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May 9, 2019

AEP-NRC-2019-17
10 CFR 50.46

Docket No.: 50-315

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Donald C. Cook Nuclear Plant Unit 1
30-DAY REPORT OF CHANGES TO OR ERRORS IN AN EVALUATION MODEL

Pursuant to 10 CFR 50.46, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1, is reporting significant changes to or errors in emergency core cooling system evaluation model (EM), or in the application of such a model that affects the calculated peak fuel cladding temperature (PCT). By report dated January 17, 2019, Westinghouse notified I&M of EM changes, for a planned plant modification, which significantly affected the Small-Break Loss-of-Coolant Accident (SBLOCA) analysis for CNP Unit 1. The effect of this EM change upon the CNP Unit 1 Large-Break Loss-of-Coolant Accident analysis is not significant as defined in 10 CFR 50.46(a)(3)(i).

Enclosure 1 to this letter provides a description of the SBLOCA EM change and the associated impact to the CNP Unit 1 SBLOCA analysis of record and the analysis performed for the CNP Unit 1 Upflow Conversion modification. The upflow conversion modification field work was completed on April 17, 2019. Based on information provided by Westinghouse, an assessment of the EM change resulted in a PCT increase of 107°F for Unit 1.

The estimated impact on the CNP SBLOCA EM represents a significant change in PCT, as defined in 10 CFR 50.46(a)(3)(i). 10 CFR 50.46(a)(3)(ii) requires the licensee to provide a report within 30 days, including a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with 10 CFR 50.46. The proposed reanalysis schedule is provided in Enclosure 2 to this letter.

A new regulatory commitment is provided as Enclosure 2 to this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Director, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Q. Shane Lies
Site Vice President

MDS/mll

A002
NRR

Enclosures:

1. Donald C. Cook Nuclear Plant Unit 1 Report of Significant Changes Related to Westinghouse Small-Break Loss-of-Coolant Analysis Emergency Core Cooling System Evaluation Model
2. Regulatory Commitment

c:
R. J. Ancona – MPSC
R. F. Kuntz – NRC Washington, D.C.
MDEQ – RMD/RPS
NRC Resident Inspector
D. J. Roberts – NRC Region III
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2019-17

Donald C. Cook Nuclear Plant Unit 1

Report of Significant Changes Related to Westinghouse Small-Break Loss-of-Coolant Analysis Emergency Core Cooling System Evaluation Model

Abbreviations:

CNP	Donald C. Cook Nuclear Plant
°F	degrees Fahrenheit
FdH	nuclear enthalpy rise hot channel factor
F_Q	heat flux hot channel factor
I&M	Indiana Michigan Power Company
LOCA	loss of coolant accident
MWt	megawatts - thermal
PCT	peak cladding temperature
SGTP	steam generator tube plugging

Summary

Pursuant to 10 CFR 50.46, I&M, the licensee for CNP, is submitting a 30-day report of LOCA evaluation model changes resulting in a significant change in calculated PCT for the CNP Unit 1 Small Break LOCA analysis. A significant change is defined as a change or error identified in the model which results in a calculated change to PCT greater than 50°F or cumulative changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.

The following page summarizes the impact of the Unit 1 Reactor Vessel Upflow Conversion modification on the CNP Unit 1 Small Break LOCA analysis of record. The upflow conversion was completed in accordance with the approved CNP Engineering Change for the Unit 1 Upflow Conversion modification. Note that the PCT impact on the Unit 1 Large Break LOCA analysis due to the Reactor Vessel Upflow Conversion modification was not significant and is not reported herein.

CNP UNIT 1

LOCA Peak Clad Temperature Summary for Appendix K Small Break

Evaluation Model: NOTRUMP
$F_Q = 2.32$ FdH = 1.55 SGTP = 10% 3.25 inch cold leg break
Analysis Date: January 6, 2012

Note: Power is 3304 MWt plus 0.34% for calorimetric uncertainty

LICENSING BASIS

	Analysis-of-Record	PCT = 1725°F

MARGIN ALLOCATIONS (Delta PCT)

A.	PREVIOUS 10 CFR 50.46 ASSESSMENTS	0°F
B.	PLANNED PLANT MODIFICATION EVALUATIONS	
	1. Reactor Vessel Upflow Conversion	107°F (a)
C.	NEW 10 CFR 50.46 ASSESSMENTS	0°F
D.	OTHER	0°F

LICENSING BASIS PCT + MARGIN ALLOCATIONS PCT = 1832°F

Notes:

a. The evaluation resulted in a minor shift in limiting break size from 3.25 inches to 3.00 inches

Enclosure 2 to AEP-NRC-2019-17

Regulatory Commitment

The following table identifies the revised action 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 U. S. Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M will submit to the NRC for review a Unit 1 small break loss of coolant accident (SBLOCA) analysis that applies NRC approved methods using Westinghouse's WCAP-16996-P, "Realistic LOCA Evaluation Methodology Applied to Full Spectrum of Break Sizes (Full Spectrum LOCA Methodology)." The date for the submittal of the analysis will align with an existing commitment to submit a Unit 1 large break loss of coolant accident analysis, (Reference letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2, U. S. Nuclear Regulatory Commission Commitment Change Related to Estimated Effect of Peak Cladding Temperature Resulting from Thermal Conductivity Degradation," dated June 9, 2015).	The Unit 1 SBLOCA analysis will be submitted 34 months from the approval of the supplement to topical report WCAP-16996-P, and any required supplements that support the new 10 CFR 50.46 rule and would be needed for the analysis.