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Indiana Michigan Power
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Bridgman, MI 49106
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July 30, 2019

AEP-NRC-2019-40
10 CFR 50.90

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Unit 1 and Unit 2
Response to Request for Additional Information Regarding License Amendment Request to Address NSAL-15-1.

References:

1. Letter from Q. S. Lies, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC), "Donald C. Cook Nuclear Plant, Unit 1 and Unit 2, License Amendment Request to Address Issues Identified in Westinghouse Document NSAL-15-1," dated February 26, 2019, Agencywide Documents Access and Management System Accession (ADAMS) No. ML19060A060.

2. E-mail from R. F. Kuntz, NRC, to M. K. Scarpello, I&M, "D. C. Cook Unit Nos. 1 & 2 – Request for Additional Information Related to LAR to address NSAL-15-1 (EPID L-2018-LLA-0246)," dated July 3, 2019.

This letter provides Indiana Michigan Power Company's (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, response to the Request for Additional Information (RAI) by the U. S. Nuclear Regulatory Commission (NRC) concerning a License Amendment Request (LAR) for an amendment to Technical Specifications for CNP, Unit 1 and Unit 2.

By Reference 1, I&M submitted the LAR to address a deficiency identified in the Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-15-1, Rev 0. By Reference 2, the NRC transmitted an RAI concerning the LAR submitted by I&M in Reference 1.

Enclosure 1 to this letter provides an affirmation statement. I&M is providing Enclosure 2 to this letter as its response to the NRC's RAI from Reference 2.

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There are no new regulatory commitments made in 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

JMT/mll

Enclosures:

1. Affirmation
2. Response to Request for Additional Information Regarding License Amendment Request to Address a Deficiency Identified in the Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-15-1, Rev 0.

c: R. J. Ancona – MPSC
R. F. Kuntz – NRC Washington D.C.
EGLE – 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-2019-40

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. 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

Quinton S. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 30 DAY OF July, 2019

Micah L Lange
Notary Public

My Commission Expires 02/20/25

Enclosure 2 to AEP-NRC-2019-40

Response to Request for Additional Information Regarding License Amendment Request to Address a Deficiency Identified in the Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-15-1, Rev 0.

The U. S. Nuclear Regulatory Commission (NRC) staff is reviewing the Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, License Amendment Request (LAR) application dated February 26, 2019, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19060A060). The proposed change will expand criteria within TS 3.2.1 Surveillance Requirements (SRs) which will apply an appropriate penalty factor to the heat flux hot channel factor ($F_Q(Z)$), specifically the measured transient ($F_Q^W(Z)$). The proposed modification of the SR was to address a deficiency identified in the Westinghouse Nuclear Safety Advisory Letter (NSAL) NSAL-15-1, Rev. 0. The NRC staff has determined that additional information is necessary in order to complete its review.

By electronic mail dated July 3, 2019, the NRC transmitted a Request for Additional Information (RAI) regarding the February 26, 2019, LAR.

SRXB RAI-1

Regulatory Basis:

10 CFR 50.34, "Contents of applications; technical information," requires that safety analysis reports analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload process, licensees perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, they confirm that the inputs to the safety analyses are conservative with respect to the current design cycle. These inputs are checked using analytical models, and if key safety analysis parameters are not bounded, further analysis of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. As discussed in 10 CFR 50.36(c)(3), surveillance requirements (SRs) are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

F_Q is an input used in safety analyses and is a parameter that is subject to the surveillance requirements of 10 CFR 50.36.

SRXB RAI-1a)

a) *Does the fuel thermal conductivity degradation (TCD) issue, as described in NRC Information Notice 2012-21, impact measured transient $F_Q(Z)$? If so,*

I&M Response to SRXB RAI-1a)

The TCD issue does not impact the measured transient $F_Q(Z)$, $F_Q^W(Z)$, and is not effected by the additional surveillance actions. Technical Specification (TS) Section 3.2.1, "Heat Flux Hot Channel Factor," ensures that $F_Q(Z)$ is maintained within the limits assumed in the plant safety analysis. Compliance with the TS Limiting Condition of Operation (LCO) is demonstrated by measuring the steady-state peak power density at each axial elevation and verifying that both the steady-state heat flux hot channel factor ($F_Q^C(Z)$), and the $F_Q^W(Z)$, are within the $F_Q(Z)$ limits specified in the cycle specific Core Operating Limits Report (COLR). The $F_Q(Z)$ limit used in the safety analysis incorporates the effect of TCD on peak cladding temperature (PCT) in the Westinghouse Electric Company furnished realistic emergency core cooling system (ECCS) evaluation models as discussed in RAI-1b).

SRXB RAI-1b)

b) Confirm that the TCD issue was addressed for CNP. Otherwise, explain how fuel TCD is accounted for.

I&M Response to SRXB RAI-1b)

The TCD issue was addressed by I&M letter dated March 19, 2012, (ML12088A104) in response to NRC request (ML12041A384) to provide information regarding the effect of a potentially significant error, as defined in 10 CFR 50.46(a)(3)(i), associated with thermal conductivity degradation (TCD), on peak cladding temperature in the Westinghouse Electric Company furnished realistic emergency core cooling system (ECCS) evaluation models. The submitted report provided an estimate of the effect of the error on the predicted peak cladding temperature (PCT) for CNP Unit 1 and Unit 2. The report was supplemented by I&M letter dated June 11, 2012, (ML12173A025), and referred to a letter from Westinghouse Electric Company dated March 7, 2012, (ML12072A035).

I&M's response was that in order to support the Loss of Coolant Accident (LOCA) evaluation of thermal conductivity degradation (TCD) on the best estimate LOCA (BELOCA) analyses of record (AOR), Westinghouse Nuclear Design proposed changes to the peaking factor limits. The Westinghouse LOCA group evaluated TCD with these proposed limits to demonstrate that the TCD objectives for both units are met. Presently, the $F_Q(Z)$ limit remains the same with a lowering of the enthalpy rise hot channel factor ($F_{\Delta H}$) limit on both units with respect to TCD. The rationale for lowering the $F_{\Delta H}$ limit was to fully take advantage of the available nuclear design margin to support the LOCA TCD. The COLR was updated appropriately to ensure the plant is operated within these new constraints. I&M's explanation of peaking factor adjustments, and rationale for each adjustment, was evaluated by the NRC in letter dated March 7, 2013, (ML13077A137).

The NRC evaluated I&M's March 19, 2012, letter, along with its supplemental information and Westinghouse letter. The NRC staff performed a detailed review of the input parameters and limiting results that were used to generate the estimate and concluded that the estimate enables the current analysis to maintain a high level of probability that the 2200 °F PCT acceptance criterion is not exceeded. The NRC staff determined that the licensee's response was acceptable because it showed that the cycle-specific core designs meet the analyzed limits.

The NRC determined in their March 7, 2013, letter that the March 19, 2012, I&M letter, along with its supplemental information and the March 7, 2012, Westinghouse letter satisfied the reporting requirements of 10 CFR 50.46(a)(3). The I&M letters dated March 19, 2012, and June 11, 2012, also enabled the NRC to: (1) determine that it agreed with the licensee's assessment of the significance of the error; (2) confirm that the evaluation model remained adequate; (3) verify that the licensee continued to meet the PCT acceptance criterion promulgated by 10 CFR 50.46(b); and (4) determine that the licensee's proposed schedule for reanalysis was acceptable in light of the information provided.

Moving forward, the TCD evaluation is maintained by being assessed for updates due to modifications and changes to the facility. For instance, the Unit 1 TCD evaluation was reassessed as part of the project to restore normal reactor coolant system pressure and temperature consistent with previously licensed conditions, and received NRC review and approval as reflected in their letter dated November 30, 2015 (ML14197A097).