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July 30, 2014

AEP-NRC-2014-56
10 CFR 50.4

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
FIRST RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION CONCERNING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

Reference:

1) Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC), "Donald C. Cook Nuclear Plant Units 1 and 2, Transmittal of Reactor Vessel Internals Aging Management Program," dated October 1, 2012. Agencywide Documents Access and Management System (ADAMS) Accession No. ML12284A320.

2) Letter from T. J. Wengert, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program Submittal (TAC Nos. MF0050 and MF0051)," dated June 6, 2014. ADAMS Accession No. ML14135A320.

This letter provides Indiana Michigan Power Company's (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, response to Requests for Additional Information (RAI) by the U. S. Nuclear Regulatory Commission (NRC) regarding CNP's Reactor Vessel Internals Aging Management Program.

By Reference 1, I&M submitted the CNP Reactor Vessel Internals Aging Management Program. By Reference 2, the NRC transmitted RAIs regarding the program. Enclosure 1 provides I&M's response to Reference 2, RAI-1, RAI-5, and RAI-7. Response to RAI-2, RAI-3, RAI-4, RAI-6, and RAI-8 will be provided in accordance with the schedule for response provided by Reference 2. Enclosure 2 contains new regulatory commitments. Enclosure 3 provides a non-proprietary Westinghouse report which contains detailed information for RAI-5. Enclosure 4 expands on selected responses contained in Enclosure 3 with proprietary information. Since Enclosure 4 contains information that is proprietary to Westinghouse, Enclosure 5 contains an Application for Withholding Proprietary Information from Public Disclosure.

A001
NRR

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

DMB/amp

Enclosures:

1. Donald C. Cook Nuclear Plant Response to Request for Additional Information Regarding The Reactor Vessel Internals Aging Management Program
2. Donald C. Cook Nuclear Plant Regulatory Commitments
3. Westinghouse Letter, LTR-RIDA-14-66-NP, Rev. 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" [Non-Proprietary]
4. Westinghouse Letter, LTR-RIDA-14-66-P, Rev. 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" [Proprietary]
5. Westinghouse Application for Withholding Proprietary Information from Public Disclosure, LTR-RIDA-14-66-P, Rev. 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" [Proprietary]

c: M. L. Chawla, NRC Washington, D.C.
J. T. King - MPSC
MDEQ- RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson – AEP Ft. Wayne

PROPRIETARY INFORMATION
Enclosure 4 to this Letter contains proprietary information.
Withhold from public disclosure under 10 CFR 2.390.
Upon removal of Enclosure 4, this Letter is decontrolled.

ENCLOSURE 1 TO AEP-NRC-2014-56

DONALD C. COOK NUCLEAR PLANT
REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

List of Acronyms:

ADAMS	Agencywide Documents Access and Management System
AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
CNP	Donald C. Cook Nuclear Power Plant
CRGT	Control Rod Guide Tube
I&M	Indiana & Michigan Power
JCO	Justification for Continued Operation
NRC	Nuclear Regulatory Commission
OE	Operating Experience
OEM	Original Equipment Manufacturer
RAI	Request for Additional Information
RFO	Refueling Outage
RVI	Reactor Vessel Internals
SG	Steam Generator

By letter dated October 1, 2012 (ADAMS Accession No. ML12284A320), I&M, the licensee for CNP, submitted an AMP for CNP, Units 1 and 2, RVI to the NRC. By letter dated June 6, 2014 (ADAMS Accession No. ML14135A320), the NRC staff reviewed the submittal and requested additional information to complete its review. The responses to RAI-1, RAI-5, and RAI-7 are provided in this enclosure. The remaining RAIs (RAI-2, RAI-3, RAI-4, RAI-6, and RAI-8) will be provided at a later date per the schedule described in the June 6, 2014 letter from the NRC.

RAI-1

In Sections 2.1 and 2.2 of the submittal, I&M states "Some examples of the results ..." in reference to reviewing operating experience (OE) and records search of plant-specific RVI AMP relevant information. Confirm that these examples include any and all relevant OE and records that will impact the CNP RVI AMPs.

Response to RAI-1

The following lists for Unit 1 and Unit 2 include any and all relevant OE and records that will impact the CNP RVI AMP. Each list is presented in the following pages as a proposed content change for the current Sections 2.1 and 2.2 of the CNP RVI AMP. Deletions are presented as ~~strikethrough~~ while additions are underlined in this response for convenience of comparison.

2.1 Unit 1 Operating Experience and Records Search

A review of plant records was performed to locate unit specific RVI OE and design changes. ~~Some examples of the results for Unit 1 are discussed in the following sections.~~ A list of relevant OE and records for the Unit 1 RVIs are discussed in chronological order in the following sections.

2.1.1 Control Rod Guide Tube and Split Pin Replacement

The original split pins were fabricated from alloy X-750. These pins were replaced with an improved stress design fabricated from alloy X-750 in 1985 (U1C9). Installation was performed through replacement of control rod guide tube (CRGT) assemblies using spares available from a CNP Unit 2 modification discussed in Section 2.2.1.

2.1.2 T_{AVG} Reduction Program

The reactor coolant system operating pressure and temperature were reduced in support of aging management for the original steam generators. This change was implemented in 1989 (U1C11).

2.1.3 Barrel-Former Bolt Inspection and Partial Replacement

A barrel-former bolt was discovered on the lower core plate after defueling in 1994 (U1C14). Inspection was performed in 1995 (U1C15) to determine the origin of the retrieved bolt. All barrel-former bolts were visually inspected and a sample of bolts was mechanically agitated to determine if additional bolts were loose. The two horizontally adjacent bolts to the vacant location were loose. A total of three bolts were replaced with oversized bolts. Replacement efforts required three holes to be machined into the core barrel for tool access. This work was completed in 1997 (U1C16).

2.1.4 Reactor Vessel Closure Head Replacement

The original reactor vessel closure head was replaced for asset management of the component. No changes were made to the RVIs. The replacement was completed in 2006 (U1C21).

2.1.5 Clevis Insert Bolt ~~Degradation~~ Replacement

~~Indications were discovered in the clevis bolts of the lower radial support system while performing the ASME 10-year ISI in 2010 (U1C23). The plant is currently operating based on a JCO provided by the OEM. Development of a repair methodology and associated tooling is under development.~~ Indications were observed in some of the Lower Radial Support System clevis insert bolts and one dowel pin during the ASME 10-year In-Service Inspection in 2010 (U1C23). A minimum bolt pattern was installed on each of the six clevis inserts in 2013 (U1C25).

2.1.6 Return to Normal Operating Pressure and Temperature

The reactor coolant system operating pressure and temperature will be returned to previously analyzed values which the plant operated at prior to the T_{AVG} Reduction Program described in Section 2.1.2. The change is driven by replacement steam generator aging management strategy. This change will be implemented in 2014 (U1C26).

2.2 Unit 2 Operating Experience and Records Search

A review of plant records was performed to locate unit specific RVI OE and design changes. ~~Some examples of the results for Unit 2 are listed in the following sections.~~ A list of relevant OE and records for the Unit 1 RVIs are discussed in chronological order in the following sections.

2.2.1 Modification from 15X15 to 17X17 Design

Modifications were made on Unit 2 to accept 17X17 fuel assemblies. Appropriate changes were made to the RVIs at the manufacturer's shop prior to operation. Spare parts generated from the modification, including the 15X15 CRGT assemblies, were retained by I&M. These spare CRGT assemblies were later installed in Unit 1 as described in Section 2.1.1.

2.2.2 Control Rod Guide Tube Split Pin Replacement

The original Unit 2 split pins were fabricated from alloy X-750. A small number of these pins failed and were retrieved from two steam generators in 1985. A JCO was provided by the OEM to operate for the remainder of the fuel cycle with this known degraded condition. The original pins were replaced during the following RFO with an improved stress design fabricated from alloy X-750. This work was completed in 1986 (U2C6).

2.2.3 Control Rod Guide Tube Cap Screw Modification

Each CRGT in Unit 2 has four hold down socket head cap screws fastening it to the support plate. A number of bolts and threaded holes were damaged during the Unit 2 CRGT split pin replacement campaign. Two CRGT hold down socket head cap screws were broken during untorquing, leaving the threaded portion of the cap screws in the tapped support plate holes. Also, the threads of two tapped holes were damaged during the split pin replacement effort. Each damaged location was on a different CRGT assembly. These bolt locations were abandoned as supported by analysis from the OEM. Three high strength bolts were installed at the remaining available locations on these four CRGT assemblies. This work was completed in 1986 (U2C6).

2.2.4 Reactor Vessel Closure Head Replacement

The original reactor vessel closure head was replaced for asset management of the component. Flow restrictors were installed on part-length control rod guide tube assemblies to accommodate the new head design. The replacement was completed in 2007 (U2C17).

2.2.5 Baffle-Former Bolt Partial Replacement

CNP Unit 2 original baffle-former bolts are internal hex with a cross tack welded lock bar. A number of baffle-former bolts were discovered on the lower core plate in 2010 (U2C19). Visual inspection revealed 18 failed bolts ranging from broken or missing lock bars to broken or missing bolt heads in a local area on the large south baffle plate. Bolts with visual indications were replaced. Replacement was expanded to bolts in adjacent rows and columns in the plate to bound the edge of the local degradation. Bolt samples were removed from the other three large baffle plates and inspected to ensure degradation was not occurring at symmetric locations. A total of 52 bolts were replaced with two locations left vacant.

RAI-5

In Section 4.4.2.3 of the submittal, I&M states that information will be provided to the NRC concerning the strategy for managing split pins prior to the period of extended operation for each unit. The NRC staff noted that the licensee intends to investigate whether replacement of the split pins is necessary. The staff requests that the licensee:

(a) *Provide a brief summary of the previous inspections thus far that were performed on the split pins including the type of inspections, frequency of inspections, and the results of the inspection. Confirm whether the split pins are binned under the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI examination category.*

(b) *If cracking was observed in the split pins, describe the corrective action that was taken to prevent recurrence of the aging degradation. If replacement of the split pins is necessary, the NRC staff requests that the licensee state the type of material that will be used for replacement.*

Response to RAI-5

Clarification of split pin aging management for each CNP unit is described below.

(a) The split pins (support pins) are not binned under the ASME Code, Section XI examination category. These components are part of the upper internals and therefore, they are not included in the B-N-1, B-N-2, or B-N-3 examination items. No specific inspections are performed on these components. However, the split pins will be replaced as addressed in the following section.

(b) The original CNP Unit 1 split pins were fabricated from alloy X-750. All pins were replaced in 1985 (U1C9) with an improved stress design fabricated from alloy X-750 with a modified heat treatment. Replacement was performed for asset management due to industry split pin OE at that time.

Two original split pins fabricated from alloy X-750 failed in CNP Unit 2 during operation in 1985. These broken pins were discovered following a metal impact monitoring system alarm in one of the SGs. Loose parts generated from broken split pins were removed from the SGs and the plant continued operation with a small number of broken pins until the next refueling outage. All split pins in CNP Unit 2 were replaced with an improved stress design fabricated from alloy X-750 with a modified heat treatment in the first refueling outage following split pin failure. This work was completed in 1986 (U2C6).

I&M has considered aging management strategies for the X-750 split pins at CNP Units 1 and 2. An evaluation of the probability of split pin cracking was performed for each unit as input to the aging management strategy. Operating history was considered for each unit specific evaluation. For example, for CNP Unit 1, the T_{avg} Reduction Program was considered, Return to Normal Operating Pressure and Temperature were considered, and specific temperature and operation records were used for the evaluation. Similarly for CNP Unit 2, specific temperature and operation records were used for the evaluation.

The requirement for split pin functionality is that the pin maintains engagement with the CRGT bottom flange and the upper core plate. This functionality is maintained even if a split pin shank and/or leaf failure occurs due to the remaining pin remnant. Therefore, split pin cracking and/or fracture does not create a condition adverse to safe operation. However, split pin separation does cause a commercial concern. Loose parts generated from split pin separation are transported to the SG channel head(s) by the reactor coolant system. These loose parts can cause surface impact damage to the SG tube sheet(s).

Replacement of CNP Unit 1 and Unit 2 split pins has been determined necessary for commercial reasons. The risk, schedule, and economics of replacement were evaluated. Split pins will be replaced in CNP Unit 1 and Unit 2 with no required inspection prior to replacement.

CNP Unit 1 split pins will be replaced during the refueling outage currently scheduled for the fall of 2017. The calculated probability of at least one pin with a crack, assuming nominal installation torque, will be 3.9% at the scheduled refueling outage of replacement.

CNP Unit 2 split pins will be replaced during the refueling outage currently scheduled for the fall of 2016. The calculated probability of at least one pin with a crack, assuming nominal installation torque, will be 70.5% at the scheduled refueling outage of replacement.

A letter report describing the split pin crack probability evaluation, Westinghouse document LTR-RIDA-14-66, Revision 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2," is included as Enclosures 3 and 4 of this letter with non-proprietary and proprietary versions, respectively.

I&M is in the process of competitively bidding split pin replacement in CNP Units 1 and 2. The material type will be selected by I&M once a vendor is chosen. I&M will provide, to the NRC, the type of replacement split pin material prior to split pin replacement in each unit.

RAI-7

Sections 2.1 and 2.2 of I&M's submittal describe a number of baffle-former and barrel-former bolt failures that have occurred at CNP, Units 1 and 2.

(a) Please discuss the root cause of the failed bolts.

(b) Given the relative frequency of these bolt failures at CNP, Units 1 and 2, above and beyond general industry experience, justify the use of the generic MRP-227-A guidelines for bolt inspection, as opposed to augmenting the CNP AMP to require more comprehensive bolt inspections.

Response to RAI-7

Bolt OE is discussed below in separate sections for clarity. Section 1 addresses the CNP Unit 1 barrel-former bolt response and Section 2 discusses the CNP Unit 2 baffle-former bolt response.

1. CNP Unit 1 Barrel-Former Bolt OE

The following sections addresses RAI-7 related to the CNP Unit 1 barrel-former bolts.

1.(a). CNP Unit 1 Barrel-Former Bolt Response to RAI-7(a)

A CNP Unit 1 barrel-former bolt was discovered on the lower core plate after defueling in 1994 (U1C14). Inspection was performed in 1995 (U1C15) to determine the original location of the retrieved bolt. All barrel-former bolts were visually inspected and a sample of bolts was mechanically agitated to determine if additional bolts were loose. The two horizontally adjacent bolts to the vacant location were loose. A total of three bolts were replaced with oversized bolts. Replacement efforts required three holes to be machined into the core barrel for tool access. This work was completed in 1997 (U1C16).

A root cause evaluation was completed following failure of the CNP Unit 1 barrel-former bolts. The following specific analyses were performed: a metallurgical evaluation of the failed bolts, an as-built flow induced vibration model of the core barrel and thermal shield in the region of the failed bolts, and a steady state thermal analysis. No single root cause was identified. The contributing causes were elevated bolt stress near the thermal shield support block, and bending stress on bolts during normal steady state operation. Stress corrosion cracking was not a factor in the mode of failure.

1.(b). CNP Unit 1 Barrel-Former Bolt Response to RAI-7(b)

The CNP Unit 1 barrel-former bolts were returned to a fully qualified condition by replacing all bolts with indications of looseness or failure. I&M and the OEM concluded that it was appropriate to return the system to its former monitoring requirements following replacement. Analytical and inspection efforts were undertaken during the investigation of the barrel-former bolt failures. Inspection at symmetrical locations in the lower internals assembly did not reveal bolt failure, which indicates an isolated, not a generic issue. I&M has not observed any abnormal conditions or symptoms related to the CNP Unit 1 barrel-former bolts following the permanent repair completed in 1997 (U1C16). The barrel-former bolts are an expansion inspection component as described in the RVI AMP, Appendix B: Expansion Inspection Components. Therefore, I&M has concluded that the barrel-former bolts are adequately managed by existing monitoring and aging management programs already in place. No augmentation is necessary for these components.

2. CNP Unit 2 Baffle-Former Bolt OE

The following sections addresses RAI-7 related to the CNP Unit 2 baffle-former bolts.

2.(a). CNP Unit 2 Baffle-Former Bolt Response to RAI-7(a)

A number of CNP Unit 2 baffle-former bolts were discovered on the lower core plate in 2010 (U2C19). Visual inspection revealed 18 failed bolts ranging from broken or missing lock bars to broken or missing bolt heads in a local area on the large south baffle plate. Bolts with visual indications were replaced. Replacement was expanded to bolts in adjacent rows and columns in the plate to bound the edge of the local degradation. Bolt samples were removed from the other three large baffle plates and inspected to ensure degradation was not occurring at symmetric locations. A total of 52 bolts were replaced with two locations left vacant.

A root cause evaluation was completed following failure of the CNP Unit 2 baffle-former bolts. The root cause of the failed baffle-former bolts was Irradiation Assisted Stress Corrosion Cracking in conjunction with loss of preload in several bolts. Contributing causes included thermal and irradiation embrittlement leading to degraded bolting material, high cycle fatigue/un-zippering/overload, and steady-state pressure gradient across the baffle plates.

2.(b). CNP Unit 2 Baffle-Former Bolt Response to RAI-7(b)

The CNP Unit 2 baffle-former bolts were returned to a fully qualified condition by replacing damaged bolts. A sampling approach was used to bound the degradation area within the large south baffle plate. Sampling was further used to confirm bolt integrity on the remaining three large baffle plates. The investigation at symmetrical locations in the lower internals assembly did not reveal bolt failure, which indicates an isolated, not a generic issue. This approach gave I&M and the OEM confidence to return the unit to operation without additional monitoring requirements. However, a voluntary visual inspection of the CNP Unit 2 baffle-former bolts was performed during the following refueling outage in 2012 with no relevant indications. I&M has not observed any abnormal conditions or symptoms related to the CNP Unit 2 baffle-former bolts following the permanent repair completed in 2010. The baffle-former bolts are a primary inspection component as described in the RVI AMP, Appendix A: Primary Inspection Components. Therefore, I&M has concluded that the barrel-former bolts are adequately managed by existing monitoring and aging management programs already in place. No augmentation is necessary for these components.

Enclosure 2 to AEP-NRC-2014-56

Donald C. Cook Nuclear Plant
Regulatory Commitments

The following table identifies the action committed to by Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP). Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Completion Date
CNP Unit 1 split pins will be replaced during the U1C28 refueling outage currently scheduled for the fall of 2017.	U1C28
CNP Unit 2 split pins will be replaced during the U2C23 refueling outage currently scheduled for the fall of 2016.	U2C23
I&M will provide, to the NRC, the type of replacement split pin material prior to split pin replacement in each unit.	U1C28 U2C23

ENCLOSURE 3 TO AEP-NRC-2014-56

Westinghouse Letter, LTR-RIDA-14-66-NP, Rev. 0, “Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2” [Non-Proprietary]

Westinghouse

To: Thomas M. Haley

cc: Charles A. Jennings, Rick J. Gradomski, Cheryl L. Boggess, Vytas J. Mickus

Date: July 10, 2014

From: Reactor Internals Design and Analysis I

Ext: 412-374-6058

Fax: 724-940-8559

Your ref: N/A

Our ref: LTR-RIDA-14-66-NP, Rev. 0

Subject: **Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2**

References: [1.

2.

3.] a, b, c

4. ASME Boiler & Pressure Vessel Code, Code Cases: Nuclear Components, Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," March 28, 2006.

[5.] a, b, c

6. AEP Design Information Transmittal, DIT-B-03551-00, "Split Pin Remaining Life Evaluation Information," August 30, 2013.
7. AEP Design Information Transmittal, DIT-S-00705-17, "Unit 1 and Unit 2 Burnup Data," October 2, 2013.

[8.] a, b, c

The purpose of this letter is to provide a response to American Electric Power (AEP) to support a commitment to the U. S. Nuclear Regulatory Commission (NRC) to provide a strategy for managing the D.C. Cook Units 1 and 2 reactor upper internals guide tube support pins as provided in the reactor vessel internals aging management plan. This response is to provide an evaluation for the replacement timing in terms of the predicted operating life of the current support pins against primary water stress corrosion cracking (PWSCC) susceptibility.

This evaluation provides the expected probabilities of stress corrosion cracking of the support pins for continued operation of Unit 1 to the end of cycle 26 in 2016 and beyond, and of Unit 2 to the end of cycle 22 in 2016 and beyond. Analyses [1] and [2] were performed to utilize support pin operational times until cracking was detected from 14 reference plants to develop crack susceptibility expectations for the D.C. Cook Unit 1 and 2 support pins at the plant's operating temperatures. Of the 14 reference plants analyzed, 12 had cracking and separation at the pin shanks (either at the bottom of the shank or near the bottom of the threads) and 2 had pins with cracked and broken leaves. Therefore, both locations were analyzed separately for crack susceptibility times.

1.0 PWSCC Susceptibility Derivation

The cracking susceptibility (Arrhenius) equation [3] below was used to relate the operating temperature and stress to the known times of operation until pin separation occurred due to cracking.

$$t \propto \exp\left[\frac{Q}{RT}\right]\sigma^{-a}$$

where,

t = operating time to crack initiation

[ ] b, c

R = universal gas constant = 0.001986 kcal/(gmole °K)

T = absolute temperature in °K

σ = stress due to preload plus temperature (for the shank region)

[ ] b, c

For the ranges for Q and a, [ ] b, c were found to be most conservative to maximize the probabilities of cracking for the D.C. Cook Unit 1 and 2 conditions. The exponent of [ ] b,c is more typically used and it gives more conservative results by minimizing the operating time if higher stress is applied.

For the case where a plant operates at different temperatures over time, a related crack initiation susceptibility equation from [4] is used to determine how to develop an equivalency constant, A, that can relate the operating times from the reference plants to the projected support pin operating time for the D.C. Cook units. The susceptibility equation from [4] is:

$$t_r = \sum\left\{t_i \exp\left[\frac{Q}{R}\left(\frac{1}{T_r} - \frac{1}{T_i}\right)\right]\right\} \qquad (1)$$

where,

t_i = time intervals at temperature T_i

T_r = a reference temperature that t_r is to be based on

From Equation (1) it can be determined that an equivalency constant, A, can be derived that relates the total operating times of reference plants where the total times of operation until cracking occurs are known to the allowable operating time of another plant. Although the stress term is not included in Equation (1), it is now included as used in the form of the Arrhenius equation above. The resulting equation for A, derived from Equation (1), is:

$$A = \sum \left\{ \frac{t_i}{\exp\left[\frac{Q}{RT_i}\right] \sigma_i^{-a}} \right\} \qquad (2)$$

For each of the 12 reference plants used in the analysis for the shank cracking susceptibility and the two plants for the leaf cracking susceptibility, the value of A is determined using the incremental times for which each plant operated at different temperatures. Ideally, for the case where the material and design of the component is the same between plants, the value for A should be the same if the operating temperatures and stresses are exactly known. However, using the operating times for these plants when cracking was detected resulted, in reality, in a distribution of the values for A. The resulting normal distribution for the 12 plants is reasonable, having a standard deviation of approximately [ ]^{a,c} of the average. Since data from only two plants is available for the leaf cracking, no statistical distribution was used for the leaf susceptibility.

Since D.C. Cook Units 1 and 2 have also operated at different temperatures for different time intervals, Equation (2) is again used to determine their values of A projected to the end of each cycle over the time range desired. From these values of A, the probability of shank cracking is determined at the end of each cycle relative to the A distribution derived from the reference plant data.

Conversely, for a selected value of A, such as at plus or minus two standard deviations from the average, or for values of A for the leaf cracking data, the future expected operational time during the last time/temperature interval for D.C. Cook Unit 1 or 2 can be derived by rearranging Equation (2):

$$t_f = \exp\left[\frac{Q}{RT_f}\right] \sigma_f^{-a} \left(A - \sum \left\{ \frac{t_i}{\exp\left[\frac{Q}{RT_i}\right] \sigma_i^{-a}} \right\} \right) \qquad (3)$$

where,

t_i = previous time intervals at temperature T_i and stress σ_i

t_f = final time interval at temperature T_f and stress σ_f

Since no A distribution is available for the pin leaves, the operating times are explicitly derived from Equation (3) by solving for the final time/temperature interval that the pins are allowed to operate using the two values of A derived from the reference plant data. The average of these times is compared to the average of the pin shank operating times to determine which area is more limiting.

2.0 Core Exit Temperatures

D.C. Cook Unit 1

The Unit 1 core exit temperature (based on T_{hot} cycle data from [6]) was derived from the cycle-specific vessel outlet temperatures from beginning of cycle 9, when the pins were installed in 1985, until possible removal in 2016 (end of cycle 26) and beyond. The T_{hot} temperatures were set to the averages of five time intervals during which the temperatures varied very little from cycle to cycle during each interval. Weighted averages were used based on the effective full power hours (EFPH) of each cycle, [6] and [7], multiplied by the temperatures. To convert to core exit temperature, an additional [ ] b, c, based on Westinghouse-derived core exit and vessel outlet temperatures for D.C. Cook Unit 1, is added. The resulting core exit temperatures are listed in Table 1.

b, c

Table 1
D.C. Cook Unit 1 Core Exit Temperature Time Intervals

Cycles	t_i (EFPH)	Core Exit Temperature, T_i (°F)

For cycle 26 and onward, D.C. Cook Unit 1 is to implement the Return to Reactor Coolant System Normal Operating Pressure (NOP)/Normal Operating Temperature (NOT) Program, which increases the core exit temperature to the value given in Table 1.

D.C. Cook Unit 2

Using the same procedure as for Unit 1, the Unit 2 core exit temperatures from [6] and cycle lengths from [6] and [7] are used, starting from beginning of cycle 6 when the pins were installed in 1986 until possible removal in 2016 (end of cycle 22) and beyond. The T_{hot} temperatures were set to the averages of five time intervals during which the temperatures varied very little from cycle to cycle during each interval. Weighted averages were used based on the EFPH of each cycle ([6], [7]) multiplied by the temperatures. To convert to core exit temperature, an additional [ ] b, c, based on Westinghouse-derived core exit and vessel outlet temperatures for D.C. Cook Unit 2, are added. The resulting core exit temperatures are listed in Table 2.

b, c

Table 2
D.C. Cook Unit 2 Core Exit Temperature Time Intervals

Cycles	t_i (EFPH)	Core Exit Temperature, T_i (°F)

3.0 Pin Shank

The shank stress, σ_t, is derived as a function of temperature. This is added to the initial preload stress. The stress as a function of temperature is based on the relative thermal expansion between the Inconel and Type 304 stainless steel materials and the stiffnesses of the pin shank and surrounding guide tube flange [5].

$$\sigma_t = \frac{\delta E}{\ell_b} = \frac{\ell_1}{\ell_b}(\alpha_1 - \alpha_2)\Delta T\left(\frac{\overline{K}}{\overline{K} + K_b}\right)E$$

where,

ℓ_1 = bottom flange thickness between pin shoulder and nut
ℓ_b = pin shank length from shoulder to thread engagement
α_1 and α_2 = coefficients of thermal expansion for Type 304 and X-750, respectively
ΔT = temperature change from room temperature to core exit normal operating temperature
$\overline{K}$ = combined axial stiffness of bottom flange and nut
K_b = pin shank axial stiffness
E = X-750 elastic modulus

The installation preload stress is derived using the minimum torque coefficient, because this maximizes the preload force that minimizes the effect of the added temperature stress when ratioed, since the D.C. Cook Units 1 and 2 operating core exit temperatures are typically less than for the reference plants. In addition, the nominal torque of [ ]^{a,c} is used for the reference plants, whereas both the nominal and maximum torques of [ ]^{a,c} are used for the D.C. Cook units to compare the effect of the nominal to maximum torque and the preload stress relative to the reference plants.

The effect of wet versus dry installation torque is also considered. Some of the reference plants had underwater torque installations and some had in-air torque installation of the support pins, all with Neolube applied to the threads and nut seating surface. D.C. Cook Unit 1 had the support pins installed in-air, whereas the Unit 2 pins were installed underwater. Different references have shown different relative effects on preloads where, on average, wet torque may produce slightly greater preload than dry and in other cases less preload. However, in all cases the effect is minor and a considerable amount of overlap in preload range can exist. The reference plant data showed no systematic effect of cracking for in-air versus underwater installation. Since the effect is small and considerable overlap of the preload range exists, it is reasonable not to correct for this effect. The use of the [ ]^{a,c} discussed above provides consideration of this effect if in-air or underwater installation should produce relatively greater preload.

<u>D.C. Cook Unit 1</u>

For the D.C. Cook Unit 1 pins, Equation (3) is used with different values for A from the resulting normal distribution to derive the operating times for the period, t_f, when the plant operates at the increased temperature of 613.8°F at the beginning of cycle 26. From [6], cycle 26 is 0.95(516 days)(24 hr/day) = 11,765 EFPH. The D.C. Cook Unit 1 support pin operating time from installation at the beginning of cycle 9 to the beginning of cycle 26 is 172,733 EFPH ([6],[7]) giving a total operating time of 184,500 EFPH from the beginning of cycle 9 to the end of cycle 26. Solving for the t_f EFPH from Equation (3) using values for A at ± 2 standard deviations, σ, about the average and T_f = 613.8°F, the projected operation times after the beginning of cycle 26 are given below.

	EFPH at 613.8°F	
	Nominal Torque	Maximum Torque
-2σ minimum hours to cracking	18,046	7,445
$+2\sigma$ maximum hours to cracking	108,267	87,177

Since the -2σ time is a little less than the cycle 26 EFPH, there is a small probability of cracking. At 11,765 EFPH at the end of cycle 26, the probability of a pin to crack is 1.1 to 3.8 percent. For end of cycle 25 and for additional cycles of operation, Equation (2) is used to derive the probabilities that at least one pin will have cracked shanks and are listed below.

End of Cycle	Outage Time	Probability of at Least One Pin with Cracked Shank, %	
		Nominal Torque	Maximum Torque
25	Fall 2014	0.3	0.9
26	Spring 2016	1.1	3.8
27	Fall 2017	3.9	11.5
28	Spring 2019	10.8	27.1
29	Fall 2020	23.9	49.6

Although not derived, the probabilities of having multiple pins with cracked shanks will be less.

D.C. Cook Unit 2

Since the Unit 2 support pins were manufactured by a non-Westinghouse supplier, a comparison was performed of the critical characteristics relative to pin shank stress to Westinghouse pins as used in the reference plants. This included comparison of the nut to ensure preload for a given torque is the same. All critical characteristics studied have either none or very minor differences and resulted in no necessary adjustments to correct for the support pin shank stress. These characteristics are all dimensional in nature. Discussion on the evaluation of the comparison of the materials is provided in Section 5.0.

As seen in Tables 1 and 2, Unit 2 has operated, and will be operating, at more consistent core exit temperatures than for Unit 1. The temperature variations shown in Table 2 are relatively small as compared to the change in temperature at the beginning of cycle 26 in Table 1 for Unit 1. Therefore, for

Unit 2, derivation of EFPH at the latest temperature, as performed for Unit 1, is unnecessary. The probabilities of shank cracking derived for the end of cycle 22 through 24 are listed below. The total support pin operating time from installation at the beginning of cycle 6 to the end of cycle 22 is 187,950 EFPH. Although the total operating time is only slightly greater than for Unit 1, since Unit 2 has operated at higher temperatures, the probability for cracking is significantly greater.

End of Cycle	Outage Time	Probability of at Least One Pin with Cracked Shank, %	
		Nominal Torque	Maximum Torque
22	Fall 2016	70.5	93.8
23	Spring 2018	84.6	98.1
24	Fall 2019	93.6	99.6

4.0 Pin Leaves

D.C. Cook Unit 1

As previously mentioned, two reference plants had only fractures of the pin leaves. The predominant bending stress in the pin leaves is due to the interference fit of the cantilevered leaves when inserted into the upper core plate hole. This "pinching" interference range is based on the tolerance stack-up of the width across the contact knobs at the bottom of the leaves and the core plate hole. Based on the reference pin design and that used for D.C. Cook Unit 1, the interference range is shifted 0.002 inch less for the D.C. Cook Unit 1 pins. So, as shown in [1], the interference range for the reference plants bounds that for D.C. Cook Unit 1, but since there is also a large overlap in the interference tolerance range, no adjustment to reduce this stress was included. Therefore, the reference plant operating times are adjusted using only the effect of temperature. The resulting operating times for D.C. Cook Unit 1 after the beginning of cycle 26 when the plant operates at 613.8°F are [1]:

Reference Plant	No. of Leaf Fractures	EFPH at 613.8°F Adjusted to D.C. Cook Unit 1	Total EFPH
A	5	73,101	245,834
B	2	85,433	258,166

The total operating times are well above the operating time even to end of cycle 29 (219,900 EFPH). The weighted average additional operating time (∑(no. broken pins per plant x EFPH)/(total pins)) is 76,624 EFPH after the beginning of cycle 26 that the pin leaves will operate before cracking occurs. Due to the small number of fractures, no standard deviation was derived. From Section 3.0, the pin shank average operating time after the beginning of cycle 26 is 47,311 to 63,157 EFPH depending on the preload. Therefore, the pin shank operating times are limiting.

D.C. Cook Unit 2

As performed for the pin shanks, a comparison was made of the critical characteristics relative to pin leaf stress to Westinghouse pins as used in the reference plants. The pin leaf pinching interference range, based on the Unit 2 pins and upper core plate hole, is identical to the reference plants interference range. All geometric critical characteristics studied have either none or very minor differences relative to the leaf stiffness and stress and resulted in no necessary adjustments to correct for the support pin leaf stress. One primary difference, however, exists between the reference plants with leaf fracturing and Unit 2. This difference is in the manufacturing process for the method of deforming the leaves to provide the interference fit with the core plate hole. The manufacturing process for some of the reference plant pins included deformation of the pin leaves to provide the interference fit with the core plate hole. This results in undesirable residual tensile stresses. The manufacturing process used for the Unit 2 pins, and for the remainder of the reference plants, did not include deformation of the pin leaves. These reference plants had no observed leaf fracturing. Since the Unit 2 pins do not have the undesirable residual tensile stress, they are bounded by the reference plant data where leaf fracturing occurred. Since there is no large change in temperature for Unit 2 operation, only the total operating times, as derived for the pin shanks, are derived for Unit 2.

Reference Plant	No. of Leaf Fractures	Total EFPH
A	5	192,760
B	2	206,428

This gives a weighted average operating time of 196,665 EFPH. The average time and the individual times are all greater than the 187,950 EFPH to the end of cycle 22; however, the average is slightly less than the 199,282 EFPH to the end of cycle 23. Although the data show, on average, that leaf cracking has less than a 50 percent chance of occurring at the end of cycle 22, there is still likely some probability that leaf cracking could occur based on the standard deviation of the shank data. However, based on the greater probabilities of the pin shank cracking at the end of cycle 22, the pin shanks are limiting.

5.0 Comparison of Support Pin Material to the Reference Plant Specification

D.C. Cook Unit 1

The support pin material was procured to Westinghouse material specification, [ ]^{a,c}, the same specification as for the reference plant pins summarized in [1]. Therefore, the reference plant support pin data are directly applicable to the D.C. Cook Unit 1 support pins.

Of the plants that had pin cracking, only one plant used pins of the same material heat as for the pins used at D.C. Cook Unit 1. These plant data, normalized to D.C. Cook Unit 1 operating conditions, gave a total operating time of 222,144 EFPH, well over the 184,500 EFPH anticipated for D.C. Cook Unit 1 at the end of cycle 26.

D.C. Cook Unit 2

Since the Unit 2 material was supplied by a non-Westinghouse supplier, a study was performed to determine if the Westinghouse support pin material used for the reference plants may be applicable to the D.C. Cook Unit 2 support pins for PWSCC operational calculations [8]. A comparison of material properties, chemistry, and processing was conducted for the Westinghouse material and the Unit 2 procured material.

The reference plant support pin heats of material that were procured as [ ][a,c] material are very similar in chemistry and physical characteristics to the D.C. Cook Unit 2 material. Both are of the basic high-temperature hydrogenation (HTH) heat treatment. The Unit 2 material was procured to MIL-N-2411B. The Westinghouse specification used for the reference plant pins included additional elements for chemistry testing and rising load tests, and was documented as a revision that made the specification more compatible with the existing revision of MIL-N-2411B.

In conclusion, this study determined that the material data presented for the Westinghouse heats of material used for the reference plants may be applied to the heats of material of Alloy X-750 used for the D.C. Cook Unit 2 material for the purposes of studying the susceptibility to PWSCC.

6.0 Summary Conclusions

Based on the preceding evaluations, inspection of the X-750 support pins is not required for Unit 1 prior to replacement. The probability of pin separation during operation until the end of cycle 26 is expected to be very low. As listed above for additional cycles, the risk of pins with cracked shanks does increase. However, if pin separation does occur, safe operation is maintained in the short term as discussed in the final paragraph.

For Unit 2, the probability of pin shank separation is considerably greater by the end of cycle 22. Again, although there is a possibility that pin separation can occur, safe operation of the reactor is maintained as discussed in the following paragraph.

The design requirement for support pin functionality is that the support pins shall maintain engagement with the guide tube bottom flange and upper core plate, even if separations of the pin shank and leaves occur. The support pins and guide tube bottom flange to upper core plate interface are designed such that if a pin shank separates, the pin shoulder remains engaged in the bottom flange counterbore when the pin drops down as far as it can go against the upper core plate. This is true for both the Westinghouse and non-Westinghouse pin designs. Therefore, lateral support of the bottom flange of the guide tube is maintained against normal, upset, and faulted condition loads and maintains acceptable guidance of the control rods into the fuel assemblies. Separation of pin leaves is also allowable relative to maintaining engagement of the pin in the upper core plate hole. The pins are designed such that the solid section of the pin, which is located above the forked length of the pin leaves where the leaves can crack and separate, extends approximately [ ][b,c] into the core plate hole. Therefore, if one or both of the leaves should separate, the solid pin section above the leaves will maintain engagement in the core plate hole. As a result of the above considerations, if there is separation at the pin shank and/or leaves, a condition adverse to safe operation does not occur. The recommendation is that if a pin shank separation does occur, loose parts monitoring indications should be heeded and plant shutdown should be performed as soon as possible to mitigate damage to the steam generator tube sheet from the separated nut/shank assembly. These are commercial concerns and do not result in an unsafe operating condition. Plants should not operate for long periods of time with broken pins because of possible wear degradation on the guide tube bottom flange.

If there are any questions or clarification needed on the information, please contact the undersigned.

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ENCLOSURE 5 TO AEP-NRC-2014-56

Westinghouse Application for Withholding Proprietary Information from Public Disclosure, LTR-RIDA-14-66-P, Rev. 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" [Proprietary]

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CAW-14-3992

July 15, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-RIDA-14-66-P, Revision 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-14-3992 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.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference CAW-14-3992, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, 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:

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 15th day of July 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 Manager, Regulatory Compliance, 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 LTR-RIDA-14-66-P, Revision 0, "Guide Tube X-750 Support Pin Operational Expectancy and Interim Management Strategy for D.C. Cook Units 1 and 2" (Proprietary), for submittal to the Commission, being transmitted by Indiana/Michigan Power 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 U.S. Nuclear Regulatory Commission letter, "Donald C. Cook Nuclear Plant, Units 1 and 2- Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program Submittal (TAC Nos. MF0050, MF0051)," ML14135A320, June 6, 2014, and may be used only for that purpose.

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

(i) Support reactor vessel internals aging management.

(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 supporting reactor internals aging management relative to guide tube support pin operational justification.

(ii) Westinghouse can sell support and defense of similar reactor internals aging management evaluations relative to guide tube support pin operational justification.

(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.