INDIANA MICHIGAN POWER

**Indiana Michigan Power Company**
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January 14, 2009

AEP-NRC-2009-2
10 CFR Part 50.55a

Docket No.: 50-316

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

Subject: Donald C. Cook Nuclear Plant Unit 2
Request for Relief to Extend the Unit 2 Inservice Inspection Interval for the Reactor Vessel Weld Examination and Request for License Amendment for Submittal of ISI Information and Analyses – Response to Request for Additional Information (TAC NO. MD9934)

References:
1. Letter from L. J. Weber, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Plant Unit 2, Request for Relief to Extend the Unit 2 Inservice Inspection Interval for the Reactor Vessel Weld Examination and Request for License Amendment for Submittal of ISI Information and Analysis," AEP-NRC-2008-41, dated October 9, 2008 (ML082980354).
2. Letter from T. A. Beltz, NRC, to M. W. Rencheck, I&M, "Donald C. Cook Nuclear Plant, Unit 2 – Request for Additional Information Regarding Relief Request (ISIR-29) for use of Risk-Informed Extension of the Inservice Inspection Interval for the Reactor Pressure Vessel Weld Examination, (TAC No. MD9934)," dated December 10, 2008 (ML083430018).

In Reference 1, Indiana Michigan Power Company (I&M) submitted a request for relief to extend the Unit 2 Inservice Inspection (ISI) Interval for the Reactor Vessel Weld Examination (ISIR-29) and request for license amendment for submittal of ISI Information and Analysis. Reference 2 transmitted the Nuclear Regulatory Commission's request for additional information (RAI) regarding the request for relief. The attachment to this letter provides I&M's response to the RAI.

There are no new or revised commitments made in this letter. Should you have any questions, please contact John A. Zwolinski, Manager of Regulatory Affairs, at (269) 466-2478.

Sincerely,

Lawrence Jweber

Lawrence J. Weber
Site Vice President

A047
NRR

RP/rdw

Attachment: Request for Relief to Extend the Unit 2 Inservice Inspection Interval for the Reactor Vessel Weld Examination and Request for License Amendment for Submittal of ISI Information and Analysis, Response to Request for Additional Information

c: T. A. Beltz - NRC Washington, DC
J. L. Caldwell - NRC Region III
K. D. Curry - AEP Ft. Wayne
J. T. King - MPSC
MDEQ - WHMD/RPS
NRC Resident Inspector

**AFFIRMATION**

I, Lawrence J. Weber, 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 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

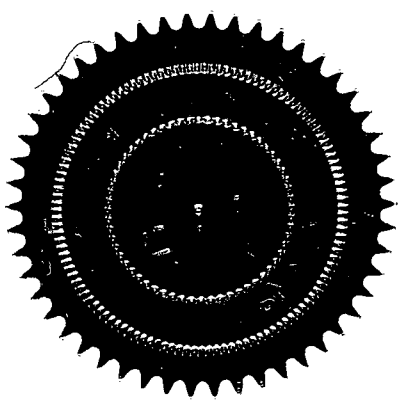
Lawrence J. Weber
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 14th DAY OF January, 2009

Sarah L. Adkins
Notary Public

My Commission Expires 9/9/2011



Request for Relief to Extend the Unit 2 Inservice Inspection Interval for the Reactor Vessel Weld Examination and Request for License Amendment for Submittal of ISI Information and Analyses, Response to Request for Additional Information

In Reference 1, Indiana Michigan Power Company (I&M) submitted a request for relief to extend the Unit 2 Inservice Inspection (ISI) Interval for the Reactor Vessel Weld Examination and request for license amendment for submittal of ISI Information and Analysis. Reference 2 transmitted the Nuclear Regulatory Commission's (NRC) request for additional information (RAI) regarding the request for relief. The requested information is provided below.

**NRC RAI 1**

*In Section 3.4 of the final safety evaluation report issued May 8, 2008 (ADAMS Accession No. ML081060045), the NRC staff notes that licensees submitting a request for an alternative based on Topical Report WCAP-16168-NP, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," must submit the following plant-specific information:*

1. *Licensees must demonstrate that the $RT_{max-x}$ and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level $\Delta T_{30}$ must be calculated using the latest approved methodology documented in Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," or other NRC approved methodology.*

*"Other NRC-approved methodology" includes equations 5, 6, and 7, as described in paragraph (g) of the proposed rule, Title 10 of the Code of Federal Regulations (CFR) 50.61 a, as published in the Federal Register, Vol. 72, No. 191, dated October 3, 2007. However, paragraph (f)(6) of the proposed rule contains a prescriptive approach to determining the validity of implementing equations 5 through 7 of paragraph (g) for the calculation of $\Delta T_{30}$ values. Data from a plant specific or integrated surveillance program shall be evaluated in accordance with 10 CFR 50.61a(f)(6) to validate use of the associated embrittlement model.*

*The licensee has implemented the use of equations prescribed in 10 CFR 50.61a(g), NUREG-1874, and WCAP-16168-NP to determine $\Delta T_{30}$ values for the beltline materials.*

*Please submit one of the following:*

1) *Information consistent with the requirements of proposed rule 10 CFR 50.61a(f)(6) that establishes the applicability of equations 5 through 7 as provided in 10 CFR 50.61a(g) for calculating $\Delta T_{30}$ values for a plant's beltline materials.*

2) *Recalculation of $\Delta T_{30}$ using RG 1.99. This would require reporting of recalculated values of both $\Delta T_{30}$ and all through wall cracking frequency (TWCF) calculations consistent with the format of Table 3 provided in the proposed relief request.*

**I&M Response to NRC RAI 1**

I&M has chosen to respond to option 2 of RAI 1. Delta T30 ($\Delta T_{30}$) has been recalculated based on Regulatory Guide (RG) 1.99, Revision 2. The through-wall cracking frequency (TWCF) was recalculated using the new $\Delta T_{30}$ values. The inputs to these calculations and the results are provided in Table 1 (next page). It should be noted that the chemistry factors in Table 1 are consistent with those used in the calculation of the Donald C. Cook Nuclear Plant (CNP) Unit 2 heatup and cooldown limit curves and $RT_{PTS}$ values. These calculations were provided to the NRC in WCAP-15047, Revision 2, and WCAP-13517, Revision 1, respectively. These two WCAP reports were approved by the NRC via the SER as noted in the response to RAI 3. As shown in Table 1, the TWCF value for CNP Unit 2 based on $\Delta T_{30}$ values calculated using RG 1.99, is 3.80E-10 events per year. This value is less than the Westinghouse pilot plant value of 1.76E-08 events per year in WCAP-16168-NP-A, Revision 2. Therefore, the application of the extended ISI interval to CNP Unit 2 remains acceptable.

| Table 1 Details of Revised TWCF Calculation – D.C. Cook Unit 2 at 60 EFPY | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Inputs | | | | | | | | |
| Reactor Coolant System Temperature, $T_{RCS}$[°F]: | | | 542.55 | | $T_{wall}$ [inches]: | | 8.50 | |
| # | Region/Component Description | Material | Cu [wt%] | Ni [wt%] | C.F. [°F] | R.G. 1.99 pos. | Un-Irradiated $RT_{NDT(u)}$ [°F] | Fluence[1] at 60 EFPY [n/cm$^2$, E > 1.0 MeV] |
| 1 | Int. Shell Axial Weld | Linde 124 | 0.056 | 0.956 | 66.3 | 2.1 | -35 | 1.38E+19 |
| 2 | Int.. Shell Axial Weld | Linde 124 | 0.056 | 0.956 | 66.3 | 2.1 | -35 | 1.38E+19 |
| 3 | Low. Shell Axial Weld | Linde 124 | 0.056 | 0.956 | 66.3 | 2.1 | -35 | 9.58E+18 |
| 4 | Low. Shell Axial Weld | Linde 124 | 0.056 | 0.956 | 66.3 | 2.1 | -35 | 9.58E+18 |
| 5 | Int./Low. Circ Weld | Linde 124 | 0.056 | 0.956 | 66.3 | 2.1 | -35 | 3.08E+19 |
| 6 | Inter. Shell Plate | A 533B | 0.125 | 0.580 | 102.3 | 2.1 | 38 | 3.08E+19 |
| 7 | Inter. Shell Plate | A 533B | 0.150 | 0.570 | 108.4 | 1.1 | 58 | 3.08E+19 |
| 8 | Lower Shell Plate | A 533B | 0.110 | 0.640 | 74.6 | 1.1 | -20 | 3.08E+19 |
| 9 | Lower Shell Plate | A 533B | 0.140 | 0.590 | 99.5 | 1.1 | -20 | 3.08E+19 |
| Outputs | | | | | | | | |
| Methodology Used to Calculate $\Delta T_{30}$: | | | Regulatory Guide 1.99, Revision 2 | | | | | |
| | Controlling Material Region # (From Above) | $RT_{MAX-XX}$ [R] | Fluence[1] [n/cm$^2$, E > 1.0 MeV] | FF (Fluence Factor) | $\Delta T_{30}$ [°F] | $TWCF_{95-XX}$ | | |
| Axial Weld – AW | 7 | 635.7 | 1.38E+19 | 1.09 | 118.1 | 3.49E-11 | | |
| Circumferential Weld - CW | 7 | 658.3 | 3.08E+19 | 1.30 | 140.6 | 3.96E-14 | | |
| Plate – PL | 7 | 658.3 | 3.08E+19 | 1.30 | 140.6 | 1.28E-10 | | |
| $TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW}$): | | | | | | 3.80E-10 | | |

Note 1: Values are for fluence at the cladding-to-base metal interface.

**NRC RAI 2**

*In Table 1 of the submittal, the license states that they are bounded by seven heatup/cooldowns per year.*

*Please cite the plant design basis for heatup/cooldowns per year.*

**I&M Response to NRC RAI 2**

According to the Updated Final Safety Analysis Report (UFSAR) Section 4.1.4. Cyclic Loads:

"A renewed operating license extends the license term an additional 20 years for Cook Nuclear Plant (CNP), Units 1 and 2. This extension was justified based on design transient cyclic loads defined in Table 4.1- 10. The reactor coolant system was originally qualified using a conservative estimate of design cycles for a 40 year life. However, design life is dependent in part on fatigue cycles, not years of service. In evaluations performed for CNP, the actual number of cycles was extrapolated to 60 years. For the major reactor coolant system components, the extrapolated numbers of cycles over a 60-year life will not exceed the design cycles noted in the UFSAR. The actual transient cycles are tracked and documented to ensure they remain below the allowable number of design cycles, as further discussed in Chapter 15 of the UFSAR."

For Unit 2 there have been 61.75 heatup and 60.75 cooldown cycles over the operating life of the plant. The number of design heatup and cooldown cycles is 200 for the life of the plant. Based on this, the remaining number of design cycles for the remaining life of the plant is approximately 140 cycles.

The renewed license permits operating Unit 2 until December 23, 2037 (DPR-74). The number of heatup and cooldown cycles allowed in the remaining 29 years of operation for Unit 2 would be 140/29 = 4.8 cycles per year. Given the previous 31 years of operation for Unit 2, the unit has averaged approximately 2 heatups and cooldowns per year. The analyses noted in the I&M submittal based on seven cycles per year would envelop the plant's remaining design basis number of cycles.

**NRC RAI 3**

*In Table 3 of the submittal, the licensee cites compositions in terms of Cu, Ni, P, and Mn for items 1-9. These compositions are not consistent with NRC records in Reactor Vessel Integrity Database.*

*Please explain where the Cu, Ni, and P compositions were accepted by the NRC or provide new data.*

*Please explain on what basis the reported Mn compositions were determined.*

**I&M Response to NRC RAI 3**

The composition of Copper (Cu), Nickel (Ni), and Phosphorus (P) elements for the reactor vessel materials are taken from the Electric Power Research Institute (EPRI) reactor vessel database called RPVDATA. This database was specifically developed to include the pressurized water reactor (PWR) reactor vessel material chemical and mechanical properties into a single integrated database. The RPVDATA database was developed by EPRI with input on the reactor vessel material information supplied by all three PWR Owners groups, i.e., Westinghouse, Combustion Engineering, and Babcock & Wilcox.

Information from the RPVDATA database has been used in surveillance capsule analysis performed by Westinghouse for CNP Unit 2 in 2002. These analyses have been documented in Westinghouse WCAP reports as noted below:

1. WCAP-13515, Revision 1, "Analysis of Capsule U from the Indiana Michigan Power Company D.C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program," April 2002.
2. WCAP-15047, Revision 2, "D.C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," May 2002.
3. WCAP-13517, Revision 1, "Evaluation of Pressurized Thermal Shock for D.C. Cook Unit 2," May 2002.

The above WCAP reports have been submitted to the NRC as part of the technical specification amendment for the revised heatup-cooldown curves. The NRC approval is documented in their Safety Evaluation Report (SER) for Amendment No. 255 for Unit 2, dated March 20, 2003.

The Manganese (Mn) composition values identified in Table 3 of the submittal, and used in the TWCF calculations, are based on the conservative percent weight estimates in Table 4 of the proposed alternate PTS Rule, 10 CFR 50.61a.

It should be noted that the P and Mn compositions are no longer relevant since TWCF has been recalculated in the response to RAI 1 using the $\Delta T_{30}$ shift correlations of RG 1.99, Revision 2, which do not consider P and Mn composition.

References:

1. Letter from L. J. Weber, I&M, to NRC Document Control Desk, "Donald C. Cook Plant Unit 2, Request for Relief to Extend the Unit 2 Inservice Inspection Interval for the Reactor Vessel Weld Examination and Request for License Amendment for Submittal of ISI Information and Analysis," AEP-NRC-2008-41, dated October 9, 2008 (ML082980354).

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