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August 15, 2003

AEP:NRC:3334-01
10 CFR 50.90

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

SUBJECT: Donald C. Cook Nuclear Plant Unit 2
Docket No. 50-316
Response to Request For Additional Information Regarding License Amendment Request to Revise Low Pressurizer Pressure Safety Injection Setpoint (TAC NO. MB8202)

REFERENCES:

1) Letter from J. E. Pollock, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2 Docket No. 50-316 License Amendment Request to Revise Low Pressurizer Pressure Safety Injection Setpoint," AEP:NRC:3334, dated March 27, 2003.

2) Letter from Mohammed Shuaibi, Nuclear Regulatory Commission, to A. Christopher Bakken III, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit 2 – Request for Additional Information, 'License Amendment Request to Revise Low Pressurizer Pressure Pressure Safety Injection Setpoint,' (TAC No. MB8202)," dated July 17, 2003.

Dear Sir or Madam:

This letter provides Indiana Michigan Power Company's (I&M's) response to a Nuclear Regulatory Commission (NRC) request for additional information (RAI) regarding a proposed license amendment. The proposed license amendment would revise the low pressurizer pressure safety injection (SI) setpoint and the engineered safety features interlock P-11 setpoint in the Donald C. Cook Nuclear Plant (CNP) Unit 2 Technical Specification (TS).

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By Reference 1, I&M, the licensee for CNP Unit 2, proposed to amend Appendix A, TS, of Facility Operating License DPR-74. I&M proposed the following:

- Revise the low pressurizer pressure SI trip setpoint in TS Table 3.3-4 from its current value of greater than or equal to 1900 pounds per square inch gauge (psig), to greater than or equal to 1815 psig.
- Revise the low pressurizer pressure SI allowable value in TS Table 3.3-4 from greater than or equal to 1890 psig, to greater than or equal to 1805 psig.
- Revise the P-11 setpoint in the TS Engineered Safety Features Interlocks table from its current value of greater than or equal to 2010 psig, to greater than or equal to 1915 psig.

I&M also proposed format changes to the affected TS pages that improve appearance but do not affect any requirements.

The primary reason for I&M to propose this TS change is to reduce unnecessary distractions for operators responding to a reactor trip.

Reference 2 transmitted an NRC RAI regarding the proposed amendment, and documented August 8, 2003, as a mutually agreeable target date for I&M's response to the RAI. In phone discussions with the NRC, I&M informed Mr. Mohammed Shuaibi of the staff that submittal of the response by the target date was not achievable, and that the response would be submitted no later than August 15, 2003.

Enclosure 1 provides an affirmation pertaining to the statements made in this letter. Enclosure 2 provides the response to the NRC RAI. The information in this letter provides supporting information for the amendment request submitted by Reference 1. The information does not alter the validity of the original evaluation of significant hazards consideration performed in accordance with 10 CFR 50.92 documented in Enclosure 2 to Reference 1. The environmental assessment provided in Enclosure 2 to Reference 1 also remains valid.

This letter contains no new commitments. Should you have any questions, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs at (269) 697-5806.

Sincerely,

A. C. Bakken III
Senior Vice President, Nuclear Operations

KS/rdw

Enclosures:

1. Affirmation
2. Response to Nuclear Regulatory Commission Request for Additional Information

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures
J. T. King, MPSC, w/o enclosures
MDEQ – WHMD/HWRPS, w/o enclosures
NRC Resident Inspector
M. A. Shuaibi, NRC Washington, DC

bc: A. C. Bakken III, w/o enclosures
M. J. Finissi
J. B. Giessner
D. W. Jenkins, w/o enclosures
J. A. Kobyra, w/o enclosures
R. J. Kohrt
J. G. Kovarik
B. A. McIntyre, w/o enclosures
J. E. Newmiller
K. J. O'Connor
D. J. Poupard
M. K. Scarpello, w/o enclosures
W. A. Wagner
T. K. Woods, w/o enclosures
J. A. Zwolinski, w/o enclosures

Enclosure 1 to AEP:NRC:3334-01

AFFIRMATION

I, A. Christopher Bakken III, being duly sworn, state that I am Senior Vice President, Nuclear Operations of American Electric Power Service Corporation and Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this response with the 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.

American Electric Power Service Corporation

A. C. Bakken III
Senior Vice President, Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15 DAY OF August, 2003

Danielle M. Schrader
Notary Public

My Commission Expires Apr. 4, 2004

DANIELLE M. SCHRADER
Notary Public, Berrien County, MI
My Commission Expires Apr 4, 2004

Enclosure 2 to AEP:NRC:3334-01

Response to Nuclear Regulatory Commission Request for Additional Information

This enclosure provides Indiana Michigan Power Company's (I&M's) response to the Nuclear Regulatory Commission (NRC) request for additional information transmitted by Reference 1.

NRC Question 1

The licensee states, "Approval of these changes will alleviate an operator concern that a safety injection (SI) actuation is imminent following a reactor trip." The staff understands this is proposed to reduce the number of SI signals that result from transients that do not require SI and to reduce necessary operator actions. Please justify how the change in the setpoint will allow the plant to successfully mitigate transients that require SI.

I&M Response to NRC Question 1

For clarity, the purpose of the proposed amendment has been summarized below, followed by a description of how the proposed setpoint will allow the plant to successfully mitigate transients that require an SI.

Purpose of the Proposed Amendment

The purpose of the proposed amendment is to increase the margin between the low pressurizer pressure safety injection (LP SI) actuation setpoint and the minimum pressurizer pressure that occurs immediately following a normal reactor trip (i.e., a reactor trip in which there are no significant perturbations of the primary and secondary coolant systems other than those caused by the trip). With the existing LP SI actuation setpoint, this margin is only 20 to 40 pounds per square inch. This small margin results in an unnecessary distraction for operators responding to the trip since it indicates that SI actuation may be imminent even though diverse indications of plant conditions reveal that SI is not necessary. I&M has determined that the small margin is the result of excess conservatism in the LP SI actuation setpoint, since:

- The minimum pressurizer pressure that occurs immediately following a normal reactor trip at Donald C. Cook Nuclear Plant (CNP) Unit 2 is similar to that which occurs at other Westinghouse four-loop plants and consistent with the design of the plant.
- The existing LP SI actuation setpoint is higher than that of other Westinghouse four-loop plants.
- The LP SI actuation setpoint proposed for CNP Unit 2 is similar to that of other Westinghouse four-loop plants.
- As described below, the unit specific accident analyses that credit LP SI actuation assume a LP SI actuation setpoint below that proposed for CNP Unit 2.

How the Proposed Setpoint Will Allow Successful Mitigation of Transients that Require Safety Injection.

There are five CNP Unit 2 accident analyses that credit initiation of SI due to an LP SI actuation. These accidents are a large break loss of coolant accident (LOCA), small break LOCA, and three non-LOCA accidents; a steamline break (SLB), a feedline break, and a SLB mass and energy (M/E) release outside containment. The analyses of these accidents demonstrate how the proposed LP SI actuation setpoint of 1815 pounds per square inch gage (psig) will allow the plant to successfully mitigate transients that require SI.

The analysis of a large break LOCA is documented in Section 14.3.1 of Unit 2 Chapter 14 in the CNP Updated Final Safety Analysis report (UFSAR). As indicated in Table 14.3.1-3, an LP SI actuation setpoint of 1715 pounds per square inch absolute (psia) (1700 psig) was assumed in the analysis. This analysis is documented in detail in Reference 2. Attachment 2, Table 3.1-3 of Reference 2 confirms that the LP SI actuation setpoint assumed in the analysis was 1715 psia.

The analysis of a small break LOCA is documented in Section 14.3.2 of Unit 2 Chapter 14 in the CNP UFSAR. As indicated in Table 14.3.2-1, an LP SI actuation setpoint of 1715 psia was assumed in the analysis. This analysis is documented in detail in Reference 3, which was approved by the NRC as documented in Reference 4. Although the analysis described in Reference 3 has been revised since NRC approval, the LP SI actuation setpoint assumed in the analysis remains 1715 psia.

The analysis of an SLB accident is described in Section 14.2.5 of Unit 2 Chapter 14 in the CNP UFSAR. This section reflects the analysis submitted by Reference 5 and approved by the NRC as documented in Reference 6. As indicated in Table 14.2.5-2, SI would actuate on low steamline pressure for cases in which there is a complete severance of the pipe. Therefore, the proposed LP SI actuation setpoint would not affect the analyses for these cases. The only analysis that credits an LP SI actuation is the analysis of the case involving the spurious opening of a steam dump or relief/safety valve. This case was reanalyzed in 1993 assuming an LP SI actuation setpoint of 1715 psia. The reanalysis demonstrated that all acceptance criteria would be met, i.e., there would be no return to criticality and the departure from nucleate boiling (DNB) transient would be bounded by the SLB cases that do not credit an LP SI actuation. The reanalysis has been incorporated into the CNP Unit 2 licensing basis in accordance with 10 CFR 50.59, and will be reflected in the next UFSAR update provided to the NRC in accordance with 10 CFR 50.71(e).

The analysis of a feedline break accident is described in Section 14.2.8 of Unit 2 Chapter 14 in the CNP UFSAR. This section reflects the analysis submitted by Reference 5 and approved by the NRC as documented in Reference 6. This accident was reanalyzed in 1993 assuming an LP SI actuation setpoint of 1715 psia. The reanalysis demonstrated that all acceptance criteria would be met, i.e., the margin to hot leg boiling would remain essentially unchanged relative to the analysis currently described in the UFSAR, no core geometry changes occur, and the core

would remain intact with no loss of cooling ability. The pressurizer water level transient would not challenge pressurizer overfill criteria, and the maximum RCS pressure would remain well below the 110% design pressure limit. DNB is not an acceptance criterion of concern in a feedline break accident. The reanalysis has been incorporated into the CNP Unit 2 licensing basis in accordance with 10 CFR 50.59, and will be reflected in the next UFSAR update provided to the NRC in accordance with 10 CFR 50.71(e).

The analysis of a SLB M/E release outside containment is described in Section 14.4.2 of Unit 2 Chapter 14 in the CNP UFSAR. This analysis is documented in detail in Reference 7, which was approved by the NRC as documented in Reference 8. A review of supporting documentation and input assumptions for the analysis identify that an LP SI actuation setpoint of 1715 psia was assumed. The proposed setpoint change is consistent with the analysis of record for an SLB M/E release outside containment. The ability to successfully mitigate this transient is demonstrated by the existing analysis.

NRC Question 2

By changing the low pressurizer pressure SI trip setpoint, a time delay is introduced from the time the reactor trips to the time SI is actuated. Are there any effects on the plant due to this time delay? Will there be changes to the emergency operating procedures with respect to the new SI setpoint? What kind of training will operators receive to these procedural changes?

I&M Response to NRC Question 2

The maximum delay in an LP SI actuation that would result from changing the actuation setpoint from 1900 psig to 1815 psig is approximately 20 seconds. This would occur in a feedline break accident. The minimum delay would be a fraction of a second. This would occur in a large break LOCA. The time delay is included in the analyses described in the response to NRC question 1 and all acceptance criteria are met. The analyses use conditions that are more limiting than actual plant conditions. The effect on the plant due to the time delay is bounded by the analyses.

The CNP emergency operating procedures do not specify any actions during the periods in which these delays occur. The only changes to the CNP emergency operating procedures that will be necessary to implement the proposed change to the LP SI actuation setpoint will be the changes to the procedure steps that explicitly state the numerical value for the setpoint. I&M intends to present the new setpoints and the basis for the change to licensed operators as either classroom or familiarization training. I&M's procedures for implementing TS changes contain provisions to ensure that the necessary procedure revisions and training will occur.

NRC Question 3

Describe the methodology used to determine the new SI setpoint and the analysis which demonstrates the new setpoint is still bounded by the uncertainty margin.

I&M Response to NRC Question 3

As described in the response to NRC question 1, the five accident analyses that credit an LP SI actuation all assume a setpoint of 1700 psig, which bounds the proposed setpoint of 1815 psig. The methodology used to determine the setpoint uncertainties in these analyses is described in Reference 9. This methodology was approved by the NRC as documented in Reference 10. In accordance with this methodology, the required LP SI actuation setpoint must be greater than or equal to 1809.6 psig. The proposed LP SI actuation setpoint of 1815 psig meets the requirements of approved methodology and is consistent with the current Unit 1 LP SI actuation setpoint.

NRC Question 4

The licensee states there was a design change that provides a 3.5 second delay to the auxiliary feedwater flow retention circuit which exacerbates the operator's concern about reactor coolant system cooling. What is the purpose of the 3.5 second delay? What would be the consequences of eliminating the 3.5 second delay?

I&M Response to NRC Question 4

The auxiliary feedwater system includes a flow retention circuit which protects the auxiliary feedwater pumps from pump runout conditions following a main steam or feedwater line break, or other condition resulting in pump runout. This 3.5 second time delay was installed in 1997 and is not related to this amendment request. Prior to adding the 3.5 second time delay, a momentary spike in auxiliary feedwater discharge pressure occurred when the two motor-driven auxiliary feedwater pumps automatically started. This resulted in a false flow retention signal since high differential pressures correspond to high flow conditions during steady state flow conditions. When the circuit is actuated, safety-related, motor-operated discharge valves automatically throttle to an intermediate position. Time delay relays were incorporated into the circuit to delay the actuation of the flow retention system by ensuring that the sensed high flow condition is more than a momentary pressure pulse. The consequences of eliminating the 3.5 second delay would be potential return of spurious actuations of safety related valves beyond the intent of the design for the flow retention circuit.

The 3.5 second time delay has no affect on the proposed change to the LP SI actuation setpoint. A reactor trip and, if required, SI would have already occurred by the time flow retention would be required to actuate.

NRC Question 5

Which licensing basis transient and accident analysis have been reevaluated to confirm lowering the SI setpoint is acceptable? Are there any effects on departure from nucleate boiling or fuel design limits due to this change?

I&M Response to NRC Question 5

As described in the response to NRC question 1, no re-analyses were performed to confirm that lowering the LP SI actuation setpoint was acceptable. Two existing analyses were incorporated into the Unit 2 design and licensing basis in support of this proposed amendment. These were an analysis of an SLB caused by a spurious opening of a steam dump or relief/safety valve, and an analysis of a feedline break accident. As described in the response to NRC question 1, these analyses demonstrated that the effects on departure from nucleate boiling and fuel design limits would be acceptable.

NRC Question 6

In your application you indicated that a loss-of-coolant accident (LOCA), feedwater line break, and an inadvertent depressurization of the main steam system are affected by the low pressurizer pressure SI setpoint. Please confirm that no other transients are affected by the low pressurizer pressure SI setpoint. Describe in detail all transients, including but not limited to LOCA, feedwater line break, and inadvertent depressurization of the main steam system, that are affected by this modification and demonstrate that the new SI setpoint doesn't hamper the system's ability to successfully mitigate them.

I&M Response to NRC Question 6

As described in the response to NRC question 1, the only accident analyses that credit an LP SI actuation are those for a large break LOCA, small break LOCA, SLB, feedline break, and SLB M/E release outside containment. The response to NRC question 1 identifies the analyses that demonstrate acceptable results assuming the proposed LP SI actuation setpoint. No other analyses are affected by the proposed change to the LP SI actuation setpoint.

References

1. Letter from Mohammed Shuaibi, Nuclear Regulatory Commission, to A. Christopher Bakken III, Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit 2 – Request for Additional Information, 'License Amendment Request to Revise Low Pressurizer Pressure Pressure Safety Injection Setpoint,' (TAC No. MB8202)," dated July 17, 2003.

2. Letter from M. W. Rencheck, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission Document Control Desk, "Donald C. Cook Nuclear Plant Unit 2 Annual Report of Loss-of-Coolant Accident Evaluation Model Changes and Submittal of New Large Break Loss-of-Coolant Accident Analysis of Record for Unit 2," C0200-08, dated February 2, 2000.

3. Letter from E. E. Fitzpatrick, Indiana Michigan Power Company, to T. E. Murley, U. S. Nuclear Regulatory Commission Document Control Desk, "Technical Specifications Change To Increase The Allowable Tolerance For Main Steam Safety Valves," AEP:NRC:1169, dated November 11, 1992.

4. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendments Re: Increased Main Steam Safety Valve Setpoint Tolerances (TAC Nos. M84979 and M84980)," dated September 9, 1994.

5. Letter from M. P. Alexich, Indiana Michigan Power Company, to T. E. Murley, U. S. Nuclear Regulatory Commission Document Control Desk, "Unit No. 2 Cycle 8 Reload Licensing, Proposed Technical Specification for Unit 2 Cycle 8, and Related Unit 1 Proposals," AEP:NRC:1071E dated February 6, 1990.

6. Letter from Timothy G. Colburn, U. S. Nuclear Regulatory Commission, to M. P. Alexich, Indiana Michigan Power Company, "Amendment No. 148 and 134 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC NOS. 75395, 75396, and 76816)," dated August 27, 1990.

7. Letter from M. P. Alexich, Indiana Michigan Power Company, to T. E. Murley, U. S. Nuclear Regulatory Commission Document Control Desk, "Technical Specification Change Request Bit Boron Concentration Reduction," AEP:NRC:1140 dated March 26, 1991.

8. Letter from William O. Long, Sr., U. S. Nuclear Regulatory Commission, to Eugene E. Fitzpatrick, Indiana Michigan Power Company, "Amendment Nos. 158 and 142 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC Nos. 80262 and 80263)," dated November 20, 1991.

9. Westinghouse Topical Report WCAP-12741, "STEPIT Menu Driven Setpoint Calculation Program," dated September 1991.

10. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Revised Safety Evaluation for Amendment Nos. 175 and 160 RE: Reactor Protection System Upgrade Project (TAC Nos. M84839 and M84840)," dated May 13, 1994.