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Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
AEP.com

December 7, 2006

AEP:NRC:6046-01
10 CFR 50.46

Docket No.: 50-315

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

Donald C. Cook Nuclear Plant Unit 1
THIRTY-DAY REPORT FOR LOSS-OF-COOLANT
ACCIDENT EVALUATION MODEL CHANGE

References:
1. Letter from Joseph N. Jensen, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, 10 CFR 50.46 Loss-of-Coolant Accident Reanalysis Schedule," submittal AEP:NRC:4046-01, Accession Number ML050040216, dated December 28, 2004.
2. Letter from Joseph N. Jensen, I&M, to U. S. NRC Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1, Thirty-day Report of Loss-of-Coolant Accident Evaluation Model Changes," submittal AEP:NRC:5046, Accession Number ML051300368, dated April 29, 2005.

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP), is submitting a 30-day report of loss-of-coolant accident (LOCA) model changes resulting in a significant change in calculated peak fuel cladding temperature (PCT) for the CNP Unit 1 large break LOCA (LBLOCA) analysis. A significant change is defined as a change or error identified in the model which results in a calculated PCT greater than 50 degree Fahrenheit (°F) or a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

Attachment 1 to this letter describes an assessment against the CNP Unit 1 LBLOCA analysis of record. Attachment 2 provides the CNP Unit 1 LBLOCA analysis of record PCT value and error assessments. Attachment 2 also demonstrates that the PCT value remains within the 2200°F PCT limit as required by 10 CFR 50.46(b)(1).

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The overall change to the Unit 1 LBLOCA analysis is classified as significant in accordance with 10 CFR 50.46(a)(3)(i). By Reference 1, I&M submitted a schedule for reanalysis of the Unit 1 and Unit 2 small break LOCA (SBLOCA) and the Unit 2 LBLOCA analyses of record. By Reference 2, I&M submitted a schedule for reanalysis of the Unit 1 LBLOCA analyses of record. I&M's current schedule for the reanalyses is provided in Attachment 3.

Should you have any questions concerning this subject, please contact Ms. Susan D. Simpson, Regulatory Affairs Manager, at (269) 466-2428.

Sincerely,

Joseph N. Jensen
Site Support Services Vice President

Attachments

KAS/rdw

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

ATTACHMENT 1 TO AEP:NRC:6046-01

ASSESSMENT AGAINST THE LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSES OF RECORD

Indiana Michigan Power Company's (I&M's) most recent annual 10 CFR 50.46 report for Donald C. Cook Nuclear Plant (CNP) has been submitted by Reference 1. The reported assessments on the Unit 1 large break loss-of-coolant accident (LBLOCA) analysis of record peak cladding temperature (PCT) in Attachment 2 of this letter is based upon the limiting fresh 15x15 Upgrade Fuel introduced into the CNP Unit 1 Cycle 21 (U1C21) core. A new PCT assessment against the CNP Unit 1 LBLOCA analysis of record is described below. The new assessment is reflected in the PCT accounting in Attachment 2.

Assessment Against the LBLOCA Analysis of Record

15x15 Upgrade Fuel

Background

I&M and members of the Nuclear Regulatory Commission (NRC) Staff began discussion of the planned introduction of the 15x15 Upgrade Fuel for CNP Unit 1 to address fuel fretting wear issues at a February 3, 2005, meeting. Subsequent discussions were noted in an NRC summary of a July 25, 2005, telephone conference that was documented in Reference 2. As planned, the 15x15 Upgrade Fuel has been introduced into the CNP U1C21 core in accordance with 10 CFR 50.59. The CNP Unit 1 LBLOCA analysis was evaluated as part of the engineering work performed to support the U1C21 core reload. The PCT assessment associated with this evaluation is greater than 50 degrees Fahrenheit (°F) and is being reported in accordance with 10 CFR 50.46(a)(3)(ii).

Affected Evaluation Models

1981 Westinghouse LBLOCA Evaluation Model with BASH

Estimated Effect

The impact on PCT was estimated using plant-specific calculations with the BASH evaluation model. As indicated in the PCT accounting in Attachment 2, the effect of the change to the 15x15 Upgrade Fuel is a 59°F reduction. The previously determined 31°F penalty associated with transition core effects due to intermediate flow mixer (IFM) grids no longer applies, since all fuel assemblies that comprise the U1C21 core have IFMs. Additionally, the previously identified 37°F penalty associated with the "Spacer Grid Blocked Area Ratio/Open Area Fraction" does not apply to the 15x15 Upgrade Fuel. The net result of these assessments and removed penalties is a 127°F reduction in the calculated PCT with assessments compared to that previously reported in Reference 1.

Conclusion

This transmittal satisfies the 30-day reporting requirement of 10 CFR 50.46(a)(3)(ii). Attachment 2 demonstrates that the PCT value remains within the 2200°F PCT limit specified in 10 CFR 50.46(b)(1).

References

1. Letter from Joseph N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Annual Report and Thirty-day Report of 10 CFR 50.46 Loss-of-Coolant Accident Evaluation Model Changes," Accession Number ML062340193, submittal AEP:NRC:6046, dated August 11, 2006.

2. Letter from L. Raghavan, NRC, to Mano K. Nazar, I&M, "Donald C. Cook Nuclear Plant Units 1 and 2 – Summary of Telephone Conference Re: Fuel Upgrade TAC Nos. MC5646 and MC5647)," Accession Number ML052650307, dated September 28, 2005.

ATTACHMENT 2 TO AEP:NRC:6046-01

DONALD C. COOK NUCLEAR PLANT (CNP) UNIT 1
LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LOCA)
PEAK CLAD TEMPERATURE (PCT) SUMMARY

TABLE 1

CNP UNIT 1

LARGE BREAK LOCA

Evaluation Model: BASH
$F_Q = 2.15$ $F_{\Delta H} = 1.55$ SGTP = 15%   Break Size: $C_d = 0.4$
Operational Parameters: RHR System Cross-Tie Valves Closed, 3250 MWt Reactor Power[1]

LICENSING BASIS		
	Analysis-of-Record, December 2000	PCT = 2038°F
MARGIN ALLOCATIONS (Delta PCT)		
A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS	
	1. LOCBART Cladding Emissivity Errors	-11°F
B.	Planned 50.59 Plant Change Evaluations	
	1. Reduced Containment Spray Temperature	+23°F
	2. 15X15 Upgrade Fuel	-59°F
C.	OTHER	
	1. Rebaseline Using PAD 4.0	+57°F
D.	LICENSING BASIS PCT + MARGIN ALLOCATIONS	PCT = 2048°F

1. The 3250 MWt power level used in the reanalysis is acceptable because it bounds the Unit 1 3304 MWt steady state power limit in the operating license after adjusting for recapture of feedwater flow measurement and power calorimetric uncertainty.

ATTACHMENT 3 TO AEP:NRC:6046-01

REGULATORY COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
A new Unit 1 small break loss-of-coolant accident (SBLOCA) analysis of the safety injection cross-tie valves closed case will be provided.	March 2007
A new Unit 1 large break loss-of-coolant accident (LBLOCA) analysis will be provided.	December 2007
A new Unit 2 SBLOCA analysis of both the safety injection cross-tie valves closed case and the safety injection cross-tie valves open case will be provided.	March 2009
A new Unit 2 LBLOCA analysis will be provided.	March 2009