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PROPRIETARY INFORMATION
WITHHOLD FROM PUBLIC DISCLOSURE UNDER 10 CFR 2.390

August 6, 2009

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 09-451
LR/MWH R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE KEWAUNEE POWER STATION LICENSE RENEWAL APPLICATION - LEAK BEFORE BREAK / BORAL

By letter dated July 7, 2009 (Reference 1), the NRC requested additional information regarding the evaluation of leak-before-break (LBB) time-limited aging analyses (TLAA), the aging management review results for spent fuel storage racks, and component screening results included in the license renewal application (LRA) for Kewaunee Power Station (KPS). The NRC staff indicated that the responses to the requests for additional information (RAIs) are needed to complete the review related to the KPS LRA.

The attachment to this letter contains the responses to the RAIs. Enclosures A, B, and D to this letter contain documents requested by the NRC staff in the RAIs.

Enclosure A contains information proprietary to Westinghouse Electric Corporation (Westinghouse). Accordingly, it is requested that this information be withheld from public disclosure pursuant to 10 CFR 2.390(a)(4) of the Commission's regulations. This request is supported by an affidavit (Enclosure C), signed by Westinghouse, that 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 2.390 of the Commission's regulations. A redacted version of this information is provided as Enclosure B.

ENCLOSURE CONTAINS PROPRIETARY INFORMATION
Enclosure A contains information to be withheld from public disclosure in accordance with 10 CFR 2.390. Upon removal of Enclosure A, this letter is decontrolled.

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NRR

Should you have any questions regarding this submittal, please contact Mr. Paul C. Aitken at (804) 273-2818. Questions regarding the proprietary aspects of the Westinghouse information and supporting Westinghouse affidavit should be addressed to: J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, PA 15230-0355.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Support Services

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and State aforesaid, today by Leslie N. Hartz, who is Vice President – Nuclear Support Services of Dominion Energy Kewaunee, Inc. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 6TH day of August, 2009.

My Commission Expires: May 31, 2010 Vicki L. Hull
Notary Public

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2010

Reference:

1. Letter from Samuel Hernandez (NRC) to David A. Heacock (DEK), "Request for Additional Information for the Review of the Kewaunee Power Station License Renewal Application – Leak Before Break/BORAL (TAC No. MD9408)," dated July 7, 2009. [ADAMS Accession No. ML091190389]

Attachment:

1. Response to Request for Additional Information Regarding Leak-Before-Break Time-Limited Aging Analyses, Spent Fuel Storage Rack Aging Management Review, and Component Screening Results

Enclosures:

A. WCAP-16738-P, Revision 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program," dated March, 2007. PROPRIETARY VERSION
B. WCAP-16738-NP, Revision 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program," dated March, 2007. NON-PROPRIETARY VERSION
C. Affidavit for Westinghouse Report WCAP-16738-P, Rev. 0, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program."
D. SIR-00-045, Revision 2, "Leak-Before-Break Evaluation 6-inch to 12-inch Safety Injection and Residual Heat Removal Piping Attached to the RCS (Kewaunee Nuclear Power Plant)"

Commitments made in this letter:

None.

cc: (without Enclosure A)

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ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LEAK-BEFORE-BREAK TIME-LIMITED AGING ANALYSES, SPENT FUEL STORAGE RACK AGING MANAGEMENT REVIEW, AND COMPONENT SCREENING RESULTS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

RAI 4.7.3-1

Background:

Section 4.7.3, page 4-36 of the Kewaunee Power Station License Renewal Application (LRA), cites NUREG-1031, Volume 3 as a source of procedures and guidance for the application of the Leak-Before-Break (LBB) methodology.

Issue:

It is the staff understanding that the procedures for LBB methodology are contained in NUREG-1061, Volume 3.

Request:

Please indicate if there is a typographical error in the description of the document referenced in the LRA.

DEK Response

The citation of NUREG-1031, Volume 3 in LRA Section 4.7.3 is the result of a typographical error. The correct reference is NUREG-1061, Volume 3.

RAI 4.7.3-2

Background:

Pressurized water reactors plants have experienced primary water stress corrosion cracking (PWSCC) in Alloy 82/182 dissimilar metal welds in the American Society of Mechanical Engineers (ASME) Class 1 piping.

Issue:

The PWSCC has an aggressive crack growth rate and is an active degradation mechanism. The LBB application prohibits active degradation mechanisms. Industry and the U.S. Nuclear Regulatory Commission (NRC) are currently working to resolve primary water stress corrosion cracking (PWSCC) in Alloy 82/182 welds with respect to the LBB analysis assumptions. It is not clear whether PWSCC is an issue for the LBB piping at Kewaunee.

Request:

- *Identify all Alloy 82/182 dissimilar metal welds in the LBB piping.*
- *If Alloy 82/812 welds exist in the LBB piping, discuss the actions that will be taken to mitigate and/or inspect the Alloy 82/182 welds to ensure that PWSCC will not affect the structural integrity of the LBB piping.*

DEK Response

The only Alloy 82/182 dissimilar metal welds in the LBB piping at Kewaunee are the steam generator primary nozzle-to-reactor coolant loop piping welds. These welds are clad internally with Alloy 52/152 weld material such that the Alloy 82/182 material is not exposed to the reactor coolant environment. Therefore, the Alloy 82/182 weld material is not susceptible to PWSCC in this application and no mitigative actions are required to ensure that PWSCC will not affect the structural integrity of the LBB piping.

RAI 4.7.3-3

<u>Background:</u>

Section 4.7.3, Leak-Before-Break, discusses the fatigue flaw growth and fracture toughness of cast austenitic stainless steel material as part of the Time-Limited Aging Analysis (TLAA).

<u>Issue:</u>

It is not clear whether there are flaws in the LBB piping. Also, it is not clear as to the current status of the LBB piping structural integrity.

<u>Request:</u>

- *Discuss the inspection history and results of the LBB piping.*
- *If indications or flaws are remaining in service in the LBB piping, discuss how the indications and flaws are monitored to the end of the period of extended operation.*
- *Discuss future inspection schedules for each of the LBB pipes (other than indications and flaws).*

DEK Response

The fatigue flaw growth and fracture toughness discussion in LRA Section 4.7.3 refers to the methodology for performance of leak-before-break (LBB) evaluations. This methodology includes postulating piping flaws, and was not intended to imply that any piping flaws currently exist.

The piping in the scope of the LBB analyses at Kewaunee has been inspected in accordance with the requirements of the approved ASME Section XI Inservice Inspection (ISI) Program since initial plant operation. This piping is subject to the inspection requirements of the ISI Program through the period of extended operation. There are no currently identified unresolved reportable indications or flaws existing in this LBB piping.

RAI 4.7.3.1-01

Background:

Section 4.7.3.1, LBB-Reactor Coolant Loop Piping, states that the LBB evaluations have been updated to support the power uprate program and steam generator replacement.

Issue:

The LRA did not discuss the impact of the power uprate and steam generator replacement on the LBB piping and fatigue flaw growth analysis, other than fracture toughness values.

Request:

Discuss the impact of the operating conditions of power uprate and steam generator replacement on all the LBB piping (including branch lines and surge line) and fatigue flaw growth analysis at the end of 60 years. The staff requests the applicant to submit the analyses (i.e., WCAP-16738-P) that supports its conclusions regarding the impact of power uprate on the reactor coolant loop piping during the period of extended operation.

DEK Response

The Kewaunee power uprate and steam generator replacement projects were completed in 2004 and 2001, respectively. The impact of power uprate and steam generator replacement on the plant, including the LBB analyses, was evaluated and incorporated into the current licensing basis at the time these projects were completed.

WCAP-16738-P, *Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Kewaunee Power Station for the License Renewal Program*, evaluated the impact that an additional 20 years of plant operation would have on the reactor coolant loop and pressurizer surge line LBB analyses, with consideration of power uprate and steam generator replacement. Enclosures A and B of this letter provide a Proprietary and non-Proprietary copy of WCAP-16738, respectively, as requested. Enclosure C is an Affidavit, which supports the basis for designating the information in Enclosure A as proprietary and requests withholding the information from public disclosure.

RAI 4.7.3.1-02

Background:

Section 4.7.3.1, LBB-Reactor Coolant Loop Piping, states that Westinghouse has updated the LBB analysis to support the steam generator replacement project in WCAP-15311 and the power uprate program in WCAP-16040-P.

Issue:

The LRA states that a review of the above documents identified that the fracture toughness values for the cast austenitic stainless steel loop piping were based on a 40-year plant service life. The LRA states that the fracture toughness for the fully aged condition was used and that mechanical properties were determined at operating temperatures. However, the LRA did not discuss whether the fracture toughness values at the 60 years were used.

Request:

- *Discuss whether the fracture toughness values used in the LBB evaluations were the values at the end of the 60-year plant life.*
- *Explain why the mechanical properties were determined at operating temperatures, not at the temperature at faulted conditions.*

DEK Response

As indicated in LRA Section 4.7.3.1, 'fully-aged' fracture toughness values were used in the updated LBB analysis. 'Fully-aged' refers to the cast stainless steel fracture toughness properties corresponding to the maximum thermal aging condition, and is determined based on the methodology presented in NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems*. Accordingly, the fully-aged material fracture toughness values used in the LBB evaluation are conservative and envelope the material condition at the end of the 60-year plant life.

The fracture toughness of cast stainless steel is adversely affected by long-term exposure to a high temperature environment, resulting in thermal embrittlement. Since 'faulted' conditions are short-lived, the long-term high temperature environment that the reactor coolant loop piping is exposed to is the normal reactor coolant operating temperature. For this reason, the short-lived 'faulted' conditions are not expected to contribute significantly to thermal embrittlement of the piping. Therefore, the mechanical properties were determined using the normal operating temperatures.

RAI 4.7.3.1-03

<u>Background:</u>

Section 4.7.3.1, LBB-Reactor Coolant Loop Piping, states that the LBB analysis for the period of extended operation is discussed in WCAP-16738. The applicant states that the report documents the plant specific geometry, operating parameters, loading, and material properties used in the fracture mechanics evaluation.

<u>Issue:</u>

It is not clear as to the impact of 60-year operation on the above parameters in the original LBB analysis. Also, it is not clear whether WCAP-16738 considered the effect of power uprate and steam generator replacement.

<u>Request:</u>

- *Discuss the impact of 60-year operation on the material properties, operating parameters, and loading of reactor coolant loop, surge line, and branch piping.*

DEK Response

The impact of 60-year operation is potentially significant to cast austenitic stainless steel (CASS) piping material properties (fracture toughness) due to the prolonged exposure to a high temperature environment resulting in thermal aging of the material. The reactor coolant loop piping includes CASS material and has been evaluated for the effects of thermal aging. As indicated in LRA Section 4.7.3.1, thermal aging resulting from 60-year operation was considered in the LBB analysis for the reactor coolant loop piping and the results of the analysis were found to remain acceptable. There is no CASS material in the pressurizer surge line piping or the reactor coolant loop branch line piping.

During the period of extended operation, there are no anticipated changes to operating parameters or loading on the reactor coolant loop piping, surge line piping, or reactor coolant loop branch line piping for which there are LBB analyses. Therefore, there is no impact due to 60-year operation on operating parameters and loadings for this piping.

RAI 4.7.3.1-04

Background:

Section 4.7.3.1, LBB-Reactor Coolant Loop Piping, references WCAP-11411 (Reference 4.8-15).

Request:

Reference 4.8-15 cites WCAP-14111 instead of WCAP-11411. Please indicate if there is a typographical error in the description of the document referenced in the LRA.

DEK Response

The citation of WCAP-14111 (Reference 4.8-15) is the result of a typographical error. The correct document number is WCAP-11411.

RAI 4.7.3.2-01

Background:

Section 4.7.3.2, LBB-Pressurizer Surge Line Piping, discusses the applicant's evaluation of pressurizer surge line.

Issue:

It is not clear whether thermal stratification events have occurred in the surge line in the past.

Request:

- *Discuss operating procedures implemented to prevent or mitigate future thermal stratification events.*

DEK Response

As discussed in LRA Section 4.7.3.2, the pressurizer surge line crack growth predictions were based on the design basis operational transients for the nuclear steam supply system (NSSS), including the effects of thermal stratification. Kewaunee operating procedures have historically limited the temperature difference between the reactor coolant loop and the pressurizer during plant start-up and shutdown, which minimizes the effects pressurizer insurge / outsurge and thermal stratification in the surge line. As stated in LRA Section 4.3.1.4, Pressurizer Lower Head and Surge Line, operating procedures were changed at the end of cycle 28 (March, 2008). These changes were implemented to further limit differential temperature between the reactor coolant loop and the pressurizer and reduce the occurrence of pressurizer insurge / outsurge and surge line thermal stratification.

RAI 4.7.3.3-01

Background:

Section 4.7.3.3, LBB – Reactor Coolant Loop Branch Piping, pages 4-39, states that the fatigue growth evaluation for the 8-inch residual heat removal (RHR) lines and the 12-inch safety injection (SI) accumulator lines show that only a limited number of RHR initiation transients could be tolerated. The applicant states further that growth of a postulated crack would remain well within critical crack size limits for a period of 10 years.

Issue:

It appears that the above LBB lines cannot tolerate transients other than a limited number of RHR initiation transients. Also, it is not clear how many years before the postulated crack would reach to half of the critical crack size in order to satisfy the margin of 2 which is recommended in SPR 3.6.3 "Leak-Before-Break Evaluation Procedures" of the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800). The staff reviewed SIR-00-045, Revision 1, which contains the LBB analysis of the subject branch lines. It is not evident that SIR-00-045, Revision 1, provides detailed information regarding the crack growth analysis that is discussed on page 4-39 of the LRA. It seems that the crack growth analysis of the subject branch lines discussed on page 4-39 is contained in SIR-00-045, Revision 2, which has not been submitted to the NRC and is identified as Reference 4.8-23.

Request:

- *Clarify why the above LBB piping can tolerated only a limited number of RHR initiation transients, but the analysis still concludes that growth of a postulated crack would remain well within the critical crack size as stated on pages 4-39 of the LRA. It seems that if the postulated crack size remains well within the critical crack size, the subject piping should be able to tolerate all RHR initiation transients.*
- *Given that all RHR initiation transients were not used in the crack growth calculation, discuss whether the subject piping is outside of the design basis.*
- *Discuss the number of years for the postulated fatigue flaw to reach half of critical crack size or the allowable flaw size, whichever is applicable, in the 8-inch RHR lines and 12-inch safety injection (SI) accumulator lines. This is to determine whether the 10-year inspection frequency is adequate to monitor the potential fatigue crack growth.*
- *Submit report SIR-00-045, Revision 2, for staff review because this report is referenced on pages 4-38 and is related to the crack growth evaluation of the 8-inch and 12-inch branch lines.*

DEK Response

The growth of postulated surface cracks by fatigue was evaluated in the reactor coolant branch lines leak-before-break (LBB) analysis, SIR-00-045, "*Leak-Before-Break Evaluation 6-inch to 12-inch Safety Injection and Residual Heat Removal Piping Attached to the RCS (Kewaunee Nuclear Power Plant),*" consistent with the guidance provided in NUREG-1061, "*Report of the U.S. Nuclear Regulatory Commission Piping Review Committee,*" Vol. 3, Section 5.6, Crack Growth Analysis.

Since Kewaunee branch line piping was designed to the requirements of ANSI B31.1-1967 code requirements (Code of record), no specific fatigue evaluation or transient definitions exist in the design basis for this piping. Transient information specific to the LBB analysis was developed to perform the crack growth evaluation. The transients used in the crack growth evaluation consist of those specified for the ASME Class 1 vessel analyses (described in LRA Section 4.3.1.1, Component Design Transient Cycles) and three additional thermal cycles specific to the operational conditions for this piping: residual heat removal (RHR) operation, refueling flood-up, and high head safety injection initiation. The RHR operation thermal cycle was assumed to occur coincident with each heat-up / cooldown cycle.

For the 12-inch safety injection (SI) accumulator line, when initial flaw sizes meeting ASME Code Section XI acceptance standards are postulated (~11% through-wall), the crack growth evaluation concluded that the Code-allowable flaw size limit could be reached after 38 heat-up / cooldown cycles at the worst-case location. For the 8-inch RHR line, the crack growth evaluation concluded that the Code-allowable flaw size limit could be reached after 123 heat-up / cooldown cycles at the worst-case location. These total allowable cycle occurrences are less than the design number of heat-up / cooldown cycles specified for the 40-year life of the plant. Therefore, the LBB analysis for the reactor coolant loop branch lines concluded that a postulated 11% through-wall flaw could potentially grow to greater than the Code-allowable flaw size within a 40-year period. However, these heat-up / cooldown cycle occurrences are greater than the number expected for a ten year period (13 for the ten years preceding the development of the LBB analysis). Therefore, the analysis concluded that the postulated flaw would not exceed the Code-allowable flaw size limit within a ten year period and that the ASME Code Section XI Inservice Inspection Program ten year inspection intervals would effectively manage the potential for flaw growth.

As noted above, the design code for the SI and RHR piping is ANSI B31.1-1967, which does not require an evaluation for crack growth or an explicit fatigue evaluation. The SI and RHR piping remains within its design basis since all of the design requirements continue to be met in accordance with the Code of record. The LBB analysis limitation with regard to heat-up / cooldown cycles results from the conservative fatigue crack growth evaluation which postulates a pre-existing large flaw. There have been no reportable indications (no flaws that met the ASME Section XI evaluation criteria) identified during the inspections performed to date.

Based on the assessment included in the LBB analysis for this piping, the time required for a postulated large flaw to reach the Code-allowable flaw size would be approximately 30 years (38 allowable heat-up / cooldown cycles ÷ 13 cycles per ten years).

Revision 1 of Calculation SIR-00-045 (reviewed by NRC and approved in letter dated September 5, 2002) included the fatigue crack growth evaluation of the 8-inch RHR and 12-inch SI piping in Section 6.0, "Evaluation of Fatigue Crack Growth of Surface Flaws." Revision 2 of this calculation includes the NRC safety evaluation and responses to NRC questions as attachments, and documents that the Kewaunee power uprate was evaluated and that the conclusions of the LBB analysis are not affected. Calculation SIR-00-045, Rev. 2, is included as Enclosure D to this letter.

RAI 4.7.3.3-02

Background:

Section 4.7.3.3, LBB – Reactor Coolant Loop Branch Piping, page 4-39, states that "...Since the time-based input for the crack growth analysis for these lines is less than 40 years, the crack growth analysis associated with these branch lines does not constitute a TLAA per 10CFR 54.3(a)(3)..."

Issue:

In general, a crack growth analysis assumes an initial flaw size. The flaw is assumed to grow based on a certain growth rate for 40 years (or for X number of years) to determine whether the final flaw size will be within the allowable flaw size. In Section 4.7.3.3, the crack growth is based on the fatigue mechanism. For the fatigue mechanism, transient cycles for 40 years should be used in combination with the fatigue crack growth rate to derive the final flaw size. Therefore it is not clear why a crack growth analysis uses time-based input that is less than 40 years unless the postulated flaw would grow to exceed the allowable flaw size.

10 CFR 54.3(a)(3) states that TLAA is applicable if it "...Involve[s] time-limited assumptions defined by the current operating term for example 40 years...". The applicant contends that because the crack growth analysis of the subject 8-inch and 12-inch piping did not use time – based input for 40 years; therefore, the crack growth analysis would not be considered as a TLAA. The staff believes that time-limited assumptions, not time-based input (that is less than 40 years), should be the criterion to satisfy the condition that the crack growth analysis is not a TLAA. It seems that the original crack growth analysis used transient cycles less than 40 years so that the final flaw size would satisfy the allowable flaw size. The less-than-40 year transient cycles are not an assumption but an input to the analysis. Therefore, the staff is not clear as to the technical basis to support the conclusion that the crack growth analysis of the subject lines is not a TLAA.

Request:

- *In light of the above, clarify what is the time-based input that is less than 40 years and why input that is less than 40 years is used in the crack growth analysis.*
- *Clarify how 10 CFR 54.3(a)(3) is applicable to the crack growth analysis of the 8-inch RHR lines and the 12-inch safety injection accumulator lines.*

DEK Response

The purpose of the review of time-limited aging analyses (TLAA) in the license renewal application is to ensure that the results of plant-specific analyses that are based on an explicitly assumed 40-year plant life remain valid for the additional 20 years of plant operation to be authorized by the renewed license. The fatigue crack growth evaluation for the 12-inch safety injection (SI) and 8-inch residual heat removal (RHR) lines was performed as part of the reactor coolant branch lines LBB analysis. Based on the ASME Code Section XI Inservice Inspection (ISI) program inspections performed during the ten year interval, only a ten year time period was ultimately considered for the fatigue crack growth evaluation of these lines. Since the fatigue crack growth evaluation considered a time period less than the current operating term (i.e., 40 years), it was initially concluded that the evaluation did not meet the criteria in 10 CFR 54.3(a)(3) and therefore, was not a TLAA.

However, based on NRC concerns, the license renewal application is amended to include the fatigue crack growth evaluation for the 12-inch SI and 8-inch RHR lines as a TLAA. As described in LRA Section 4.7.3.3, the 12-inch SI and 8-inch RHR lines are inspected in accordance with the ASME Section XI ISI program (described in LRA Appendix B, Section B2.1.2, ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD).

Therefore, fatigue crack growth is managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program and the LBB analysis crack growth evaluation TLAA for the 12-inch SI and 8-inch RHR lines is acceptable for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

RAI 4.7.3.3-03

<u>Background:</u>

Section 4.7.3.3, LBB - Reactor Coolant Loop Branch Piping, page 4-39, states that "...The fatigue crack growth conclusions are not affected by the extended plant service life since the original design basis transient have been shown to be bounding for the period of extended operation in Section 4.3.1.1, Component Design Transient Cycles...".

<u>Issue:</u>

It is not clear how the applicant can assure that the actual operating transients at the end of 60 years are bounded by the design transient cycles.

<u>Request:</u>

Discuss how the actual transient cycles can be monitored to verify that the design transient cycles used in the LBB evaluations bound the actual operating transients. This question applies to all LBB piping and associated LBB evaluations.

DEK Response

As indicated in LRA Sections 4.7.3.1, 4.7.3.2, and 4.7.3.3, the leak-before-break (LBB) analyses fatigue crack growth evaluations were based on the design basis operational transients for the nuclear steam supply system (NSSS). The branch line LBB analysis also included additional transients defined specifically for these lines. The design basis operational transients have been shown to be bounding for the period of extended operation in LRA Section 4.3.1.1, Component Design Transient Cycles.

LRA Section 4.3.1.1 provides the approach to monitoring design basis operational transients using the Metal Fatigue of Reactor Coolant Pressure Boundary Program. As described in LRA Appendix B3.2, the program monitors actual transients and components. Specifically, the design basis operational transients are tracked and the number of occurrences is evaluated against the design basis to assure that actual plant operation remains bounded by the assumptions used in the design analyses.

RAI 4.7.3.3-04

Background:

Section 4.7.3.3, LBB – Reactor Coolant Loop Branch Piping, page 4-39, states that "...for the 8-inch RHR lines and the 12-inch SI accumulator lines...It was further concluded that the American Society of Mechanical Engineers Boiler & Pressure Vessel code, Section XI – required inspections every 10 years would effectively manage cracking in this piping such that a crack greater in size than that postulated would not be present at the start of the ten year interval..." The applicant stated further that the crack growth analysis associated with the 8-inch RHR line and 12-inchs SI accumulator lines does not constitute a TLAA.

Issue:

It appears that the 10-year inservice inspection is a part of the applicant's technical basis for not considering the crack growth analysis as a TLAA. However, it is not clear how these two piping systems will be inspected during the license renewal period to ensure their structural integrity to the end of 60 years.

Request:

- *Identify exactly the number of 8-inch RHR lines and 12-inch safety injection accumulator lines that will be inspected during the period of extended operation.*
- *Specify the total number of the welds in each of the RHR and safety injection accumulator lines that will be inspected in each 10-year inspection interval during the period of extended operation.*
- *Specify the total number of welds that are in the subject lines. This information is used to determine the percentage of the welds that will be inspected.*
- *Discuss which nondestructive examination method will be used (e.g., penetrant testing, ultrasonic testing).*
- *Discuss the criteria for weld selection for examination (e.g., high stress locations, or fatigue crack growth calculation of the subject piping).*
- *If the above information has been submitted to the NRC, provide the references.*

DEK Response

It should be noted that in response to RAI 4.7.3.3-02, DEK is amending the license renewal application to include an evaluation of the fatigue crack growth evaluation for the 12-inch safety injection (SI) and 8-inch residual heat removal (RHR) lines, performed as part of the reactor coolant branch lines LBB analysis, as a TLAA.

There are two 12-inch SI lines and two 8-inch RHR lines that were evaluated for fatigue crack growth in the LBB analysis and that are subject to inspection in accordance with the ASME Code Section XI Inservice Inspection (ISI) Program. There are a total of six welds in the two 12-inch SI lines and a total of 17 welds in the two 8-inch RHR lines. As indicated in LRA Section 4.7.3.3, LBB – Reactor Coolant Loop Branch Piping, the inspections of these lines are performed in accordance with the ASME Code Section XI ISI Program. Weld selection for inspection is performed in accordance with ASME Code Section XI requirements for sample size and selection and does not depend upon calculated stress levels or fatigue crack growth evaluation results. The examination method is currently ultrasonic testing (UT).

Kewaunee is currently in the fourth ASME Code Section XI inspection interval. The ISI Program has been updated for the fourth Inspection interval and was submitted for NRC review by letter dated December 16, 2003 (ADAMS ML033580734). Specific details of the planned inspections for these lines in the fourth interval are included in the ISI Program document.

Performance of ASME Code Section XI-required inspections is a current licensing basis commitment that will continue to apply through the period of extended operation, thus there is no new commitment established herein specifically for license renewal.

Each of the welds in these lines was ultrasonically examined in accordance with ASME Code Section XI preservice inspection requirements. Additionally, each of these weld locations, with the exception of one inaccessible weld in an 8-inch RHR line, have either been inspected in the first three intervals or are scheduled for inspection in the fourth inspection interval. Therefore, each accessible weld location will have been inspected twice ensuring a low probability of a pre-existing fabrication flaw. There were no reportable indications (no flaws that met the ASME Code Section XI evaluation criteria) identified during the inspections that have been performed.

Since the fifth and sixth ten-year ISI program plans are to be developed and approved in the future, it is not currently known which RHR or SI accumulator pipe welds will be inspected during the period of extended operation. However, the ISI program is in place and will continue to remain in place during the period of extended operation.

RAI 2.3-01(a)

Background:

In RAI 2.3-01, LRA Section 2.3.4.2, dated April 03, 2009, the staff noted that the license renewal drawing LRM-211 locations B-6 and B-7 show continuations of 10 CFR 54.4(a)(2) pipe sections from cylinder heating steam supply system without drawing numbers or grid locations. The applicant was requested to submit additional information to identify the license renewal boundaries.

Issue:

In its response, dated April 27, 2009, the applicant stated that the drain lines shown on drawing LRM-211, locations B-6 and B-7, continue on drawing LRXK-101-17A at locations C-3 and C-7. Based on its review, the staff finds the applicant's response to this RAI 2.3.4.2-02 acceptable because the staff located the continuations. However, staff noted on LRXK-101-17A, location C-3, a piping section continued to the GLD STM LEAKOFF TO GLAND CONDENSER that was not included in-scope. Similar piping at location C-7 was included in-scope for 10 CFR 54.4 (a)(2).

Request:

The applicant is requested to provide a basis for not including the piping continuing to the GLD STM LEAKOFF TO GLAND CONDENSER in the scope of license renewal for 10 CFR 54.4(a)(2).

DEK Response

The GLD STM LEAKOFF TO GLAND CONDENSER piping on LRXK-101-17A at location C-3 should have been highlighted within the scope of license renewal for 10 CFR 54.4(a)(2) and is included in the component type "Pipe" in LRA Table 2.3.4-2.

RAI 2.3.3:26-01(a)

Background:

Follow-up RAI 2: In RAI 2.3.3.26-01, dated April 3, 2009, the staff noted drawing LRM-350, locations D-5, D-6, and D-10, show non-safety related piping connected to safety-related piping components at valves MD(R)-250A&B, MD(R)-251A&B, MD(R)-260, MD(R)-261, MD(R)-270, MD(R)-271, MD(R)-272, MD(R)-273 and MD(R)-262. The applicant was requested to provide the location of the seismic restraint for the non safety-related 1" lines connected to the safety-related heat exchangers #1A and #1B, the letdown exchanger and seal water heat exchanger piping.

Issue:

In its response dated April 27, 2009, the applicant clarified that the bounding scoping methodology was applied wherein the sludge interceptor tank was used as an equivalent anchor. Based on its review, the staff finds the applicant's response to RAI 2.3.3.26-01 acceptable for the piping to the sludge interceptor tank, however the response was incomplete because it did not identify the seismic anchor for the branch piping continued to the waste area sump pumps. Therefore, the staff's concern described in RAI 2.3.3.26-01 is not resolved.

Request:

The applicant is requested to provide the location of the seismic anchor for the non safety-related for the branch piping continued to the waste area sump pumps.

DEK Response

There is no identified structural anchor associated with the branch piping to the waste area sump pumps. Therefore, the bounding methodology was applied as described in LRA Section 2.1.3.6.2. As such, the waste area sump pumps and the associated discharge piping below the sump cover plate should have been included in scope for 10 CFR 54.4(a)(2) and highlighted on license renewal drawing LRM-350, at location G-10. The waste area sump pumps are constructed of the same material and exposed to the same environment as the screen house sump pumps. The aging management review results for the waste area sump pumps are identical to the results for the screen house sump pumps that are shown in LRA Table 3.3.2-26: Auxiliary Systems – Miscellaneous Drains and Sumps-Aging Management Evaluations. The carbon steel pump discharge piping aging management review results are the same as those presented for the component group "Pipe" in LRA Table 3.3.2-26.

RAI 3.3.2.2.6-1

Background/Issue:

Boron carbide plates have been used at KPS for over twenty five years with no evidence of bulging, reduction in neutron absorption or loss of material; however this justification is not sufficient in stating that there will not be any bulging, reduction in neutron absorption of loss of material in the period of license extension. The staff requires additional information in order to determine if aging management would be required:

Request:

- *Please provide the operating experience of the boron carbide plates at KPS, including the following:*
  - *What was the location of coupons relative to the spent fuel racks? What was the neutron flux of the coupons relative to that for the rods?*
  - *How were the coupons mounted and were they fully exposed to the spent fuel pool water (both sides exposed or bolted to a wall)?*
  - *What specific testing procedures were used for determining areal density, verifying surface corrosion (if any) and examining for blister formation?*
  - *To demonstrate that the boron carbide plate integrity has been maintained, the staff requests the applicant to provide the test results for the coupons, including areal density measurements.*
  - *What are the acceptance criteria for these results?*
  - *After removal from the pool for inspection were the coupons inserted back at the same locations in the pool?*
  - *What was the subcritical margin used in the criticality analysis? In order to prevent excess degradation, the potential degradation should be accounted for in the subcritical margin. How is the potential degradation during the time in between surveillance periods accounted for in the subcritical margin?*
  - *Please describe the corrective actions that would be implemented if coupon test results are not acceptable.*
- *Please describe how the neutron-absorbing capacity will be monitored in the period of extended operation. Please include a description of the parameters, calculations, and acceptance criteria. If coupon testing will still be used:*
  - *Discuss the schedule for coupon removal and testing during the period of extended operation to demonstrate continued boron carbide performance.*
- o *Please discuss any other industry operating experience of boron carbide, and discuss how that experience is applicable to KPS and any potential safety concerns identified in the boron carbide operating experience.*

DEK Response

- The boron carbide (B_4C) neutron absorber in the Kewaunee spent fuel pool storage racks was manufactured by Electroschmelzwerk Kempten (ESK) and the racks were initially installed in 1979. In mid–December, 1980, personnel at Kewaunee determined that a number of spent fuel cells, and two test specimens had experienced some amount of wall deflection. Analysis indicated the deflection was the result of B_4C off-gassing. The racks were modified by drilling vent holes in the top portion of the cells. Additional racks installed in the mid-1980's were designed with a vent path and did not experience the deflection problem. There have been no instances of wall deflection identified since these modifications were performed. In addition, test coupon blackness testing has been performed since 1982 with no detected degradation of the neutron absorber.
  - The B_4C test coupons are located vertically above the location of the installed neutron absorber coupons in an empty cell adjacent to a freshly discharged fuel storage location. The orientation of the coupon is on a diagonal across the empty cell. Although the numeric value of the neutron flux test coupons to spent fuel rod ratio has not been calculated, the test coupons are located in a position that is expected to have the highest neutron flux relative to the spent fuel rods.
  - The coupons are mounted in a non-watertight stainless steel enclosure such that both sides of the coupon are fully exposed to the spent fuel pool water. The design of the coupon holder is such that it hangs from the top of the spent fuel rack.
  - The boron-10 (B-10) areal density is calculated based on blackness testing results. The test procedure compares the test rig control plate data of different thicknesses to test coupons from the spent fuel pool to calculate the areal density of each test coupon. In addition, the surface condition of the B_4C test coupons is visually examined for signs of corrosion or other degradation. If blistering did occur, it would be detected and documented as a part of the visual inspection. Based on industry experience, blistering is not expected since it is typically associated with Boral neutron absorber, not B_4C.

- The following table provides the areal density test results since 1982 for each test coupon (gm/cm^2 B-10 loading):

Date	#79-1	#79-2	#79-3	#81-1	#81-2	#88-1	#88-2	Source
3/82	0.114	0.107	0.107	0.101	0.095	*	*	AmBe 3.38
8/82	0.118	0.118	0.118	0.103	0.118	*	*	AmBe 3.2
3/83	0.117	0.117	0.112	0.128	0.134	*	*	AmBe
9/83	0.117	0.134	0.125	0.117	0.134	*	*	AmBe 3.4
6/84	0.136	0.126	0.126	0.130	0.131	*	*	AmBe 3.4
1/85	0.104	0.100	0.108	0.110	0.112	*	*	AmBe 3.4
8/85	0.100	0.109	0.131	0.138	0.125	*	*	AmBe 3.4
2/86	0.100	0.100	0.105	0.108	0.103	*	*	AmBe 3.4
9/86	0.110	0.980	0.110	0.092	0.110	*	*	AmBe 3.4
1/87	0.119	0.107	0.094	0.105	0.119	*	*	AmBe 3.4
1/87	0.097	0.100	0.095	0.094	0.090	*	*	PuBe
2/88	0.099	0.102	0.093	0.097	0.104	*	*	AmBe 3.4
10/88	0.096	0.103	0.095	0.095	0.095	0.102	0.093	AmBe 3.4
10/89	0.092	0.100	0.094	0.098	0.097	0.097	0.102	AmBe 3.4
7/90	0.098	0.088	0.089	0.094	0.088	0.094	0.098	AmBe 3.4
10/91	0.097	0.101	0.094	0.094	0.101	0.098	0.098	PuBe 4.6
9/92	0.115	0.124	0.123	0.116	0.123	0.117	0.121	PuBe 4.6
10/93	0.104	0.108	0.101	0.104	0.107	0.102	0.102	PuBe 4.6
10/94	0.111	0.104	0.101	0.104	0.106	0.108	0.113	PuBe 4.6
11/95	0.099	0.098	0.095	0.100	0.096	0.107	0.097	PuBe 4.15
7/99	0.100	0.106	0.104	0.103	0.106	0.102	0.102	PuBe 4.6
6/02	0.093	0.091	0.096	0.094	0.091	0.097	0.098	PuBe 3.4
2/05	0.106	0.107	0.106	0.108	0.108	0.103	0.105	PuBe 4.6
8/08	0.089	0.087	0.094	0.102	0.090	0.097	0.094	PuBe 4.59

* These test coupons were installed when additional racks were installed at a later date.

- o The test procedure acceptance criterion for the minimum B-10 loading is 0.086 gm/cm^2.
- o The B_4C test coupons are generally not returned to the same location in the spent fuel pool. Following testing and inspection, the coupons are placed in locations that are near freshly discharged fuel to ensure continued exposure to the highest pool temperature, gamma radiation, and neutron flux.
- o The subcritical margin determined by the spent fuel pool criticality analysis is 7.996%. In order to address potential degradation of the neutron absorber between surveillance periods, which is typically a slow process, the subcritical margin calculation includes the following conservative assumptions:
  - A minimum boron-10 loading of 0.0863 gm/cm^2 in the neutron absorber plate (the nominal boron-10 loading in the neutron absorber plates is 0.0959 (+0.00 / -0.0096) gm/cm^2).
  - No soluble boron in the spent fuel pool (the spent fuel pool soluble boron concentration is procedurally controlled at ≥ 2500 ppm).
  - Minimum thickness and widths of the stainless steel storage rack structure.
  - Minimum borated plate thickness.

  Based on plant specific operating experience, there has been no measured degradation of the B_4C neutron absorption capability. In the event that degradation does occur, there is adequate subcritical margin to ensure the Kewaunee technical specification limit of $K_{eff} \leq 0.95$ is met during the time between surveillance periods.
- o If acceptance criteria are not met, the surveillance procedure requires that the condition is documented in the Corrective Action Program. The condition will be evaluated and the cause and appropriate corrective actions determined. In addition, the following actions are required by the surveillance procedure:
  - Reactor Engineering will review the spent fuel rack loading pattern.
  - One or more of the 1 inch x 4 inch x 0.24 inch samples will be sent to an independent lab and analyzed for B-10 areal density.
  - Test results will be evaluated to determine the acceptability of continued unrestricted use of the spent fuel racks.

- Blackness testing, to demonstrate B_4C neutron absorption capability, is planned to continue during the period of extended operation. Neutron absorbing capacity is monitored by placing a neutron source, with a moderating material, on one side of a test plate and a detector with a counter on the other. Five control plates of varying thicknesses, and proportional boron-10 loading, are tested to determine a linear function to fit the results from the test plates. The number of counts measured for each of the test plates is substituted into the linear function to determine boron-10 loading. Physical characteristics of each test plate such as dimensions, thickness, surface finish, cracking, and pitting are also noted. If any physical characteristic abnormalities are noted or the boron-10 loading decreases below 0.086 gm/cm^2, evaluations of the abnormalities are performed at time of discovery. This coupon testing is performed every three years.

- Based on neutron absorber blackness (Badger) testing performed in 2009, Palisades has identified degradation of their B_4C neutron absorber plates, which were manufactured by Carborundum. Palisades determined that the loss of B_4C was related to aging of the base material (phenolic resin), which was caused by three possible effects: 1) residual polymers from manufacturing, 2) gamma radiation exposure, and 3) water ingress. A combination of the environment, material, and exposure resulted in hydrogen removal from the phenolic resin which allowed pool water to cause a chemical reaction and the B_4C to go into solution. Based on the Palisades OE, Kewaunee inspected a spent fuel storage rack that was removed from the pool for maintenance. The conditions noted at Palisades were not found.

  Columbia Generating Station also has spent fuel storage racks with B_4C that are fabricated by ESK. Columbia has accumulated more than 20 years of blackness testing results indicating no degradation of the surface condition or B-10 areal density.

  In summary, conditions similar to those found at Palisades are not expected to occur with the Kewaunee spent fuel storage racks, since:

  - Both Kewaunee and Columbia have performed visual inspections and blackness testing of test coupons for greater than 20 years and have not observed any degradation on the surface of the test coupons or the ability of the test coupons to attenuate neutrons.

  - The B_4C neutron absorbers used in the spent fuel storage racks at Kewaunee and Columbia were manufactured by ESK, while the Palisades B_4C neutron absorbers were manufactured by Carborundum, and there is no industry OE to indicate that B_4C manufactured by ESK has degraded.

  Events related to neutron absorber degradation at Palisades and other stations continue to be monitored and evaluated on an ongoing basis in accordance with the Operating Experience Program.

RAI 3.3.2.2.6-2

Background/Issue:

The Generic Aging Lessons Learned (GALL) report recognizes the possibility of the existence of aging effects in the Boral used in the spent fuel storage racks and the need for having a plant specific aging management program. However, the applicant has indicated in the KPS submittal that degradation of the Boral is insignificant and no aging management program is required. The applicant provided several justifications for not having management program. In order to determine if aging management is required the staff requires the following additional information:

Request:

- *It is unclear to the staff which Holtec and industry testing is referenced in section 3.3.2.2.5 of the submittal. Please describe the testing performed and how that relates to a Boral period of performance of over thirty years. Also please provide a copy of the reference/report.*
- *The staff requests the applicant to provide the installation date of the Boral currently in the spent fuel pool. In addition, to demonstrate that Boral integrity has been maintained, the staff requests the applicant to provide the test results for the coupons, including areal density measurements.*
- *Please provide the following specifications of the Boral panels in the spent fuel pool racks:*
  - *Geometry of the Boral panels*
  - *Areal density of boron*
- *Please describe how the neutron-absorbing capacity will be monitored. Please include a description of the parameters, calculations, and acceptance criteria.*

DEK Response

- The informational sources referenced in LRA Section 3.3.2.2.6 are as follows. [Note that LRA Section 3.3.2.2.5 is incorrectly referenced in RAI 3.3.2.2.6-2.]

  1. The results of accelerated exposure testing and summaries of industry operating experience documented in EPRI Report 1013721, *Handbook on Neutron Absorber Material for Spent Nuclear Fuel Applications.*
  2. Operating experience provided through discussions with cognizant personnel from other nuclear stations.
  3. Information provided by the Boral vendor (Holtec) technical representative.

The accelerated exposure tests described in EPRI Report 1013721 were conducted at the University of Michigan 2MW$_{th}$ Ford Nuclear Reactor. The tests ran for nine years and periodically three samples were removed for inspection and analysis, including:

- Visual inspection
- Neutron radiography
- Neutron attenuation
- Tensile properties
- Chemical analysis for Boron-10

The test samples were exposed to 7 x 10^{11} rads of gamma radiation. The samples were also exposed to fast and thermal neutrons. Other than localized oxidation, the test samples showed no signs of physical deterioration. Neutron attenuation testing and neutron radiography showed no loss of boron carbide as confirmed by chemical analysis. It was also concluded that the test conditions were far more severe than conditions in spent fuel storage applications. Therefore, the test results are considered to be enveloping for the Kewaunee spent fuel racks. Note: Per a telephone discussion between members of the DEK staff and NRC staff on July 30, 2009, it was confirmed that EPRI Report 1013721 is available to the NRC staff at NRC headquarters.

Discussions with Holtec representatives and with cognizant engineering personnel at other stations utilizing Boral and implementing a Boral coupon surveillance program support EPRI conclusions related to Boral (i.e., the areal density of B-10 has not degraded, general corrosion of Boral does not occur in spent fuel pools containing boric acid concentrations of 2500 ppm, and the occurrence of localized corrosion is primarily an aesthetic effect that does not affect the neutron attenuation or the structural integrity of the racks).

The thirty years of service referenced in LRA Section 3.3.2.2.6 is related to the spent fuel pool racks containing boron carbide plates (B_4C) not the spent fuel racks containing Boral. The spent fuel racks containing Boral were placed in service in September 2001.

- The Boral spent fuel storage racks were placed in service in September 2001. Kewaunee has no Boral surveillance program and no in-service areal density measurements have been obtained.

When the Kewaunee spent fuel storage racks were installed and licensed, the industry considered Boral to be stable and chemically inert. This was based on tests simulating the radiation exposure to the storage racks and the thermal and chemical environment of the spent fuel pools.

Boral spent fuel storage racks have been installed at a large number of nuclear stations and have been in service for several years. Although there has been industry experience with blistering and bulging, there have been no instances where the structural integrity or the neutron attenuation capability of the Boral panels has been adversely affected in spent fuel storage racks similar in design to the Kewaunee racks. In the spent fuel racks that utilize flux trap design, water between the fuel storage rack cells thermalizes neutrons, enhancing the neutron absorber effect. In the event that blistering occurs in this design, there is a potential for a reduction in thermalization of neutrons. The Kewaunee spent fuel racks that contain Boral are a non-flux trap design and, therefore, are not subject to this concern. In addition, the Kewaunee Boral spent fuel racks are currently limited by the licensing basis to storage of fuel that was removed during or before the 1984 refueling outage. As a result, stored spent fuel in these racks has decayed for greater than 24 years resulting in a relatively low gamma radiation exposure and cooler spent fuel temperatures.

- The Boral panels have a thickness of 0.075 inch, a width of 5 inches and a length of 146 inches. The panels are held in place and protected against damage by 0.035 inch stainless steel sheathing. The Boral panels have been sized to fully shadow the active fuel height of all spent fuel assembly designs stored in the fuel transfer canal storage racks. The nominal B-10 areal density is 0.0216 g/cm^2.

- Based on the industry operating experience at the time the Boral spent fuel racks were installed, a monitoring program was not established as a part of the licensing basis. Based on current industry operating experience at Beaver Valley, Humboldt Bay, and Seabrook, there have been no adverse effects to the structural integrity of the spent fuel racks or ability of the Boral in the spent fuel racks to perform its neutron-absorbing function. Therefore, it was concluded that a monitoring program is not needed. Kewaunee will continue to monitor industry operating experience related to Boral through the Operating Experience Program and any necessary actions will be initiated through the Corrective Action Program.

RAI 3.3.2.2.6-3

<u>Request:</u>

- *In the submittal, it is unclear whether a surveillance program will still be used in the period of extended operation. Please confirm the existence of a surveillance program in period of extended operation. If a surveillance program will be in use please address the following:*
    - *Please confirm that KPS has sufficient Boral coupon samples to maintain the sampling frequency through the period of extended operation.*
    - *Please provide a detailed description of the Boral coupons and the tests performed on them during their examination:*
        - *What was the location of coupons relative to the spent fuel racks?*
        - *How were the coupons mounted and were they fully exposed to the spent fuel pool water?*
        - *What specific testing procedures were used for determining Boral-10 areal density, verifying surface corrosion (if any) and examining for blister formation?*
        - *After removal from the pool for inspection were the coupons inserted back at the same locations in the pool?*
    - *What are the acceptance criteria for these results?*
    - *Please discuss the correlation between measurements of the physical properties of Boral coupons and the integrity of the Boral panels in the storage racks.*
    - *What was the subcritical margin used in the criticality analysis? In order to prevent excess degradation, the potential degradation should be accounted for in the subcritical margin. How is the potential degradation during the time in between surveillance periods accounted for in the subcritical margin?*
    - *Please describe the corrective actions that will be implemented if coupon test results are not acceptable.*
    - *Discuss the schedule for coupon removal and testing during period of extended operation to demonstrate continued Boral performance.*
- *In September 2003, inspection of Boral test coupons at Seabrook Nuclear Station revealed bulging and blistering of the aluminum cladding. Please discuss the impact, if any, that this event is considered to have on the Boral surveillance program at KPS. Industry experience has indicated that during long-term exposure such blisters may form. Since formation of blisters may affect the efficiency of the Boral panels to attenuate neutrons (through flux trap formation) and may cause deformation of the fuel cells, the applicant should explain why in its plant blistering will not be a safety concern.*
- *Please discuss any other industry operating experience of Boral, and discuss how that experience is applicable to KPS and any potential safety concerns identified in the boron carbide operating experience.*

DEK Response

- As described in the response to RAI 3.3.2.2.6-2, a monitoring program was determined not to be required for the Boral spent fuel storage racks. There are no current plans to implement a monitoring program for these racks. However, industry OE will continue to be evaluated on an ongoing basis through the period of extended operation in accordance with the Operating Experience Program and any actions determined to be necessary to ensure that the intended functions of the Boral spent fuel storage racks are maintained will be evaluated through the Corrective Action Program.

  The subcritical margin determined by the spent fuel pool criticality analysis is 5.58%. The subcritical margin calculation includes the following conservative assumptions:

  - A minimum boron-10 loading of 0.020 gm/cm^2 in the neutron absorber plate (the actual nominal boron-10 loading in the neutron absorber plates is 0.0216 gm/cm^2).
  - No soluble boron in the spent fuel pool (the spent fuel pool soluble boron concentration is procedurally controlled at ≥ 2500 ppm).
  - 40° F in the spent fuel pool (the pool is typically >60° F).

- The Seabrook operating experience report and 10 CFR Part 21 notification concerning bulging and blistering of a Boral test coupon have been reviewed for impact on Kewaunee. Although Seabrook identified bulging and blistering in September, 2003 via inspection of their test coupon, the evaluation of this condition concluded that the acceptance criteria for B-10 areal density were met and that there was no impact on the structural integrity of the racks. Based on these conclusions, no safety concerns have been identified related to the Kewaunee Boral spent fuel storage racks.

- Additional operating experience related to bulging and blistering of the Boral aluminum cladding identified at Beaver Valley and Humboldt Bay was also reviewed. Each of these stations also concluded that blistering did not affect neutron attenuation or the structural integrity of the spent fuel storage racks.

  Therefore, since there has been no industry operating experience that identifies that observed blistering has had an adverse affects on neutron absorber performance or storage rack structural integrity, no safety concerns have been identified.