table 1, of the LAR.

The NRC staff concludes that the licensee clearly identified the specific pipe segments in the accumulator lines, RHR lines, and SI lines that are subject to this LBB analysis. The NRC staff acknowledges that the licensee analyzed a broader scope of piping segments in the accumulator, RHR, and SI lines than what is being requested for this LBB application. The NRC staff concludes that the licensee has clearly defined the scope of the LBB application and the scope of the LBB analysis for the accumulator, RHR and SI lines.

The NRC staff concludes that the licensee may only credit this LBB analysis for the piping segments that are within the scope of this evaluation (i.e., as bounded by the piping segments identified in Sections 3.1.1 thru 3.1.3 above).

3.2 Screening Criteria for Degradation Mechanisms

The SRP, Section 3.6.3.III, specifies that the piping requested for the LBB application should not experience active degradation mechanisms such as erosion/corrosion (wall thinning), stress corrosion cracking (SCC), water hammer, creep and cleavage failure, brittle failure, cycle fatigue, thermal stratification and aging. The following summarizes the NRC staff's evaluation of the acceptability of the subject piping with respect to the degradation screening criteria specified in the SRP, Section 3.6.3.III, as it pertains to this LBB evaluation.

3.2.1 Erosion/Corrosion (Wall Thinning)

The licensee stated that wall thinning by erosion and erosion-corrosion effects should not occur in the accumulator, RHR, and SI, piping because of the low velocity, typically less than 1.0 foot per second, and the stainless-steel material which is highly resistant to these degradation mechanisms. Per NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors" (PWRs), a study on pipe cracking in PWR piping reported only two incidents of wall thinning in stainless steel pipe and these were not in the accumulator RHR or SI lines. The licensee explained that the cause of wall thinning is related to the high water velocity and is, therefore, clearly not a degradation mechanism that would affect the accumulator, RHR, and SI, piping at CNP, Unit 1.

Based on the evaluation of operating experience, the NRC staff has no concerns regarding wall thinning in the subject piping. Furthermore, the NRC staff concludes that due to the low velocity flow in the subject piping, wall thinning is not considered an active degradation mechanism at CNP, Unit 1, and, therefore, has no impact on this LBB analysis.

3.2.2 Stress Corrosion Cracking

SCC could occur in piping when the following three conditions exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. Because inherent residual stresses and certain material susceptibility exist in any stainless-steel piping, stress corrosion can be minimized by properly selecting a material resistant to SCC and preventing the occurrence of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other materials in the system, applicable American Society of Mechanical Engineers (ASME) Code rules, fracture toughness, welding, fabrication, and processing.

The licensee stated that the elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are: oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides, sulfites, and thionates). The licensee noted that strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. The licensee further noted that prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications.

The licensee stated that requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping. During plant operation, the reactor coolant water chemistry is monitored and maintained within specific limits. The licensee explained that 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 as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is maintained in the parts per billion (ppb) range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus, during plant operation, the likelihood of SCC is minimized.

Operating experience has shown that primary water stress corrosion cracking (PWSCC) has occurred in nickel-based Alloy 82/182 dissimilar metal butt welds in the PWR environment. The licensee stated that this susceptible material is not found in the accumulator, RHR, and SI lines of CNP, Unit 1.

The NRC staff recognizes that based on operating experience, PWSCC in nickel-based Alloy 82/182 dissimilar metal butt welds have been the prevailing active degradation mechanism in Class 1 piping in PWRs. However, the subject piping does not contain Alloy 82/182 dissimilar metal butt welds. The NRC staff concludes that PWSCC is not an active degradation mechanism of concern at CNP, Unit 1, and, therefore, has no impact on this LBB analysis.

3.2.3 Water hammer

The licensee indicated 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 coolant-filled lines. The licensee stated that the accumulator, RHR, and SI lines are designed for normal, upset, emergency, and faulted condition transients. The licensee further stated that the design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening are also considered in the system design. Other valve and pump actuations are relatively slow transients with no significant effect on the system dynamic loads.

The licensee 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 controlled also 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, Westinghouse has instrumented the RCS to verify the flow and vibration characteristics of the system and the connecting auxiliary lines. The licensee stated that preoperational testing and operating experience have verified that the Westinghouse approach is effective. The operating transients of the RCS primary piping and connected accumulator, RHR, and SI lines ensure that no significant water hammer can occur.

The NRC staff determines that the subject piping under review is designed and operated to minimize the water hammer event and would not likely occur in a water solid pipe. The staff concludes that water hammer is not an active degradation mechanism of concern at CNP, Unit 1, and, therefore, has no impact on this LBB analysis.

3.2.4 Creep and Cleavage Failure

The licensee stated that the maximum normal operating temperature of the accumulator piping is about 549 degrees Fahrenheit (°F). The maximum normal operating temperature of the RHR piping is about 617 °F. The maximum normal operating temperature of the SI piping is about 618 °F. These temperatures are well below the temperature that would cause any creep damage in stainless steel piping. The licensee further stated that brittle cleavage-type failures are not a concern for the operating temperatures and the material used in the stainless-steel piping of the accumulator, RHR, and SI, lines.

The NRC staff determines that the operating temperature of the subject piping is well below the temperature that would cause either (brittle) creep or cleavage damage in stainless steel material. The NRC staff concludes that (brittle) creep and cleavage damage are not active degradation mechanisms of concern at CNP, Unit 1, therefore, have no impact on this LBB analysis.

3.2.5 Brittle Fracture

The licensee stated that brittle fracture for stainless steel material occurs when the operating temperature is about minus 200 °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 these lines.

The NRC staff concludes that brittle fracture of the subject piping is not an active degradation mechanism at CNP, Unit 1, because the operating temperature of the subject piping is outside the temperature range that would cause brittle fracture, and, therefore, has no impact on this LBB analysis.

3.2.6 Low Cycle and High Cycle Fatigue

The licensee noted that the 1967 Edition of the American National Standards Institute B31.1 Code, does not contain requirements for an explicit piping low cycle fatigue analysis. The B31.1 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 evaluations. The licensee 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 1, the equivalent full temperature cycles for the applicable design transients are less than 7000, no reduction is required.

The licensee reported that pump vibrations during operation would result in high cycle fatigue loads in the piping system. During operation, an alarm signals the exceedance of the reactor coolant pump shaft vibration limits. Field vibration measurements have been made on the RCL piping in several plants during hot functional testing. The licensee noted that stresses in the elbow below the reactor coolant pump have been found analytically to be very small, between 2 and 3 kilopound per square inch (ksi) at the highest. Field measurements on typical PWR plants indicate vibration stress amplitudes less than 1 ksi. The licensee stated that when translated to the accumulator, RHR, and SI, lines connected to the RCS line, these stresses would be even lower, well below the fatigue endurance limit for the materials of the subject piping and would result in an applied stress intensity factor below the threshold for fatigue crack growth.

The NRC staff concludes that low cycle and high cycle fatigue are not active degradation mechanisms in the subject piping because the licensee has shown that its low cycle thermal fatigue and high cycle fatigue from pump vibration are below allowable levels and, therefore, have no impact on this LBB analysis.

3.2.7 Thermal Stratification

The licensee stated that thermal stratification occurs when conditions permit hot and cold layers of water to exist simultaneously in a horizontal pipe. This can result in significant thermal loadings due to the high fluid temperature differentials. The licensee noted 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," dated April 11, 1989, identifies a scenario that could lead to stratification conditions which would jeopardize piping integrity. The licensee 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 detailed in NRC Bulletin 88-08. The licensee noted that the RHR lines are not identified as one of the three susceptible lines.

The NRC staff concludes that the CNP, Unit 1, RHR piping is not identified as susceptible to thermal stratification; therefore, thermal stratification is not an active degradation mechanism of concern in the RHR piping and has no impact on this LBB analysis.

3.2.8 Thermal Aging

The licensee stated in the LAR that the accumulator, RHR, and SI piping and associated fittings are forged products which are not susceptible to toughness degradation due to thermal aging.

Following the NRC staff's review of the licensee's application, the NRC staff concludes that the susceptibility of forged products to be impacted by thermal aging degradation mechanisms does not exist on the subject piping at CNP, Unit 1, and, therefore, has no impact on this LBB analysis.

3.2.9 Conclusion

Based on the overall evaluation of the associated screening criteria as discussed above, the NRC staff determines that the analyzed piping segments of the accumulator, RHR, and SI piping at CNP, Unit 1, do not experience active degradation mechanisms. The NRC staff identified that the licensee's inservice inspection (ISI) examination records show no recordable indications in the piping segments under review for this SE. Therefore, the NRC staff concludes that, given the absence of any active degradation mechanisms and recordable piping indications, the CNP, Unit 1, LBB analysis meets the acceptance requirements for a fracture mechanics analysis to be used for the determination of the probability of pipe failure, as required by GDC 4 (discussed above in Section 2.3).

3.3 Limit Load and Fracture Mechanics Analysis

3.3.1 Material Properties

SRP, Sections 3.6.3.III.11.A and 3.6.3.III.11.B, specify that plant-specific material specifications and material properties be used.

In the LAR, the licensee stated that the material for fabrication of the accumulator lines is A376 TP316 (seamless pipe) and A403 WP316 (fittings). Similarly, the material for fabrication of the RHR and SI lines is A376 TP316 (seamless pipe) and A403 WP316 (fittings).

The licensee stated that certified material test reports with mechanical properties were not readily available for the accumulator, RHR, and SI lines. The licensee used the ASME Code mechanical properties to establish the tensile properties for the LBB analyses.

For the A376 TP316 piping and A403 WP316 fitting material types, the representative properties at operating temperatures are established from the tensile properties given by the 2007 Edition of ASME Code, Section II. The licensee obtained the code tensile properties at temperatures for the operating conditions and considered the material properties in the LBB analyses by performing linear interpolations. The licensee also interpolated material modulus of elasticity from ASME Code values for the operating temperatures considered. The Poisson's ratio[7] was taken as 0.3. Mechanical properties such as yield strength, ultimate strength, and modulus of elasticity, for the materials of each subject piping segments are provided in Table 4-1 of the corresponding technical basis WCAP report as referenced in Section 3.0 above.

The welding processes used in these pipes are submerged arc weld and shielded metal arc weld.

The NRC staff recognizes that the material properties in the certified material test report were not available for the material in the subject piping. However, as an alternative, the licensee used the appropriate material properties from the ASME Code. The NRC staff verified that the licensee correctly applied the use of the ASME Code regarding the material properties corresponding to the fabrication of the subject piping segments and is, therefore, acceptable.

3.3.2 Load Combinations

SRP, Section 3.6.3.III.1, specifies that the LBB evaluation uses design basis loads which are based on the as-built piping configuration as opposed to the design configuration. The NRC staff verified that the licensee used the as-built piping configurations and associated piping loads of the subject piping, included in the WCAP reports referenced in Section 3.0 above.

SRP, Section 3.6.3.III.11.C, specifies the application of pipe loads in deriving critical and leakage flaw sizes in the LBB evaluation. SRP, Section 3.6.3.III.11.C.v, specifies that the size of leaking cracks will not become unstable if 1.4 times the normal plus safe shutdown earthquake loads are applied. The 1.4 margin can be reduced to 1.0 if the deadweight, thermal expansion, pressure, safe shutdown earthquake (SSE), and seismic anchor motion loads, are combined based on individual absolute values.

To calculate the leakage crack size, the licensee used the algebraic sum method to combine the applied loads based on normal operating conditions (deadweight, normal thermal expansion and pressure).

To calculate the critical crack size, the licensee used loadings under the faulted conditions. As such, the licensee used the absolute sum method to combine the applied loads, which include deadweight, thermal expansion, pressure, SSE, and earthquake anchor movement. Following the SRP's guidance, the licensee applied a margin of 1 to applied loads based on the absolute sum of faulted condition load combinations. The NRC staff verified that the loads in the faulted condition are bounding.

The licensee considered the torsional moment and bending moment to obtain the limiting total applied moment. The licensee calculated the applied moment based on the square root of the sum of squares of the torsional and bending moments which is consistent with SRP, Section 3.6.3.III.11.C.v.

The NRC staff finds that the load combinations in the licensee's LBB evaluation are acceptable because the licensee has followed the load combinations to calculate the critical flaw size and leakage flaw size per SRP, Section 3.6.3. III.11.C.

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[7] Poisson's ratio is a measure of the Poisson effect, the phenomenon in which a material tends to expand in directions perpendicular to the direction of compression.

3.3.3 Leakage Crack Size Calculation

The SRP, Section 3.6.3.III.11.C.iii, specifies that the leakage crack size be sufficiently large so that the estimated leak rate during normal operation is 10 times greater than the minimum leak rate that the RCS leakage detection systems can detect.

The licensee noted that the flow of hot pressurized coolant through an opening to the outside of the subject piping at atmosphere pressure is subject to flashing which can result in choking of the exiting flow. The licensee considered the potential for choking and pressure drop due to friction in calculating the leak rate through the leakage crack.

The licensee used a relationship between the mass velocity versus stagnation enthalpy from the book M. M. El-Wakil, "Nuclear Heat Transport, International Textbook Company," New York, NY, 1971, to estimate the critical pressure for the subject piping enthalpy condition and an assumed flow. Once the critical pressure was found for a given mass flow, the licensee obtained the pressure at stagnation from the graph of the critical pressure ratio vs. length/diameter ratio. The licensee explained that for all cases considered, this method will yield the two-phase pressure drop due to momentum effects. Using the assumed flow rate, the licensee calculated the frictional pressure drop based on a friction factor and crack relative roughness.

The frictional pressure drop was then calculated for the assumed flow rate to obtain the total pressure drop from the inside of the subject piping to the atmosphere. The procedure was repeated until the absolute pressure equation was satisfied to within an acceptable tolerance leading to a flow rate value for a given crack size. The licensee stated that the leak rate was calculated with the crack opening area obtained by the method from Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, "Application of Fracture-Proof Design Methods using Tearing-Instability Theory to Nuclear Piping Postulating Circumferential Through-Wall Cracks," September 1983.

The licensee calculated leakage flaw sizes that correspond to leak rate of 8 gpm at the governing locations. The licensee stated that the RCS pressure boundary leak detection system can detect a leak rate of 0.8 gpm and meets the intent of RG 1.45, Revision 1. Thus, to satisfy the margin of 10 on the leak rate, the leakage flaw sizes are determined to yield a leak rate of 8 gpm. Once the leakage flaw size was determined, the licensee compared the critical flaw size to the leakage flaw size to verify that a margin of 2 exists as shown in Table 7-1 as specified in WCAP reports WCAP-18295-P/NP, WCAP-18302-P/NP, and WCAP-18309-P/NP.

The NRC staff determines that the licensee demonstrated that the critical crack size (length) is at least twice of the leakage crack size (length) at the worst pipe location. Therefore, the NRC staff concludes that the licensee satisfied the margin of 2 between the critical crack size and leakage crack size as specified in SRP, Section 3.6.3, Revision 1. The NRC staff further concludes that the leakage crack sizes were calculated based on a leak rate that is 10 times the leak rate that the RCS leak detection systems can detect, as specified in SRP, Section 3.6.3, and is, therefore, acceptable.

3.3.4 Crack Size Calculation

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

As stated above in Section 2.4 of this SE, SRP, Section 3.6.3, Revision 1, discusses that LBB evaluation margins are to be demonstrated for the leakage flaw at critical locations (governing locations) in the subject piping. The licensee states in the LAR that each critical location represents the highest stress for each categorization based on piping geometry, operating conditions (pressure and temperature), material properties, material of fabrication, and the highest faulted stresses at the welds, and welding process. Such critical locations are established based on the loads and the pipe material properties. The licensee further stated that it calculated the highest faulted stresses and the corresponding weld location node for each welding process type in each segment of the accumulator, RHR, and SI lines.

The licensee selected the worst locations in each of the pipe systems, derived the critical crack size at the worst locations, derived the leakage flaw size at these locations that would result in an 8 gpm leak and evaluated the stability of these flaws under various faulted conditions. The licensee performed a stability analysis to demonstrate that the postulated leakage cracks are stable with a margin of at least two between the leakage flaw size and the critical flaw size exists and a margin of 10 between the leak rate and the RCS leakage detection system capability of 0.8 gpm.

The NRC staff notes that the licensee used the limit load method to calculate the critical crack size for most of the critical pipe locations. However, for those critical locations that may not satisfy the margin of 2 in crack size based on the limit load method, the licensee used the elastic-plastic fracture mechanics method (J-integral method) to satisfy the margin of at least 2. The NRC staff notes that SRP, Section 3.6.3, permits the use of elastic-plastic fracture mechanics to calculate crack size.

The NRC staff notes that the critical flaw and leakage flaw are all circumferential oriented because the circumferential flaw, not the axial flaw, is the limiting orientation in the LBB analysis. From the above evaluation, the NRC staff concludes that the licensee used the appropriate fracture mechanics method and load combinations in accordance with SRP, Section 3.6.3. III.11.C, to obtain the critical crack sizes in each of the subject piping and is, therefore, acceptable.

3.3.5 Crack Stability Analysis

The licensee stated that determination of the conditions which lead to failure in stainless steel piping should be done with plastic fracture mechanics methodology because of the large amount of deformation accompanying fracture. The licensee used two methods to demonstrate crack stability. One method is the limit load method which is based on traditional plastic limit load concepts, but accounts for strain hardening and the presence of a flaw. The second method is the elastic-plastic fracture mechanics using the concept of the J-integral.

The licensee explained that the limit load method is based on the failure of the degraded pipe when the remaining net section of the pipe reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is called the flow stress, which is the average of the sum of yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through many experiments.

The licensee used the limit load method to predict the critical flaw size for most of the critical locations in the subject piping. The failure criterion has been obtained by requiring equilibrium of the section containing the 100 percent through wall circumferential flaw when loads are applied considering internal pressure, axial force, and imposed bending moments. The limiting moment for the degraded pipe is based on the flow stress, internal pressure, axial force, pipe size, and flaw configuration. For the limit load method, the licensee multiplied the pipe faulted loads by the Z factor because the shop and field welds used shielded metal arc weld and submerged arc weld processes. The licensee stated that it derived Z factors for shielded metal arc weld and submerged arc weld in accordance with SRP, Section 3.6.3, Revision 1. The crack stability analysis using the limit load method confirms the stability of the flaws in most of the critical locations.

However, the licensee could not satisfy the safety margin as specified in SRP, Section 3.6.3, using the limit load method for a few critical locations. Therefore, the licensee used the elastic-plastic fracture mechanic (J-integral) method to confirm crack stability for those few critical locations. Regarding the J-integral method, the licensee stated that the local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally cracks instability. The licensee further stated that the local stability will be assumed if the crack does not initiate at all. The NRC staff accepts that the initiation toughness measured in terms of J_{Ic} from a J-integral resistance curve is the material's fracture toughness parameter defining the crack initiation as discussed in NUREG-1016, Volume 3. If the applied J-integral value is less than the J_{Ic} of the material, the crack will not initiate. To demonstrate crack stability of those few critical locations, the licensee used the J-integral method to analyze through-wall circumferential cracks using the procedure in the Fracture Mechanics Handbook (Kumar, V., German, M.D. and Shih, C. P., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI Report NP-1931, Project 1237-1, Electric Power Research Institute, July 1981).

The NRC staff notes that the fracture toughness value, J_{Ic} from a J-integral resistance curve represents material resistance to crack initiation. If, for a given load, the applied J-integral value, $J_{applied}$ is less than the J_{Ic} of the material, the crack will not likely to initiate. In the supplement dated September 27, 2018, the licensee clarified that it used the minimum J_{Ic} value from the pipe base metal and weld metal as the threshold limit. The licensee demonstrated that $J_{applied}$ values at the worst locations in the subject piping are less than the minimum J_{Ic} value of the subject piping material. The NRC staff finds that the licensee demonstrated that cracks will not likely initiate in the accumulator, RHR, and SI lines. The NRC staff notes that if a flaw exists in the subject piping and the flaw size is smaller than the critical flaw, it will be stable as demonstrated in the elastic-plastic fracture mechanics analysis. Therefore, the NRC staff concludes that the licensee demonstrated crack stability in the subject piping by performing the appropriate analyses based on either the limit load method or elastic-plastic fracture mechanics.

3.3.6 Fatigue Crack Growth Analysis

Application of fracture-proof design methods using tearing-instability theory to nuclear piping postulating circumferential through-wall cracks, the licensee stated that fatigue crack growth (FCG) analysis is not a requirement for the LBB analysis because the LBB analysis is based on the postulation of a through-wall flaw, whereas FCG analysis is performed based on the surface flaw. In addition, the licensee referred to 10 CFR Part 50, "Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule," *Federal Register*, Volume 52, No. 207, Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295, which stated that, "the Commission deleted the fatigue crack growth analysis in the proposed rule. This requirement was found to be unnecessary because it was bounded by the crack stability analysis." The licensee noted that because the growth of a flaw which leaks 8 gpm would be expected to be minimal between the time that leakage reaches 8 gpm and the time that the plant would be shut down; therefore, only a limited number of cycles would be expected to occur. As such, the licensee did not perform a fatigue crack growth calculation for the auxiliary piping in the LBB submittal.

However, the NRC staff noted that in the LBB applications for the primary loop piping in 1999 and pressurizer surge line in 2000 for CNP, Units 1 and 2, the licensee (via Westinghouse) performed fatigue crack growth calculations for the primary loop piping and pressurizer surge piping as shown in WCAP-15131-P/NP, Revision 1, "Technical Justification for Eliminating large Primary Loop Pipe Rupture as the Structural Design Basis for the D.C. Cook Units 1 and 2 Nuclear Power Plants," and WCAP-15434-P/NP, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for D.C. Cook Units 1 and 2 Nuclear Power Plants," respectively.

In addition, the NRC staff notes that other licensees have performed FCG calculations as part of LBB applications at other nuclear plants in previous years. The NRC staff notes that SRP, Section 3.6.3.III.11.C.v, states, in part that the licensee is to, "...[d]emonstrate that the crack growth is stable, and the final crack size is limited such that a double-ended pipe break will not occur."

In a letter dated September 27, 2018, the licensee included FCG calculations for the subject piping as documented in WCAP-18394-P/NP, Revision 1, "Fatigue Crack Growth Evaluations of D.C. Cook Units 1 and 2 RHR, accumulator, and Safety Injection Lines Supporting Expanded Scope Leak-Before-Break," September 2018 (proprietary and non-proprietary versions).

The licensee demonstrated FCG for the subject piping based on representative FCG evaluations. The licensee stated that these representative FCG evaluations used both generic PWR piping system design and the design of an operating plant with comparable design considerations. Because these representative FCG evaluations are not plant-specific to CNP, Unit 1, the licensee reviewed the generic FCG evaluation input parameters to identify and assess differences between the representative plant design and CNP, Unit 1. The licensee compared the following factors: the piping geometry and material properties, operating temperature and pressure of the piping systems, operating transients for the design life of the plants, and piping loads experienced at the evaluated locations. The licensee stated that for each of these analysis parameters, the representative FCG evaluations are shown to be bounding or equivalent to the applicable CNP, Unit 1, piping systems.

The licensee stated in the LAR that the representative FCG evaluations are performed following the methodology of the ASME Code, Section XI, Appendix A. The FCG evaluations considered a set of initial flaw sizes which typically range from 10 percent up to 35 percent of the approximate pipe wall thickness. These ranges of initial flaw sizes are based on acceptance standards from Section XI of the ASME Code for flaw inspections and detectability. Although flaw detectability is not a specific consideration for the demonstration of LBB, this same initial flaw basis is considered due to the use of these representative FCG evaluations. The NRC staff finds that these ranges of initial flaw sizes are appropriate for demonstrating that flaw growth is stable.

The NRC staff concludes that the licensee demonstrated that postulated small surface flaws would not develop to though-wall flaws and the growth of a flaw will be very slow. The NRC staff determines that an existing flaw would not grow significantly between reaching the limiting leak rate of 8 gpm and the time that the plant would be shut down. Based on the licensee's calculations, the NRC staff concludes that fatigue crack growth will not affect the structural integrity of the subject piping at CNP, Unit 1, and, therefore, is acceptable.

3.3.7 Power Uprate

The licensee stated that the effects due to a measurement uncertainty recapture (MUR) power uprate [8] were previously evaluated for each of the subject piping systems for CNP, Unit 1, The licensee determined that the operating parameters considered in the piping stress calculations for each system were either bounded by the power uprate operating parameters or would result in a negligible impact to the pipe loading conditions. Therefore, the NRC staff concludes that the licensee appropriately accounted for additional loading on the subject piping associated with the unit's power uprate.

3.3.8 License Renewal

In Section 3.3, Enclosure 2 of the LAR, the licensee stated that the LBB analysis does not include any time dependencies and is not being used to discontinue any activities credited for age management of components in CNP, Unit 1, license renewal. The licensee further stated that there are no impacts to existing calculations associated with CNP, Unit 1, the LBB analysis does not include any time dependencies, and the LBB application would not impact CNP, Unit 1, license renewal.

The NRC approved the license renewal application for CNP, Unit 1, in 2005.[9] The staff noted that the 40-year operating license for CNP, Unit 1, expired in 2014, and that the plant is currently in the period of extend operation. The NRC notes that licensees have performed FCG assessments of RCS piping that NRC has approved for LBB application.

In a letter dated September 27, 2018, the licensee included FCG calculations for the subject piping.[10] Based on its calculations the licensee indicated that the application for LBB for the subject piping considers potential time dependencies of the LBB analyses. The licensee stated that the FCG evaluations for each subject piping system bound the transient projections for 60-years including the period of extended operation.

The NRC staff concludes that the licensee has satisfactorily addressed the validity of the LBB application during the period of extended operation by submitting appropriate FCG calculations of the subject piping.

3.4 RCS Leakage Detection System Capability

As indicated in Section 3.1 of this SE, the licensee is permitted to take advantage of the proposed LBB methodology only for the piping segments reviewed as part of this evaluation. The scope of the evaluated piping segments is discussed in Sections 3.1.1 thru 3.1.3, above.

As previously discussed, these lines are RCS piping attached up to the first check valve or closed valve and are considered high-energy lines. The NRC staff's evaluation included only the RCS piping within the revised scope of the proposed LBB analysis which is exposed to full reactor system pressure and radioactivity, at normal operating conditions of the reactor.

The licensee 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 licensee performed a stability analysis to demonstrate that the calculated leakage crack size yielding 8.0 gpm are stable with

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8 ADAMS Accession No. ML023470126, dated December 20, 2002.

9 ADAMS Accession No. ML052230442, dated July 31, 2005.

10 ADAMS Accession No. ML18274A095, dated September 2018.

a margin of at least 2.0 between the calculated crack size and the critical crack size. The licensee stated that the RCS containment air particulate detectors (APDs) can detect a 0.8 gpm RCS leak in 1 hour.

The UFSAR states that the containment APDs can detect particulate radioactivity in concentrations as low as 10^{-9} micro Curies/cc (cubic centimeter). The containment gas monitor can detect 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 licensee stated that their calculations show that the containment APDs can detect a 0.8 gpm leak within 1 hour.

During the NRC's review, the staff requested that the licensee explain the calculation used to determine containment APDs detection of a 0.8 gpm leak within 1 hour, using realistic reactor coolant activity, and whether the calculation accounted for holdup time due to piping insulation and the delay transport time of the activity to the containment APDs (detector response time). The staff's question 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. In its May 6, 2019, response,[11] the licensee stated that the leak detection system capability calculation considers both the effects of RCS piping insulation holdup time and detector 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 1, lower primary containment would be less than 2-minutes. The licensee 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 APD for the LBB analysis used realistic reactor coolant activity levels and considered the attenuating factors of piping insulation and detector delay and transport time. In a letter dated May 6, 2019, the licensee stated that the leakage calculation ensures 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 1, main control room. Therefore, the NRC staff concludes that the licensee's containment APDs can detect a 0.8 gpm RCS leak within 1 hour based on the licensee's calculation, thereby meeting the requirements of RG 1.45, Revision 1.

The RG 1.45, Revision 1, states that plants should 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 CNP, Unit 1, TS 3.4.15, "RCS Leakage Detection Instrumentation," requires, (a) one containment sump monitor in each sump, (b) one containment atmosphere containment air particulate radioactivity monitor, and (c) one containment humidity or containment atmosphere gaseous radioactivity monitor to be operable when the unit is in Modes 1, 2, 3 and 4.

11 ADAMS Accession No. ML19129A126, dated May 6, 2019.

In a letter dated May 6, 2019, the licensee discussed the capabilities of the gaseous detectors, the containment sump monitoring system, and the RCS water inventory balance as required by SR 3.4.13.1. The licensee explained the gaseous monitors will respond to 1.0 gpm leak in 4 hours. The acceptance criterion for the gas monitors is a final reading that is greater than 2 sigma above background within 4 hours after the leak is initiated which is sufficient above background to indicate a signal 95 percent of the time. The licensee stated that this evaluation represents an upscale or actual increase in activity and used realistic RCS activity levels. Therefore, the licensee stated that the gas monitors will respond to leakages in the range of 0.8 to 1.0 gpm considering normal operational RCS activity. The gaseous activity monitors are not as sensitive as the containment APDs. There are, however, other methods which can be used in conjunction with RCS leakage indicators (i.e., RCS mass balance and increase in containment sump level indications) and are available for a confirmatory analysis of RCS leakage.

Also, as part of the licensee's response (see footnote 13), the licensee stated that the CNP, Unit 1, 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 licensee 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 licensee discussed RCS inventory balance, as required by SR 3.4.13.1, as an alternate method of detecting changes in RCS leakage of less than 0.8 gpm. 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. The licensee may also use the RCS inventory mass balance if the gaseous or containment APDs are inoperable instead of the backup grab samples identified in TS 3.4.13. Plant procedures and TSs requires performance of either SR 3.4.13.1 or grab samples of the containment atmosphere if the containment APDs 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, the existing procedures used to perform TS SR 3.4.13.1 ensure that the mass balance can be used to back up the containment APDs for LBB detection.

The licensee 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 and that leakage of this magnitude would likely be detected by routine main control room control board monitoring, within a few hours.

The NRC staff concluded that the likelihood of a crack growth resulting in a loss of coolant accident in the RCS piping segments associated with this LAR, prior to detection by one of the RCS detection systems discussed above, is very low. This conclusion may be drawn from 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.

3.4.1 Summary

The NRC staff evaluated the licensee's proposed LBB analysis for CNP, Unit 1, and determined that the credited leakage monitoring capabilities are sufficiently redundant, diverse, and sensitive, to support the analysis for the applicable piping segments for the accumulator, RHR, and SI piping. The staff determined the licensee provided sufficient information demonstrating that the containment APDs are sensitive to a 0.8 gpm leak within 1 hour and provide alarm indication in the man control room. Also, CNP, Unit 1, has diverse backup methods to detect small leaks including the radioactive gas monitor in containment, sump level and sump pump run time monitoring, and sensitive RCS mass balance analysis that are performed daily at normal power conditions. Based on licensee-furnished information regarding plant operations, the main control room operators would identify a 0.8 gpm leak during routine monitoring within a few hours. Finally, the NRC staff concluded that the probability of crack growth resulting in a loss of coolant accident, prior to detection by the containment APDs or the backup detection systems, is very low. This conclusion is based on the plants design consisting of a defense-in-depth approach by having diverse and redundant leak detection systems, slow rates of stable crack growth, and the low probability of an abnormal loading event (e.g., earthquake), which would lead to catastrophic failure of the evaluated RCS piping. Therefore, the NRC staff concluded that the licensee's RCS leakage detection capabilities of the plant support the LBB fracture mechanic calculations and analysis, thereby, providing reasonable assurance of public health and safety. Furthermore, since the proposed change to CNP, Unit 1, TS 3.4.13 is based on analyses and evaluations as discussed within this SE, the staff concludes the proposed TS change meets the requirements of 10 CFR 50.36 and is, therefore, acceptable.

4.0 TECHNICAL CONCLUSION

On the basis of its review of the LBB LAR submittal for CNP, Unit 1, the NRC staff concludes that the licensee has demonstrated for the subject accumulator, RHR, and SI, piping that: (1) a margin of 10 exists between the calculated leak rate from the postulated leakage flaw sizes and the RCS leakage detection system capability; (2) a margin of 2 exists between the critical flaw sizes and the leakage flaw sizes; (3) analysis input parameters (e.g., loadings and crack morphology) are applied consistent with SRP, Section 3.6.3, Revision 1; (4) the screening criteria of SRP, Section 3.6.3, Revision 1, are satisfied; (5) the postulated cracks have been demonstrated to be stable; and (6) the RCS leakage detection systems are adequate to detect the required leak rate in the specified period of time.

Additionally, the NRC staff concludes that the licensee has demonstrated in the LBB analysis that the subject accumulator, RHR, and SI, piping lines have an extremely low probability of rupture. However, as described in Section 3.1 of this SE, the licensee has limited the scope of the LBB application to certain RCS piping segments associated with the accumulator, RHR, and SI, lines.

Pursuant to GDC 4 of Appendix A to 10 CFR Part 50, the NRC staff concludes that the licensee is permitted to exclude consideration of the dynamic effects associated with the postulated rupture of the accumulator, RHR and SI, pipe segments, as specified in Section 3.1 of this SE, from the licensing basis at CNP, Unit 1.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Michigan official was notified of the proposed issuance of the amendment on June 20, 2019. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 published a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (83 FR 20862). Accordingly, this 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 this amendment.

7.0 CONCLUSIONS

The NRC staff concludes, 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.

Principal Contributors: John Tsao, NRR
GPurciarello, NRR
MHamm, NRR

Date of issuance: August 1, 2019

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT NO. 346 RE: "APPROVAL OF APPLICATION OF PROPRIETARY LEAK-BEFORE-BREAK METHODOLOGY FOR REACTOR COOLANT SYSTEM SMALL DIAMETER PIPING" (EPID L-2018-LLA-0054) DATED AUGUST 1, 2019

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