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April 29, 2014

AEP-NRC-2014-27
10 CFR 50.90

Docket No.: 50-315

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

Donald C. Cook Nuclear Plant Unit 1
Response to "Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)"

References:

1. Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Unit 1 Docket No. 50-315, License Amendment Request Regarding Restoration of Normal Reactor Coolant System Operating Pressure and Temperature Consistent with Previously Licensed Conditions," dated October 8, 2013, ADAMS Accession Number ML13283A121.

2. Letter from T. J. Wengert, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Unit 1, Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)," dated March 31, 2014, ADAMS Accession Number ML14066A311.

By letter dated October 8, 2013 (Reference 1), Indiana Michigan Power Company (I&M) submitted an application for a license amendment to restore the normal reactor coolant system operating pressure and temperature consistent with previously licensed conditions for the Donald C. Cook Nuclear Plant, Unit 1. By letter dated March 31, 2014, the U. S. Nuclear Regulatory Commission (NRC) staff requested additional information (Reference 2) to complete the review of Reference 1.

Enclosure 1 to this letter provides an affirmation statement. Enclosure 2 provides a cross-reference for I&M's response to the NRC Request for Additional Information (RAI). Enclosure 3 provides responses to RAIs Reactor Systems Branch (SRXB) 7.c, SRXB 7.d, SRXB 7.e, Component Performance, Nondestructive Examination and Testing Branch (EPNB) RAI 1 and EPNB RAI 2. Enclosure 5 contains a non-proprietary Westinghouse report for the remaining RAIs.

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Enclosure 6 expands on selected responses contained in Enclosure 5 with proprietary information. Since Enclosure 6 contains information that is proprietary to Westinghouse, Enclosure 4 contains an Application for Withholding Proprietary Information from Public Disclosure.

This letter contains no new or revised commitments. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

JJV/amp

Enclosures:

1. Affirmation
2. Request for Additional Information Response Enclosure Cross-Reference
3. Responses to Request for Additional Information Reactor Systems Branch (SRXB) 7.c, SRXB 7.d, SRXB 7.e, Component Performance, Nondestructive Examination and Testing Branch (EPNB) RAI 1 and EPNB RAI 2
4. Application for Withholding Proprietary Information from Public Disclosure for Enclosure 6, Westinghouse Letter, LTR-PL-14-17, P-Attachment
5. Westinghouse Letter, LTR-PL-14-17 NP-Attachment [non-Proprietary]
6. Westinghouse Letter, LTR-PL-14-17 P-Attachment [Proprietary]

c: J. T. King, MPSC
MDEQ-RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
T. J. Wengert, NRC Washington, DC
A. J. Williams, AEP Ft. Wayne, w/o enclosures

ENCLOSURE 1 TO AEP-NRC-2014-27

AFFIRMATION

I, Joel P. Gebbie, being duly sworn, state that I am 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

Joel P. Gebbie
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 29 DAY OF April, 2014

Notary Public

My Commission Expires 04-04-2018

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2018
Acting in the County of Berrien

Enclosure 2

Donald C. Cook Nuclear Plant Unit 1

Request for Additional Information Response Enclosure Cross-Reference

The Request for Additional Information (RAI) responses are contained in separate Enclosures. Table 1 below provides a cross-reference between the RAI and the Enclosure providing the RAI Response.

Table 1

RAI Response Enclosure Cross Reference

Reactor Systems Branch (SRXB) RAI-1)	Section 5.1.1, "Best-Estimate Large-Break LOCA [loss of coolant accident]," of WCAP-17762-NP indicates that the proposed normal operating pressure (NOP)/normal operating temperature (NOT) restoration was evaluated using the analysis of record (AOR), approved in 2008, as a baseline. The WCAP is clear that the AOR included the 571 °F value within its range of reactor coolant system (RCS) average temperature (T_{ave}); however, the hot full power RCS pressure is pressure is presented, from the AOR, at both 2100 and 2250 psia. The 2008 ASTRUM implementation LAR (ADAMS Accession No. ML080090268) also includes an allowance for both pressure bands (see Table 1 of Enclosure 2 to ASTRUM LAR), but it is not clear how the analysis accounts for these pressure bands.	
1.a)	Explain how the AOR accounts for the two pressure bands.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
1.b)	Explain whether the AOR peak clad temperature (PCT) case reflects the higher RCS pressure.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
1.c)	Explain why the AOR value provided in WCAP-17762-NP-A 3 (2128 °F) differs from that contained in the ASTRUM implementation LAR (2106 °F).	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
1.d)	Explain whether the thermal conductivity degradation (TCD) estimate (ADAMS Accession No. ML12088A104) treated the RCS pressure consistently with the AOR and/or the WCAP.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary

SRXB RAI-2)	Section 5.1.1, "Best-Estimate Large-Break LOCA," of WCAP-17762-NP, contains the following passage: "Due to the non-linear effects of the design input changes (which were updated relative to the assessment *reported in Reference 3 [ADAMS Accession No. ML12088A104 - estimated effects of thermal conductivity* degradation {TCD}]), the return to NOP/NOT evaluation is being assessed against the Cook Unit 1 BE [best estimate] LB [large break] LOCA analysis of record (AOR), which was submitted in Reference 7 [ADAMS Accession No. ML080090268 - ASTRUM LAR] and approved by the USNRC ... Additionally, due to different cases becoming limiting at NOP/NOT conditions, the prior PCT assessment reported in Reference 10 [August 30, 2013, 30-day report of significant emergency core cooling system (ECCS) Evaluation Model error/change] is also re-considered ..." The method of selecting limiting cases to determine the effect of a model change on the PCT prediction has been previously reviewed and accepted by the NRC staff; however, the method of identifying and analyzing the case sub-set is a topic of plant-specific review (see, for example, ADAMS Accession No. ML12173A025 - D.C. Cook Response to Request for Additional Information related to TCD estimate). Please provide information to enable NRC staff review of the case subset selection and validation process.	
2.a)	Provide a matrix of the significant sampled input parameters from the AOR and the various cases executed to estimate the effects of TCD, model changes and error corrections, and the restoration of NOP/NOT conditions.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
2.b)	Provide a summary of the case sub-set selection process: explain how the limiting cases were identified and what attributes were identified for the newly limiting cases.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
2.c)	Explain how the case sub-set selection method was validated, and how the results were verified to be limiting.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary

SRXB RAI-3)	The licensee presents its evaluation of the NOP/NOT restoration with respect to the small break (SB) LOCA analysis in Section 5.1.2 of WCAP-17762-NP. The evaluation is based on an SBLOCA analysis that was provided to the commission by letter dated August 31, 2012 (ADAMS Accession No. ML12256A685). The succinct evaluation provided in WCAP-17762-NP concludes that the revised SBLOCA analysis explicitly accounts for the restored NOP/NOT conditions. Noting that the August 2012, SBLOCA analysis was provided to the Commission, rather than submitted for review and approval, the NRC staff is reviewing the SBLOCA analysis as part of the NOP/NOT review effort to verify that it satisfies applicable regulatory requirements and confirm that it accounts for the proposed NOP/NOT operating conditions.	
3.a)	Figure 6 provides the core mixture level for the 3.25-inch (limiting) break. The figure shows that the core mixture level remains below 20 feet for a significant period of time (i.e., about 2500 seconds), despite that the PCT node is located at 11.75 feet (NRC staff infers that this elevation corresponds to approximately 21.8 feet on Figure 6). At the time of PCT, 1483 seconds, the hot node does not appear to be covered. The mixture level appears closer to 14.5 feet. Additionally, the rod film heat transfer coefficient depicted in Figure 15 shows that the coefficient is reasonably stable below approximately 50 BTU/hr/ft^2/°F, from 1000 through 3000 seconds of the transient. Furthermore, the accumulators begin to empty 200 seconds prior to time of PCT. Please explain how the PCT temperature excursion is being terminated and provide supporting tables and plots with additional data from the NOTRUMP and LOCTA runs.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
3.b)	Describe the loop seal clearing behavior depicted in Table 6 in greater detail. For all the breaks, provide the thermal-hydraulic conditions present in the intact loop seals. For the limiting break in particular, describe the reactor coolant conditions immediately prior to and following the loop seal clearing, especially with regard to the effect that the loop seal clearing has on the mixture level transient and system pressure.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary

3.c)	The greatest fraction of pumped safety injection flows into the broken loop. Due to the large variation in liquid flow out the break throughout the duration of the transient, it is difficult to evaluate the broken loop flow behavior. Please provide detailed plots of liquid and vapor flow rates at the junctions or links connecting the broken loop to pumped safety injection sources, the break, the reactor coolant pump, and the vessel, for the first 2000 seconds of the limiting break. Include scaling appropriate prior to and following the loop seal clearing.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
3.d)	Describe the modeling of flow paths between the downcomer and the upper plenum and core barrel.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
3.e)	Provide plots of the hot assembly void fraction as a function of height for the limiting break at the time of minimum core level, and again at the time of PCT.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
3.f)	Provide the mass flow rate at the core exit as a function of time for the limiting break.	Enclosure 5, Non-Proprietary Enclosure 6, Proprietary
SRXB RAI-4)	The D.C. Cook post-LOCA long term cooling (LTC) analyses demonstrate that boric acid concentration control measures are adequate, and that the ECCS recirculation flows "dilute the core and replace core boil-off, thus keeping the core quenched." WCAP 17762-NP refers to an analysis (ADAMS Accession No. ML11195A025), which is performed for D.C. Cook Unit 2.	
4.a)	Please explain how the calculation concludes that the ECCS recirculation flow is adequate.	Enclosure 5, Non-Proprietary
4.b)	Please address differences between the D.C. Cook Units.	Enclosure 5, Non-Proprietary
4.c)	The Unit 2 analysis states the following: "The current hot leg switchover time and plant operating procedures result in ECCS flows that temporarily drop below the injected flow necessary to replace core boil-off (plus entrainment) during the HLSO process."	
	4.c (i) Please explain what consequence, if any, the hot leg swap over evolution could have on maintaining a stable core quench.	Enclosure 5, Non-Proprietary

	4.c (ii) Please explain how this calculation accounts for entrainment.	Enclosure 5, Non-Proprietary
4.d)	Section 5.1.3 of WCAP-17762 does not appear to indicate, as other sections of the WCAP do, that the boric acid precipitation analysis reflects the Unit 1 NOP/NOT values. The WCAP states, "The inputs used to perform post-LOCA LTC analyses include core power levels, fuel dimensions, and RCS and ECCS volumes, temperatures, pressures, and boron concentrations." Explain whether, and how, these inputs are affected by the NOP/NOT restoration, and whether, and how, the analysis accounts for the NOP/NOT restoration.	Enclosure 5, Non-Proprietary
SRXB RAI-5)	Subsection 5.2.1, "Introduction and Background," to Section 5.2, "Non-LOCA Transients," discusses evaluations for events that take credit for the lower temperature/pressure, stating, "In particular were the overtemperature ΔT (OT ΔT) and overpower ΔT (OP ΔT) setpoints, which utilized T' and T" values that were restricted below the full power Tavg primarily to provide overpower protection while maintaining the same ΔT setpoints." Subsection 5.2.3.2 discusses the Uncontrolled Rod Withdrawal at Power, and states, "Additionally, it was confirmed as part of the Return to RCS NOP/NOT Program that the OT ΔT setpoints modeled in the current analysis remain valid at NOP/NOT conditions." Please explain how this confirmation was performed and provide additional detail regarding the results of the confirmation. In particular, explain whether the T' and T" values assumed in the Rod Withdrawal at Power analyses reflect the more restrictive values, and if so, how the setpoints remain valid for the proposed operating conditions.	Enclosure 5, Non-Proprietary
SRXB RAI-6)	The Steam Generator Tube Rupture Margin to Overfill (MTO) analysis discussed in Section 5.3.4 refers to NSAL 07-11. Please provide a copy of NSAL 07-11.	Enclosure 5, Non-Proprietary

SRXB RAI-7)	WCAP-17762-NP, Section 5.3.4.4, indicates that the present MTO analysis was re-assessed to address the proposed changes in RCS conditions, and to address the issues identified in NSAL 07-11. This analysis was *incorporated into the Cook Unit 1 licensing basis via Amendment 256. The approving safety evaluation notes that* the methods based on those described in WCAP-10698-P-A, along with the LOFTTR2 code, were used to evaluate SGTR MTO. However, analytic assumptions were consistent with the Cook licensing basis and not necessarily the analysis approved in WCAP-10698-P-A.	
7.a)	The SE approving WCAP-10698-P-A states, "The design basis SGTR analysis assumes LOOP, the most reactive stuck rod, conservative initial conditions, safeguards capacities and setpoints, turbine runback, 120% of 1971 ANS decay heat rate, and the worst single failure." The NRC staff understands that these assumptions are not necessarily employed in the revised Cook MTO analysis. Explain which of these assumptions are applied to the Cook MTO analysis. Provide specific information regarding the "conservative initial conditions."	Enclosure 5, Non-Proprietary
7.b)	The Updated Final Safety Analysis Report indicates that the secondary volume of the replacement steam generators, which is smaller than that of the original steam generators, remains sufficient to accommodate the integrated leakage during the SGTR event. Explain whether the updated MTO analysis reflects the smaller volume of the replacement steam generators.	Enclosure 5, Non-Proprietary
7.c)	The MTO analysis approved in 2001 includes the assumption of time-critical operator actions. Explain what effect the NOP/NOT restoration will have on the time available to complete these actions.	Enclosure 3
7.d)	The 2000-2001 NRC staff review of the MTO LAR included an assessment of the licensee's ability to execute the time-critical operator actions credited in the analysis. Provide an update to this assessment.	Enclosure 3

7.e)	The 2000-2001 NRC staff review of the MTO LAR revealed that the analysis credited a substantial amount of non-safety grade equipment. For example, remote manual operation (i.e., using a switch in the control room) of SG PORVs is credited in the analysis. Nitrogen bottles are provided to ensure operability of the PORVs as added safety margin in the event of a coincident loss of offsite power. Explain whether the available bottled nitrogen supply would permit SG PORV operation for the duration of the analyzed MTO event. In addition, explain whether the SG PORV can be operated by local, manual action. If so, provide a quantitative estimate of the time required to execute a local PORV operation.	Enclosure 3
Component Performance, Nondestructive Examination and Testing Branch (EPNB) RAI-1)	Enclosure 7 to the October 8, 2013, submittal discusses the assessment of the impact of increased temperature and pressure on various systems. However, the assessment did not discuss any previously identified degradation that was evaluated and found acceptable under the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI flaw evaluation methods. Identify all evaluations that were completed using the lower temperature and pressure and identify if these evaluations will be revised to take into account the higher temperature and pressure. If not, provide justification.	Enclosure 3
EPNB RAI-2)	The examination frequency of control rod drive mechanism (CRDM) penetration nozzles is related to the temperature of the reactor vessel head and is based on the Effective Degradation Years (EDY) and Reinspection Years (RIY) calculations as specified in ASME Code Case N-729-1 as conditioned in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(g)(6)(ii)(D). Discuss whether the lower temperature was used in calculating RIV and EDY parameters for the examination frequency of the CRDM penetration nozzles in the previous licensing actions. If yes, discuss whether the RIY and EDY calculations will be revised to take into account the higher temperature to obtain the examination frequency for the CRDM nozzles. If not, provide justification.	Enclosure 3

ENCLOSURE 3 TO AEP-NRC-2014-27

DONALD C. COOK NUCLEAR PLANT UNIT 1

Responses to Request for Additional Information (RAI)
Reactor System Branch (SRXB) 7.c, SRXB 7.d, SRXB 7.e, Component Performance Nondestructive Examination and Testing Branch (EPNB) RAI 1 and EPNB RAI 2

ENCLOSURE 3 TO AEP-NRC-2014-27

Responses to Request for Additional Information (RAI)
Reactor System Branch (SRXB) 7.c, SRXB 7.d, SRXB 7.e, Component Performance Nondestructive Examination and Testing Branch (EPNB) RAI 1 and EPNB RAI 2

Reactor Systems Branch (SRXB)

SRXB RAI-7) WCAP-17762-NP, Section 5.3.4.4, indicates that the present MTO analysis was re-assessed to address the proposed changes in RCS conditions, and to address the issues identified in NSAL 07-11. This analysis was incorporated into the Cook Unit 1 licensing basis via Amendment 256. The approving safety evaluation notes that the methods based on those described in WCAP-10698-P-A, along with the LOFTTR2 code, were used to evaluate SGTR MTO. However, analytic assumptions were consistent with the Cook licensing basis and not necessarily the analysis approved in WCAP-10698-P-A.

7.a) The SE approving WCAP-10698-P-A states, "The design basis SGTR analysis assumes LOOP, the most reactive stuck rod, conservative initial conditions, safeguards capacities and setpoints, turbine runback, 120% of 1971 ANS decay heat rate, and the worst single failure." The NRC staff understands that these assumptions are not necessarily employed in the revised Cook MTO analysis. Explain which of these assumptions are applied to the Cook MTO analysis. Provide specific information regarding the "conservative initial conditions."

Response: See Enclosure 5, Non-Proprietary

7.b) The Updated Final Safety Analysis Report indicates that the secondary volume of the replacement steam generators, which is smaller than that of the original steam generators, remains sufficient to accommodate the integrated leakage during the SGTR event. Explain whether the updated MTO analysis reflects the smaller volume of the replacement steam generators.

Response: See Enclosure 5, Non-Proprietary

7.c) The MTO analysis approved in 2001 includes the assumption of time-critical operator actions. Explain what effect the NOP/NOT restoration will have on the time available to complete these actions.

Response: The time available to perform operator actions was not changed. The evaluation performed for the normal operating pressure/normal operating temperature (NOP/NOT) program confirmed that the existing times were acceptable at NOP/NOT operating conditions.

7.d) The 2000-2001 NRC staff review of the MTO LAR included an assessment of the licensee's ability to execute the time-critical operator actions credited in the analysis. Provide an update to this assessment.

Response: The operator Time Critical Actions (TCA) required in the steam generator tube rupture (SGTR) event, which are unchanged from those reviewed by the U. S. Nuclear Regulatory Commission (NRC) in 2000-2001, are now controlled in a formal TCA Program. The program is implemented by the three procedures listed after this paragraph. These documents describe the program requirements and responsibilities, validate each TCA at least every five years, and provide a controlled engineering source document that contains all plant TCAs.

- PMI-4075, Operator Time Critical Actions
- PMP-4075-TCA-001, Time Critical Action Validation and Verification
- 12-EHP-4075-TCA-001, Operator Time Critical Actions

In addition, the performance of licensed operators during SGTR training scenarios on the simulator is continuously examined to ensure that design basis analysis acceptance criteria, including avoidance of steam generator (SG) overfill, are satisfied. Crew remediation is conducted if simulator results indicate a valid failure to maintain margin to overfill. The periodic TCA validation program and the continuous active monitoring of simulator training exercises, including crew remediation when necessary, provide a high confidence level that the operating crews have the ability to execute the time-critical operator actions credited in the SGTR accident analysis.

7.e) The 2000-2001 NRC staff review of the MTO LAR revealed that the analysis credited a substantial amount of non-safety grade equipment. For example, remote manual operation (i.e., using a switch in the control room) of SG PORVs is credited in the analysis. Nitrogen bottles are provided to ensure operability of the PORVs as added safety margin in the event of a coincident loss of offsite power. Explain whether the available bottled nitrogen supply would permit SG PORV operation for the duration of the analyzed MTO event. In addition, explain whether the SG PORV can be operated by local, manual action. If so, provide a quantitative estimate of the time required to execute a local PORV operation.

Response: For background, it is agreed that the original Unit 1 and Unit 2 licensing basis for SGTR accident mitigation and the supplemental Margin to Overfill analysis methodology, the latter of which was the subject of a license amendment request (LAR) submittal in October 2000, and approved Unit 1 and Unit 2 License Amendments in October 2001, credits the use of non-safety related equipment.

The list of systems and components credited in the analysis was provided to the NRC on June 29, 2001, in response to Question Number 4 of a Request for Additional Information (RAI) submitted during staff review of the October 2000 LAR. It was noted in the response to Question Number 4 that SG power operator relief valves (PORV), which are used for reactor coolant system cool down, rely on non-safety grade electrical and control air appurtenances and do not have safety grade backups.

In response to the specific question in RAI-7e regarding credit for nitrogen bottles for SG PORV operability in the event of a coincident loss of offsite power, it is noted that SG PORV operability at Donald C. Cook Nuclear Plant (CNP) for SGTR accident mitigation has never depended on the availability of nitrogen bottles with or without offsite power available to the affected unit. When the original NRC Safety Evaluation Review for CNP was issued in September 1973, a nitrogen system backup for the SG PORVs did not exist, thereby confirming that the plant's licensing basis for SGTR accident mitigation does not include the availability of this system.

The nitrogen backup system to support local-manual operation of SG PORVs was added to the plant design after Unit 1 and Unit 2 began power operation in response to commitments made to new fire protection requirements in NRC Bulletin 75-04 and 75- 04A. This Bulletin was issued as a result of the Browns Ferry fire event in March 1975. The nitrogen backup system was part of the new alternate shutdown system that was installed in Unit 1 and Unit 2 at local control stations throughout the plant using critical instrumentation on local shutdown indication panels. Although the alternate shutdown system provides the ability to operate the SG PORVs by local manual operator actions, this design feature has not been credited in past or current SGTR accident analysis.

Since the response to the first question is that SG PORV operability for SGTR accident mitigation does not credit nitrogen bottles, detailed responses to the additional RAI-7e questions regarding the volume of nitrogen available to support SG PORV operability, the ability to position the SG PORV by local manual action, and the amount of time it would take to position the PORVs locally are not applicable.

Component Performance, Nondestructive Examination and Testing Branch (EPNB)

EPNB RAI-1) Enclosure 7 to the October 8, 2013, submittal discusses the assessment of the impact of increased temperature and pressure on various systems. However, the assessment did not discuss any previously identified degradation that was evaluated and found acceptable under the requirements of the American Society of Mechanical Engineers (ASME)

Boiler and Pressure Vessel Code, Section XI flaw evaluation methods. Identify all evaluations that were completed using the lower temperature and pressure and identify if these evaluations will be revised to take into account the higher temperature and pressure. If not, provide justification.

Response: Unit 1 currently has no instances of previously identified degradation of pressure-retaining components that rely on American Society of Mechanical Engineers (ASME) Section XI flaw evaluation methods for acceptability of service.

EPNB RAI-2) The examination frequency of control rod drive mechanism (CRDM) penetration nozzles is related to the temperature of the reactor vessel head and is based on the Effective Degradation Years (EDY) and Reinspection Years (RIY) calculations as specified in ASME Code Case N-729-1 as conditioned in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(g)(6)(ii)(D). Discuss whether the lower temperature was used in calculating RIY and EDY parameters for the examination frequency of the CRDM penetration nozzles in the previous licensing actions. If yes, discuss whether the RIY and EDY calculations will be revised to take into account the higher temperature to obtain the examination frequency for the CRDM nozzles. If not, provide justification.

Response: CNP conducts examinations of the Unit 1 and Unit 2 Reactor Vessel Closure Heads (RVCH) in accordance with 10 CFR 50.55a (g)(6)(ii)(D), which states in part:

"All licensees of pressurized water reactors shall augment their inservice inspection program with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section."

- ASME Code Case N-729-1, "Alternative Examination Requirements for Pressurized Water Reactor Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," March 2006, provides alternative examination requirements to the requirements of ASME Section XI for Pressurized Water Reactor RVCH having pressure-retaining partial-penetration welds. The augmented alternative examination requirements fall into one of two categories based on RVCH nozzle and weld materials:
- Section 1210(a) – Heads having nozzles fabricated from UNS N06600 (Alloy 600) material with UNS N06082 (Alloy 82) or UNS W86182 (Alloy 182) partial-penetration welds.
- Section 1210(b) – Heads having nozzles fabricated from Primary Water Stress Corrosion Cracking (PWSCC) resistant materials, such as UNS N06690 (Alloy 690) base metal with UNS N-06052 (Alloy 52) or UNS W86152 (Alloy 152) partial-penetration welds.

The original Unit 1 RVCH, which had Alloy 600 penetration nozzles and Alloy 82/182 welds, was replaced in the fall of 2006 with a new RVCH having Alloy 690 nozzles and Alloy 52/152 welds. Therefore, the new RVCH is characterized as PWSCC-resistant.

Per Table 1 of ASME Code Case N-729-1, PWSCC-resistant heads must implement inspection items B4.30 for visual exams and B4.40 for volumetric and surface exams. The normal examination frequency for items B4.30 and B4.40 is not determined by EDY or RIY calculations, but rather is a constant time interval irrespective of unit operating conditions:

- B4.30 – Every third refueling outage or 5 calendar years, whichever is less
- B4.40 – All nozzles, not to exceed one inspection interval (nominally 10 calendar years)

For PWSCC-resistant RVCH nozzles, EDY and RIY calculations only apply in the event flaws are detected during a normal examination, which then invokes Note 8 of Table 1 to determine a new, shorter examination frequency.

A visual inspection of the Unit 1 RVCH was performed in 2011, with no adverse findings. Since no flaws have been detected on the new Unit 1 RVCH, operating conditions are not a factor in determining examination frequency.

ENCLOSURE 4 TO AEP-NRC-2014-27

DONALD C. COOK NUCLEAR PLANT UNIT 1

Application for Withholding Proprietary Information from Public Disclosure for Enclosure 6, Westinghouse Letter, LTR-PL-14-17, P-Attachment

Westinghouse

Westinghouse Electric Company
Engineering, Equipment and Major Projects
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Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
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e-mail: greshaja@westinghouse.com
Proj letter: AEP-14-14

CAW-14-3936

April 14, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Responses to NRC, "Donald C. Cook Nuclear Plant Unit 1 – Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)," (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-14-3936 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Indiana Michigan Power Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference CAW-14-3936, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

for

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared Bradley F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Bradley F. Maurer, Principal Engineer
Plant Licensing

Sworn to and subscribed before me
this 14th day of April 2014

Anne M. Stegman
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Principal Engineer, Plant Licensing, in Engineering, Equipment and Major Projects, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Westinghouse Responses to NRC, 'Donald C. Cook Nuclear Plant Unit 1 - Request for Additional Information on the Application for Amendment to Restore Normal Reactor Coolant System Pressure and Temperature Consistent with Previously Licensed Conditions (TAC No. MF2916)'" (Proprietary), for submittal to the Commission, being transmitted by Indiana Michigan Power Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with NRC approval of WCAP-17762-NP, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Provide input to Indiana Michigan Power Company for input to the U. S. Nuclear Regulatory Commission in response for Additional Information regarding Restoration of Normal Operating Pressure and Normal Operating Temperature.

(ii) Provide licensing support for customer submittal.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of the information to its customers for the purpose of obtaining license changes for a Westinghouse pressurized water reactor (PWR).

(ii) Westinghouse can sell support and defense of the technology to tis customer in the licensing process.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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