



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

October 17, 2008

Mr. Michael W. Rencheck
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 TO RENEWED FACILITY OPERATING LICENSE REGARDING USE OF THE WESTINGHOUSE ASTRUM LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY (TAC NO. MD7556)

Dear Mr. Rencheck:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 306 to Renewed Facility Operating License No. DPR-58 for Unit 1 of the Donald C. Cook Nuclear Plant. The amendment changes the Technical Specifications (TS) in response to your application dated December 27, 2007, as supplemented by letter dated July 14, 2008.

The amendment revises TS Section 3.4.1, "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and modifies the analytical method used for determining core operating limits for a large break loss-of-coolant accident (LBLOCA) specified in TS 5.6.5, "Core Operating Limits Report (COLR)." The proposed amendment also requested NRC approval of a new LBLOCA analysis using a plant-specific adaptation of Westinghouse topical report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)."

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

Sincerely,

A handwritten signature in black ink, appearing to read "Terry A. Beltz", written over a horizontal line.

Terry A. Beltz, Senior 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. 306 to DPR-58
2. Safety Evaluation

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NUCLEAR REGULATORY COMMISSION
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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
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 December 27, 2007, as supplemented by letter dated July 14, 2008, 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 Appendix B, as revised through Amendment No. 306, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Lois M. James, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Operating License
and Appendix A

Date of Issuance: October 17, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 306
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 revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE

- 3 -

INSERT

- 3 -

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

REMOVE

3.4.1-1
3.4.1-2
5.6.3

INSERT

3.4.1-1
3.4.1-2
5.6.3

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A and Appendix B, as revised through Amendment No. 306, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(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) Indiana Michigan Power Company shall implement and maintain, in effect, all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated December 12, 1977, July 31, 1979, January 10, 1981, February 7, 1983, November 22, 1983, December 23, 1983, March 16, 1984, August 27, 1985

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

- a. Pressurizer pressure is greater than or equal to the limit specified in the COLR;
- b. RCS average temperature is less than or equal to the limit specified in the COLR; and
- c. RCS total flow rate is greater than or equal to the limit specified in the COLR. The minimum RCS total flow rate shall be $\geq 354,000$ gpm.

APPLICABILITY: MODE 1.

-----NOTE-----
Pressurizer pressure limit does not apply during:

- a. THERMAL POWER ramp > 5% RTP per minute; or
- b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is $\geq 354,000$ gpm and greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	<p>-----NOTE-----</p> <p>Not required to be performed until 24 hours after $\geq 90\%$ RTP.</p> <p>-----</p> <p>Verify by precision heat balance that RCS total flow rate is $\geq 354,000$ gpm and greater than or equal to the limit specified in the COLR.</p>	24 months

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. LCO 3.1.6, "Control Bank Insertion Limits";
 6. LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
 7. LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)";
 8. LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
 9. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," Functions 6 and 7 (Overtemperature ΔT and Overpower ΔT , respectively) Allowable Value parameter values;
 10. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
 11. LCO 3.9.1, "Boron Concentration."
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Westinghouse Proprietary);
 2. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," (Westinghouse Proprietary);
 3. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/ F_Q Surveillance Technical Specification," (Westinghouse Proprietary);
 4. Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary);
 5. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," (Westinghouse Proprietary);
 6. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," (Westinghouse Proprietary); and
 7. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," (Westinghouse Proprietary).



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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 306

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 December 27, 2007 (Agencywide Document and Access Management System (ADAMS) Accession No. ML080090268), as supplemented by letter dated July 14, 2008 (ADAMS Accession No. ML082040584), the Indiana Michigan Power Company (the licensee) requested a license amendment to revise the Technical Specifications (TS) for the Donald C. Cook Nuclear Plant, Unit 1 (DCCNP-1). The supplemental letter of July 14, 2008, provided additional information to clarify the application; did not expand the scope of the application as originally noticed; and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 29, 2008 (73 FR 5223).

The proposed amendment would increase the required minimum Reactor Coolant System (RCS) flow rate specified in TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and modify the analytical method used for determining core operating limits for a large break loss-of-coolant accident (LBLOCA) specified in TS 5.6.5, "Core Operating Limits Report (COLR)" for DCCNP-1.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 includes the NRC's requirement that TSs shall be included by applicants for a license authorizing operation of a production or utilization facility. 10 CFR 50.36 (d) requires that TSs include items in five specific categories related to station operation. These categories are (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operations (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS. This amendment is within categories (2), (3), and (5).

Title 10 of the *Code of Federal Regulations* (10 CFR), Paragraph 50.46(a)(1) identifies the calculation methodology requirements for nuclear power plant LOCA methodologies. 10 CFR 50.46(c) identifies the types of processes which are required to assure that LOCA analyses performed for a given plant actually represent the plant. Section 50.46(a)(3)(i) and (ii) specify criteria to be applied and actions to be taken when significant changes or errors in parts of the plant-specific LOCA methodology are found to have accumulated.

Enclosure

In accordance with 10 CFR 50.46(a)(3)(ii), the licensee submitted a new DCCNP-1 LBLOCA analysis to the NRC due to an accumulation of changes and errors requiring a scheduled reanalysis. The new LBLOCA analysis uses a higher RCS flow rate value to provide a greater margin to DNB limits and requires a change to the minimum flow rate specified in LCO 3.4.1.c. and surveillance requirements 3.4.1.3 and 3.4.1.4. The new LBLOCA analysis is referenced in the COLR, which is one of the administrative controls to assure operation of the facility in a safe manner and is referenced in the DCCNP-1 TSs, Section 5.6.5.

3.0 TECHNICAL EVALUATION

3.1 Description of Proposed Changes

The licensee requested approval to use a plant-specific version of the enhanced adaptation of the WCAP-16009-P-A methodology containing an enhanced treatment of the downcomer. The licensee also requested approval of a DCCNP-1 inclusion of this enhanced version of the WCAP-16009-P-A in the DCCNP-1 TSs.

The licensee proposed to revise TS 5.6.5 by replacing the existing LBLOCA methodology specified in TS 5.6.5.b.4 with the following: "Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)."

The licensee also proposed to revise TS Limiting Condition for Operation 3.4.1.c and Surveillance Requirements 3.4.1.3 and 3.4.1.4 to reflect a higher minimum RCS flow rate of 354,000 gallons per minute.

3.2 Evaluation of Proposed Changes

The current DCCNP-1 LBLOCA analysis of record was performed in calendar year 2000 using the BASH evaluation model methodology documented in WCAP-10266-P-A, "the 1981 Version of Westinghouse ECCS [emergency core cooling system] Evaluation Model Using the BASH Code." A new LBLOCA analysis for DCCNP-1 was provided to the NRC due to an accumulation of changes and errors requiring a schedule reanalysis in accordance with 10 CFR 50.46(a)(3)(ii). The ASTRUM methodology received NRC approval for referencing in licensing applications by the safety evaluation in WCAP-16009-P-A.

The licensee provided the following statement to confirm that the reference generically-approved ASTRUM LOCA analysis methodology applies specifically to DCCNP-1: "I&M and Westinghouse have ongoing processes that assure that the ranges and values of the input parameters for the DCCNP-1 LBLOCA analyses bound the ranges and values of the as-operated plant parameters."

To address the effects, if any, of the mixed core on peak cladding temperature and oxidation for the pre-resident fuel, the licensee states that both the pre-resident fuel and the fresh fuel are of the same design and therefore there are no mixed core effects to consider.

The results of the DCCNP-1 LBLOCA analyses are provided in Table 1.

Table 1

D.C. Cook Unit 1: Large Break LOCA Analysis Results Using ASTRUM

Parameter	ASTRUM LBLOCA Analyses Results	10 CFR 50.46 Limits
Cladding Material	ZIRLO	(Cylindrical) Zircaloy or ZIRLO
Peak Cladding Temperature	2128 °F	≤ 2200 °F
Maximum Local Oxidation Percentage	11.1 %	≤ 17 %
Core-Wide H ₂ Generation (Oxidation)	0.40 %	≤ 1.0 %

The burnup effects on oxidation for the pre-resident fuel were addressed by providing the expected maximum values of LOCA oxidation at beginning of life (~10%) and end of life (~0%). The NRC staff agrees with the conclusion that regardless of what time in the “fuel life” the postulated LBLOCA were to occur, the anticipated total LOCA oxidation would be less than 17%.

To verify that the treatment of the vessel wall (radial noding, etc.) during reflood remains as historically approved in addressing the issue of downcomer boiling, the licensee indicated that the number of downcomer nodes in the DCCNP-1 analyses of downcomer boiling had been increased versus the previous generic noding. The NRC staff finds that the DCCNP-1 downcomer noding is consistent with downcomer noding for other similar Westinghouse plants that addressed the issue and, therefore, is acceptable.

Limits on RCS pressure, temperature, and flow rate ensure that the minimum DNB ratio will be met for each of the safety analysis transients analyzed. RCS flow rate normally remains constant during an operational fuel cycle with all reactor coolant pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to determine a value for comparison to the limit. The new LBLOCA analysis uses a higher RCS flow rate value that provides a greater margin to DNB limits and requires a change to the minimum flow rate specified in TS 3.4.1.

3.3 Summary

The NRC staff finds the ASTRUM LBLOCA methodology and associated amended TSs are acceptable for licensing application to DCCNP-1. The specific DCCNP-1 LBLOCA analyses performed using ASTRUM and the results of those analyses show compliance with the provisions of 10 CFR 50.46. The NRC staff finds the TS changes to be acceptable, as they accurately reflect the changes found acceptable in this safety evaluation.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (73 FR 5223). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The staff 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) 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.

Principal Contributor: Frank Orr, NRR

Date: October 17, 2008

Mr. Michael W. Rencheck
 Senior Vice President and
 Chief Nuclear Officer
 Indiana Michigan Power Company
 Nuclear Generation Group
 One Cook Place
 Bridgman, MI 49106

October 17, 2008

SUBJECT: 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)

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Sincerely,

/RA/

Terry A. Beltz, Senior 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. 306 to DPR-58
 2. Safety Evaluation

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Package Accession Number: **ML082670379**

Amendment Accession Number: **ML082670351** TS: **ML082670393**

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