From:	Scott Wall
Sent:	Wednesday, June 26, 2024 8:25 AM
To:	Michael K. Scarpello
Cc:	Helen L Levendosky; Bradford M Culwell
Subject:	FINAL RAI - D.C. Cook 1 & 2 - License Amendment Request Regarding Neutron Flux Instrumentation (EPID No. L-2023-LLA-0011)

Dear Michael Scarpello,

By letter dated January 26, 2023, (Agencywide Documents Access and Management System Accession No. ML23026A284), as supplemented by letters dated August 2, 2023 (ML23214A289), and February 27, 2024 (ML24058A357), Indiana Michigan Power company (I&M, the licensee) submitted a license amendment request for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (CNP). The amendment would revise Technical Specification (TS) Table 3.3.3-1, “Post-Accident Instrumentation.” Specifically, the proposed changes would remove Function 1, Neutron Flux, from the list of required post-accident monitoring (PAM) instrumentation.

The NRC staff has reviewed the submittals and determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). During a telephone call on June 20, 2024, the I&M staff indicated that a response to SNSB- RAI-2, SNSB- RAI-3, and SNSB- RAI-4, would be provided by August 2, 2024. Given the potential scope of the response to SNSB- RAI-1, the I&M staff indicated that a final response date would be provided by August 2, 2024.

If you have questions, please contact me at 301-415-2855 or via e-mail at Scott.Wall@nrc.gov.

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Docket Nos. 50-315 and 50-316

Enclosure:
Request for Additional Information

cc: Listserv

RAI (Neutron Flux Instrumentation)

REQUEST FOR ADDITIONAL INFORMATION

NEUTRON FLUX INSTRUMENTATION

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

INTRODUCTION

By letter dated January 26, 2023, (Agencywide Documents Access and Management System Accession No. ML23026A284), as supplemented by letters dated August 2, 2023 (ML23214A289), and February 27, 2024 (ML24058A357), Indiana Michigan Power company (I&M, the licensee) submitted a license amendment request (LAR) for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 (CNP). The amendment would revise Technical Specification (TS) Table 3.3.3-1, "Post-Accident Instrumentation." Specifically, the proposed changes would remove Function 1, Neutron Flux, from the list of required post-accident monitoring (PAM) instrumentation.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the application and has determined that the following additional information is required in order to complete the review.

Regulatory Basis

Title 10 *Code of Federal Regulations* (CFR) Part 50, Section 50.36, "Technical Specifications," requires, in part, that the TS shall be included by applicants for a license authorizing operation of a production or utilization facility. 10 CFR 50.36(c) requires that TS 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 operation (LCOs) (3) Surveillance requirements, (4) Design features, and (5) Administrative controls.

The regulation at 10 CFR 50.36(c)(2)(ii)(C) requires that TS LCOs of a nuclear reactor be established for a structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design-basis accident (DBA) or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

10 CFR 50.46(b)(5), "Long-term cooling," requires that after any calculated successful initial operation of the Emergency Core Cooling System (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In CNP Updated Final Safety Analysis Report (UFSAR), Section 1.4, "Plant Specific Design Criteria (PSDC)" (ML22340A150), states that the CNP specific design is committed to meet the intent of the proposed GDC published in the *Federal Register* on July 11, 1967.

CNP PSDC CRITERION 12, "Instrumentation and Control Systems," states, in part:

> Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

> Instrumentation and controls are provided to monitor and maintain all operationally important reactor operating parameters such as neutron flux, system pressures, flow rates, temperatures,levels and control rod positions within prescribed operating ranges. The quality and types of instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant.
>
> Process variables, which are required on a continuous basis for the startup, power operation and shutdown of the plant, are indicated in, recorded in, and controlled as necessary from the control room, which is a controlled area.

CNP PSDC CRITERION 13, “Fission Process Monitors and Controls,” states, in part:

> Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.
>
> The primary function of nuclear instrumentation is to safeguard the reactor by monitoring the neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. It also provides a secondary control function and indicates reactor status during startup and power operation.

<u>Background</u>

By letter dated February 27, 2024, the licensee response to the NRC staff’s request for additional information (RAI) (ML23321A122) regarding a request to use alternate means of fulfilling the requirements of Regulatory Guide (RG) 1.97, Revision 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident,” dated May 1983 (ML003740282), with regards to the plant safety function of reactivity control.

<u>Nuclear Systems Performance Branch (SNSB) Questions</u>

SNSB- RAI-1

In the section “I&M Response to EICB-RAI-4” of February 27, 2024, supplement, the licensee states that for Steam Line Break Inside Containment, in part:

> …a postulated return to power for this type of cooldown event is self-limiting.
>
> Assuming the initial reactor trip verification was successful, any return to criticality from an uncontrolled RCS [Reactor Coolant System] cooldown during a steam line break event would be terminated through temperature feedback as the RCS heats up. The RCS temperature following the heat up would be below the temperature of the RCS at the time of the initial reactor trip, since boron would have been added by ECCS injection during the initial event response, and since control rods would insert during the reactor trip. The plant UFSAR accident analysis considers the return to power possibility from a steam line break, where

> the core is ultimately shut down by boric acid delivered by the ECCS to the RCS, which remains intact.

The NRC staff notes the licensee's conclusion that any return to criticality from an uncontrolled RCS cooldown during Steam Line break event would be eventually terminated through temperature feedback due to RCS temperatures going up and the core will ultimately shut down by boric acid injection through EECS. However, there is no evaluation provided to show that following a large reactivity insertion due to steam line break inside Containment, the return to criticality will be detected and mitigated timely and sufficiently due to the RCS heat-up and the boric acid injection.

- Please provide a quantitative evaluation to show that following any accident scenario the temperature feedback and the boron injection from ECCS are timely and sufficient to detect and indicate whether a return to criticality is occurring and enabling reactor operators to take appropriate mitigative actions to address a return to criticality during such an event, especially with lack of environmentally qualified nuclear instrumentation to monitor the criticality.

SNSB- RAI-2

In the section "I&M Response to EICB-RAI-4" of February 27, 2024, supplement, the licensee states that for Steam Line Break Inside Containment, in part:

> During post-accident recovery with the RCS intact, in a situation where Gamma-Metrics instruments are not available, control room operators are trained and directed by emergency operating procedures to monitor RCS temperature indication as a key variable to identify any postulated return to criticality and rising core power level. One or more indications of RCS temperature would be available to control room operators, including CET temperature, RCS Hot Leg temperature, and RCS Cold Leg temperature.

- Please provide the expected uncertainties on the core exit thermocouple (CET), cold leg and hot leg system temperatures during any postulated post-accident neutron flux increase for situations where the wide range neutron flux monitoring may be rendered inoperable. In the response, indicate the differences in uncertainty for situations where flow in the hot leg and cold leg of the primary system are stagnant, under natural circulation, or when reactor coolant pumps running.

SNSB- RAI-3

In the section "I&M Response to EICB-RAI-4" of February 27, 2024, supplement, the licensee states that for Steam Line Break Inside Containment, in part:

> …Pressurizer Level with an intact RCS is very responsive to small changes in RCS temperature and would provide defense in depth for monitoring a return to criticality in this scenario.

The NRC staff notes the licensee's conclusion that the pressurizer level is very responsive to changes in RCS temperatures. However, there is no evaluation provided to show that the core will not go critical due to return to criticality event prior to the temperature increase leading to a pressurizer level change.

- Please provide a quantitative assessment of how pressurizer level change feedback provides defense in depth against a fast criticality change post any accidents with an intact RCS, especially with lack of environmentally qualified nuclear instrumentation to monitor the criticality.

SNSB- RAI-4

During post-accident condition, the reactor must remain in a subcritical state, and if it becomes critical, the reactor must be safely returned to sub-critical state. The times involved with its safe return to subcriticality are:

- Detection time, i.e., instrumentation response time to detect criticality,

- Mitigation time, i.e., Operator action time for the mitigation of criticality post detection according to the plant emergency operating procedures (EOPs)

Consider the following two scenarios:

Scenario 1 - The wide range neutron flux instruments are OPERABLE to detect core criticality.

Scenario 2 - The wide range neutron flux instruments are INOPERABLE. The proposed means, i.e., the core exit thermocouples plus other devices are used to detect criticality.

- Please provide responses to the following for Scenarios 1 and 2:

  (a) The criticality detection time following any type of accident.

  (b) The mitigation time for returning the core to subcriticality following any type of accident.

  (c) The OPERATOR ACTIONS involved for returning to criticality according to the current EOPs following any type of accident.

  (d) Confirm 10 CFR 50.46(b)(1) through (b)(5) continues to be satisfied following a LOCA event.

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