INDIANA MICHIGAN POWER

Indiana Michigan Power Company
Nuclear Generation Group
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July 14, 2008

AEP-NRC-2008-10
10 CFR 50.90

Docket No. 50-315

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Response to Request for Additional Information Regarding Reanalysis of Large Break Loss-of-Coolant Accident (TAC No. MD7556)

REFERENCES:
1. Letter from Joseph N. Jensen, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," AEP:NRC:7565-01, dated December 27, 2007 (ML080090268).

2. Letter from Peter S. Tam, NRC, to Michael W. Rencheck, I&M, "D. C. Cook Nuclear Plant, Unit 1 (DCCNP-1) - Request for Additional Information, Regarding Re-analysis of Large-Break Loss-of-Coolant Accident (TAC No. MD7556)," dated June 5, 2008 (ML081570070).

Dear Sir or Madam:

By Reference 1, Indiana Michigan Power Company proposed to amend Facility Operating License DPR-58, for the Donald C. Cook Nuclear Plant, Unit 1. The proposed amendment would revise the Technical Specifications (TS) to increase the required minimum Reactor Coolant System 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)." The proposed amendment also requested U. S. Nuclear Regulatory Commission (NRC)-approval of a new Unit 1 LBLOCA analysis using a plant-specific adaptation of topical report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)." Reference 2 transmitted an NRC Request for Additional Information regarding the proposed amendment. This letter provides the requested information.

Enclosure 1 to this letter provides an affirmation statement regarding information in this letter. Enclosure 2 provides information requested by Reference 2. Enclosure 3 provides information on modeling errors that have been corrected for the Residual Heat Removal and Safety Injection Systems resulting in a reduction in the minimum calculated Emergency Core Cooling System flow

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rates used in the LBLOCA analysis submitted by Reference 1. Copies of this letter and its enclosures are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91. This letter contains no new regulatory commitments. Should you have any questions, please contact Mr. John A. Zwolinski, Regulatory Affairs Manager, at (269) 466-2478.

Sincerely,

Lawrence J. Weber

Lawrence J. Weber
Site Vice President

KAS/rdw

Enclosures:

1. Affirmation
2. Response to Request for Additional Information
3. Evaluation of Reduction in Emergency Core Cooling System Flow

c: J. L. Caldwell, NRC Region III
K. D. Curry, AEP Ft. Wayne, w/o enclosures
J. T. King, MPSC
MDEQ – WHMD/RPS
NRC Resident Inspector
P. S. Tam, NRC Washington, DC

AFFIRMATION

I, Lawrence J. Weber, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Lawrence Weber

Lawrence J. Weber
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 14th DAY OF July, 2008

Notary Public

My Commission Expires 6/10/2013

Enclosure 2 to AEP-NRC-2008-10

Response to Request for Additional Information

Documents referenced in this enclosure are identified on Page 3.

By Reference 1, Indiana Michigan Power Company (I&M) proposed to amend Facility Operating License DPR-58, for the Donald C. Cook Nuclear Plant (CNP), Unit 1. The proposed amendment would revise the Technical Specifications (TS) to increase the required minimum Reactor Coolant System 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)." The proposed amendment also requested U. S. Nuclear Regulatory Commission (NRC)-approval of a new Unit 1 LBLOCA analysis using a plant-specific adaptation of topical report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)." Reference 2 transmitted an NRC Request for Additional Information (RAI) regarding the proposed amendment. Each question in the RAI is restated below followed by I&M's response.

RAI Question 1

To show that the referenced generically approved ASTRUM LOCA analysis methodology applies specifically to the DCCNP-1, please provide a statement that Indiana Michigan Power and its vendor (Westinghouse) have ongoing processes which assure that the ranges and values of the input parameters for DCCNP-1 LOCA analyses bound the ranges and values of the as-operated plant parameters.

I&M Response to Question 1

Both I&M and Westinghouse have ongoing processes that assure that the ranges and values of the input parameters for the CNP Unit 1 LBLOCA analysis conservatively bound the ranges and values of the as-operated CNP Unit 1 parameters.

RAI Question 2

The discussion in the December 27, 2007, submittal did not address the effects of the mixed core on peak cladding temperature (PCT) and oxidation for the pre-resident fuel, but it does seem to address the PCT and oxidation for the new fuel. Please clarify if this is a whole core reload. In its Rulemaking Hearing dated December 28, 1983, the Nuclear Regulatory Commission stated, regarding the performance criteria of 10 CFR 50.46 (b): "In view of the lack of experience in this hypothetical situation, we think it prudent to apply our criteria to all of the core and not to exempt any part."

Please address PCT and oxidation results for "pre-resident" fuel in the core, if any.

[Note: In a letter to NEI dated November 8, 1999, Gary M. Holahan, reiterated the NRC position that "total oxidation" encompasses accident and pre-accident oxidation. This position continues to apply. Therefore, in response to this question, please provide total oxidation for the "other" (pre-resident) fuel (if any), including pre-accident oxidation, plus LOCA cladding outside oxidation, plus cladding inside oxidation.]

I&M Response to Question 2

The current CNP Unit 1 core, Cycle 22, which has been operating since spring 2008, consists entirely of fuel assemblies of the 15x15 Upgrade Fuel design with ZIRLO™ cladding. The Reference 1 LBLOCA analysis modeled a full core of this same design. Since the core does not include a mixture of different fuel assembly designs, there are no mixed core effects to consider.

The ASTRUM methodology includes treatment of core burnup effects on stored energy and PCT as described in Section 11-2-2 of Reference 3. The NRC described this aspect of the methodology in Section 3.3.1.2 of Reference 4 and found it to be acceptable. The CNP Unit 1 analysis (Reference 1) has addressed PCT over a range of burnup values applicable to core life consistent with the approved methodology.

The pre-accident oxidation was not factored into the local maximum oxidation results presented in Reference 1. The maximum expected total of the normal operation (pre-accident or pre-transient) and LOCA transient oxidation, for any time in life, is considered in the CNP Unit 1 analysis as described below. The pre-transient oxidation increases with burnup, from zero at the beginning of life (BOL) to a maximum value at fuel assembly end of life (EOL). The transient oxidation decreases from 10.0 percent (%) near the BOL for CNP Unit 1 ASTRUM analysis to a negligible value at EOL. The transient oxidation is calculated within the ASTRUM methodology, including the contribution of both outside and inside (following rod burst) cladding oxidation. This transient oxidation and the plant-specific pre-transient oxidation were used to assess total oxidation. It has been confirmed that the sum of the pre-transient plus transient oxidation remains below 17% at all times in life for the 15x15 Upgrade Fuel design with ZIRLO™ cladding.

RAI Question 3

Please verify that the treatment of the vessel wall (radial noding, etc.) during reflood remains as historically approved in addressing the issue of downcomer boiling.

I&M Response to Question 3

The detailed radial noding of the vessel wall remains unchanged from the approved ASTRUM LBLOCA Evaluation Model (References 3 and 5) and therefore, does not change the historically approved method for addressing downcomer boiling during reflood. The only difference from the previous noding method is that the vessel wall is partitioned into twelve segments connected to twelve downcomer channel "stacks" versus the four in the generic noding.

References

1. Letter from Joseph N. Jensen, I&M, to NRC Document Control Desk, "License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," AEP:NRC:7565-01, dated December 27, 2007 (ML080090268).

2. Letter from Peter S. Tam, NRC, to Michael W. Rencheck, I&M, "D. C. Cook Nuclear Plant, Unit 1 (DCCNP-1) - Request for Additional Information, Regarding Re-analysis of Large-Break Loss-of-Coolant Accident (TAC No. MD7556)," dated June 5, 2008 (ML081570070).

3. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," dated January 2005.

4. Letter from Herbert N. Berkow, NRC, to James A. Gresham, Westinghouse, "Final Safety Evaluation for WCAP-16009-P, Revision 0, 'Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)' (TAC No. MB9483)," dated November 5, 2004 (ML043100073).

5. WCAP-12945-P-A, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis," dated March 1998.

Enclosure 3 to AEP-NRC-2008-10

Evaluation of Reduction in Emergency Core Cooling System Flow

The document referenced in this enclosure is identified on Page 2.

The ASTRUM large break loss-of-coolant accident (LBLOCA) analysis assumption for Emergency Core Cooling System (ECCS) flow is the minimum flow rate with the loss of one ECCS train. Since submittal of the Donald C. Cook Nuclear Plant (CNP) Unit 1 LBLOCA analysis by the referenced letter, errors in the calculated flow rates for the Residual Heat Removal (RHR) and Safety Injection (SI) Systems have been identified. The correction of these errors has resulted in a reduction in the minimum calculated ECCS flow rates applicable to the CNP Unit 1 LBLOCA analysis.

The maximum total reduction for the combined RHR and SI flow rate, due to the identified errors, is approximately 200 gallons per minute. The Charging System flow rate remains unchanged.

The reduction in the minimum calculated ECCS flow rate requires an estimated effect of the error to be determined. From the 124 cases performed for the CNP Unit 1 ASTRUM LBLOCA analysis as discussed in the referenced letter, the same case identified in the referenced letter proved to be the limiting case with respect to peak cladding temperature (PCT), local maximum oxidation (LMO), and core wide oxidation (CWO). The approved WCOBRA/TRAC and HOTSPOT codes were used to assess the impact on PCT, LMO, and CWO by using the corrected ECCS flow rate as input to the limiting transient. The resulting PCT penalty of the reduced ECCS flow rate was 22 degrees Fahrenheit (°F). The revised limiting PCT (2128°F) continues to remain below the 2200°F acceptance criterion. The resulting LMO of 11.1 percent (%) is a 1.1% increase from the analysis presented in the referenced letter. The resulting CWO of 0.40% is a 0.05% increase from the analysis presented in the referenced letter.

For the LBLOCA analysis, limiting calculated transient oxidation is obtained from fresh fuel (first cycle of irradiation). The maximum expected total of the normal operation (pre-transient) and the transient oxidation, for any time in life, was considered for CNP Unit 1. The pre-transient oxidation increases with burnup, from zero at the beginning of life (BOL) to a maximum value at fuel assembly end of life (EOL). The transient oxidation decreases from 11.1% (determined for the ECCS flow reduction) near the BOL to a negligible value at EOL. The transient oxidation is calculated within the ASTRUM methodology, including the contribution of both outside and inside (following rod burst) cladding oxidation. This transient oxidation and the plant-specific pre-transient oxidation were used to assess total oxidation. It has been confirmed that the sum of the pre-transient plus transient oxidation remains below 17% at all times in life for the 15x15 Upgrade Fuel design with ZIRLO™ cladding with the reduced ECCS flow. An update to the 10 CFR 50.46 acceptance criteria provided in Table 2 of the referenced letter is provided below:

LBLOCA Fuel Cladding Results

ASTRUM Result	Value	Criteria
95/95 PCT (°F)	2128	< 2,200
95/95 LMO (%) [1]	11.1	< 17
95/95 CWO (%)	0.40	< 1

1. The maximum total oxidation of any fuel in the core, including pre-transient oxidation, is less than 17% throughout the life of the fuel.

Reference

Letter from Joseph N. Jensen, I&M, to NRC Document Control Desk, "License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," AEP:NRC:7565-01, dated December 27, 2007 (ML080090268).