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

November 30, 2015

Mr. Lawrence J. Weber
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 1 - ISSUANCE OF AMENDMENT REGARDING RESTORATION OF NORMAL REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE CONSISTENT WITH PREVIOUSLY LICENSED CONDITIONS (CAC NO. MF2916)

Dear Mr. Weber:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 329 to Renewed Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1, in response to your application dated October 8, 2013, as supplemented by letters dated April 29, 2014, June 5, 2014, July 3, 2014, September 30, 2014, and September 18, 2015. The amendment revises various technical specification (TS) surveillance requirements and Section 6.3.2 of the Updated Safety Analysis Report to increase normal reactor coolant system temperature and pressure consistent with previously licensed conditions.

The amendment approves changes to the Unit 1 reactor coolant system normal operating pressure and full power average temperature to 2250 pounds per square inch absolute and 571 degrees Fahrenheit, respectively. Specific changes to the TSs include a revised start time for the Containment Air Recirculation/Hydrogen Skimmer System to support best-estimate loss-of-coolant accident (BELOCA) peak cladding temperature (PCT) analysis; adoption of a modified set of accident analysis inputs for BELOCA PCT thermal conductivity degradation evaluation; and a revision to the Updated Final Safety Analysis Report to describe the containment spray system actuation delay time as an input to BELOCA PCT.

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

Sincerely,

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Amendment No. 329 to DPR-58
2. Safety Evaluation

cc w/ Enclosure: Distribution via 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 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 329
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 October 8, 2013, as supplemented by letters dated April 29, 2014, June 5, 2014, July 3, 2014, September 30, 2014, and September 18, 2015, 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. 329, 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 180 days of issuance. Implementation of the amendment shall include revision of the Updated Final Safety Analysis Report as described in the licensee's application dated October 8, 2013, and the NRC staff's safety evaluation attached to this amendment. Implementation of the amendment shall also include implementation of the analysis for containment pressure based on WCAP-17721-P, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology," and the associated NRC Safety Evaluation dated August 24, 2015, into the plant licensing basis.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed
Operating License No. DPR-58
and Technical Specifications

Date of Issuance: November 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 329

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of Renewed Facility Operating License No. DPR-58 with the attached revised page. The change area is identified by a marginal line.

REMOVE	INSERT
- 3 -	- 3 -

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The changed areas are identified by marginal lines.

REMOVE	INSERT
3.4.14-2	3.4.14-2
3.5.5-1	3.5.5-1
3.6.10-1	3.6.10-1

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. 329, 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,

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. RHR System interlock function inoperable.	C.1	Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	----------NOTE---------- Only required to be performed in MODES 1 and 2. ---------- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	In accordance with the Inservice Testing Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow resistance shall be ≥ 0.227 ft/gpm^2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow resistance not within limit.	A.1 Restore seal injection flow resistance to within limit.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	----------NOTE---------- Not required to be performed until 4 hours after the pressurizer pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig. ---------- Verify seal injection flow resistance is ≥ 0.227 ft/gpm^2.	31 days

3.6 CONTAINMENT SYSTEMS

3.6.10 Containment Air Recirculation/Hydrogen Skimmer (CEQ) System

LCO 3.6.10 Two CEQ trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CEQ train inoperable.	A.1 Restore CEQ train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.10.1	----------NOTE---------- Only required to be met in MODES 1, 2, and 3. ---------- Verify each CEQ System fan starts on an actual or simulated actuation signal, after a delay of ≥ 270 seconds and ≤ 300 seconds, and operates for ≥ 15 minutes.	92 days

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 329

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application dated October 8, 2013 (Reference (Ref). 1), as supplemented by letters dated April 29, 2014, June 5, 2014, July 3, 2014, September 30, 2014, and September 18, 2015 (Refs. 2 through 6), respectively Indiana Michigan Power Company (I&M, the licensee) requested an amendment to Renewed Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1 (CNP-1). The proposed amendment would revise various technical specification (TS) surveillance requirements (SRs) and Section 6.3.2 of the Updated Final Safety Analysis Report (UFSAR) to increase normal reactor coolant system (RCS) temperature and pressure consistent with previously licensed conditions. The proposed normal operating pressure (NOP) would increase from 2100 pounds per square inch absolute (psia) to 2250 psia, and the proposed normal full-power average temperature (NOT, T_{avg}) would increase from 556 degrees Fahrenheit (°F) to 571 °F.

Specific changes to the CNP-1 TSs would include a revised start time for the Containment Air Recirculation/Hydrogen Skimmer (CEQ) System to support best-estimate loss-of-coolant accident (BELOCA) peak cladding temperature (PCT) analysis; adoption of a modified set of accident analysis inputs for BELOCA PCT thermal conductivity degradation (TCD) evaluation; and a revision to the CNP-1 UFSAR to describe the containment spray system actuation delay time as an input to BELOCA PCT.

The supplemental letters dated April 29, 2014, June 5, 2014, July 3, 2014, September 30, 2014, and September 18, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 19, 2014 (79 FR 9495).

1.1 Background

CNP-1 was originally licensed at an RCS operating pressure of 2250 psia, a nominal core inlet temperature of 536 °F, and a nominal core outlet temperature of 599 °F. Beginning in the late 1980s, steam generator tube degradation led I&M to institute a steam generator preservation

program. This program included reductions in T_{avg} from a high end of 576.3 °F to 556 °F and pressurizer pressure from 2250 to 2100 psia. The CNP-1 steam generators were replaced in 1999, but operation continued at the reduced RCS pressure and temperature.

During its refueling outage in late 2011, the licensee's eddy current examinations of steam generator tubing showed tube wear in the vicinity of fan bar supports in the U-bend region of each steam generator. In early 2013, the CNP-1 steam generator inspection confirmed that tube wear at fan bar intersections is continuing, although the wear rate is low. The licensee attributed the tube wear to vibration in the steam generator U-bend region, due to the low steam generator pressure associated with the current RCS temperature and pressure. An I&M root cause analysis suggested that restoring the plant to roughly its original operating conditions, and increasing steam generator secondary side pressure from 675 psig to 800 psig, would resolve the vibration issue.

In the license amendment request, the licensee asserted that raising the steam generator pressure will help ensure long-term reliability of the steam generators by mitigating ongoing wear between fan bar supports and steam generator tubes in the U-bend region. The licensee determined that the desired steam pressure can be accomplished by increasing RCS pressure from 2100 psia to 2250 psia, and increasing full power average coolant temperature from 556 °F to 571 °F.

The licensee proposed the following changes:

- SR 3.4.14.1, RCS Pressure Isolation Valve (PIV) Leakage is being changed to account for the higher RCS pressure.
- SR 3.5.5.1, Seal Injection Flow is being changed to account for the higher RCS pressure.
- SR 3.6.10.1, CEQ System timing is being changed to gain accident analysis margin.
- CNP-1 UFSAR, Section 6.3.2 is being changed to acknowledge use of containment spray system timing to support achievement of accident analyses acceptance criteria.

2.0 ACCIDENT AND TRANSIENT ANALYSIS

This section of the safety evaluation addresses the postulated events of loss-of-coolant accident (LOCA) transients, non-LOCA transients, and steam generator tube rupture (SGTR).

2.1 Loss-of-Coolant Accidents

2.1.1 Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant inventory from a failure of the reactor coolant pressure boundary (RCPB) at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. A loss of significant quantities of reactor coolant prevents heat removal from the reactor core, unless the water is replenished.

The reactor protection system and emergency core cooling system (ECCS) are designed to mitigate these events.

The proposed increase in reactor coolant temperature and pressure could potentially affect the licensing basis ECCS evaluations in several ways. For example, the increased temperature and pressure will cause the initial energy present in the reactor coolant to increase. The added energy could cause the break and coolant discharge characteristics to change, which in turn would affect the coolant and fuel behavior during the blowdown phase of the LOCA. In addition, the different initial coolant conditions could have a slight, but discernible, effect on the transient reactor physics behavior.

The NRC staff reviewed (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of PCT, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection system and ECCS; and (7) operator actions.

The construction permits for CNP were issued and the majority of construction was completed prior to issuance of Appendix A, General Design Criteria (GDC), to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, in 1971 by the Atomic Energy Commission. CNP was designed and constructed to comply with the GDC as proposed on July 10, 1967. Section 1.4 of the CNP UFSAR, "Plant Specific Design Criteria" (PSDC), defines the principal criteria and safety objectives for the design of CNP. The current Appendix A of 10 CFR Part 50 GDC differ both in numbering and content from the PSDC for CNP.

The NRC's acceptance criteria are based on the following:

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance;

- Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA;

- PSDC-30, "Reactivity Holddown Capability," insofar as it requires that reactivity control systems shall be capable of making the core subcritical under credible accident conditions;

- PSDC-42, "Engineered Safety Features Components Capability," insofar as it requires that engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a LOCA; and

- PSDC-44, "Emergency Core Cooling System Capability," insofar as it requires that a system with the capability for accomplishing adequate emergency core cooling shall be

provided. The system shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function.

Specific review criteria are provided in NUREG 0800, standard review plan (SRP) Sections 6.3 and 15.6.5 (Refs. 7 and 8, respectively).

2.1.2 Technical Evaluation

The licensee's evaluation of the ECCS/LOCA analyses supporting the restoration of NOP and NOT is provided in Section 5.1 of WCAP-17662-NP (Ref. 1, Encl. 6). The licensee uses two separate ECCS evaluation models: The Automated Statistical Treatment of Uncertainty Method (ASTRUM) is used to evaluate ECCS performance for postulated large break LOCAS (LBLOCAs), and the NOTRUMP code is used to evaluated ECCS performance for postulated Small Break (SB) LOCAs.

The NRC staff notes that, although the change in RCS pressure is reflective of proposed facility operation, nothing as proposed in the license amendment request would explicitly obligate operation at an RCS nominal pressure of 2250 psia. However, completion of TS SRs 3.4.14.1 and 3.5.5.1, revised as proposed, would become impracticable if the facility were operated significantly off of the analyzed 2250 psia value. Such operation would, in addition to being unanalyzed, be inconsistent with the 10 CFR 50.36 requirements intended to ensure that facility operation is within safety limits and that the necessary level of quality and safety is maintained.

2.1.2.1 Large Break Loss-of-Coolant Accidents

2.1.2.1.1 Background

As stated above, the licensee uses the NRC-approved ASTRUM, documented in WCAP-16009-NP-A (Ref. 9), to evaluate ECCS performance. ASTRUM relies on an approach based on order statistics, in which a set number of cases with randomly varied initial conditions are analyzed using the WCOBRA/TRAC (WC/T) reactor system analysis code. The number of cases is chosen so that the highest predicted PCT within the case set becomes a predictor of the 95/95 upper tolerance limit for the PCT associated with a hypothetical population of LOCA scenarios. This result is used to show compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT.

The NRC approved ASTRUM implementation at CNP-1 in October 2008 (Ref. 10). The proposed NOP/NOT restoration was evaluated by estimating the effect the restoration would have on the results of the analysis of record (AOR). This evaluation also includes model changes and corrections to errors that have been identified and implemented in the CNP-1 LBLOCA analysis since its approval in 2008. In addition, the evaluation includes input changes that are intended to compensate for, or offset, increases in predicted PCT associated with both the NOP/NOT restoration and the post-2008 model changes and errors.

2.1.2.1.2 Technical Information Provided by the Licensee

The licensee made the following changes to its application of the ASTRUM ECCS Evaluation Model in order to account for the NOP/NOT restoration and other changes:

- Nuclear fuel TCD
- Nuclear design peaking factor changes
- WC/T and HOTSPOT data sharing
- HOTSPOT update to Version 8.0
- Hot full power nominal average reactor coolant temperature
- Hot full power RCS pressure
- Containment performance:
    - Revised LOTIC2 containment backpressure calculation
    - Maximum spray flow
    - Minimum spray flow actuation delay time
    - Minimum air recirculation fan delay time
- Accumulator temperature
- Safety injection:
    - Flow rate
    - Temperature
    - Delay time

Specific details regarding the parametric ranges assumed for the evaluation are provided in Table 5.1.1-3 of WCAP-17762-NP.

The changes were evaluated by identifying 31 WC/T cases from the original AOR that had the potential to be most adversely affected by the PCT-increasing changes contained in the above list. These 31 cases were used as a basis to execute 45 new cases, which were created using newly adjusted input parameters. From the analysis of the 45 new WC/T cases, the licensee determined a new estimated upper tolerance limit PCT that accounted for the model changes and the NOP/NOT restoration. This analytic process, including the case set selection, is a Westinghouse proprietary evaluation that has previously been applied as described in a letter from Westinghouse to the NRC dated March 7, 2012 (Ref. 11). In consideration of new peaking factor limits, the revised RCS temperature and pressure conditions, revised containment backpressure input, and revised safety injection parameters, the licensee concluded that the revised PCT was 2043 °F.

2.1.2.1.3 NRC Review of AOR and NOP/NOT Evaluation

The NRC staff reviewed the evaluation supporting the LBLOCA analysis applicability at the proposed NOP/NOT conditions. The NRC staff considered the means by which the licensee evaluated its LBLOCA analysis, the specific changes made to the application of the ASTRUM evaluation model, and the overall results. The NRC staff reviewed both the AOR and the proposed NOP/NOT evaluation to determine how the AOR accounted for operation at the reduced pressure level, and to determine whether the new NOP/NOT evaluation acceptably accounted for facility operation at the increased pressure.

In its supplement dated April 29, 2014 (Ref. 2), the licensee explained that the AOR reflects consideration of RCS nominal pressures at both 2100 psia and 2250 psia. The NRC staff determined that the NOP/NOT evaluation based on the restored initial RCS pressure of 2250 psia has a generally PCT-lowering effect. This change to the analysis eliminates excess

model conservatism and generally lowers the PCT prediction. The NRC staff determined that this is acceptable, since it accurately reflects proposed facility operation.

The NRC staff observed that the AOR PCT, as discussed in the license amendment request, was identified as 2128 °F. By contrast, the PCT contained in the ASTRUM implementation license amendment request (Ref. 12) was 2016 °F. In its supplement dated April 29, 2014, the licensee clarified that the ASTRUM implementation license amendment request had been supplemented with an evaluation of a reduction in ECCS flow, which was the cause for the apparent discrepancy.

Using the AOR as the starting point, the licensee evaluated the effect of proposed NOP/NOT operation on the predicted PCT. Since the licensee had implemented a number of significant changes and error corrections to its application of the ECCS evaluation model, the effects of these changes needed to be incorporated in the evaluation of the NOP/NOT effects. The licensee's 45-case evaluation identified those cases with the potential to exhibit the greatest effect on PCT resulting from the NOP/NOT and other model changes.

In the April 29, 2014 supplement, the licensee provided a matrix of input data from its AOR, a recent estimate of the effects of nuclear fuel TCD (the most recent quantitative evaluation of estimated PCT at CNP-1), and the present evaluation. The NRC staff verified that the licensee is implementing mild reductions in peaking factors when compared to the most recent quantitative estimation of PCT, and that RCS temperatures, which had been lowered to support the original TCD estimate, were restored to a value bounding of NOT operation. Based on its confirmatory review of the licensee's analytic inputs, the NRC staff concluded that the licensee had identified an acceptable subset of cases to determine the limiting PCT when considering previous model changes along with NOP/NOT operation. The NRC staff also concluded that the analytic inputs support the proposed NOP/NOT operation.

Since the licensee has identified the limiting WC/T cases and re-executed them using the revised NOP/NOT input parameters, the NRC staff concludes that the licensee's estimate of the effects of NOP/NOT operation is acceptable. The NRC staff also notes that this approach has been applied previously at other licensed facilities and accepted by the NRC staff.

2.1.2.1.4 Large Break Loss-of-Coolant Accident Conclusion

The NRC staff reviewed the licensee's assessment of the effect that the proposed NOP/NOT restoration would have on ECCS performance. The NRC staff determined that the licensee has acceptably evaluated the effect that the proposed change would have on the plant's compliance with the 2200 °F acceptance criterion contained in 10 CFR 50.46(b)(1). The NRC staff concludes, therefore, that the proposed change is acceptable with respect to ECCS performance.

The licensee's AOR value for maximum local and core-wide oxidation, 11.1 percent, and 0.40 percent, respectively, remain the licensing basis values. The NRC staff determined that the licensee's approach of estimating the effect of NOP/NOT restoration on the PCT was acceptable without adjustments for oxidation because there is sufficient margin to regulatory limits in both cases, because the licensee has an existing commitment to reanalyze ECCS

performance for LBLOCA, and because cladding oxidation values could reasonably be expected to decrease, since the current PCT is slightly lower than the AOR PCT.

The licensee did not evaluate the effect of the proposed NOP/NOT restoration on LBLOCA performance using an acceptable evaluation model as required by 10 CFR 50.46(a)(1)(i). Therefore, the acceptable evaluation model for LBLOCA remains ASTRUM as implemented in accordance with License Amendment No. 306 (Ref. 10). By letter dated March 19, 2012 (Ref. 13), the licensee committed to reanalyze ECCS performance to address the effects of any significant changes or errors previously reported. The commitment was changed by letter dated June 9, 2015 (Ref. 14). The updated commitment states:

> I&M will submit to the NRC for review unit-specific LBLOCA analyses that apply NRC approved methods that include the effects of fuel TCD. The date for the submittal of the analyses is projected based upon the following milestones needed to perform a revised licensing basis LBLOCA analysis with an NRC approved Emergency Core Cooling System Evaluation Model that explicitly accounts for TCD:
>
> 1) Submittal by Westinghouse, to the NRC for review and approval, of a revised fuel performance and LBLOCA Evaluation model methodologies that include the effects of TCD.
>
> 2) NRC approval of WCAP-17642-P, a fuel performance analysis methodology that includes the effects of TCD.
>
> 3) NRC approval of WCAP-16996-P, and any required Supplements thereto, a LBLOCA Evaluation Model that includes the effects of TCD and accommodates the ongoing 10 CFR 50.46(c) rulemaking process.

The NRC staff's consideration of this amendment request does not affect this commitment to reanalyze ECCS performance.

2.1.2.2 Small Break Loss-of-Coolant Accidents

The licensee presents its evaluation of the NOP/NOT restoration with respect to the SBLOCA analysis in Section 5.1.2 of WCAP-17762-NP. The evaluation is based on an SBLOCA analysis that was provided to the NRC by letter dated August 31, 2012 (Ref. 15). The evaluation provided in WCAP-17762-NP concludes that the revised SBLOCA analysis explicitly accounts for the restored NOP/NOT conditions. The NRC staff reviewed the SBLOCA analysis to verify that it satisfies applicable regulatory requirements and to confirm that it accounts for the proposed NOP/NOT operating conditions.

The licensee used the NRC-approved NOTRUMP evaluation model. The model conforms to the required and acceptable features of ECCS evaluation models set forth in Appendix K to 10 CFR Part 50. The CNP-1-specific application of NOTRUMP includes explicit representation of each reactor coolant loop. Remaining features of the evaluation are as described in WCAP-10081-A (Ref. 16). Since the licensee used an NRC-approved evaluation model that conforms to the required and acceptable features of ECCS evaluation models set forth in

Appendix K to 10 CFR Part 50, the NRC staff concluded that the licensee evaluated the NOP/NOT restoration in accordance with an acceptable evaluation model, in accordance with 10 CFR Sections 50.46(a)(1)(i) and 50.46(a)(1)(ii).

The SBLOCA analysis for CNP-1 considers a spectrum of break sizes that range from a 1.5-inch break of a reactor coolant loop pipe to an 8.75-inch break of an accumulator line. Also considered is a high pressure safety injection cross tie. The restored NOP/NOT operating conditions are explicitly modeled, meaning that the evaluation inherently addresses the proposed NOP/NOT operating conditions. The NRC staff determined that the licensee evaluated the SBLOCA impact of the proposed NOP/NOT restoration using an ECCS evaluation that considered a variety of break sizes, locations, and other properties sufficient to provide assurance that the most severe hypothetical LOCA has been calculated, as required by 10 CFR 50.46(a)(1)(i).

The NRC staff reviewed the results for the limiting event, which was the 3.25-inch break. Based on the results provided, the NRC staff determined that the PCT appeared to occur at a time when the hot node was being vapor-cooled, and the NRC staff requested additional information to confirm that additional results confirmed that the vapor cooling could reasonably terminate the PCT transient. In addition, the staff requested additional information that the licensee provided, and that the staff used, to verify that the licensee acceptably modeled loop seal clearing phenomena, that hydraulic communication between the core, upper and lower plena, and downcomer were modeled with appropriate conservatism, and that the model was depicting an acceptable hydraulic balance among reactor coolant loop, break, and safety injection flows.

2.1.2.2.1 Hot Node Vapor Cooling

The results contained in the SBLOCA analysis appeared to indicate that the hot node was not stably quenched at the time the PCT was reached. Although this is not an unacceptable result, the potential for vapor cooling at the time of PCT is an indication that the evaluation model may be calculating cooling phenomena that are beyond the capability of its computer codes to predict accurately or conservatively. Therefore, the NRC staff requested that the licensee provide additional explanation of the thermal hydraulic phenomena occurring at the hot node at the time of PCT, in order to justify the validity of its result.

The licensee's April 29, 2014 supplement, demonstrated that the calculated presence of accumulator injection fluid was contributing to cool the vapor near the hot node, which, in turn, was driving an increasing temperature difference between the fuel cladding and the core water vapor at the time of PCT. The licensee identified this large temperature difference as being a significant contributor to the termination of the cladding heatup. Since the licensee identified the causes of the end of the PCT transient, and supported its explanation with results from the analysis, the NRC staff determined that the PCT calculated from the limiting SBLOCA analysis was an acceptable result.

2.1.2.2.2 Loop Seal Clearing

The loop seal is the depressed piping region located between the steam generator and the reactor coolant pump. Loop seal clearing is the process by which sufficient vapor pressure is generated in the coolant loop in order to clear the reactor coolant loop seal of liquid water. The

presence of liquid water in the loop seals allows the buildup of hydrostatic pressure above the reactor core, which can depress the core liquid level. The clearing itself is a two-phase flow phenomenon that is very challenging for thermal hydraulic codes to predict. Therefore, prescriptive approaches that force the codes to agree with experimental data are sometimes necessary to ensure that the codes are providing results that are sufficiently conservative.

Integral effects tests have been performed to help improve understanding of loop seal clearing behavior in reactor coolant systems. One example of such tests was performed at the ROSA-IV Large Scale Test Facility and documented in scholarly journals and conference proceedings. Other tests also support the development of appropriate modeling approaches to capture loop seal clearing phenomena. An important observation from these large scale tests is that loop seal clearing behavior is significantly limited for small breaks below a threshold size.

In its April 29, 2014 supplement, the licensee stated that only the loop seals in the broken loop are permitted to clear for small break sizes. The licensee's approach ensures that the smaller breaks treat loop seal clearing in a manner that agrees with the appropriate integral effects tests. Since the licensee applies this approach to limiting loop seal clearing predictions within NOTRUMP, the NRC staff determined that the application of NOTRUMP for CNP-1 was acceptable for the proposed NOP/NOT evaluation with respect to loop seal clearing.

2.1.2.2.3 Safety Injection Flow Balancing

Based on its review of the licensee's results, the NRC staff determined that the greatest fraction of pumped safety injection flows into the broken loop. Due to the large variation in liquid flow out of the break throughout the duration of the transient, it is difficult to evaluate the broken loop flow behavior. In the April 29, 2014 supplement, the licensee provided plots of liquid and vapor flow rates at the junctions or links connecting the broken loop to pumped safety injection sources, the break, the reactor coolant pump, and the vessel, for the first 2,000 seconds of the limiting break. The plots showed that the safety injection flow into the broken loop was more than an order of magnitude less than the total break flow prior to loop seal clearing, and that after loop seal clearing, the safety injection flow remained less than one-fourth the total break flow. Plots provided by the licensee also illustrated that total liquid flow from the broken loop into the vessel inlet is very near zero. The plots indicate that, while the broken loop safety injection flow may contribute to reducing the enthalpy of vapor flowing into the vessel from the broken loop, the majority of the safety injection liquid is being spilled from the break and not pumped into the vessel.

The NRC staff considers this result acceptable because the removal of safety injection liquid from the RCS increases the severity of the system performance prediction. The opposite result, of more safety injection liquid entering the vessel through the broken loop, would improve, or lower, the PCT prediction and give cause to question the validity of the thermal-hydraulic modeling. The staff considers removing the pumped safety injection flow from the system through the broken loop to be conservative, and since the results showed this behavior, the NRC staff determined the results were acceptable.

2.1.2.2.4 Hot Leg Nozzle Gaps

Hot leg nozzle gaps, if modeled during a SBLOCA analysis, allow steam to vent from the core through the core barrel to the break in the cold leg. As opposed to venting through the higher resistance reactor coolant loop piping, direct venting through the cold leg can allow the core liquid level to increase artificially. This can non-conservatively introduce an artificial cooling mechanism into the analysis, since hydraulic forces associated with a LOCA can cause the hot leg nozzle gaps to distort and close.

In the April 29, 2014 supplement, the licensee provided a list of the flow links employed when modeling the vessel, and confirmed that the NOTRUMP model was being applied in such a way as to treat the hydraulic communication between the upper core and the break in a conservative manner. Since the hot leg nozzle gaps were modeled in a conservative fashion, the NRC staff concluded that the licensee acceptably applied the NOTRUMP evaluation model at CNP-1 for the proposed NOP/NOT evaluation with respect to modeling the hot leg nozzle gaps.

2.1.2.2.5 Small Break Loss-of-Coolant Accident Conclusion

The licensee determined that the limiting SBLOCA event occurs with a break diameter of 3.25 inches, and the PCT is 1725 °F. The local oxidation is 2.08 percent, and the core-wide oxidation is 0.30 percent. The 10 CFR 50.46(b)(1-3) acceptance criteria for these parameters are 2200 °F, 17.0 percent, and 1.0 percent, respectively. Since all results are less than their respective acceptance criteria, the NRC staff determined that they are acceptable. As described in the preceding sections, the staff also concluded, based on its review, that the licensee has demonstrated that its application of the NOTRUMP evaluation model is acceptable in accordance with 10 CFR Section 50.46(a)(1), Section 50.46(a)(2), and 10 CFR 50 Appendix K requirements. Based on these considerations, the NRC determined that the licensee has performed an acceptable evaluation of the effects of the proposed NOP/NOT operation on the SBLOCA analyses. Hence, the NRC staff concluded that the proposed NOP/NOT operation is acceptable with respect to SBLOCA.

2.1.2.3 Post-Loss-of-Coolant Accident Long-Term Core Cooling

For post-LOCA long-term core cooling, the licensee evaluated the effect that NOP/NOT operation would have on post-LOCA subcriticality, boric acid precipitation control, and long-term decay heat removal. These evaluations are used to show that the post-LOCA core conditions remain in a coolable geometry, and that decay heat can be removed from the core, in accordance with the acceptance criteria contained in 10 CFR Sections 50.46(b)(4) and 50.46(b)(5). Coolable geometry is a qualitative acceptance criterion that is also addressed, in part, by demonstrating acceptable PCT and oxidation results in the LBLOCA and SBLOCA analyses.

In its April 29, 2014 supplement, the licensee explained that its long-term cooling calculations determine that the ECCS recirculation flows are adequate by considering the safety injection sources and equipment alignments necessary to provide long-term recirculation cooling for different break locations. For instance, the licensee considered the capacity of charging flow and existing core liquid inventory following an RCS cold leg break to conclude that safety injection flow could be interrupted for the 15 minutes required for hot leg swap-over, an

evolution necessary to redirect safety injection flow from the RCS cold leg to the hot leg to prevent concentration of boric acid in the core. This, along with other examples provided by the licensee, indicated that a variety of different LOCA scenarios had been identified in the long-term cooling analyses.

The licensee stated that the CNP-1 and CNP-2 long-term cooling analyses are based on a hybrid of both units, and draw on limiting aspects of either unit to provide a bounding calculation. The licensee identified two examples that explain how this calculation is accomplished. First, the higher Unit 2 core power level is used. Second, the smaller Unit 1 RCS liquid volume is used. Based on its review of the information provided by the licensee, the NRC staff concluded that the post-LOCA long-term cooling calculation was bounding for both units.

The licensee provided information explaining how the hot leg swap-over calculations address stable core quench and account for entrainment. The licensee explained that two semi-mechanistic models were being used to model liquid entrainment in the vapor flow in the swap-over calculations. The licensee indicated that the CNP-1 calculations indicated that the boiling behavior in the core dropped below the calculated entrainment thresholds significantly before the swap-over evolution. The NRC staff reviewed the licensee's explanation, and determined that the licensee's reliance on these semi-mechanistic models supported its conclusion that the hot leg swap-over evolution would not be susceptible to excessive liquid entrainment. The NRC staff therefore concluded that the licensee had acceptably addressed this phenomenon in its swap-over calculations.

The licensee described how the post-LOCA long-term cooling calculations account for proposed NOP/NOT operation. In the boric acid precipitation calculation, for example, RCS pressure was assumed to be 2250 psia, but sensitivity studies were performed to determine that a lower RCS temperature assumed in the calculation was effectively the same as NOT. The licensee stated that ECCS design and operation are not changed by NOP/NOT restoration, and thus that ECCS flow rate assumptions remain valid.

Evaluations that the licensee performed concluded that proposed NOP/NOT operation was supported by the post-LOCA analyses in the current licensing basis. The post-LOCA environment is largely dependent on the core decay heat — strongly a function of the core power — and the safety injection and residual heat removal system capabilities. The RCS operating state points have a more significant effect on the immediate consequences of the LOCA event, which are analyzed in the large and small break LOCA analyses. Based on its review of the licensee's analyses, the NRC staff determined that the licensee has acceptably addressed the coolable geometry and long-term decay heat removal acceptance criteria set forth in 10 CFR Sections 50.46(b)(4) and 50.46(b)(5). Therefore, the NRC staff concluded that the proposed NOP/NOT restoration is acceptable with respect to post-LOCA long-term core cooling.

2.1.3 Loss-of-Coolant Accident Conclusion

Based on the licensee's acceptable large and small break LOCA evaluations, as well as the licensee's evaluation of the impact of restored NOP/NOT operation on the post-LOCA long-term core cooling, the NRC staff determined that the licensee has acceptably addressed the effect of

restored NOP/NOT operation on ECCS evaluation requirements, and that the proposed NOP/NOT restoration is acceptable with respect to ECCS performance and evaluation.

2.2 Anticipated Operation Occurrences and Non-Loss-of-Coolant Accident Events

2.2.1 Regulatory Evaluation

WCAP-17762-NP, Section 5.2.1, indicated that the only non-LOCA licensing basis event explicitly analyzed for the NOP/NOT restoration was the steam line break core response from hot full power. As such, this regulatory evaluation addresses the basis, requirements, and acceptance criteria applicable to the main steam line break.

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase, and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions; (2) methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed responses of the reactor coolant and auxiliary systems; (5) functional and operational characteristics of the reactor protection system; (6) operator actions; (7) core power excursion due to power demand created by excessive steam flow; (8) variables influencing neutronics; and (9) the results of the transient analyses.

The NRC's acceptance criteria are based on:

- PSDC-30, "Reactivity Holddown Capability," insofar as it requires that reactivity control systems shall be capable of making the core subcritical under credible accident conditions;

- PSDC-32, "Maximum Reactivity Worth of Control Rods," insofar as it requires that the reactivity control systems be designed to assure that the effects of a sudden or large change in reactivity cannot (a) rupture the RCPB or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core;

- PSDC-34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," insofar as it requires that the RCPB be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure; and

- PSDC-44, "Emergency Core Cooling System Capability," insofar as it requires that a system with the capability for accomplishing adequate emergency core cooling shall be provided.

Specific review criteria are provided in SRP Section 15.1.5 (Ref. 17).

2.2.2 Technical Evaluation

The licensee evaluated its licensing basis non-LOCA accidents and anticipated operational occurrences (AOOs). The results of the evaluation are discussed in Section 5.2, "Non-LOCA Transients," of WCAP-17762-NP. The licensee reanalyzed the hot full power steam line break and concluded remaining licensing basis events were bounded by existing analyses.

2.2.2.1 Hot Full Power Main Steam Line Break

An explicit analysis of the hot full power main steam line break accident was required because, while original licensing basis safety analyses largely assumed NOP/NOT operation, revisions to the nominal T_{avg} at rated thermal power (T") component of the overpower-delta temperature (OPΔT) setpoints crediting the reduced pressure and temperature state points had been assumed. Performance of a main steam line break analysis at full power conditions demonstrates that the OPΔT setpoint provides adequate protection to prevent exceeding fuel overpower conditions that could result in fuel melt. This analysis conservatively applies AOO acceptance criteria, meaning that the analysis is performed to show that fuel does not undergo departure from nucleate boiling (DNB), and that fuel melt is precluded. This study demonstrates adequate functional and operational characteristics of the reactor protection system.

The licensee analyzed the hot full power main steam line break using initial conditions intended to present the biggest challenge to the OPΔT trip setpoints. The licensee performed analyses to determine the largest break size that would be terminated by the OPΔT trip. Larger breaks can result in a low steam pressure condition that causes a safety injection trip instead. Smaller break sizes generally result in a less severe transient, which may not even demand a reactor trip. The licensee identified the 0.89 ft^2 break as the limiting case for the NOP/NOT analysis. Since the licensee selected initial conditions and determined the break size that would present the biggest challenge to the efficacy of the OPΔT trip setpoint, the NRC staff determined that the licensee's initial conditions were acceptable.

The licensee analyzed the event using the NRC-approved Revised Thermal Design Procedure documented in WCAP-11397-P-A (Ref. 18), which governs the selection of initial conditions and the treatment of uncertainties. The analyses were performed using the NRC-approved LOFTRAN computer code documented in WCAP-7907-P-A (Ref. 19). Further core physics and DNB analyses are performed using the Advanced Nodal Computer (ANC) code, WCAP-10965-P-A (Ref. 20) and VIPRE, WCAP-14565-P-A (Ref. 21). The ANC code is used to determine the peak linear heat rate, in order to confirm that fuel melt limits are not exceeded. The NRC staff observed that each of these methods and computer codes is approved for non-LOCA safety analysis for Westinghouse plants, and on that basis determined that the licensee's analytic methods were acceptable.

The licensee stated that the results of the analysis indicated that DNB was not calculated to occur, and that fuel melt conditions were not calculated, for the limiting break. Based on these results, the NRC staff determined that the main steam line break analyses provided an adequate demonstration of the acceptability of the OPΔT setpoints at proposed NOP/NOT conditions, and on that basis, concluded that the proposed NOP/NOT restoration was acceptable with respect to the main steam line break analysis and the OPΔT setpoint selection.

2.2.2.2 Overtemperature Delta-Temperature Setpoints

In a manner similar to the OPΔT setpoints, restrictions to the nominal T_{avg} at rated thermal power (T') term of the overtemperature delta-temperature (OTΔT) equation had been applied, which credited the reduction in RCS operating pressure and temperature. Subsection 5.2.1, "Introduction and Background," to Section 5.2, "Non-LOCA Transients," of WCAP-17662-NP, discusses evaluation for events that take credit for the lower temperature and pressure, stating, "In particular were the [OTΔT] and [OPΔT] setpoints, which utilized T' and T'' values that were restricted below the full power T_{avg} primarily to provide overpower protection while maintaining the same ΔT setpoints." Subsection 5.2.3.2 discusses an uncontrolled rod withdrawal at power, and states, "Additionally, it was confirmed as part of the Return to RCS NOP/NOT Program that the OTΔT setpoints modeled in the current analysis remain valid at NOP/NOT conditions."

In its supplement dated April 29, 2014, the licensee stated that no credit had been taken in the analysis for the benefit of restricted T' or T'' values, such that removing the restrictions on T' and T'' (as was done to restore NOP/NOT operating conditions) had no impact on the rod withdrawal at power analysis. Since the removal of the T' and T'' restrictions had no effect on the rod withdrawal at power analysis, the NRC staff found that the licensee's conclusion that the current rod withdrawal at power analysis remains valid at proposed NOP/NOT conditions, and that the OTΔT trip setpoints support RCS NOP/NOT operation, was valid.

2.2.2.3 Disposition for Remaining Events

For the remaining licensing basis events, the licensee evaluated each in turn and either confirmed that the existing analysis already employed initial conditions that were bounding of proposed NOP/NOT operation, or performed sensitivity studies using NRC-approved codes and methods to conclude that the results of the current licensing basis analysis would be bounding of one performed at proposed NOP/NOT conditions. Based on its review of the licensee's evaluations, the NRC staff confirmed that the remaining licensing basis events would be unaffected by the proposed NOP/NOT restoration, and concluded that a detailed review of each of these events was unnecessary.

2.2.3 Anticipated Operation Occurrences and Non-Loss-of-Coolant Accident Conclusion

Based on its review, performed as described above, the NRC staff determined that the proposed NOP/NOT restoration was acceptable with respect to non-LOCA accident and transient events. In particular, the licensee evaluated the continued adequacy of OPΔT and OTΔT trip setpoints, performed an explicit analysis of the main steam line break event, and determined that remaining licensing basis events were unaffected by the proposed NOP/NOT restoration.

2.3 <u>Steam Generator Tube Rupture</u>

2.3.1 Regulatory Evaluation

A SGTR event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured steam generator tube and main steam safety or atmospheric relief valves. The reactor protection system and ECCS systems are actuated to

mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 100. The NRC staff's review covered (1) postulated initial core and plant conditions, (2) the method of thermal and hydraulic analysis, (3) the sequence of events, assuming offsite power either available or unavailable, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions consistent with the plant's emergency operating procedures, and (7) the results of the accident analysis. A single failure of a mitigating system is assumed for this event. The NRC staff's review of the SGTR is focused on the thermal and hydraulic analysis for the SGTR in order to confirm that the faulted steam generator does not experience overfill. Preventing steam generator overfill is necessary in order to prevent the failure of the main steam lines. Specific review criteria are contained in SRP Section 15.6.3 (Ref. 22).

2.3.2 Technical Evaluation

The SGTR is described in Section 5.3 of WCAP-17762-NP. The licensee describes its current licensing basis, which is comprised of several different analyses of SGTR scenarios.

The licensing basis input to the dose analysis assumes that a double-ended guillotine rupture of a single steam generator tube occurs, and primary-to-secondary leakage through the ruptured tube continues, unmitigated, for thirty minutes following the break. The attendant radiological consequences are determined assuming that the primary-to-secondary leakage is released to the environment. This licensing basis input to dose does not credit any operator actions, but it also neglects the potential that the ruptured steam generator could overfill, or that the break will realistically continue for longer than thirty minutes, until operators can equalize primary and secondary pressures to terminate the break.

Experience has shown that the operator intervention required to terminate the SGTR event would likely exceed the thirty minute release assumed in the radiological consequence analysis, and that the associated extended blowdown could result in an overfill of the ruptured generator, which could in turn challenge the structural integrity of the main steam lines. Other assumptions in the radiological consequence analysis, such as the assumption that the break flow continues unmitigated at the full rate associated with the double-ended guillotine rupture, are generally understood to result in a dose estimate that bounds an event that proceeds for a longer period of time, but is mitigated by operator intervention.

Confirmation that the radiological analysis is adequately conservative relies, in part, on supplemental analyses that demonstrate that the mass release calculated for the radiological consequence analysis is conservative, and that there is adequate margin to steam generator overfill. These supplemental analyses credit operator actions, and reflect a more realistic time to terminate the event. The licensee addressed proposed NOP/NOT operation with respect to each of the SGTR analyses.

2.3.2.1 Licensing Basis Input to Dose

The licensee evaluated the existing SGTR analysis, which assumes a 30-minute blowdown in order to determine the input to the radiological consequence analyses. The licensee determined that the analyses reflect the RCS pressure and temperature conditions proposed for NOP/NOT restoration. In particular, the analyses assume a nominal T_{avg} range of 553.7 °F to

575.4 °F, and a nominal RCS pressure range from 2100 psia to 2250 psia. The proposed NOT and NOP are 571.0 °F and 2250 psia, respectively. Since the proposed operating conditions fall within the range analyzed in the CNP-1 current licensing basis, the NRC staff concluded that the licensing basis input to dose need not be reanalyzed. The NRC staff therefore determined that the proposed NOP/NOT restoration was acceptable with respect to the SGTR licensing basis radiological consequence analysis, based on the fact that the current licensing basis analysis bounds the proposed NOP/NOT operation.

2.3.2.2 Supplemental Input to Dose

The supplemental input to dose analysis acknowledges that operators require more than 30 minutes to terminate the primary-to-secondary leakage. The analysis also credits operator actions that reduce the primary-to-secondary leakage rates. These assumptions together are analyzed to confirm that the mass release associated with a longer blowdown and operator intervention is still less than that predicted in the licensing basis input to dose analysis.

To address the effects of NOP/NOT restoration on the supplemental input to dose analysis, the licensee modeled the proposed NOP/NOT operating conditions in a sensitivity study employing the same methodology as that used in the current licensing basis supplemental analysis. The sensitivity study confirmed that the current analysis remains bounding of NOP/NOT operation. Based on its review of the results of the licensee's sensitivity study, the NRC staff concluded that the supplemental input to dose analysis in the current licensing basis acceptably supports the proposed NOP/NOT operation.

In the April 29, 2014 supplement, the licensee stated that time-critical operator actions assumed in the supplemental analysis are unaffected by NOP/NOT restoration, and that these actions, and station operator ability to complete them, are maintained in accordance with an administrative program. Since the licensee confirmed that NOP/NOT operation does not affect the operator actions to mitigate the SGTR event, and that these actions are controlled by a formal program, the NRC staff determined that the operator actions assumed in the supplemental analysis were acceptable.

2.3.2.3 Margin to Overfill

According to the CNP-1 UFSAR, Section 14.2.4.4, "The termination of break flow occurs before the steam generator would overfill into the main steam piping. A specific overfill calculation was not included in the original analysis; however, a more recent analysis described below confirms the continued validity of this assumption." Section 14.2.4.4 of the CNP-1 UFSAR also states:

> Demonstration that the ruptured steam generator does not overfill during the accident has more recently been performed by utilizing an NRC-approved thermal hydraulic analysis code. Reference 2 ["Acceptance for Referencing of Licensing Topical Report WCAP-10698 'SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,' December 1984," letter from C. E. Rossi, NRC, to Alan E. Ladieu, Westinghouse Owners' Group SGTR Subgroup Chairman, March 30, 1987] includes the NRC's approval of the break flow model contained within the LOFTTR2 computer code that has been used for the Cook unit-specific supplemental overfill analysis. The approved code

simulates the plant response, and models specific operator actions. Thus, a more realistic representation of the break flow during the accident is obtained. The supplemental analysis demonstrates that break flow following the complete severance of a steam generator tube is terminated within 52 minutes after initiation of the tube rupture and overfill of the steam generator does not occur. The resultant mass release data from this recent overfill analysis has been confirmed to remain bounded by the mass release data calculated with the current licensing basis analysis methodology, which assumes break flow persisting for 30 minutes from the initiation of the accident.

The discussion on SGTR contained in WCAP-17762-NP also includes reference information to the license amendment and review documents associated with the incorporation of the margin-to-overfill analysis into the CNP-1 licensing basis. These documents indicate that, although the UFSAR specifically references the NRC-approval of a topical report, the NRC staff did not approve the incorporation of this topical report into the CNP-1 licensing basis, because the CNP-1 margin-to-overfill analysis did not reflect the set of conservative assumptions that had been applied in the generic analysis documented in the topical report.

This supplemental analysis, however, is based on assumptions that are non-bounding of permissible plant operation, do not consider uncertainties, and do not include a limiting single active component failure in the mitigating safety system. The analyses also demonstrate a very small available margin to steam generator overfill. Such assumptions, which reduce analytic margin in favor of improving the final result, compromise the ability of the analysis to demonstrate that the plant can maintain margin-to-overfill during an SGTR event in the same range of variations in possible scenarios required of a typical licensing basis safety analysis. In practice, the NRC staff has not accepted analytic approaches that refer to the amendment approving these reductions in analytic margins as a precedent. However, since these assumptions were previously analyzed and approved by the NRC staff, specifically for CNP, the NRC staff presently accepts the previously approved analytic assumptions.

Generic studies documented in WCAP-10698-P-A (Ref. 23) indicate that final liquid volume in the steam generator shell side will be greater, by about 12 to 13 percent, when employing conservative input values. The NRC staff accepted the technical basis underlying WCAP-10698-P-A, since it described an acceptable methodology for performing a conservative margin-to-overfill analysis for a postulated SGTR.

The licensee concluded that modeling the margin-to-overfill event at the restored NOP/NOT operating conditions provides a benefit to the margin-to-overfill analysis. However, the licensee also incorporated into its analysis a corrected set of assumptions regarding decay heat to introduce additional conservatism in response to operating experience suggesting that modeling maximum decay heat would not necessarily be conservative.

Noting the above issues with the nominal input assumptions, but also acknowledging the NRC staff's specific, prior approval of the use of these assumptions for the CNP-1 licensing basis, the NRC staff concludes that the licensee has evaluated the effect of NOP/NOT restoration on SGTR margin-to-overfill in a manner consistent with the CNP-1 licensing basis. If an SGTR were to occur in the scenario that the plant is operating exactly at its nominal statepoint, the analysis indicates that there would be 69 cubic feet of margin-to-overfill.

2.3.3 Steam Generator Tube Rupture Conclusion

Based on its review, as described above, the NRC staff concludes that the licensee adequately addressed the effect of NOP/NOT restoration on the SGTR licensing basis input to dose analysis. Additional, supporting analyses remain consistent with the current licensing basis and provide a similar level of assurance that the licensing basis input to dose analysis is bounding.

2.4 Accident and Transient Analysis Conclusion

Based on the NRC staff's conclusions, discussed in Sections 2.1.3, 2.2.3, and 2.3.3, the staff concluded that overall, the licensee has acceptably evaluated the effects of proposed NOP/NOT restoration on the CNP-1 ECCS evaluation, and on the licensing basis accident and transient analyses.

3.0 REACTOR CORE DESIGN

This section of the safety evaluation addresses the proposed amendment's effects on the CNP-1 nuclear design, fuel design and performance, and core thermal and hydraulic design.

3.1 Regulatory Evaluation

The current design and licensing bases of CNP-1 are documented in the UFSAR, as required by 10 CFR 50.34. The supporting licensing report provided by Westinghouse, WCAP-17762-NP, provides a number of evaluations to demonstrate that the existing UFSAR analyses will remain valid at the proposed operating pressure and temperature. For transients where the existing analysis is found to no longer be valid with the proposed operating conditions, a reanalysis is provided.

These analyses are also used to support the CNP-1 TS. Section 50.36 of 10 CFR provides the NRC requirements for TS, which include operating limits in the form of safety limits, limiting safety system settings, limiting control settings, and limiting conditions for operation. In order to provide for the safe operation of the plant, these TS limits must be based on accurate analysis that fully accounts for the plant's design basis.

At CNP-1, several cycle-specific core operating limits have been moved out of the TS and into the administratively controlled Core Operating Limits Report, in accordance with the NRC guidance of Generic Letter 1988-16 (Ref. 24). These cycle-specific limits, such as peaking factors, are to be calculated with NRC-approved methodologies and must be consistent with the applicable limits of the safety analysis provided in the CNP-1 UFSAR.

The NRC's acceptance criteria are based on:

- PSDC-6, "Reactor Core Design," insofar as it requires that the reactor core, with its related control and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated;

- PSDC-29, "Reactivity Shutdown Capability," insofar as it requires that one of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition to prevent exceeding fuel damage limits; and

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance.

3.2 Technical Evaluation

3.2.1 Nuclear Design

As stated in WCAP-17762-NP, Section 6.1, the licensee's process for nuclear design includes cycle-specific calculations which "are performed for each reload cycle" and are used to demonstrate that Reload Safety Analysis Checklist criteria are satisfied given the specific operating conditions of the cycle, in accordance with the methodology presented in WCAP-9272-P-A, (Ref. 25). For each reload cycle, the core operating limits presented in the Core Operating Limits Report are determined and the operating conditions of the plant are confirmed to be bounded, with appropriate margin, by the existing AOR as described in the CNP-1 UFSAR.

The licensee stated that "the standard set of reload core design criteria has been confirmed via evaluation or explicit analysis for the transition to [NOP/NOT] conditions." Explicit calculations of Doppler power coefficient and burnup dependent peaking factors were performed for the hot full power steam line break transient, the analysis of which is described in WCAP-17762-NP Section 5.2.2. The licensee also stated that "The impact of NOP/NOT conditions on all [Reload Safety Analysis Checklist] parameters that were not explicitly analyzed was evaluated against typical margins to their limits. Adequate margin was confirmed to be available."

Based on its review of the licensee's analysis, the NRC staff has determined that the licensee's approach is acceptable. This determination was made because the nuclear design was confirmed to be acceptable at NOP/NOT conditions within the licensee's existing core reload design and evaluation process, because the licensee will continue to use this process to confirm the acceptability of future core designs at the new NOP and NOT, and because evaluations were able to confirm adequate margin where Reload Safety Analysis Checklist parameters were not explicitly analyzed.

3.2.2 Nuclear Peaking Factors and Thermal Conductivity Degradation

The existing LBLOCA AOR for CNP-1 was analyzed using the ASTRUM method, as documented in WCAP-16009-NP-A (Ref. 9). WCAP-17762-NP includes an evaluation supporting the applicability of the LBLOCA AOR, in which the impacts of restoration of NOP and NOT on the AOR results are estimated. This evaluation also includes estimates of the impacts of model changes and errors that have been identified and implemented in the CNP-1 LBLOCA analysis since the original NRC approval in 2008. Input changes, made to offset increases in PCT associated with both restorations of NOP/NOT and model errors, are also included in the evaluation.

This portion of the review focused on adjustments made to offset the effects of nuclear fuel TCD with fuel burnup. TCD caused the largest estimated effect on PCT at CNP-1 since the implementation of ASTRUM at CNP-1 in 2008. This error has a significant impact on the nuclear design of the plant, in both a previous evaluation and the one presented in WCAP-17762-NP. The effect of TCD on the CNP-1 PCT was offset by changes to the nuclear peaking factors. These changes are made to the heat flux hot channel factor, which is the ratio of the peak heat generation rate to the average heat generation rate, and the enthalpy rise hot channel factor, which is the ratio of the maximum integrated rod power to the average integrated rod power. The changes included decreases in the beginning-of-life peaking factor limits as well as peaking factor burndown, where the peaking factor limits decrease as a function of burnup.

The NRC staff observed that burnup dependence of peaking factors was not included in the CNP-1 TS or the Core Operating Limits Report, but was required to maintain margin to the 10 CFR 50.46(b)(1) limit on peak cladding temperature in the LBLOCA analysis. To demonstrate that the LBLOCA analysis would remain bounding of plant operating conditions without including peaking factor burndown in the TS, the licensee provided information regarding the treatment of peaking factor limits. In its June 5, 2014 supplement (Ref. 3), the licensee provided information regarding the peaking factor limits. The licensee stated that, in accordance with the engineering procedure controlling the process for reactor core design, predicted power distributions and peaking factor burndown limits are analytically confirmed and verified during the reload design process. As required by TS 3.1.2, 3.2.1, and 3.2.2, the licensee measures reactivity and peaking factors periodically throughout each cycle, to verify that the values are within their design limits. These confirmations, along with startup physics testing, ensure that the LBLOCA analysis remains applicable. The peaking factor limits for lower burnup fuel are more limiting than for higher burnup fuel. Therefore, no TS or Core Operating Limits Report changes are required to account for burnup. If the core is found to not be operating as designed, adjustments are made, as required by the TSs, to ensure operation remains within LBLOCA analysis limits.

The NRC staff concluded that this approach is acceptable, because it maintains the applicability of the LBLOCA analysis throughout the operating cycle, while continuing to use the existing core reload design procedures and TS SRs. The NRC staff therefore concludes that the nuclear design for revised NOP/NOT conditions is acceptable.

3.2.3 Fuel Design and Performance

3.2.3.1 Applicability of Current Methodologies

WCAP-17762-NP, Section 6.2.1, states that CNP-1's current fuel, Westinghouse 15x15 Optimized Fuel Assembly fuel with ZIRLO® and Optimized ZIRLO™ cladding, will be maintained for the transition to NOP/NOT conditions. Optimized ZIRLO™ was approved for use at CNP-1 by the NRC in a letter dated August 25, 2011 (Ref. 26). Fuel rod design evaluations are performed using the methodologies described in WCAP-10851-P-A (Ref. 27) and WCAP-15064-NP-A (Ref. 28). These methodologies are NRC-reviewed and approved and have been confirmed by the NRC staff to be part of CNP-1's current licensing basis. The licensee also uses WCAP-9272-P-A (Ref. 25), to confirm that the specified acceptable fuel design limits are met for reload cores.

The NRC staff concluded that this approach is acceptable, because it continues to use fuel design analysis methodologies that have been previously approved for use at CNP-1 by the NRC. As well, no changes being made as part of the return to NOP/NOT program impact the applicability of the existing methodologies.

In their LBLOCA evaluation, the licensee analyzed the fuel thermal-mechanical response with the PAD 4.0-TCD methodology in order to evaluate the ECCS performance. PAD 4.0-TCD is a model that has been found acceptable by the NRC staff for limited, plant-specific estimates which includes TCD effects within the existing PAD 4.0 framework. It was previously used to estimate the impact of TCD at CNP-1 in letters dated March 19, 2012, and June 11, 2012 (Refs. 13 and 29, respectively), which in turn made Reference to the Westinghouse March 7, 2012 letter (Ref. 11), for the methodology used for the estimate. While the PAD 4.0-TCD model remains bound to estimates applied to specific plants, the NRC accepted the CNP-1 estimate of TCD effects in a March 7, 2013 letter (Ref. 30).

For the fuel performance in LBLOCA analysis, the NRC staff concluded that the licensee's approach is acceptable for estimating the effects of TCD at NOP/NOT conditions due to the previous acceptability of the methodology. However, the NRC staff considers the updated evaluation presented in WCAP-17762-NP to be only an estimate of the impacts of TCD. The staff continues to expect the licensee to perform a reanalysis for LBLOCA that includes the effects of fuel TCD, in accordance with the commitments made in the letter dated March 19, 2012, as updated by letter dated June 9, 2015 (Refs. 13 and 14).

3.2.3.2 Fuel Rod Acceptance Criteria, Analyses, and Evaluations

WCAP-17762-NP, Section 6.2.3, presents four basic conditions and requirements for fuel rod design analyses, summarizing the licensing basis requirements for the fuel currently in use at CNP-1. This list includes fuel burnup limits, waterside corrosion and calculated metal-oxide interface temperature limits, and the conditions and limitations placed in NRC safety evaluations on ZIRLO®, zircaloy-4, and Optimized ZIRLO™ cladding fuel analyses. These conditions, requirements, and limitations imposed on the fuel will be maintained for the return to NOP/NOT program.

Section 6.2.3 also discusses the fuel design acceptance criteria for CNP-1 and presents evaluations of the fuel at the proposed increased RCS pressure and temperature against these criteria. The criteria addressed in WCAP-17762-NP include all of the relevant design bases from the CNP-1 UFSAR, Section 3.5.1.3, which discusses the Westinghouse Optimized Fuel Assembly fuel rod thermal and mechanical design.

Increasing the RCS temperature and pressure will have competing effects on the fuel temperature – higher coolant temperature drives the fuel-to-coolant heat transfer down and therefore drives fuel temperatures up, but increased coolant pressure causes pellet-clad contact to occur earlier in the cycle, which increases the overall fuel rod conductivity and drives fuel temperature down. These competing effects were evaluated by the licensee for their influence on rod internal pressure, cladding oxidation and hydriding, and fuel rod end plug weld integrity, and found to have minimal impacts that could be offset by available margin. Cladding stress, strain and fatigue were determined to be negligibly affected by the increase in RCS temperature

and pressure, and any effect on these parameters would be able to be accounted for with available margin. According to WCAP-17762-NP, this is due to the continued use of the constant axial offset control bands, which constrain plant operation to avoid significant xenon transients and restrict the number of limiting events for cladding stress and strain.

The licensee plans to mitigate many of the effects of the restoration of NOP and NOT by using available margin. In the June 5, 2014 supplement, the licensee stated that "while no explicit, quantitative calculations were performed to evaluate the fuel rod design criteria, a qualitative assessment was done as part of the Engineering Report to compare the estimated impact of NOP/NOT operation with the adequate available margin for recent cycles of D.C. Cook Unit 1." This assessment determined that "the available margin for D.C. Cook Unit 1 is sufficient to offset both the estimated impacts of transition to NOP/NOT operation, as well as the maximum impacts of TCD, which have been calculated generically." The licensee also stated that the fuel rod design criteria are evaluated on a cycle-specific basis, similar to the nuclear design criteria discussed previously, and are ensured to be met each cycle.

The NRC staff determined that the licensee's approach to fuel design at NOP/NOT conditions is acceptable, because the licensee will continue to use currently approved methodologies to evaluate the fuel performance against the acceptance criteria that are documented in the CNP-1 UFSAR. Furthermore, evaluations based on recent cycles have shown minimal or insignificant impact on the acceptance criteria due to NOP/NOT conditions, and have determined that sufficient margin is available to offset any impact.

3.2.4 Core Thermal and Hydraulic Design

The only event explicitly analyzed in WCAP-17762-NP is the hot full power steam line break, which has previously taken credit for the current CNP-1 operating temperature and pressure and thus needed to be reanalyzed at the proposed operating conditions. The main steam line break is classified as an American Nuclear Society (ANS) Condition IV event, which means that it is postulated but not expected to occur during the life of a plant. The NRC staff's acceptance criteria for main steam line breaks are described in SRP 15.1.5 (Ref. 17).

As discussed in WCAP-17762-NP, Section 5.2.2.3, the CNP-1 hot full power steam line break analysis is performed according to the more restrictive acceptance criteria associated with AOOs. As such, the NRC staff has applied the AOO acceptance criteria and the CNP-1 PSDCs for AOOs to its evaluation. These acceptance criteria include a requirement to keep the departure from nucleate boiling ratio (DNBR) above the 95/95 limit throughout the transient, and peak linear heat generation rate below a value that would cause centerline fuel melt. Satisfying these criteria will satisfy the existing licensing basis for main steam line break analysis for CNP-1, where the Standby Safeguards Analysis is described in UFSAR, Section 14.2 as follows:

> The analyses presented in in this section demonstrate that adequate provisions are included in the design of the plant and its engineered safeguards which restrict potential exposures to below the appropriate limits for the fault conditions resulting in the fission product release to the environment listed as follows: [...]
> 5. Rupture of a Steam Pipe

Meeting the AOO criteria for the hot full power steam line break will therefore demonstrate that the SRP 15.1.5 and ANS Condition IV acceptance criteria are met while aligning with the CNP-1 licensing basis, and providing a more appropriate and relevant evaluation than the SRP 15.1.5 criteria. The NRC staff's approach in this evaluation should not be construed to alter the CNP-1 licensing basis or the main steam line break classification as an ANS Condition IV event.

To meet the AOO acceptance criteria, a thermal-hydraulic analysis was performed with the subchannel code VIPRE-W, a Westinghouse quality assurance configured version of the NRC-approved VIPRE-01 code. The critical heat flux correlation used for the analysis was the WRB-1 correlation, which was combined with the Revised Thermal Design Procedure statistical methodology to determine the design limit DNBR. The WRB-1 DNBR correlation limit of 1.17 was used to calculate the design limit DNBR of 1.21. All of the codes and methods used for the DNB analysis of the hot full power steam line break transient have been reviewed and approved for generic use by the NRC, and have also been previously approved for incorporation into the CNP-1 licensing basis. The WRB-1 correlation limit DNBR of 1.17 and the design limit DNBR of 1.21 are both consistent with the CNP-1 UFSAR.

Application of the proper DNBR acceptance criteria, based on these limits and codes, will satisfy both the regulations and the PSDCs. As such, the NRC staff concluded that the thermal-hydraulic design presented in WCAP-17762-NP, as used to analyze the DNBR effects of a hot full power steam line break transient at CNP-1, is appropriate for the plant and is consistent with the plant's licensing basis.

3.3 Reactor Core Design Conclusion

Based on the NRC staff's conclusions discussed in Sections 3.2.1, 3.2.2, 3.2.3, and 3.2.4, the staff concluded that overall, the licensee has acceptably evaluated the effects of proposed NOP/NOT restoration on the CNP-1 nuclear design, fuel design and performance, and core thermal and hydraulic design, and has fulfilled the requirements of the applicable regulations as discussed in the regulatory evaluation.

4.0 ELECTRICAL POWER SYSTEM

This section of the safety evaluation addresses the impact on environmental qualification of electrical equipment and the capacity of electrical power systems. In its supplement dated June 5, 2014, the licensee provided the following information regarding the impact of the license amendment request on the electrical power system and environmental qualification (EQ) of electrical equipment.

Restoring the CNP-1 RCS to NOP/NOT conditions does not affect the loading on the emergency diesel generators (EDGs) since the containment design pressure is not affected by the proposed changes. In addition, safety-related motor loads used in the CNP-1 EDG evaluations bound maximum loading conditions consistent with maximum design performance of the driven equipment. Process-driven 600-volt motor loads, such as the CEQ fan motors that are actuated following a containment pressure "Hi" signal, are conservatively assumed to start simultaneously as a block load at the most critical time in the automatic sequence.

The load flow analysis for CNP-1 includes verification that each individual LOCA sequence run has steady state voltages that meet the acceptance criteria for the connected equipment, and a LOCA Sequence output voltage vs. time plot on the 4.16 kilovolt T-Buses (T11A, T11D, T21A, and T21D), where the degraded voltage relays are located, that does not trip the degraded voltage relays.

The reserve auxiliary transformers can provide the required voltage to engineered safety features loads without actuating protective devices and meet the accident analysis assumptions.

For CNP-1, the EQ limiting condition for containment temperature occurs following a main steam line break. The results of the CNP-1 main steam line break reanalysis show that the peak post-accident temperature value considered in the EQ program (324.7 °F) is not affected by the proposed change to restore NOP/NOT conditions. The EQ envelope with regard to pressure for equipment in containment is not affected. The current qualification criteria for other EQ parameters for equipment inside containment, specifically containment post-accident radiation levels, flood level, and pH, are unaffected by implementation of NOP/NOT conditions on CNP-1.

4.1 Electrical Power System Conclusion

Based on the above evaluations, the NRC staff concludes that there are no changes to the electric power system and EQ of electrical equipment as a result of the proposed license amendment request. Therefore, the changes are acceptable with respect to the electrical power system.

5.0 REACTOR COOLANT PRESSURE BOUNDARY

The NRC staff reviewed the application to determine if it adequately demonstrated that the structural integrity and leak tightness of RCPB materials will be maintained under the increased temperature and pressure.

5.1 Regulatory Evaluation

The proposed NOP and NOT increase may affect the structural integrity of the RCPB which defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff reviewed the impact of the proposed temperature and pressure increase as it applies to RCPB materials in terms of design specification, fabrication, testing, susceptibility to degradation, and inspections. The NRC's acceptance criteria for RCPB materials are based on the following:

- PSDC-1, "Quality Standards," insofar as it requires that SSCs important to safety be designed, fabricated, erected, tested, and inspected to quality standards that reflect the importance of the safety functions to be performed;

- PSDC-9, "Reactor Coolant Pressure Boundary," insofar as it requires the RCPB to be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime;

- PSDC-33, "Reactor Coolant Pressure Boundary Capability," insofar as it requires the RCPB to be capable of accommodating without rupture the static and dynamic loads imposed as a result of an inadvertent and sudden release of energy to the coolant;

- PSDC-34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention," insofar as it requires that the RCPB be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure;

- PSDC-36, "Reactor Coolant Pressure Boundary Surveillance," insofar as it requires that RCPB components have provisions for inspection, testing, and surveillance to assess the structural and leaktight integrity of the boundary;
- Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," specifies the fracture toughness requirements for ferritic components of the RCPB;

- 10 CFR 50.55a(g)(6)(ii)(D), insofar as it requires the control rod drive mechanism nozzle penetrations be inspected in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-729-1, "Alternative Examination Requirements for Pressurized Water Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," as conditioned;

- 10 CFR 50.55a(g)(6)(ii)(E), insofar as it requires components fabricated with Alloy 600/82/182 metal be inspected in accordance with ASME Code Case N-722, "Additional Examinations for [Pressurized Water Reactor] Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1," as conditioned; and

- 10 CFR 50.55a(g)(6)(ii)(F), insofar as it requires Alloy 82/182 welds be inspected in accordance with ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 [Pressurized Water Reactor] Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities Section XI, Division 1," as conditioned.

5.2 Technical Evaluation

The NRC staff has reviewed (1) the impact of the increased NOP and NOT on the RCPB materials with respect to the requirements of PSDC-1, 9, 33, 34 and 36, 10 CFR 50.55a, and Appendix G to Part 50, and (2) material degradation of the RCPB materials.

The NRC staff notes that it approved the design bases for the RCPB materials based on the original temperature and pressure as part of initial licensing review process. The original NOP and NOT were higher than the current NOP and NOT. This means that the design, fabrication, construction, and testing of the RCPB materials has satisfied PSDC-1, 9, 33, 34 and 36, 10 CFR 50.55a, and 10 CFR Part 50, Appendix G, based on the original, higher temperature and pressure. Therefore, increasing the current NOP and NOT to the original licensed values will not affect the original design bases for the RCPB.

The NRC staff notes that pressurized water reactors have experienced primary water stress corrosion cracking (PWSCC) in piping components that contain nickel-based Alloy 600 base metal and Alloy 82/182 weld metal. PWSCC tends to initiate and grow in Alloy 600/82/182 metal at high temperatures. Therefore, temperature and pressure increase may degrade the RCPB materials that are made of Alloy 600/82/182 metal.

In its April 29, 2014 supplement, the licensee stated that CNP-1 currently has no evidence of previously identified degradation to pressure-retaining components that rely on ASME Code, Section XI, flaw evaluation methods for acceptability of service. In addition, the licensee is required to follow 10 CFR 50.55a(g)(6)(ii)(E) and 10 CFR 50.55a(g)(6)(ii)(F) to periodically inspect Alloy 600/82/182 components. The NRC staff notes that the potential degradation of the RCPB materials is being monitored through implementation of the above regulations such that any impact of the increased temperature and pressure, if evident, will be detected in a timely manner. Therefore, the NRC staff finds that the structural integrity of the RCPB materials will be maintained with respect to the increased NOP and NOT.

The licensee replaced the original CNP-1 reactor vessel closure head (RVCH), which had Alloy 600 penetration nozzles and Alloy 82/182 welds, in the fall of 2006 with a new RVCH having Alloy 690 nozzles and Alloy 52/152 welds. The Alloy 690/52/152 material is less susceptible to PWSCC. The licensee stated that it has examined the RVCH in accordance with 10 CFR 50.55a(g)(6)(ii)(D). The licensee visually inspected the RVCH in 2011, with no adverse findings.

The NRC staff notes that the design temperature and pressure for the reactor vessel are 650 °F and 2485 psig as documented in Table 4.1-3 of CNP-1 UFSAR, Revision 17, respectively. The new RVCH is part of the reactor vessel and, therefore, is fabricated to the reactor vessel design conditions. The NRC staff finds that the design temperature and pressure for the new RVCH are higher than the increased NOP and NOT of 571 °F and 2250 psia.

The NRC staff determines that based on the design conditions, the new RVCH is qualified to be used for the increased temperature and pressure conditions. In addition, the NRC staff notes that the licensee has followed 10 CFR 50.55a(g)(6)(ii)(D) in the examination of the control rod drive mechanism nozzle penetrations. Therefore, the staff determines that the increased NOP and NOT is not likely to affect the structural integrity of the RVCH significantly because (1) the RVCH is designed with a higher temperature and pressure than the proposed NOP and NOT, and (2) the licensee will periodically inspect the RVCH in accordance with the NRC regulations.

The licensee stated that the systems and components, including interface systems and control systems, will function as designed and all performance requirements for these systems remain acceptable. The licensee further stated that there are no physical changes being made to the RCPB. The NRC staff finds that no physical changes were made to the RCS systems and that the existing RCS piping system was originally designed based on the higher NOP and NOT. Therefore, the NRC staff finds that the change from the current to higher NOP and NOT will not affect the structural integrity and leak tightness of the RCS piping system.

5.3 Reactor Coolant Pressure Boundary Conclusion

On the basis of information submitted, the NRC staff determines that the licensee has demonstrated that the structural integrity and leak tightness of RCPB materials will be maintained under the increased temperature and pressure. The staff further determines that the licensee will monitor any potential material degradation caused by the increased temperature and pressure via the periodic inspections as required by ASME Code, Section XI, 10 CFR 50.55a(g)(6)(ii)(D), 10 CFR 50.55a(g)(6)(ii)(E), and 10 CFR 50.55a(g)(6)(ii)(F).

The NRC staff concludes that under the increased RCS temperature and pressure, the RCPB materials will continue to satisfy the requirements of PSDC-1, 9, 33, 34 and 36, and Appendix G to 10 CFR Part 50. Any potential degradation of the RCPB materials as a result of increased temperature and pressure will be monitored by the periodic inspections as mandated by the existing regulations, such that the structural integrity of the RCPB materials will be maintained. Therefore, the staff finds that the proposed NOP and NOT increase is acceptable with respect to the RCPB materials.

6.0 CONTAINMENT AND VENTILATION

6.1 Regulatory Evaluation

The NRC staff acceptance criteria for the primary containment functional design is based on the following:

- PSDC-42, "Engineered Safety Features Components Capability," insofar as it requires that Engineered Safety Features be designed so that the capability of these features to perform their required function is not impaired by the effects of a LOCA. According to the CNP-1 UFSAR, Section 6.0, the containment structure is an engineered safety feature that is designed to maintain containment integrity when subjected to accident temperatures and pressure.

- PSDC-10, "Reactor Containment," insofar as it requires that the containment structure be designed to sustain the initial effects of a large reactor coolant pipe break without loss of required integrity, and to retain the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

- PSDC-49, "Reactor Containment Design Basis," insofar as it requires that the reactor containment structure be designed so that leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from a LOCA will not result in undue risk to the health and safety of the public.

- PSDC-52, "Containment Heat Removal Systems," insofar as it requires that where active heat removal systems are needed under accident conditions to prevent exceeding containment pressure, at least two systems with full capacity shall be provided.

- SRP Section 6.2.1.1.B (Ref. 31) provides criteria for containment integrity analysis for ice condenser containments.

- SRP Section 6.2.1.3 (Ref. 32) provides criteria for Mass and Energy (M&E) release analysis for LOCAs.

- SRP Section 6.2.1.4 (Ref. 33) provides criteria for M&E release analysis for steam line breaks.

6.2 Technical Evaluation

6.2.1 Description of the Containment

CNP-1 is a 4-loop Westinghouse Pressurized Water Reactor (PWR) having an ice condenser containment. The containment structure which is lined with steel plate on the inside is a reinforced concrete vertical right cylinder with a slab base and a hemispherical dome. It is divided into three main compartments: (a) the lower compartment, (b) the upper compartment, and (c) the ice condenser compartment. The lower compartment encloses the reactor system and associated auxiliary systems equipment. The upper compartment contains the refueling cavity, refueling equipment and polar crane used during refueling and maintenance operations. The upper and lower compartments are separated by a divider barrier. The ice condenser compartment is a completely enclosed annular compartment located around approximately 300° of the perimeter of the upper compartment, but penetrating the operating deck so that a portion extends into the lower compartment. The ice condenser is a refrigerated compartment containing borated ice provided to absorb the energy release during a pipe break accident.

6.2.2 Function of the Ice Condenser

The primary purpose of the ice condenser is to absorb energy released in the event of a postulated design basis LOCA or a main steam line break accident inside the containment. Sufficient ice is included in the ice condenser to absorb the energy released in a rupture of the largest RCS pipe and, in conjunction with the other engineered safety features, maintain the containment pressure below its design value.

6.2.3 Containment Mass and Energy Release and Containment Integrity Analysis

6.2.3.1 Purpose

CNP-1 currently operates at a full power pressurizer pressure of 2100 psia and T_{avg} of 556 °F. In the license amendment request, the licensee proposes to operate at a full power pressurizer pressure of 2250 psia, and T_{avg} of 571 °F.

The sections below provide the NRC staff's evaluation of the licensee's technical evaluation and analyses of the following events under the proposed RCS NOP/NOT conditions: (a) Short term LOCA M&E release and containment pressure and temperature response analysis, (b) Main steam line break accident M&E release and containment pressure and temperature response analysis, (c) Sump temperature response and Net Positive Suction Head (NPSH) analysis, and (5) Minimum containment back pressure for emergency core cooling analysis.

6.2.3.2 Short Term LOCA Mass and Energy Release and Containment Pressure and Temperature Response Analysis

The mass flow rate released due to an RCS high energy line break inside the containment results in local short term pressure build-up at a faster rate than the overall containment pressure. Therefore the structural integrity of the containment and its subcompartments should be re-evaluated due to changes in the RCS operating conditions. The subcompartments to be re-evaluated are the (a) pressurizer enclosure, (b) loop subcompartments, and the (c) upper and lower reactor cavity.

The short-term LOCA M&E release and the containment pressure and temperature response depend on the two key parameters of RCS operating pressure and temperature. Table 1 provides a comparison of the key parameter values used in the AOR and the proposed short term LOCA analysis, as well as the current and proposed normal operating values.

Table 1: Comparison of Core Power, Pressure, and Temperature Values

Parameter	**Current Normal Operating Values**	**Proposed Normal Operating Values**	**AOR Values**	**Proposed Short Term Analysis Values**
Core Power (megawatts thermal (MWt)	3304	3304	3315 (Note 1)	3315 (Note 1)
RCS Full Power Pressurizer Pressure (psia)	2100	2250	2317 (Note 2)	2317 (Note 2)
Vessel/Core Inlet Temperature (°F)	511.7	519.2	506.6 (Note 3)	514.1 (Note 3)
Note 1: Includes +0.34% uncertainty Note 2; Includes +67 psi uncertainty Note 3: Includes -5.1 °F uncertainty				

The licensee stated that the proposed short term M&E release and containment pressure and temperature response for the most limiting LOCA are bounded by the AOR, even though the proposed analysis vessel/core inlet temperature of 514.1 °F is greater than the AOR vessel/core inlet temperature of 506.6 °F. The licensee stated that the short term M&E releases depend on the break critical mass flux, which increases with decreasing RCS temperature because of the higher fluid density at lower temperature. The AOR is bounding because the higher fluid density results in higher mass release, and therefore the AOR M&E release is greater than the M&E release that would be calculated using the proposed analysis values. The containment pressure and temperature response, which depends on the M&E release in the AOR, would therefore bound the response using the proposed analysis values. Therefore the structural integrity of the containment and its subcompartments due to changes in the RCS operating conditions is not affected. The NRC staff finds that the proposed RCS NOP/NOT change meet

the requirements of PSDC-42 because the licensee showed that there is no impact on the engineered safety features and they are protected against dynamic effects during a LOCA.

6.2.3.3 Main Steam Line Break Accident Mass and Energy Release and Containment Pressure and Temperature Response Analysis

The licensee performed a full-spectrum analysis of the main steam line break inside containment and calculated M&E releases for input to the containment pressure and temperature response analysis using the NRC-approved methodology contained in WCAP-8860 (Ref. 34). The M&E calculation considered break size, water entrainment in the break flow (dryness fraction less than 1.0), and the type of protection signal actuated for split breaks. The licensee identified 18 break sizes at power level of 100.34 percent, 70 percent, 30 percent, and 0 percent of the nominal full-load power. These break cases are based on the selection of similar breaks analyzed to support the results in UFSAR Section 14.3.4.4. The M&E release analysis also considered changes associated with the fuel and the replaced Model BWI-Series 51 steam generators.
The licensee used the following conservative assumptions for the proposed blowdown M&E release analysis:

- Consistent with SRP 6.2.1.4, assumed zero pipe flow resistance, and used Moody's correlation for critical flow calculation from the double ended rupture (DER) of the steam line.

- Consistent with NRC-approved WCAP-8860 methodology, realistically assumed water entrainment in the steam from the steam generator in the faulted loop only and added an uncertainty of 0.1 to the steam quality. The licensee stated that entrainment is expected for all DER cases which effectively reduced the energy content in the break steam compared to the energy released in the dry saturated steam. The water entrainment in the steam release during a DER is independent of the steam generator design because the high steam flow associated with the DER greatly exceeds the capacity of moisture separator equipment, thereby making the separators ineffective in preventing water entrainment in the exiting steam.

- Included steam flow from the secondary steam piping by assuming a conservative steam piping volume representing the main steam piping from the four steam generators up to the inlet to the turbine.

- For split-rupture steam line breaks, included the unisolable steam mass in the piping as part of the initial inventory in the faulted-loop steam generator since the break is not large enough to cause a sudden decompression of the piping.

The licensee selected the break sizes 1.4 ft^2, 0.865 ft^2, 0.857 ft^2, 0.834 ft^2, 0.808 ft^2, and 1.0 ft^2 for the M&E release analysis. WCAP-8860 describes the basis for selection of the break sizes and the initial power levels for the M&E releases inside the containment. The description includes four plant power levels of 102 percent, 70 percent, 30 percent, and 0 percent of nominal full load and three categories of break sizes which are the full DER, the small DER, and the small split rupture.

The licensee stated that the full DER forward flow direction is represented by the 1.4 ft^2 cross-sectional area of the flow restrictor integral with the discharge nozzle of the faulted steam generator. The reverse flow direction during the initial blowdown is represented by the cross-sectional flow area 4.7465 ft^2 of the main steam line.

The licensee stated that the small 1.0 ft^2 DER is analyzed at 0 percent power initial conditions assuming no entrainment to show that this break size is more limiting than the full DER at 0 percent power with entrainment.

The licensee stated that each of the small split rupture break area is determined as the largest cross-sectional area that does not produce a steam line isolation signal from the primary protection equipment nor results in water entrainment in the break effluent as discussed in Section 2.1 of WCAP-8860. These areas were determined for each initial power level based on the CNP-1 plant specific values for the secondary-side protection system setpoints.
The licensee conservatively assumed dry saturated steam release from all breaks except from the full DER at 0-percent initial power. The licensee stated that analysis for the full DER at 0-percent power with dry saturated steam release resulted in a peak temperature that exceeded the containment peak transient temperature limit of 324.7 °F. Therefore, consistent with NRC-approved methodology in WCAP-8860, the licensee realistically analyzed the full DER break with entrainment and calculated the peak temperature of 323.8 °F. For small 1.0 ft^2 DER, the licensee followed the guidance in SRP 6.2.1.4. The NRC staff finds the analysis acceptable because the licensee used NRC-approved methodology for the full DER and followed NRC guidance in SRP 6.2.1.4 for the remaining breaks.

The protection system signals that actuate a reactor trip for a DER are the low steam pressure setpoint for the full DER, and the high-1 containment pressure setpoint for the small DER and split ruptures. As the protection functions are actuated in the transient steam line break M&E analysis, reactor trip and rod motion occurs following the applicable delays for the specific protection function. Control and shutdown rods are released from the grippers and begin to fall into the core. The occurrence of reactor trip, which reduces the core power and therefore the energy of the reactor coolant system, is significant in the M&E analysis.

For limiting results for the main steam line break containment pressure and temperature response analysis, some of the input parameters for the split breaks are different from the DERs. The licensee used a containment upper compartment initial temperature of 57 °F for split breaks and 55 °F for DERs. The licensee used an ice condenser compartment initial temperature of 27 °F for splits breaks and 5 °F for DERs. The licensee used the TS maximum value of 120 °F in the lower and dead-ended compartments for both splits breaks and DERs.

The most limiting split break from the peak containment temperature standpoint was 0.865 ft^2 at 100 percent power with a single failure of a main steam isolation valve (MSIV), assuming maximum TS limits of 100 °F in the upper compartment, 120 °F in the lower compartment, and 100 percent relative humidity in all compartments. For this case, the licensee calculated a peak containment temperature of 324.96 °F. This temperature is approximately 0.3 °F greater than the acceptable maximum containment temperature limit of 324.7 °F. To offset the impact of assuming the higher upper compartment initial temperature, the licensee lowered the initial relative humidity in the upper compartment from a conservative value of 100 percent to a

realistic value of 65 percent based on plant operational data. The licensee ran three sensitivity cases with the upper compartment assuming relative humidity of 75 percent, 60 percent, and 50 percent while assuming the same initial maximum TS limits of 100 °F in the upper compartment, 120 °F in the lower compartment, and 100 percent relative humidity in the lower compartment. The calculated peak containment temperatures are 323.95 °F, 324.04 °F, and 324.12 °F for the upper compartment relative humidity of 75 percent, 60 percent, and 50 percent respectively. The sensitivity analysis result show that with initial relative humidity of 75 percent or less, the peak containment temperature is below the maximum acceptable limit of 324.7 °F.

The licensee stated that even though the split main steam line break is limiting for the proposed conditions, analysis was performed to determine the impact of a 100 °F upper containment temperature initial condition on the two most limiting DER cases which are: (1) 1.0 ft^2 at 0 percent power with an MSIV single failure, and (2) 1.4 ft^2 at 30 percent power with an AFW runout control single failure. The remaining inputs to the analysis remained unchanged. The peak containment temperature was calculated to be 323.26 °F for case (1) and 324.14 °F for case (2).

The licensee calculated the peak containment pressure for the most limiting split and DER breaks, assuming the TS maximum value 100 °F as the upper compartment initial temperature, while the remaining inputs remained the same. The licensee compared the pressure transient results for these breaks for the cases with 100 °F and 60 °F as initial upper compartment temperature. For a split break with 100 °F as the initial upper compartment temperature, the short term pressure transient and the peak pressure is slightly more limiting and becomes less limiting in the long term. For the two DER cases with 100 °F as the initial upper compartment temperature, the short term pressure transients, the peak pressures, and the long term pressure transients are less limiting.

The licensee stated that regardless of these results, the containment pressure response for LOCA is more limiting than the containment pressure response for the main steam line break. The above results justify that the initial conditions for the main steam line break containment analyses are defined to maximize the transient temperature and not the transient pressure response.

6.2.3.4 Sump Fluid Temperature Response and Net Positive Suction Head Analysis

By letter dated June 5, 2014 (Ref. 3), the licensee stated that the maximum recirculation sump water temperature used in the NPSHA analysis is not based on the accident analysis. It is based instead on WCAP-8282 and WCAP-8110, Supplement 6, which contain the test data taken during Waltz Mill Facility ice condenser testing performed in 1974 (Ref. 35). The sump water temperature is therefore not impacted by the changes in M&E releases. The UFSAR Revision 25, Section 14.3.4.1.3.1.3, "Peak Containment Pressure Transient," Item 4, identifies an ice condenser drain temperature of 190 °F as the sump fluid temperature value used for containment pressure analysis. The licensee's calculation of sump temperature less than 170 °F for the main steam line break at the initiation of the switchover to cold leg recirculation confirms that the assumption of 190 °F maximum sump temperature is conservative. Therefore the NPSH analysis for a main steam line break accident is acceptable.

6.2.3.5 Minimum Containment Back Pressure for Emergency Core Cooling Analysis

By letter dated June 5, 2014 (Ref. 3), the licensee described the method of analyzing the minimum containment back pressure for ECCS analysis, and the method of calculation to be used as an input to the ECCS analysis. Using the NRC-approved ASTRUM methodology (Ref. 9) for LBLOCA evaluation, the licensee analyzed a conservatively low containment back pressure. In addition, the licensee calculated the minimum containment back pressure by updating the revised inputs which are: (a) the air recirculation fan delay time, (b) containment spray initiation delay time, and (c) the containment spray flow, and by modeling the heat sinks and pressure-reducing equipment consistent with WCAP-8860. The latter method resulted in a slightly higher minimum containment back pressure. Figure 5.1.1-1 of Enclosure 6 to the license amendment request (Ref. 1) shows the minimum containment back pressure profiles obtained from the two methods.

6.2.4 Long Term LOCA Mass and Energy Release and Containment Integrity Analysis

The long term LOCA M&E release and containment integrity analysis is performed to confirm that the containment internal pressure, temperature, and the wall temperature do not exceed their design limits. The analysis should ensure that the containment heat removal system can remove the most limiting LOCA M&E released into the containment without exceeding the CNP-1 containment internal design pressure of 12 psig and the wall design temperature of 250 °F. The analysis should also ensure that the sump temperature change does not adversely impact the NPSH available for the pumps that draw fluid from the sump during the LOCA recirculation phase.

The LOCA M&E release analysis calculates the M&E of the break fluid, which is an input to the containment integrity pressure and temperature response analysis. The licensee revised the CNP-1 containment AOR for the following reasons.

<u>Reasons for Changes to M&E Release Analysis</u>

- To change the values of the input parameters for long term LOCA M&E release analysis, due to the change in the RCS NOP/NOT conditions.

- To correct the errors in the WCAP-10325-P-A (Ref. 36) methodology reported in Westinghouse Nuclear Safety Advisory Letter (NSAL)-06-6 (Ref. 37), NSAL-11-5 (Ref. 38), and NSAL-14-2 (Ref. 39).

- To resolve of the issue reported in Westinghouse InfoGram IG-14-1 (Ref. 40) which states that the volumetric heat capacity value used for the RCS metal mass in the current M&E analysis did not bound the values published by the ASME. Using the ASME published data for the volumetric heat capacity would increase the stored energy of the RCS metal during normal plant operation, and consequently increases the M&E release into the containment during a design basis LOCA.

Reasons for Changes in Inputs to Containment Pressure and Temperature Response

- To correct biasing of the containment compartments' initial air temperature inputs to the LOTIC1 code (Ref. 41) for a conservative containment pressure and temperature response.

- To correct biasing of the initial ice bed temperature input to the LOTIC1 code for conservative analysis. The 15 °F ice bed temperature assumed in the AOR is non-conservative; the TS SR 3.6.11.1 specifies a maximum value of 27 °F.

The licensee stated that the revised analysis has been performed and will be implemented into the CNP-1 licensing basis. The licensee further stated that the revised M&E analysis was performed using the WCAP-17721-P methodology (Ref. 42) which has been recently approved by NRC in a safety evaluation dated August 24, 2015 (Ref. 43). In the analysis, consistent with the NRC safety evaluation, the licensee assumed a conservative initial accumulator pressure which bounds the plant operating conditions and accounts for measurement uncertainty. Using the M&E release results, the licensee performed the revised containment response analysis using the AOR LOTIC1 methodology (Ref. 41). The licensee stated that the calculated peak containment pressure is determined to be 10.37 psig, which is bounded by the containment internal design pressure of 12 psig.

The NRC staff finds that the licensee used methods consistent with applicable regulatory requirements and guidance. The licensee performed the LOCA long term containment M&E release analysis using the NRC-approved methodology contained in WCAP-17721-P, and containment integrity analysis according to the methodology contained in WCAP-8354-P-A. Therefore the long term LOCA M&E release and containment integrity analysis is acceptable.

6.3 Containment and Ventilation Conclusion

The NRC staff determines that the licensee has demonstrated that containment integrity will be maintained under the increased temperature and pressure. Therefore, the staff finds that the proposed NOP and NOT increase is acceptable with respect to containment and ventilation.

7.0 BALANCE-OF-PLANT SYSTEMS

In its license amendment request, the licensee stated that balance-of-plant systems and components were analyzed by Westinghouse in WCAP-17762-NP for the effects of operation at the original licensed pressure of 2250 psia and T_{avg} of 571 °F. The following presents a summary of the evaluation for the CNP-1 balance-of-plant systems which are safety related or important to safety.

7.1 Main Steam System

The CNP-1 UFSAR, Section 10.2, "Main Steam System," provides a detailed system description, performance analysis, and testing and inspection for the system. According to the licensee's evaluation for the proposed increases to RCS pressure, the main steam system, consisting of piping and valves from the steam generator outlet nozzle to the main turbine

stop/control valves, safety and power operated relief valves, and turbine bypass piping and associated valves, would not be adversely impacted. Also, with the proposed changes in RCS conditions, the full power main steam pressure will increase to approximately 800 psig at steam generator outlet with a saturation temperature of 520.4 °F. However, the proposed conditions would remain within the main steam design parameters of 1100 psig and 550 °F. Furthermore, according to an evaluation of plant control systems in Section 4.1 of WCAP-17762-NP, operation at the proposed conditions would be bounded by existing control system analysis.

Based on its review of the above referenced documents and justifications provided by the licensee, the NRC staff determined that the main system would not be adversely affected by the proposed RCS NOP and NOT.

7.2 Condensate and Main Feedwater Systems

The CNP-1 UFSAR, Section 10.5, "Condensate and Feedwater System," describes the condensate (CS) and main feedwater (FW) system, its design basis, and testing and inspection requirements. The CS, in conjunction with the FW system, returns the condensed steam from the low-pressure turbines in the main condensers through the FW heaters to the steam generators while maintaining the overall water inventory throughout the cycle. The loss of steam and water during power operation is compensated from the condensate storage tanks. According to the licensee's evaluation, the slightly changed flow, pressure, and temperature due to the proposed operating conditions would remain within the limits of the CS and FW system design parameters. Furthermore, the thermal load on the condenser, and the condensate temperature, would remain essentially the same under the proposed changes.

Based on its review of the licensee's evaluation, the NRC staff determined that the CS and FW systems would not be adversely affected by the proposed increases in RCS NOP and NOT.

7.3 Auxiliary Feedwater System

The auxiliary feedwater (AFW) system is designed to remove residual heat from the reactor core upon loss of FW. The system design and related aspects are provided in the CNP-1 UFSAR, Section 10.5.2. The AFW system provides water to the steam generators under the transients of a loss of main FW system, loss of offsite power, main steam line break, or LOCA. The sources of water for AFW are the condensate storage tanks and the essential service water system (ESW).

As the licensee stated, the design basis flow rates for the AFW pumps were established prior to original plant licensing, when the RCS NOP and NOT were the same as those proposed in the current license amendment request. Based on its review of the licensee's justifications, the NRC staff determined that the AFW system would not be adversely affected by the proposed changes for RCS NOP and NOT.

7.4 Chemical and Volume Control System

The CNP-1 UFSAR, Section 9.2 provides the design details of the chemical and volume control system (CVCS). The CVCS provides reactivity control by regulating the concentration of the boric acid solution (the neutron absorber) for reactivity control in the RCS.

The licensee's evaluation states that the proposed increase in RCS T_{avg} does impact the pressurizer level control program, which could impact the CVCS normal operation. The function of the pressurizer level control system is to maintain the pressurizer level at or near its programmed level. During load changes, the pressurizer level setpoint varies automatically with T_{avg}, compensating partially for the expansion or contraction of reactor coolant associated with T_{avg} changes. The purpose of the pressurizer level program is to maintain an approximate constant mass inventory in the RCS during load changes, so that the CVCS charging rate will remain relatively constant. Therefore, in order to maintain a relatively constant CVCS charging rate, the licensee stated that the pressurizer water level control program will be revised consistent with existing Westinghouse design basis, to meet the proposed T_{avg} increase to 571 °F.

Based on the above discussion, the NRC staff determined that the CVCS system would not be adversely affected by the proposed changes for RCS NOP and NOT.

7.5 Component Cooling Water System

The CNP-1 UFSAR, Section 9.5 describes the Component Cooling Water (CCW) system. CCW is a safety-related system. It is designed to remove residual and sensible heat from the RCS via residual heat removal system during shutdown, cool the spent fuel pool water and the letdown flow to the CVCS during power operation, and provide cooling to various primary plant components and safeguards equipment.

The licensee performed an evaluation of the proposed increases to the RCS pressure and T_{avg}. According to the evaluation, these changes to RCS will have little, if any, effect on the CCW heat loads during normal operation. The pressurizer water level control program will be revised in order to be consistent with existing design basis and to maintain a relatively constant CVCS charging rate. Normal CVCS letdown flow rates are not expected to change due to the change in NOP and NOT. Therefore, similar CCW heat loads from the letdown heat exchanger will be maintained. Based on the above explanation, the NRC staff finds that the proposed changes to RCS T_{avg} will have little or no effect on the CCW heat loads.

7.6 Steam Generator Blowdown

The steam generator blowdown function is to remove the impurities from the secondary side water, and also to drain the steam generators during a plant outage. According to the licensee's evaluation, the current blowdown rate is controlled by the operator, depending on the system conditions. The steam generator blowdown system will continue to operate based on the original design when the plant NOP and NOT are restored. Based on its review, the NRC staff determined that the proposed changes to the NOP and NOT do not adversely affect the steam generator blowdown operation.

7.7 Chemical Feed System

The CNP-1 UFSAR, Section 10.10 provides information about the chemical injection into the CS and FW system. The chemical injection maintains a non-corrosive, non-scale forming within the CS and FW system. The proposed increases in RCS pressure and temperatures match the

original design conditions of the RCS. Therefore, the NRC staff determined that the proposed increase in RCS NOP and NOT will not adversely affect the FW flow rate or chemical injection from the Chemical Feed System into the CS and FW system.

7.8 Essential Service Water System

The function of the ESW system is to provide cooling water to the CCW heat exchangers, the containment spray heat exchangers, the EDG coolers, the AFW pump enclosure coolers, and the control room air conditioning condensers. The ESW cooling requirements for the above equipment are not affected by the RCS NOP and NOT. Therefore, the NRC staff finds that the ESW system is not adversely affected by the proposed increases in RCS NOP and NOT.

7.9 Fire Protection Systems

The fire protection functions are independent of the RCS operating characteristics, and therefore the proposed changes have no impact on the fire protection systems.

7.10 High Energy Line Break

According to the licensee, the high energy line break analysis was performed at a power of 3426 megawatts thermal (MWt) and a steam pressure of 820 psia, which bounds the proposed changes. Accordingly, no adverse effects are identified due to the proposed increase in RCS NOP and NOT. Therefore, the NRC staff finds that the high energy line break analysis is not adversely affected by the proposed increase in RCS NOP and NOT.

7.11 Spent Fuel Pool Cooling System

As stated in WCAP-17762-NP, the primary function of the spent fuel pool (SFP) cooling system is to remove decay heat that is generated by the spent fuel assemblies stored in the pool. The decay heat generation is proportional to the power level, and as the reactor power level of 3304 MWt is not increasing, the demands on the SFP cooling system are not increased. Therefore, the NRC staff concludes that the SFP cooling system is not adversely affected by the proposed changes to the RCS NOP and NOT.

7.12 Ice Condenser Refrigeration System

The CNP-1 UFSAR, Section 5.3 provides a detailed description of the CNP-1 Ice Condenser. In the ice condenser, ice is held in baskets, and a refrigeration system maintains the ice in the solid state. The ice condenser is sufficiently subcooled and maintains the desired equilibrium temperature in the ice compartment. Suitable insulation surrounding both the ice condenser volume and the refrigeration ducts serve to minimize the heat transfer to the ice condenser boundaries. Even a complete breakdown of the refrigeration system or air-handling system would not cause ice melting for one week. According to the licensee's evaluation, the allowable containment temperature during operation according to TS 3.6.5 and 3.6.11 are not changed by the proposed amendment to return to the original NOP/NOT conditions. Therefore, the NRC staff finds that the ice condenser refrigeration system is not adversely affected by the proposed increase in RCS NOP and NOT.

7.13 Balance-of-Plant Systems Conclusion

Based on its review of the licensee's evaluation of the proposed changes to the RCS NOP and NOT to the original operating conditions and the justifications provided in WCAP-17762-NP, the NRC staff concluded that these changes submitted in its license amendment request will have no adverse impact on the CNP-1 balance-of-plant systems.

8.0 RADIATION PROTECTION

8.1 Regulatory Evaluation

The NRC staff evaluated the impact of the proposed changes on the previously analyzed radiological consequences of design basis accidents (DBAs). The regulatory requirements are the accident dose criteria in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and PSDC-11, "Control room." The NRC staff also considered the relevant information in the CNP-1 UFSAR. The applicable acceptance criteria are 5 rem total effective dose equivalent in the control room, 300 rem thyroid and 25 rem whole body at the exclusion area boundary, and 300 rem thyroid and 25 rem whole body at the outer boundary of the low population zone. The NRC staff used the regulatory guidance provided in Regulatory Guide (RG) 1.195, (Ref. 44). RG 1.195 provides guidance to licensees on acceptable methods and assumptions for performing evaluation of fission product releases and radiological consequences of several DBAs.

The NRC approved the selective implementation of the alternative source term limited to control room habitability assessments at CNP-1 by License Amendment No. 271 (Ref. 45). This amendment replaced the current accident source term used in design-basis radiological analyses for control room habitability with an alternative source term pursuant to 10 CFR 50.67, "Accident Source Term."

8.2 Technical Evaluation

The proposed NOP and NOT conditions will impact the RCS mass, which is used as a design input for the dose consequence analyses and its supporting calculations. With the proposed increase to RCS pressure and temperature, RCS fluid density will decrease, therefore affecting the previous maximum RCS liquid mass value used in some of the dose consequence analyses. The licensee stated that for the current AOR, the upper-bound liquid mass is 2.3874E+08 grams and the lower-bound liquid mass is 2.2649E+08 grams. At NOP/NOT conditions, the CNP-1 liquid mass is calculated to be 2.3696E+08 grams. Because the calculated value is within the current AOR upper-bound and lower-bound liquid mass values, the current AOR remains bounding for the proposed change.

Therefore, NRC staff finds that the current offsite and control room habitability dose AORs remain bounding and are not affected by the proposed return to NOP/NOT conditions for Unit 1.

In support of the proposed amendment change, the licensee would also revise the follow parameters:

- CEQ fan actuation delay time from 180 seconds to 300 seconds.

- Containment Spray actuation delay time, assuming a loss of offsite power, from 180 seconds to 300 seconds.

The licensee reviewed the offsite and control room habitability radiological dose consequence analyses and supporting calculations. It was determined that only the LBLOCA analysis required further evaluation due to proposed changes to the CEQ and containment spray fan actuation times. The remaining dose analyses are not affected since they do not model CEQ or containment spray fan actuation. The increased containment spray and CEQ delay times had a minimal effect on the offsite radiological dose results. Table 2 shows the LBLOCA calculated dose results provided by the licensee.

Table 2: LBLOCA Calculated Dose

Control Room	4.26 rem total effective dose equivalent
Offsite exclusion area boundary	236 rem* / 2.64 rem (thyroid / whole body)
Offsite low population zone	176 rem / 0.864 rem (thyroid / whole body)

The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its LBLOCA radiological dose consequence analysis and finds that that they are consistent with the guidance provided in RG 1.195. Additionally, the small increase in dose is within the regulatory guideline values. Therefore, the NRC staff finds the proposed change is acceptable with respect to the radiological consequences of DBAs.

8.3 Radiation Protection Conclusion

As described above, the NRC staff reviewed the proposed changes, assumptions, and parameters used by the licensee to assess the radiological consequences of the postulated DBA analyses at CNP-1. The NRC staff finds that the licensee used methods consistent with applicable regulatory requirements and guidance. The NRC staff finds that CNP-1, as modified by this license amendment request, would continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the radiological consequences of postulated DBAs.

9.0 STATE CONSULTATION

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

10.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or 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 NRC has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published February 19, 2014 (79 FR 9495). 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.

11.0 CONCLUSION

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

12.0 REFERENCES

1. Letter from J.P. Gebbie, I&M, to NRC, "License Amendment Request Regarding Restoration of Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions," October 8, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML13283A121 and ML13283A122).

2. Letter from J.P. Gebbie, I&M, to NRC, "Response to 'Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916),'" April 29, 2014 (ADAMS Accession No. ML14121A422).

3. Letter from J.P. Gebbie, I&M, to NRC, "Response to 'Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916),' Dated May 6, 2014," June 5, 2014 (ADAMS Accession No. ML14181A537).

4. Letter from Q.S. Lies, I&M, to NRC, "Response to 'Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916),' Dated May 6, 2014," July 3, 2014 (ADAMS Accession No. ML14189A102).

5. Letter from J.P. Gebbie, I&M, to NRC, "Response to 'Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916),'" September 30, 2014 (ADAMS Accession No. ML14275A453).

6. Letter from Q.S. Lies, I&M, to NRC, "Supplemental Response to 'Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System

Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916),'" September 18, 2015 (ADAMS Accession No. ML15265A036).

7. NUREG-0800, NRC SRP, Section 6.3, "Emergency Core Cooling System," Rev. 3, March 2007 (ADAMS Accession No. ML070550068).

8. NUREG-0800, NRC SRP, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Rev. 3, March 2007 (ADAMS Accession No. ML070550016).
9. WCAP-16009-NP-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Westinghouse Electric Company, January 2005 (ADAMS Accession No. ML050910159).

10. Letter from T.A. Beltz, NRC, to M.W. Rencheck, I&M, "Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment to Renewed Facility Operating License Regarding Use of the Westinghouse ASTRUM Large Break Loss-of-Coolant Accident Analysis Methodology (TAC No. MD7556)," October 17, 2008 (ADAMS Accession No. ML082670351).

11. Letter from J.A. Gresham, Westinghouse Electric Company, to NRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012 (ADAMS Accession No. ML12072A035).

12. Letter from J.N. Jenson, I&M, to NRC, "License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," December 27, 2007 (ADAMS Accession No. ML080090268).

13. Letter from J.P. Gebbie, I&M, to NRC, "Response to Information Request Pursuant to 10 CFR 50.54(f) Related to the Estimated Effect on Peak Cladding Temperature Resulting from Thermal Conductivity Degradation in the Westinghouse-Furnished Realistic Emergency Core Cooling System Evaluation (TAC No. M99899)," March 19, 2012 (ADAMS Accession No. ML12088A104).

14. Letter from J.P. Gebbie, I&M, to NRC, "U.S. Nuclear Regulatory Commission Commitment Change Related to Estimated Effect of Peak Cladding Temperature Resulting from Thermal Conductivity Degradation," June 9, 2015 (ADAMS Accession No. ML15162A095).

15. Letter from J.P. Gebbie, I&M, to NRC, "Revised Small Break Loss-of-Coolant Accident Analysis," August 31, 2012 (ADAMS Accession No. ML12256A685).

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17. NUREG-0800, NRC SRP, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Rev. 3, March 2007 (ADAMS Accession No. ML070550006).

18. WCAP-11397-P-A (proprietary), "Revised Thermal Design Procedure," Westinghouse Electric Corporation, April 1989 (ADAMS Accession No. ML080630437).

19. WCAP-7907-P-A (proprietary), "LOFTRAN Code Description," Westinghouse Electric Corporation, April 1984 (ADAMS Accession No. ML080650325).

20. WCAP-10965-P-A (proprietary), "ANC: A Westinghouse Advanced Nodal Computer Code," Westinghouse Electric Corporation, December 1985 (ADAMS Accession No. ML080630392).

21. WCAP-14565-P-A (proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Westinghouse Electric Company, October 1999 (ADAMS Accession No. ML993160153).

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24. Generic Letter 1988-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," October 4, 1988 (ADAMS Accession No. ML031200485).

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30. Letter from T.J. Wengert, NRC, to L.J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Evaluation of Report Concerning Significant Emergency Core Cooling System Evaluation Model Error Related to Nuclear Fuel Thermal Conductivity Degradation (TAC Nos. ME8322 and ME8323)," March 7, 2013 (ADAMS Accession No. ML13077A137).

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Date: November 30, 2015

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

Sincerely,

/RA/

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures:
1. Amendment No. 329 to DPR-58
2. Safety Evaluation

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