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

Donald C. Cook Nuclear Plant Unit 1 and Unit 2
Response to Request for Additional Information for Neutron Flux
Instrumentation License Amendment Request

References:

1. Letter from K. J. Ferneau, Indiana Michigan Power Company (I&M), to U.S. Nuclear Regulatory Commission (NRC), "Request for Approval of Change Regarding Neutron Flux Instrumentation," dated January 26, 2023, Agencywide Documents Access and Management System (ADAMS) Accession No. ML23026A284.

2. Letter from Q. S. Lies, I&M, to NRC, "Supplement to Request for Approval of Change Regarding Neutron Flux Instrumentation," dated August 2, 2023, ADAMS Accession No. ML23214A289.

3. E-mail from S. P. Wall, NRC, to M. K. Scarpello, I&M, "Final RAI - D.C. Cook 1 & 2 - License Amendment Request Regarding Neutron Flux Instrumentation (EPID No. L-2023-LLA-0011)," dated November 17, 2023, ADAMS Accession No. ML23321A122.

4. Letter from K. J. Ferneau, I&M, to NRC, "Response to Request for Additional Information on Requested Change Regarding Neutron Flux Instrumentation," dated February 27, 2024, ADAMS Accession No. ML24058A357.

5. E-mail from S. P. Wall, NRC, to M. K. Scarpello, I&M, "Final RAI - D.C. Cook 1 & 2 - License Amendment Request Regarding Neutron Flux Instrumentation (EPID No. L-2023-LLA-0011)," dated June 26, 2024, ADAMS Accession No. ML24178A043.

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) submitted by the U.S. Nuclear Regulatory Commission (NRC) on June 26, 2024, regarding a request to modify CNP Unit 1 and Unit 2 Technical Specification (TS) 3.3.3, Post Accident Monitoring Instrumentation, to remove the environmental qualification requirements from TS Table 3.3.3-1, Function 1, Neutron Flux.

By Reference 1, I&M submitted a request to reclassify the wide range neutron flux instrumentation at CNP Unit 1 and Unit 2 to Category 3 and requested a corresponding change to CNP Unit 1 and Unit 2 TS Table 3.3.3-1, Post Accident Monitoring Instrumentation, Function 1, Neutron Flux. By Reference 2, I&M submitted a supplement to Reference 1. By Reference 3, the NRC submitted an RAI concerning the letter submitted by I&M as Reference 1. By Reference 4, I&M responded to the RAI submitted as Reference 3 and revised the scope of the request such that Neutron Flux remains as Function 1 in TS Table 3.3.3-1, but is exempt from the requirement to maintain environmental qualification. By Reference 5, the NRC submitted an additional RAI. As discussed during a teleconference with NRC staff on July 29, 2024, this response to the NRC RAI is being submitted by August 16, 2024.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 to this letter provides I&M's response to the NRC's RAI from Reference 5.

The changes proposed in this letter do not impact the conclusions provided in Reference 1 that a finding of "no significant hazards consideration" is justified. 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,

Kelly J. Ferneau

Kelly J. Ferneau
Site Vice President

BMC/sjh

Enclosures:

1. Affirmation
2. Response to Request for Additional Information for Neutron Flux Instrumentation License Amendment Request

c: EGLE – RMD/RPS
J. B. Giessner – NRC Region III
NRC Resident Inspector
N. Quilico – MPSC
R. M. Sistevaris – AEP Ft. Wayne, w/o enclosures
S. P. Wall – NRC Washington, D.C.
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

AFFIRMATION

I, Kelly J. Ferneau, 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

Kelly J. Ferneau

Kelly J. Ferneau
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15 DAY OF August 2024

Notary Public

My Commission Expires 5/23/2030

Enclosure 2 to AEP-NRC-2024-61

Response to Request for Additional Information for Neutron Flux Instrumentation License Amendment Request

By letter dated January 26, 2023 (Reference 1), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, submitted a request to use alternate means of fulfilling the requirements of Regulatory Guide (RG) 1.97 with regards to the plant safety function of reactivity control at CNP Unit 1 and Unit 2. The request would reclassify the wide range neutron flux instrumentation at CNP Unit 1 and Unit 2 as Category 3 instrumentation, and would modify Technical Specification (TS) Table 3.3.3-1, Post Accident Monitoring Instrumentation, to remove Function 1, Neutron Flux, from the list of required post-accident monitoring (PAM) instrumentation.

By letter dated August 2, 2023 (Reference 2), I&M submitted a supplement to Reference 1. By e-mail dated November 17, 2023 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC) submitted a request for additional information (RAI) concerning the letter submitted by I&M as Reference 1. By letter dated February 27, 2024 (Reference 4), I&M responded to the RAI submitted as Reference 3 and revised the scope of the request such that Neutron Flux remains as Function 1 in TS Table 3.3.3-1, but is exempt from the requirement to maintain environmental qualification.

The NRC staff is currently reviewing the change request and has determined that additional information is needed in order to complete the review (Reference 5). I&M's response to Reference 5 is provided below.

<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 ECCS. 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.*

I&M Response to SNSB-RAI-1

The proposed changes to CNP Unit 1 and Unit 2 TS Table 3.3.3-1 do not impact the availability or functionality of the Post-Accident Monitoring (PAM) Wide Range Nuclear Instrumentation (WR NIS), which satisfies TS Table 3.3.3-1 Function 1, Neutron Flux, for accident scenarios which do not result in an adverse containment environment. Steamline Break Inside Containment Accident (SLB) and Loss of Coolant Accident (LOCA) design basis events can result in adverse containment conditions where the WR NIS, if not environmentally qualified, cannot be relied upon to function for the duration of the event. However, the accident analysis in the CNP Updated Final Safety Analysis Report (UFSAR) for SLB and LOCA does not credit WR NIS. Even more broadly, none of the accident analyses in the CNP UFSAR credit the WR NIS.

The SLB accident analysis is the only accident scenario described in the UFSAR where the reactor core is postulated and analyzed for a return to criticality and power, which is one of the functions of WR NIS post-accident. Unit 1 UFSAR Section 14.2.5 and Unit 2 UFSAR Section 14.2.5 both provide quantitative SLB analyses for this return to power scenario. The UFSAR SLB analysis demonstrates that:

- there is no consequential damage to the core,
- the core remains in place and intact,
- Departure from Nucleate Boiling (DNB) and clad perforation criteria are met,
- bounding inputs for reactor temperature feedback effects are used,
- the core is ultimately shutdown with credit for an automatic reactor trip and automatic delivery of boric acid by the Emergency Core Cooling System (ECCS), and
- there is no credit for operator action.

CNP Unit 1 and Unit 2 UFSAR Table 14.2.5-2 describes the analyzed sequence of events for a double-ended SLB inside containment case with offsite power available and hot zero power initial core conditions. For this Unit 1 case, criticality is re-attained after 14.20 seconds, the core reaches a peak core average heat flux of 22.3%, and the core becomes subcritical after 134.6 seconds. For this Unit 2 case, criticality is re-attained after 22.6 seconds, the core reaches a peak core average heat flux of 17.3%, and the core becomes subcritical after 121.0 seconds.

As stated above, in all design basis LOCA scenarios the core remains subcritical following the reactor trip and injection of boric acid by the ECCS. Therefore, the function for detection of criticality by WR NIS post-accident would not need to be relied on.

Additional discussion of expected instrument response following an accident scenario, though not quantitative in nature, is contained in I&M's response to SNSB-RAI-2.

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.*

I&M Response to SNSB-RAI-2

As was discussed on a July 29, 2024, phone call between I&M staff and NRC staff, quantitative analyses of the CNP CET, Wide Range Reactor Coolant System (WR RCS) hot leg and WR RCS cold leg temperature instrument responses during postulated post-accident conditions are considered not feasible and therefore the requested uncertainties are not provided with this response.

SLB and LOCA design basis events can result in adverse containment conditions where the WR NIS, if not environmentally qualified, cannot be relied upon to function for the duration of the event. This response will clarify how the CET, WR RCS hot leg Resistance Temperature Detectors (RTDs) and WR RCS cold leg RTDs are physically arranged at CNP Unit 1 and Unit 2 and how they are expected to respond during the SLB and LOCA accidents.

<u>Arrangement of Applicable RCS Temperature Instruments</u>

Current CNP Unit 1 and Unit 2 Technical Specifications (TSs), Table 3.3.3-1, "Post Accident Monitoring Instrumentation," lists:

- Two required channels for Function 3, RCS Hot Leg Temperature (Wide Range),
- Two required channels for Function 4, RCS Cold Leg Temperature (Wide Range), and
- Two required channels in each of the four quadrants for Functions 15, 16, 17 and 18, Core Exit Temperature (where the table notes that one core exit temperature channel consists of one core exit thermocouple).

For TS Table 3.3.3-1, Functions 3 and 4, the RCS hot leg and RCS cold leg channels each receive input from one RTD. In each of RCS loops 1 and 3, there is one WR RCS hot leg RTD and one WR RCS cold leg RTD that satisfy the guidance of RG 1.97, Revision 3.

CNP Unit 1 and Unit 2 both have a four-loop RCS with a single WR hot leg RTD and single WR cold leg RTD in each of the four loops. The WR RCS RTDs are located in thermowells that extend into the cold and hot leg piping. The WR cold leg RTDs are downstream of the Reactor Coolant Pump (RCP) discharge. While each of the four loops has WR RTDs, only the hot leg and cold leg RTDs in Loops 1 and 3 are credited by CNP Unit 1 and Unit 2 TS 3.3.3 for post-accident monitoring. In addition to WR RTDs, CNP Unit 1 and Unit 2 also have narrow range hot leg and cold leg RTDs installed in each of the four loops. While these narrow range RTDs do support the reactor protection system, they are not credited by TS 3.3.3 for post-accident monitoring.

The CNP Unit 1 and Unit 2 in-vessel instrumentation systems provides up to 65 CETs, positioned to measure fuel assembly coolant outlet temperature at preselected locations. Thermocouples are threaded into guide tubes that penetrate the reactor vessel head through seal assemblies and terminate in the upper core support assembly above the exit flow end of the fuel assemblies. The CETs are approximately 1.3 feet above the active fuel line, which is below the bottom of the cold leg and hot leg vessel nozzles. Note that CNP Unit 1 and Unit 2 TS 3.3.3 for post-accident monitoring only require the use of eight total CETs at a given time (two for each of the four core quadrants).

Discussion of RCS Temperature Instrument Response During a SLB

Steam releases from a SLB event increases the steam energy release from the steam generators, which causes an increase in the heat extraction rate from the RCS. The result is a reduction of primary coolant temperature and pressure. The CNP UFSAR SLB analysis credits an automatic reactor trip, insertion of Rod Cluster Control Assemblies (RCCAs), and automatic injection of boric acid by the ECCS. The CNP Unit 1 and Unit 2 SLB analysis considers an accident both with and without offsite power available (i.e., both with and without RCPs running). The SLB analysis demonstrates that DNB ratio criteria are met for local power peaking that can occur during a return to power with RCCAs inserted (where a single RCCA is conservatively assumed not to insert into the core). The acceptable SLB analysis DNB ratio results demonstrate that the bulk fluid circulating through the core and RCS is subcooled during post-SLB conditions. If the RCPs are stopped, heat generated from the core is transferred to the Steam Generators (SGs) through single-phase natural circulation. Density differences between the cooler water in the reactor vessel downcomer annulus and the hotter water in the core region act in a manner that helps drive flow around the RCS loops. During post-SLB conditions, the SG for the RCS loop with the secondary system fault may become unavailable for cooling the RCS. During plant cooldown, the plant procedures are designed to prevent stagnation from occurring. However, even if stagnation were postulated to occur in the inactive loop, natural circulation flow would still occur through the other active RCS loops. WR RCS hot leg and cold leg RTDs are required in Loops 1 and 3 per CNP Unit 1 and Unit 2 TS 3.3.3, which ensures that a WR RCS hot leg and cold leg RTD would be present in an active loop without stagnation.

Included in the CNP Unit 1 and Unit 2 UFSAR SLB analysis is credit for Overpower Delta-Temperature (OPΔT) or Overtemperature Delta-Temperature (OTΔT) reactor protection logic for a reactor trip, which relies on the narrow range cold and hot leg RTD measurements. The total reactor trip delay time in the OPΔT/ OTΔT response is 8 seconds or less. While the physical arrangement of the narrow range RCS loop RTDs differ from the wide range RCS loop RTDs (for example, the hot leg narrow range RTD signal isn't from a single RTD but is an electronically averaged signal from three RTDs separated by 120 degrees intervals around the periphery of the hot leg), the narrow range RTD responsiveness supports the judgment that WR RCS hot leg and cold leg RTDs would be relatively responsive to changes in RCS temperature in post-SLB conditions.

In summary, the WR RCS hot leg and cold leg RTDs are judged to be responsive to a return to power given that the bulk fluid circulating from the core through the RCS loops is subcooled during post-SLB conditions and the observation that the narrow range RCS loop RTDs are responsive for post-SLB OPΔT/ OTΔT reactor protection. CETs are also judged to be as responsive or more responsive to changes in core temperature during post-SLB conditions than the WR RCS cold leg and hot leg RTDs given that they are directly above the fluid exiting the active fuel region and the bulk fluid circulating is subcooled.

Discussion of RCS Temperature Instrument Response Following a LOCA

Following a postulated design basis LOCA event, the pressure in the RCS will decrease and the ECCS will automatically initiate safety injection, thereby replenishing RCS inventory and reducing cladding temperatures to acceptable levels. When the RCS depressurizes to the accumulator gas cover pressure, the accumulators also begin to inject borated water into the reactor coolant loops. The ECCS fills the reactor vessel to an elevation above the core, where the actual level is controlled by the break location and size. The ECCS pumps take suction from the Refueling Water Storage Tank (RWST) during the initial injection phase following a LOCA. Once the RWST drains to a low level, the control room operator will transfer the ECCS pumps' suctions to the containment recirculation sump. For a typical large break LOCA, the CNP UFSAR describes the transient in terms of a sequence of phases where Long-Term Core Cooling (LTCC) follows the reflood phase. In a general sense, LTCC can be defined to start after the core has completely quenched, all fuel rod cladding temperatures are near the event-specific saturation conditions, and liquid inventory has been reestablished in the reactor pressure vessel.

As is discussed in Enclosure 2 of Reference 4, in response to EICB-RAI-4, a postulated post-LOCA boron dilution would be relatively slow. Therefore, a postulated return to criticality and power from a boron dilution during post-LOCA conditions is only considered to be relevant for the LTCC phase of the accident.

Initially during LTCC, ECCS injection flow is directed through any intact cold legs, into the downcomer and upwards through the core which is covered by the two-phase mixture flow. The CETs, which are positioned closely above the top of the fuel (the CET tips are at an elevation lower than the bottom of the cold leg and hot leg vessel nozzles), are judged to be thermally responsive to changes in the two-phase core exit temperature and steaming, given that the CETs will be effectively submerged in the two-phase mixture.

Between 5.5 and 7.5 hours after the initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control boric acid concentration in the reactor vessel. During this LTCC hot leg recirculation, the flow direction of the core is effectively reversed, and therefore CETs are not expected to be as responsive to changes in core temperature, given that at this point the CETs would be primarily measuring the temperature of the recirculated ECCS flow going into the reactor core. If a rising core power due to a dilution event is postulated, then it is expected to impact the entirety or a majority of the core, due to the core being a well-mixed environment in this scenario. Therefore, the CETs would be expected to respond to steam released from bulk boiling in the core.

It should be noted that, during LTCC hot leg recirculation, if conditions permit, it is expected that periodic sampling for boric acid concentration would be occurring, which would be capable of providing indication of a dilution event prior to a return to criticality.

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.*

I&M Response to SNSB-RAI-3

As was discussed on the July 29, 2024, phone call between I&M staff and NRC staff, quantitative analysis of how Pressurizer level indication would respond to a postulated "fast" criticality event at CNP Unit 1 and Unit 2 is not provided with this response, as Pressurizer level is only considered as defense-in-depth for identifying an increase in RCS temperature associated with a return to power.

SLB and LOCA design basis events can result in adverse containment conditions where the WR NIS, if not environmentally qualified, cannot be relied upon to function for the duration of the event. Defense in depth monitoring of the pressurizer level is only applicable when the RCS is intact and pressurizer level is on instrument scale. The RCS is intact during a SLB but not during a LOCA. Therefore, the SLB accident is the RCS intact accident of consideration for this response.

As discussed in Enclosure 2 of Reference 4, in response to EICB-RAI-4, plant response and operator actions for a SLB event in the initial accident mitigation stage do not rely on WR NIS or the proposed use of RCS temperature indications. The reactor trip and subsequent ECCS boron injection that ensure reactor shutdown occurs during this accident mitigation stage are irrespective of WR NIS or the proposed use of RCS temperature indications and any associated operator actions. Once ECCS injection has been secured and the accident recovery stage is entered, indications of a return to criticality and power would be monitored by use of CETs, WR RCS hot leg RTDs, WR RCS cold leg RTDs, and (defense in depth) Pressurizer level, and would drive operator actions. While unlikely, the applicable mechanism for the return to criticality and power would be dilution of the RCS boron concentration. As is discussed in Reference 4, this dilution mechanism would be relatively slow. Therefore, consideration of WR NIS or RCS temperature indication responsiveness to "fast" criticality changes is not applicable.

Pressurizer level increase is typically the first indication of the addition of nuclear heat. This applies for all phases of power operation, including the initial point of adding heat following a reactor startup. Pressurizer level will increase by approximately 0.6 percent for each one degree Fahrenheit increase in RCS temperature, which is easily observable with installed control room indication.

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 [sub]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.*

I&M Response to SNSB-RAI-4

SNSB-RAI-4 Parts (a) and (b)

As was discussed on the July 29, 2024, phone call between I&M staff and NRC staff, quantitative analyses of the CNP Unit 1 and Unit 2 WR NIS, CET, WR RCS hot leg and WR RCS cold leg temperature instrument responses during postulated post-accident conditions are considered not feasible and therefore are not provided with this response. As was also discussed on the July 29, 2024, phone call, quantitative analyses for the mitigation time for returning the core to subcriticality are considered not feasible and therefore are not provided with this response.

Detection Time

SLB and LOCA design basis events can result in adverse containment conditions where the WR NIS, if not environmentally qualified, cannot be relied upon to function for the duration of the event. For a description of how the CET, WR RCS hot leg and WR RCS cold leg temperature instruments respond during SLB or LOCA events, please see I&M's response to SNSB-RAI-2 above. A detailed description or analysis of how the WR NIS responds during SLB or LOCA events is not provided. For the purposes of comparing environmentally qualified WR NIS response to alternative indications during a SLB or LOCA event, it may be conservatively assumed that the WR NIS provides indication of a return to criticality condition without delay.

Mitigation of a Return to Criticality Event

The primary reactivity control mechanisms for responding to an unplanned criticality event are the rapid insertion of RCCAs and the injection of borated water into the reactor core. RCCAs are unavailable for this function post-accident as they will already be inserted into the core, leaving the injection of borated water as the only available reactivity control mechanism. This mechanism is most effective with an intact RCS (such as following a SLB event) where only the volume of the RCS would require boration.

The RCS intact configuration is when the WR NIS, CET, WR RCS hot leg RTD, WR RCS cold leg RTD, and (defense in depth) Pressurizer level indications are all most responsive to a return to power condition. While the WR NIS can provide an advanced indicator of criticality during RCS intact conditions, small changes in RCS temperature would support timely control room operator response to a return to power condition by addition of boric acid.

In the case of accidents where the RCS is not intact (i.e., following a LOCA event), success is based on deterministically demonstrating that the post-accident boron concentration is sufficient under postulated accident conditions to ensure the reactor remains subcritical through the recirculation phase of the accident. Step 11 of procedure OHP-4023-E-1 (E-1), Loss of Reactor or Secondary Coolant, directs operators to request the Plant Evaluation Team (PET) to evaluate the need for chemistry samples to support long term recovery. In the event that WR NIS indication is not available, it is expected that, if conditions permit, chemistry sampling will be recommended by the PET.

One of the final operational steps of procedure E-1 is to transfer to hot leg recirculation once approximately 7 hours have lapsed from the initial event. Following transfer to hot leg recirculation, operators are directed to monitor "core exit temperatures" and reactor vessel level indication system (RVLIS) levels and either consult the PET or adjust ECCS flow if needed. There are no subsequent procedurally-directed actions and this effectively ends the accident mitigation phase of the event.

While OHP-4023-E-1 would nominally be considered the governing procedure, the plant has completed the accident mitigation phase of the event and has entered the long-term recovery phase where it will be expected to remain for the foreseeable future. In this condition subsequent operator actions will be as recommended by the PET.

Beyond the analysis and design to assure that post-LOCA criticality does not occur, prevention of re-criticality during the long term recovery phase is assured by the early detection and elimination of dilution sources. This strategy is not dependent on early detection by the WR NIS. It is accomplished by monitoring containment level, system flow rates, interfacing system response (such as component cooling water surge tank level), and periodic sampling for boric acid concentration.

SNSB-RAI-4 Part (c)

Critical Safety Function Status Tree for Subcriticality, OHP-4023-F-0.1 (F-0.1), directs control room operators to monitor WR NIS for a potential return to criticality following entry into the Emergency Operating Procedures. F-0.1 may direct control room operators to enter one of two Function Restoration Procedures, depending on the status of the Subcriticality Critical Safety Function, OHP-4023-FR-S.1 (FR-S.1), Response to Nuclear Power Generation / ATWS, or OHP-4023-FR-S.2 (FR-S.2), Response to Loss of Core Shutdown.

If wide range (WR) log power is less than 10E-5% but WR start up rate is positive, or if WR log power is not less than 10E-5% and WR start up rate is greater than or equal to -0.2 decades per minute, a YELLOW condition exists, and control room operators are directed to enter FR-S.2, Response to Loss of Core Shutdown, which directs operators to initiate emergency boration. The initiation of emergency boration requires the operation of four control room switches within reach of each other and can reasonably be completed in as little as one minute upon direction by the Unit Supervisor.
In the event that wide range log power is not less than 5% (RED condition), or in the event that wide range log power has a positive start up rate and is not less than 10E-5% (ORANGE condition), F-0.1, directs control room operators to enter Function Restoration Procedure FR-S.1. The first several steps of FR-S.1 are to:

- Check reactor trip

- Manually actuate AMSAC (the Anticipated Transient without Scram Mitigation System Actuation Circuitry)
- Check turbine trip
- Check that auxiliary feedwater pumps are running
- Initiate emergency boration of the RCS

In the event that WR NIS are inoperable, as described in Scenario 2 of SNSB-RAI-4, wide range log power less than 5% cannot be confirmed, and operators would be directed to enter FR-S.1 and would continue to borate until it could be determined that the reactor is subcritical via an aggregate assessment using multiple diverse indications.

It is important to note that the proposed changes to TS Table 3.3.3-1, Function 1, would still require neutron flux channels to be operable in Modes 1, 2, and 3, and would only exempt the channels from the requirement to maintain environmental qualification. Therefore, Scenario 2 would only occur as a result of an adverse containment environment, resulting from a SLB or LOCA, and emergency boration would already have been initiated.

SNSB-RAI-4 Part (d)

As is described in the above response to SNSB-RAI-1, the LOCA analysis in the CNP UFSAR does not credit the WR NIS. Therefore, there would be no impact to the LOCA analysis if the PAM WR NIS indication was not available or was replaced with a PAM RCS temperature indication.

During a LOCA event the core is initially shutdown due to core voiding. Boron injection through the ECCS and accumulators then occurs automatically. Long-term core cooling includes long-term criticality control. The UFSAR describes post-LOCA criticality control and two separate supporting calculations; during cold leg recirculation, and at the time ECCS-recirculation is realigned from cold leg injection to hot leg injection. Criticality control during cold leg recirculation is achieved by determining the RWST and accumulator concentration necessary to maintain subcriticality without credit for RCCA insertion. RCCA insertion credit has been assumed to provide negative reactivity at the time of hot leg switchover following a cold leg break. The necessary RWST and accumulator boron concentration for post-LOCA criticality control is a function of each core design and is checked each cycle and is controlled by plant Technical Specifications.

The UFSAR-described quantitative LOCA evaluations, including the post-LOCA criticality control evaluations, which demonstrate compliance with all the 10 CFR 50.46 criteria (10 CFR 50.46 (b)(1) through (b)(5)) do not rely on WR NIS indication to monitor for a return to criticality or power. Deterministically assuming that re-criticality occurs post-LOCA is not required for demonstrating compliance with 10 CFR 50.46.

It is also noted that the NRC issued amendments for CNP Unit 1 and Unit 2 on December 23, 1999 (Reference 6), which approved the request submitted by I&M on September 17, 1999 (Reference 7). The 1999 submittal makes it clear that the LOCA analysis licensing basis assures that subcriticality is maintained. As one example, Attachment 1 to Reference 7 explicitly stated that "The current LOCA analyses requirements are based upon the core remaining subcritical after shutdown following the occurrence of such an accident. The cycle-specific reload safety evaluations confirm that post-LOCA subcriticality requirements are met." Similar discussion can also be found in Attachment 4 of Reference 7.

References:

1. Letter from K. J. Ferneau, Indiana Michigan Power Company (I&M), to U.S. Nuclear Regulatory Commission (NRC), "Request for Approval of Change Regarding Neutron Flux Instrumentation," dated January 26, 2023, Agencywide Documents Access and Management System (ADAMS) Accession No. ML23026A284.

2. Letter from Q. S. Lies, I&M, to NRC, "Supplement to Request for Approval of Change Regarding Neutron Flux Instrumentation," dated August 2, 2023, ADAMS Accession No. ML23214A289.

3. E-mail from S. P. Wall, NRC, to M. K. Scarpello, I&M, "Final RAI - D.C. Cook 1 & 2 - License Amendment Request Regarding Neutron Flux Instrumentation (EPID No. L-2023-LLA-0011)," dated November 17, 2023, ADAMS Accession No. ML23321A122.

4. Letter from K. J. Ferneau, I&M, to NRC, "Response to Request for Additional Information on Requested Change Regarding Neutron Flux Instrumentation," dated February 27, 2024, ADAMS Accession No. ML24058A357.

5. E-mail from S. P. Wall, NRC, to M. K. Scarpello, I&M, "Final RAI - D.C. Cook 1 & 2 - License Amendment Request Regarding Neutron Flux Instrumentation (EPID No. L-2023-LLA-0011)," dated June 26, 2024, ADAMS Accession No. ML24178A043.

6. Letter from J. F. Stang, NRC, to R. P. Powers, I&M, "Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," dated December 23, 1999, ADAMS Accession No. ML003672677.

7. Letter from R. P. Powers, I&M, to NRC, "License Amendment Request for Credit of Rod Cluster Control Assemblies for Cold Leg Large Break Loss-of-Coolant Accident Subcriticality," dated September 17, 1999, ADAMS Accession Nos. ML17326A142 (submittal letter), and ML17326A145 and ML17326A146 (attachments to the letter).