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INDIANA
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January 21, 2005

AEP:NRC:5034
10 CFR 54

Docket Nos. 50-315
50-316

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

Donald C. Cook Nuclear Plant, Units 1 and 2
RESPONSE TO OPEN AND CONFIRMATORY ITEMS IN THE
DRAFT SAFETY EVALUATION REPORT RELATED TO THE LICENSE
RENEWAL OF DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2
(TAC NOS. MC1202 AND MC1203)

Dear Sir or Madam:

By Reference 1, Indiana Michigan Power Company (I&M) submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant, Units 1 and 2. Based on information provided in the license renewal application (LRA), subsequent responses to U. S. Nuclear Regulatory Commission (NRC) requests for additional information (RAIs) and other questions related to the LRA, the NRC staff developed a draft safety evaluation report (DSER) titled, "Safety Evaluation Report With Open Items related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2." By letter dated December 21, 2004 (Reference 2), the NRC staff issued the DSER to I&M. [Note: References are listed on page 2 of this cover letter.]

The NRC staff identified two open items and two confirmatory items in its review of the LRA and requested a timely submittal of the information required to satisfactorily resolve these items so that it can make a final determination on the application. This letter provides the information required to resolve the DSER open and confirmatory items. In addition, this letter provides I&M's response to an RAI regarding the Boral Surveillance Program, which was issued in an NRC letter dated January 12, 2005 (Reference 3).

The enclosure to this letter provides an affirmation pertaining to the statements made in this letter. The attachment to this letter provides I&M's responses to the

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DSER open and confirmatory items and the Boral Surveillance Program RAI. There are no new commitments made in this submittal.

Should you have any questions, please contact Mr. Richard J. Grumbir, Project Manager, License Renewal, at (269) 697-5141.

Sincerely,

Joseph N. Jensen
Site Vice President

NH/rdw

Enclosure: Affirmation

Attachment: Response to Boral Surveillance Program Request for Additional Information and Open and Confirmatory Items Identified in the "Safety Evaluation Report with Open Items Related to the License Renewal of Donald C. Cook Nuclear Plant, Units 1 and 2"

References:

1. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Application for Renewed Operating Licenses," AEP:NRC:3034, dated October 31, 2003 [Accession No. ML033070177].
2. Letter from P. T. Kuo, NRC, to M. K. Nazar, I&M, "Safety Evaluation Report (SER) with Open Items Related to the License Renewal of Donald C. Cook Nuclear Plant, Units 1 and 2," dated December 21, 2004 [Accession No. ML043570535].
3. Letter from J. Rowley, NRC, to M. K. Nazar, I&M, "Request for Additional Information (RAI) for the Review of the Donald C. Cook Nuclear Plant, Units 1 and 2, License Renewal Application," dated January 12, 2005 [Accession No. ML050120254].

c: J. L. Caldwell – NRC Region III
K. D. Curry – AEP Ft. Wayne, w/o attachment
J. T. King – MPSC, w/o attachment
C. F. Lyon – NRC Washington DC
MDEQ – WHMD/HWRPS, w/o attachment
NRC Resident Inspector

AFFIRMATION

I, Joseph N. Jensen, 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

Joseph N. Jensen
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 21st DAY OF January, 2005

Notary Public

My Commission Expires 6/10/2007

Response to Boral Surveillance Program Request for Additional Information and Open and Confirmatory Items Identified in the "Safety Evaluation Report with Open Items Related to the License Renewal of Donald C. Cook Nuclear Plant, Units 1 and 2"

By Reference 1, Indiana Michigan Power Company (I&M) submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. Based on information provided in the license renewal application (LRA), subsequent responses to U. S. Nuclear Regulatory Commission (NRC) requests for additional information (RAIs) and other questions related to the LRA, the NRC staff developed a draft safety evaluation report (DSER) titled, "Safety Evaluation Report With Open Items related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2." By letter dated December 21, 2004 (Reference 2), the NRC staff issued the DSER to I&M.

This attachment provides I&M's responses to the two open items and two confirmatory items identified in the DSER. In addition, this attachment provides I&M's response to the Boral Surveillance Program RAI that was issued in an NRC letter dated January 12, 2005 (Reference 3).

Draft SER Open Item 3.3.2.1.11-1:

The staff did not find the applicant's initial response to RAI 3.3.2.1.11-1 acceptable. The staff's specific concern, as described in RAI 3.3.2.1.11-1, is the apparent aging management of the internal environments of components/systems by visual inspection of external surfaces if environmental differences exist between internal and external surfaces. While external inspection of component condition (e.g. pipes, valves) can indicate the components' internal condition, this is generally not the case until internal degradation results in loss of component integrity as might be indicated by a system leak. The applicant has not provided sufficient information to demonstrate that aging effects on internal surfaces of various components in miscellaneous systems will be effectively managed by the System Walkdown Program. The staff asked the applicant to provide further justification for the use of the System Walkdown Program to manage aging effects for all components identified in LRA Table 3.3.2-11 with different internal and external environments sufficiently to maintain the intended function of the components and ensure that operation of safety-related equipment will not be jeopardized during the period of license renewal.

I&M Response to Draft SER Open Item 3.3.2.1.11-1:

By letter dated October 18, 2004 (Reference 4), I&M supplemented the initial response to RAI 3.3.2.1.11-1 provided in our June 8, 2004, letter (Reference 5). The supplemental response provided additional information regarding the aging management programs that I&M credits for managing aging effects for the components identified in LRA Table 3.3.2-11 with different internal and external environments. Based on discussions with the NRC staff subsequent to issuance of the DSER, it is I&M's understanding that the supplemental RAI response was reviewed and determined to be acceptable to the NRC staff. Consequently, no additional information is required in response to the issue identified in DSER Open Item 3.3.2.1.11-1.

Draft SER Open Item B.1.12-1 (from SER Section 3.0.3.1):

[Note: The open item pertaining to FAC Program inspection sample expansion is presented in DSER Section 1.5, Summary of Open Items, and Section 3.0.3.1, AMPs That Are Consistent with the GALL Report. A comparison of these two sections identified minor differences in how the issue was presented in the two sections. The NRC staff clarified that DSER Section 3.0.3.1 accurately portrays the staff's concern; therefore, I&M's response reflects the issue presented in DSER Section 3.0.3.1 rather than the summary in Section 1.5. It should be noted that I&M has no record of having received RAI B.1.12-1, as stated in the open item.]

In CNP LRA, Appendix B, Section B.1.12, the applicant states that CNP AMP [Aging Management Program] B.1.12, "Flow-Accelerated Corrosion Program," is consistent with GALL [Generic Aging Lessons Learned Report] AMP XI.M17. During the audits and inspections, the staff noted that CNP's Flow-Accelerated Corrosion (FAC) Program is consistent but with an exception. The Monitoring and Trending element of GALL AMP XI.M17 requires that if degradation is detected such that the wall thickness is less than the minimum predicted thickness, additional examinations are performed in adjacent areas to bound the thinning. However, CNP's FAC program bases its sample expansion determination on a threshold criteria rather than on predicted thickness. Sample size is increased when inspections detect significant FAC wear resulting in a wall thickness threshold of less that or equal to 60 percent of nominal wall thickness. In RAI B.1.12-1, the staff requested that the applicant provide a description of the FAC Program, as modified by the exception, and justification for the exception regarding the criteria for performing additional examinations by expanding the sample size. The concern was not resolved by the time this SER was issued, this concern is Open Item B.1.12-1.

I&M Response to Draft SER Open Item B.1.12-1:

In the FAC Program description in NUREG-1801 Section XI.M17, the Monitoring and Trending section states, in part, that "If degradation is detected such that the wall thickness is less than the minimum predicted thickness, additional examinations are performed in adjacent areas to bound the thinning." Literal implementation of this statement from the NUREG-1801 program description is not practical in many cases. If very little degradation is predicted, measured wall thickness may be less than the predicted thickness even though the calculated life of the affected component may exceed the operating life of the plant. In this case, sample expansion would not be warranted. Therefore, I&M takes exception to the Monitoring and Trending attribute of NUREG-1801, Section XI.M17.

The CNP FAC Program is based on industry guidance in the Electric Power Research Institute (EPRI) report NSAC-202L-R2, *Recommendations for an Effective Flow-Accelerated Corrosion Program*, dated April 1999, which recommends increasing the sample size when inspections of the sample detect significant FAC wear. In the CNP FAC Program, significant FAC wear is defined as FAC resulting in a wall thickness of less than or equal to 60 percent of nominal wall thickness. Sample expansion is typically required if any component is determined to have a wall thickness of less than or equal to 60 percent of nominal wall thickness. In addition, CNP FAC procedures require that a sample expansion be performed when inspection results indicate that a component has a remaining life less than one operating cycle. This covers situations where the minimum wall thickness required may be greater than 60 percent of nominal wall thickness.

An exception to NUREG-1801 is warranted because literal implementation of the sample expansion criterion is not practical. The CNP FAC program criterion for sample expansion is acceptable because it specifies a wall thickness criterion and requires projection of inspection results to the next inspection opportunity consistent with industry guidance.

Draft SER Confirmatory Item 4.3-1:

The applicant provided a UFSAR [Updated Final Safety Analysis Report] supplement description of the Fatigue Monitoring Program (FMP) in Section A.2.1.12 of the LRA and a description of its TLAA [Time-Limited Aging Analysis] evaluation for Class 1 and non-Class 1 component fatigue analyses in Section A.2.2.2 of the LRA. The applicant was requested to update Section A.2.2.2 to include a discussion of its proposed actions to evaluate the auxiliary spray line piping and a discussion of its proposed actions to evaluate the environmental fatigue of the safety injection nozzles, charging nozzles, and the residual heat removal (RHR) line.

Draft SER Confirmatory Item 4.6-1:

The applicant provided a UFSAR supplement describing its TLAA evaluation for containment liner plate and penetration fatigue analyses in Section A.2.2.4 of the LRA. The applicant committed to updating the UFSAR Supplement to capture its commitment to analyze the containment penetrations.

I&M's Response to Draft SER Confirmatory Items 4.3-1 and 4.6-1:

By letters dated June 16, 2004, and September 21, 2004 (References 6 and 7), I&M committed to evaluate the environmentally-assisted fatigue of the Class 1 RHR piping and the safety injection and charging nozzles, respectively. I&M's commitment to review the piping loads on hot containment penetrations was also provided in the June 16, 2004, letter (Reference 6). These one-time commitments are tracked and managed by CNP's Commitment Management Program, in accordance with plant procedures. Although one-time commitments are typically not addressed in the UFSAR, I&M has opted to include a summary of relevant license renewal commitments in the appropriate aging management program and TLAA summaries within the UFSAR Supplement in LRA Appendix A.

In response to DSER Confirmatory Item 4.3-1, I&M has revised the Environmentally-Assisted Fatigue subsection of the UFSAR Supplement in LRA Section A.2.2.2 and has included the commitment to address environmental fatigue of the pressurizer surge line, Class 1 safety injection and charging nozzles, and Class 1 portions of RHR piping in LRA Section A.2.1.12. In response to DSER Confirmatory Item 4.6-1, I&M has revised the Containment Penetration Fatigue subsection of the UFSAR Supplement, in LRA Section A.2.2.4. Additionally, to ensure a consistent approach is followed in the treatment of license renewal commitments, I&M reviewed all commitments that were provided in correspondence related to the NRC review of the LRA. The following table identifies the UFSAR Supplement sections that are revised to reflect one-time commitments that were made in I&M's RAI responses and that were not previously included in RAI responses modifying LRA Appendix A.

UFSAR Supplement Changes for One-Time License Renewal Commitments

Summary of Commitment	I&M Reference	UFSAR Supplement Changes
• Environmentally-assisted fatigue of Class 1 portions of RHR piping	June 16, 2004 (Reference 6)	A.2.1.12 and A.2.2.2, under Environmentally-Assisted Fatigue heading
• Environmentally-assisted fatigue of Class 1 charging and safety injection nozzles	Sept. 21, 2004 (Reference 7)	A.2.1.12 and A.2.2.2, under Environmentally-Assisted Fatigue heading
• Hot containment penetration fatigue	June 16, 2004 (Reference 6)	A.2.2.4, under Containment Penetration Fatigue heading
• Submit Alloy 600 Aging Management Program inspection plan to NRC three years prior to period of extended operation	August 11, 2004 (Reference 8)	A.2.1.1
• Submit Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program to NRC three years prior to period of extended operation	October 18, 2004 (Reference 4)	A.2.1.30
• Evaluate auxiliary spray line piping to address isolation valve leakage per WCAP-14070	October 18, 2004 (Reference 4)	A.2.2.2, under Class 1 Metal Fatigue heading

A mark-up of the affected sections of the UFSAR Supplement follows. New text that was added to reflect the incorporation of these one-time commitments is shown in *italics* and deleted text is shown in ~~strikeout~~.

A.2.1.1 ALLOY 600 AGING MANAGEMENT PROGRAM

This program will manage aging effects of Alloy 600/690 components and Alloys 52/152 and 82/182 welds in the reactor coolant system that are not addressed by the following aging management programs:

- The Control Rod Drive Mechanism and Other Vessel Head Penetration Inspection Program, Section A.2.1.9;
- The Steam Generator Integrity Program, Section A.2.1.34; and
- The Reactor Vessel Internals Programs, Sections A.2.1.30 and A.2.1.31.

The Alloy 600 Aging Management Program will detect cracking from primary water stress corrosion cracking (PWSCC) by using the examination and inspection requirements specified in ASME Section XI. Guidance developed by the EPRI Material Reliability Program and the owners groups will be used to identify susceptibility rankings and program inspection requirements regarding Alloy 82/182 pipe butt welds.

The Alloy 600 Aging Management Program inspection plan will be submitted for NRC staff review and approval three years prior to the period of extended operation to determine if the program demonstrates an ability to manage the effects of aging per 10 CFR 54.21(a)(3). This program will be implemented prior to the period of extended operation.

A.2.1.12 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program monitors and tracks the number of critical thermal and pressure transients for selected reactor coolant system components in order not to exceed the design limit on fatigue usage. The program maintains the basis for component analyses containing explicit thermal cycle count assumptions. Components managed by this program are those shown to be acceptable by analyses that explicitly addressed thermal and pressure fatigue transient limits. *As discussed in Section A.2.2.2, the Fatigue Monitoring Program will be enhanced* ~~This program requires enhancements that will be implemented~~ prior to the period of extended operation *to address environmentally-assisted fatigue for the pressurizer surge line, Class 1 portions of the RHR piping, and the Class 1 charging and safety injection nozzles.*

A.2.1.30 <u>REACTOR VESSEL INTERNALS PLATES, FORGINGS, WELDS, AND BOLTING PROGRAM</u>

The Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program will manage aging effects of reactor vessel internals plates, forgings, welds, and bolting. This program will supplement the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Programs. This program will manage the effects of:

- Crack initiation and growth due to stress corrosion cracking or irradiation-assisted stress corrosion cracking,
- Loss of fracture toughness due to neutron irradiation embrittlement, and
- Distortion due to void swelling.

This program will provide visual inspections and non-destructive examinations of the reactor vessel internals. I&M will participate in industry-wide programs designed by the PWR Materials Reliability Project Reactor Internals Issues Task Group for investigating the impacts of aging on PWR vessel internal subcomponents. The Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program will be *submitted for staff review and approval three years prior to the period of extended operation and* implemented prior to the period of extended operation.

A.2.2.2 <u>METAL FATIGUE</u>

The analysis of metal fatigue is a TLAA for Class 1 and selected non-Class 1 mechanical components within the scope of license renewal.

<u>**Class 1 Metal Fatigue**</u>

Fatigue evaluations performed in the design of the Class 1 reactor coolant system (RCS) components were based on a number of design cycles assumed for the life of the plant. The RCS design transients used in the fatigue evaluations for the Class 1 components were reviewed for both units. The numbers of actual RCS design transients from plant operating history were extrapolated to 60 years of operation. ~~In all instances~~ *Except for auxiliary spray line piping thermal cycling transient described in WCAP-14070, Page 6-3,* the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Therefore, *except for auxiliary spray line piping thermal cycling transient fatigue evaluation,* the fatigue evaluations for the Class 1 components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The RCS design transients are monitored through the Fatigue Monitoring Program, which is discussed in Section A.2.1.12.

Class 1 piping has been qualified in accordance with USAS B31.1. The allowable stress limits for the piping implicitly assumes a limit of 7000 equivalent full-temperature thermal cycles. To identify the specific locations where extended operation could invalidate the stress limits, the design temperatures and operating conditions of the Class 1 piping systems were reviewed. This review determined that, based on assumptions of fewer than 7000 equivalent full-temperature thermal cycles, the analyses for all locations are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

In response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and its supplements, a fatigue evaluation of the auxiliary spray line was performed and reported in WCAP-14070 (Reference A.2.3.10). The fatigue evaluation is based on CNP design transients. ~~As described above, the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Thus, the auxiliary spray line thermal stratification analysis and its results are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).~~

The transient development assessment in WCAP-14070 evaluated the effects of a cyclic leak in an auxiliary spray isolation valve. The WCAP-14070 evaluation assumed the cyclic leakage would continue throughout the 40 years of plant operation. Therefore, this frequency is time-dependent and constitutes a TLAA.

The WCAP-14070 auxiliary spray line piping thermal fatigue TLAA will be addressed prior to the period of extended operation by performing one or more of the following:

(1) Perform a plant-specific fatigue reanalysis of the auxiliary spray line piping prior to entering the period of extended operation to ensure that cumulative usage factors (CUFs) are below 1.0;

(2) Repair piping at the affected locations;

(3) Replace piping at the affected locations;

(4) Manage the effects of fatigue of the auxiliary spray line piping by an NRC-approved inspection program (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC). It is expected that the inspections will be able to detect cracking due to thermal fatigue prior to loss of function. Replacement or repair, if necessary, will then be implemented such that the intended function will be maintained for the period of extended operation.

A plant-specific structural analysis of the pressurizer surge line performed and reported in WCAP-12850 (Reference A.2.3.11) supports the conclusion that CNP is in compliance with the requirements of NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." The surge line stratification analysis was based on the CNP design transients. As described above, the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Thus, the existing pressurizer surge line thermal stratification analysis and its results are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Non-Class 1 Metal Fatigue

Non-Class 1 piping within the scope of license renewal was designed to USAS B31.1. Piping components that may have Normal or Upset Condition operating temperatures in excess of 220°F for carbon steel, or 270°F for austenitic stainless steel, were evaluated for fatigue. These piping components were evaluated for their potential to exceed the limiting number of equivalent full-temperature cycles used for the original design in 60 years of plant operation. With one exception, the review determined that none of the piping or components would exceed the limit of equivalent full-temperature thermal cycles. Thus, for all but the one exception, fatigue considerations for the original piping and component design are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). For the exception, RCS sampling system piping, a calculation was prepared to justify a new limit to support RCS sampling for 60 years of operation.

Only non-Class 1 pressure vessels, heat exchangers, storage tanks, and pumps designed and fabricated in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, Division 2 or Section III, NC-3200 (Class B) require evaluation for thermal fatigue. Of these components, consideration of thermal fatigue is not required unless specifically directed by the equipment specification. A review of the components designed to the above Code requirements determined that the components with equipment specifications requiring consideration of thermal fatigue used design transients identified consistent with the RCS transients defined in Table 4.1-10 of the UFSAR. As described for Class 1 metal fatigue in this section, the assumed number of RCS design transients is acceptable for 60 years so the fatigue evaluation considered in the original design of these components will remain valid during the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Environmentally-Assisted Fatigue

Recent test data indicates that certain environmental effects (such as temperature, oxygen, and stress rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. Although the NRC has concluded that the environmental effects associated with fatigue life are not safety significant through the end of the initial license term, they also determined that the effects of fatigue should be addressed for license renewal.

The effects of environmentally-assisted thermal fatigue for the limiting locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, have been evaluated for CNP in accordance with 10 CFR 54.21(c)(1)(i and ii). ~~and all locations were determined to be acceptable for the period of extended operation with the exception of the pressurizer surge line.~~ *The evaluations determined that the reactor vessel shell, lower head, and inlet and outlet nozzles are acceptable for the period of extended operation.* ~~Aging management of cracking by environmentally-assisted fatigue of the pressurizer surge line is addressed by~~

Prior to the period of extended operation, the Fatigue Monitoring Program, which is discussed in Section A.2.1.12, *will be enhanced to address environmentally-assisted fatigue of the pressurizer surge line, Class 1 portions of the RHR piping, and the Class 1 charging and safety injection nozzles,* in accordance with 10 CFR 54.21(c)(1)(iii). *The approach for addressing environmentally-assisted fatigue for these components will include one or more of the following:*

(1) A plant-specific fatigue analysis that includes environmental effects will be performed to ensure that cumulative usage factors remain below 1.0;

(2) Repair of the affected locations;

(3) Replacement of the affected locations;

(4) Manage the effects of fatigue on the affected locations by an NRC-approved inspection program;

(5) Monitor ASME Code activities to use the environmental fatigue methodology approved by the ASME Code Committee and the NRC.

A.2.2.4 CONTAINMENT LINER PLATE AND PENETRATION FATIGUE ANALYSES

TLAAs applicable to the containment structure are the containment liner plate and the containment penetration fatigue analyses.

Containment Liner Plate Fatigue

The fatigue life of the liner was evaluated in 1999 after discovery of localized thinning of the liner. The evaluation, based on testing, determined a fatigue cyclic loading limit for the uncorroded liner plate, of 180,000 cycles at an amplitude of ± 20 ksi. The amplitude of a thermal stress cycle based on an enveloping assessment of the liner design cyclic loads (UFSAR, Section 5.2.3) is well within the amplitude of the evaluated limit. Additionally, the number of containment load and thermal cycles expected during the plant life including the period of extended operation is insignificant compared to 180,000 cycles. Therefore, the analysis of fatigue for the containment liner will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Containment Penetration Fatigue

Analyses for the main steam and residual heat removal (RHR) penetrations were developed using the operating transients listed in Table 4.1-10 of the UFSAR. The analyses determined that the requirements of ASME III, Section N-415.1 (exemption from fatigue) were met and that fatigue evaluations were not required for the main steam and RHR penetrations. The analyses supporting the exemption-from-fatigue analyses are TLAAs, since the evaluation is based on selected design thermal and loading cycles. As described for Class 1 metal fatigue in Section A.2.2.2, the assumed number of RCS design transients is acceptable for 60 years. Therefore, the exemption-from-fatigue evaluations for the main steam and RHR containment penetration analyses remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

I&M will review the piping loads on the remaining hot penetrations to establish the base loads for the fatigue exemption provisions of ASME Section III, N-415.1. The penetrations will be grouped based on their duty cycle during normal operations including inservice testing duty. The cycle loads and stresses will be added to the piping analysis loads as appropriate and the resultant loads will be compared to the fatigue exemption provisions of ASME Section III, N-415.1. Any penetration group that does not meet the exemption provisions will be analyzed for fatigue using the most limiting penetration to represent the group. This evaluation will be completed prior to entering the period of extended operation, and will be projected to the end of the period of extended operation.

RAI B.1.3-2, Part 1:

In recent discussions between the applicant, NRC Regional III inspector and NRC DE staff, the licensee explained that trending of the Boral coupon measurements is not performed because the measurement uncertainty is equivalent to the acceptance criteria (5% B-10 decrease and 10% thickness increase). The staff's understanding is that the coupon either passes or fails the acceptance test based on these two criteria. According to the Boral Surveillance Program (12-THP-6020-CSP-203), failure would require an investigation, engineering evaluation, and perhaps additional testing (such as blackness testing of the storage racks). Also according to the Boral Surveillance Program, the remaining measurement parameters are used to detect early indications of degradation and may prompt a change in the measurement schedule.

In a letter dated August 11, 2004, the applicant stated that the most recent coupon thickness change ranged from -0.67% to 1.19%. This suggests a measurement precision better than ± 10%. The staff asks that the licensee respond to the following:

1) *Clarify the capability to measure and evaluate coupon thickness.*

2) *Provide the results of the coupon evaluations. How did the measured neutron attenuation and thickness compare to the acceptance criteria? What were the results and conclusions from the other measurement parameters used to detect early indications of Boral degradation? If early indications of degradation were detected, what actions were taken?*

3) *In a clarification to RAI B.1.3-1, the applicant states 5% variation in B-10 areal density is within the "usual uncertainty tolerance applied in the nuclear criticality safety analyses." Please confirm that this value was used in the most recent criticality safety analysis for CNP.*

I&M Response to RAI B.1.3-2, Part 1:

1) The calibrated measuring instruments provide the capability to measure length, width, and thickness at predetermined locations to an accuracy that meets or exceeds CNP's program specifications (±0.04 inches, ±0.02 inches, ±0.002inches, respectively). These measurements are recorded and compared to the initial (baseline) measurements. An increase in thickness at any point that exceeds 10 percent of the baseline thickness requires investigation and engineering evaluation. By using this process, the cumulative environmental effects (radiation, thermal, chemistry) on the coupon, and indirectly the Boral panels, can be monitored as subsequent coupons are removed and evaluated.

   The Boral Surveillance Program identifies the areas on the coupon where the measurements are to be taken. The table included at the end of this response provides a comparison of the as-measured length, width, and thickness dimensions with coupon baseline dimensions for measurements performed to-date. The difference in thickness is presented as a percentage change with respect to the initial thickness. The thickness measurements were taken at the same locations as the five baseline measurements. No investigation or engineering evaluation was performed because the measured thicknesses did not exceed the acceptance criteria of 10 percent of the baseline thickness at any location.

2) Coupons ID 213616-1-3 and ID 213616-1-5 were removed and evaluated in 1994. These coupons were reattached to the coupon tree in January 1995. The next coupon (ID 213616-1-3) was not removed until 2001.

   The coupon evaluation results are summarized in the table included at the end of this response. As shown in the table, the Boron-10 areal density corresponds to neutron attenuation measurements. The percent differences between the baseline and measured areal density values are within the areal density acceptance criteria (i.e., areal density decrease of no more than five percent in Boron-10 content). In addition, the range of percentages between the baseline and measured thicknesses is within the thickness acceptance criteria (i.e., increase in thickness at any point of no more than 10 percent of the initial thickness at that point).

Regarding other measurement parameters, the table shows that there was no appreciable percent difference between the baseline and measured dry weight values. Additional testing, such as neutron radiograph (confirmation of uniform boron distribution within coupon), has not been performed on the evaluated coupons because no evidence of Boral degradation has been identified. Visual or photographic results are not available for the coupons evaluated in 1994. However, the visual inspection after the coupon was removed in 2001 indicated minor corrosion pitting, which had not progressed to the extent that it would affect the Boral function. No unusual surface pitting, corrosion, or edge deterioration was identified. As no early coupon degradation has been observed, no engineering evaluations or actions have been taken.

3) A five percent variation in Boron-10 areal density is conservative with respect to the corresponding assumption in the most recent spent fuel pool criticality analysis. The nominal Boron-10 density in the Boral absorber panel is 0.0345 grams per square centimeter (g/sq cm) and the minimum Boron-10 density assumed in the uncertainty analysis is 0.030 g/sq cm, a variation of approximately 15 percent.

RAI B.1.3-2, Part 2:

The "Schedule of Coupon Surveillance" in the applicant's Boral Surveillance Program specifies a range of years over which the first 5 test coupons can be removed from the rack for evaluation. According to the schedule, the time between coupon evaluations can range from 1 year to 5 years. For example, Coupon #3 and #4 could be pulled 3 years and 8 years respectively after removal of coupon #1. Starting with coupon #6 however, the evaluation interval is 5 years.

To determine the significance of establishing a 5 year test interval, the staff asks the applicant to respond to the following:

1) *Please provide the dates that coupons were actually removed and evaluated.*

2) *Please explain how the coupon removal/evaluation times are determined. For example, how did the applicant decide if coupon #4 would be removed and evaluated 6, 7, or 8 years after removal of coupon #1?*

I&M Response to RAI B.1.3-2, Part 2:

1) Coupon removal and evaluation dates are as follows:

Coupon Number	Coupon removal date	Evaluation completion date
ID213616-1-3	October 1994	December 1994
ID213616-1-5	October 1994	December 1994
ID213616-1-3[a]	November 2001	March 2002

(a) Coupon ID213616-1-3 was removed in 1994, and reinstalled in 1995. This coupon was also removed for evaluation in 2001.

As indicated in LRA Section B.1.3, Operating Experience, on Page B-25, insufficiently defined responsibilities in the controlling procedure resulted in missed samples (i.e., the Boral coupon removal and evaluation tasks were not performed twice between 1994 and 2001, as required by the Boral Surveillance Program procedure). This condition was identified in 1999 during the extended plant shutdown. The three-year plant restart effort that concluded in 2000 resulted in numerous plant process improvements. Significant process improvements were implemented in response to a comprehensive assessment of work control activities. These improvements included upgrades to the work control process and improvements in process and program ownership. For example, procedures were enhanced to more clearly define roles and responsibilities for activity task owners and to apply more stringent controls to the process for requesting, evaluating, and approving task deferrals. Additionally, for the Boral Surveillance Program, ownership has been improved by reassigning administration of this program to the design organization. These process and program changes provide assurance that future sampling requirements will not be missed.

2) The coupon removal/evaluation schedule was based on vendor recommendations. The guidance for the removal/evaluation schedule is intended to allow coupons to accumulate more radiation dose than the expected lifetime dose for normal storage. Accelerated dose is accomplished by re-installing the coupon tree in a new location surrounded by freshly discharged fuel assemblies that have been among the higher specific power assemblies in the core. In accordance with plant procedures, surveillance coupons are typically removed for evaluation one or two months prior to a reactor refueling for either unit. At that time, the coupon tree is moved to a region where it will be surrounded by freshly discharged fuel assemblies upon completion of fuel off-load. This coupon tree relocation process was initiated at the time of the first fuel off-load following installation of the coupon tree and has been repeated for each coupon removal/evaluation, at a minimum. If future evaluations determine that dose acceleration is no longer required, coupon evaluations may continue without relocating the coupon tree. In accordance with the Boral Surveillance Program procedure, the next coupon is scheduled to be removed prior to the end of 2005, and the remaining coupons will be removed every five years for the duration of wet storage. The five-year removal frequency is also based on vendor recommendations, and is further

justified by the lack of coupon degradation noted when the coupon was evaluated after being in the spent fuel pool for seven years. Periodicity of coupon removal may be adjusted depending on coupon inspection results.

Boral Coupon Evaluation Results

Coupon Number	Removal Date	L1 (in.)	L2 (in.)	L3 (in.)	W1 (in.)	W2 (in.)	W3 (in.)	T1 (in.)	T2 (in.)	T3 (in.)	T4 (in.)	T5 (in.)	Dry Weight (gm)	Density (gm/cm^3)	Areal Density (gm B-10 per cm^2)
ID213616-1-3	Baseline	15.015	15.022	15.028	7.522	7.52	7.523	0.104	0.102	0.102	0.103	0.101	468.08	2.4974	0.0345
	Oct-94	15.023	15.031	15.038	7.521	7.521	7.524	0.1015	0.1005	0.102	0.102	0.1005	470.5	2.515	0.0351
	Difference (%)	0.05	0.06	0.07	-0.01	0.01	0.01	-2.40	-1.47	0.00	-0.97	-0.50	0.52	0.70	1.74
ID213616-1-5	Baseline	15.022	15.025	15.029	7.53	7.53	7.534	0.102	0.101	0.102	0.104	0.102	469.13	2.508	0.0345
	Oct-94	15.035	15.034	15.038	7.533	7.538	7.531	0.101	0.1	0.101	0.1025	0.101	469.4	2.508	0.0351
	Difference (%)	0.09	0.06	0.06	0.04	0.11	-0.04	-0.98	-0.99	-0.98	-1.44	-0.98	0.06	0.00	1.74
ID213616-1-3	Baseline	15.015	15.022	15.028	7.522	7.52	7.523	0.104	0.102	0.102	0.103	0.101	468.08	2.4974	0.0345
	Nov-01	15.019	15.025	15.021	7.524	7.5255	7.526	0.1033	0.1026	0.1032	0.1028	0.1022	470.6	2.51	0.035
	Difference (%)	0.03	0.02	-0.05	0.03	0.07	0.04	-0.67	0.59	1.18	-0.19	1.19	0.54	0.50	1.45

Difference (%) is the percent difference between the baseline and the as-measured dimensions.

References:

1. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Application for Renewed Operating Licenses," AEP:NRC:3034, dated October 31, 2003 [Accession No. ML033070177].
2. Letter from P. T. Kuo, NRC, to M. K. Nazar, I&M, "Safety Evaluation Report (SER) with Open Items Related to the License Renewal of Donald C. Cook Nuclear Plant, Units 1 and 2," dated December 21, 2004 [Accession No. ML043570535].
3. Letter from J. Rowley, NRC, to M. K. Nazar, I&M, "Request for Additional Information (RAI) for the Review of the Donald C. Cook Nuclear Plant, Units 1 and 2, License Renewal Application," dated January 12, 2005 [Accession No. ML050120254].
4. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2; License Renewal Application – Response to Requests for Additional Information (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-17, dated October 18, 2004 [Accession No. ML042960028].
5. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2; License Renewal Application – Response to Requests for Additional Information on Electrical and Auxiliary Systems (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-06, dated June 8, 2004 [Accession No. ML041680255].
6. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2; License Renewal Application – Response to Requests for Additional Information on Time-Limited Aging Analyses (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-08, dated June 16, 2004 [Accession No. ML041750561].
7. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2; License Renewal Application – Supplemental Responses to Requests for Additional Information (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-16, dated September 21, 2004 [Accession No. ML042740439].
8. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2; License Renewal Application – Response to Requests for Additional Information on Aging Management Programs (TAC Nos. MC1202 and MC1203)," AEP:NRC:4034-10, dated August 11, 2004 [Accession No. ML042470410].