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Dominion®

June 28, 2012

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

Serial No. 12-295
LIC/JG/R0
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST
SUPPLEMENT AND RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By application dated December 12, 2011 (Reference 1), Dominion Energy Kewaunee, Inc. (DEK), requested approval, pursuant to the provisions of Renewed Operating License DPR-43, of the inspection plan for reactor vessel internal (RVI) components at Kewaunee Power Station (KPS). This inspection plan submittal was to fulfill certain requirements of Renewed Operating License DPR-43, Section 2.C(15)(b); specifically, Commitment Items 1 and 2 of Appendix A of NUREG-1958, "Safety Evaluation Report Related to the Kewaunee Power Station," dated January 2011.

Subsequently, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the inspection plan (References 2, 3, and 4). The NRC questions were discussed with NRC staff to obtain clarification during a telephone conference on April 19, 2012. The DEK responses to the NRC questions are provided in Attachment 1 to this letter. Attachment 2 provides revised RVI components inspection plan tables.

If you have questions or require additional information, please feel free to contact Mr. Jack Gadzala at 920-388-8604.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

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NRR

Attachments:

1. Response to Request for Additional Information, KPS RVI Inspection Plan
2. KPS RVI Inspection Plan, Tables 1 and 2 (Revision 1)

References:

1. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request," dated December 12, 2011.
2. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 - Kewaunee - Review of RVI Inspection Plan RAI Set 1 of 2 (Rev 1)," dated March 28, 2012.
3. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 (not 7277) - Kewaunee - Review of RVI Inspection Plan - (1) RAI Set 2 of 2, adding to RAI Set 1 of 2 (Rev 1); and (2) request for clarification conference call," dated April 9, 2012.
4. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 Kewaunee Request for Additional Information (RAI) ME7727-RAII-EVIB-Cher-014-2012-04-27," dated April 11, 2012.

Commitments made by this letter:

1. If evaluations of the CASS materials (discussed in the response to NRC Question ME7727-RAII-EVIB-Cher-012-2012-05-09) require implementation of enhanced visual testing (EVT-1), then the results of these examinations of the CASS materials will be provided to the NRC.

cc: Regional Administrator, Region III
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Kewaunee Power Station

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INSPECTION PLAN FOR THE AUGMENTED INSERVICE INSPECTION PROGRAM FOR EXAMINATION OF REACTOR VESSEL INTERNALS

On March 28, April 9, and April 11, 2012, the NRC transmitted to Dominion Energy Kewaunee, (DEK) a request for additional information (RAI) (References 1, 2, and 3) concerning the inspection plan for reactor vessel internal (RVI) components at Kewaunee Power Station (KPS). This inspection plan submittal was to fulfill certain requirements of Renewed Operating License DPR-43, Section 2.C(15)(b); specifically, Commitment Items 1 and 2 of Appendix A of NUREG-1958, "Safety Evaluation Report Related to the Kewaunee Power Station," dated January 2011.

These questions were discussed with NRC staff to obtain clarification during a telephone conference on April 19, 2012.

Each of the RAI questions are provided below, followed by the corresponding DEK response.

NRC Question ME7727-RAII-EVIB-Cher-001-2012-04-27

Applicant/Licensee Action Item 1 from the NRC staff's final SE of MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," requires that applicants/licensees submit an evaluation that demonstrates that their plant is bounded by the assumptions regarding plant design and operating history that were made in the failure modes, effects and consequences analyses (FMECA) and functionality analyses for reactors of their design.

KPS's response to Applicant/Licensee Action Item 1 in the RVI inspection plan addresses the core loading assumptions (switch to a low-leakage core) and operational (base loaded plant) aspects of design and operation that are mentioned in MRP-227-A, Section 2.4. An additional assumption listed in Section 2.4 of MRP-227-A is that there have been no design changes to the RVI beyond those identified in general industry guidance or recommended by the original vendors. Section 2.4 of MRP-227-A indicated that these assumptions are considered to represent any U.S PWR operating plant provided that these three assumptions are met, given the information on design and operation known to the MRP as of May 2007.

MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR [pressurized water reactor] Designs" (proprietary), documents the screening for susceptibility to aging effects, the FMECA results, and the categorization and ranking of the RVI components. In addition to the assumptions listed in Section 2.4 of MRP-227-A, MRP-191 documents additional assumptions that were used. In particular, neutron

fluence range, temperature, and material grade for each generic component of the Westinghouse design internals were used for input to the screening process. These values were determined based on an "expert elicitation" process. Stress values were not explicitly tabulated, but were recorded as either above the stress threshold (>30 ksi) or not based on the expert interviews.

MRP-232, Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals" (proprietary), reported more specific stress, temperature and neutron fluence values based on finite element analyses for selected high consequence of failure components identified in MRP-191.

The EPRI-MRP did not verify that the values of fluence, temperature, stress, and material, documented in MRP-191 and MRP-232 were bounding for all individual plants, and in fact MRP-227 states, "These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter."

The NRC staff expects that the licensee should have access to design information enabling verification that the material for each RVI component is bounded by the design assumptions of the MRP. In this context, the NRC staff requests that the licensee provide the following information:

1) To provide reasonable assurance that the RVI components are bounded by assumptions in the FMECA and functionality analyses supporting the development of MRP-227-A, the licensee is requested to respond to either part a) or part b) of this RAI:

   a) Provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for a sample of RVI components. The components selected should represent a range of neutron fluences, and temperatures. This information should identify whether the stress is greater or less than 30 ksi. Values of neutron fluence and temperature may be estimated or analytical values. The values should be the peak values of each parameter for each component (e.g., peak end-of-life value for fluence). Provide the method used to estimate the values, or describe the analysis method. An acceptable sample of components is:

      i) Lower Core Plate

      ii) Core Barrel Flange

      iii) Barrel-Former Bolts

      iv) Upper Core Barrel Welds

      v) Lower Core Barrel Welds

      vi) Upper Core Plate Alignment Pins

b) If the sample verification approach in Part a) is not used, describe the process used to verify that the RVI components at KPS are bounded by the assumptions regarding the neutron fluence, temperature, stress values, and materials that were made for each component in the FMECA and functionality analyses supporting the development of MRP-227-A.

2) If there are any components at KPS not bounded by assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A, describe how the differences were addressed in the plant-specific RVI Inspection Plan. The NRC staff requests that the licensee, as a part of its demonstration, discuss whether there would be any changes to the screening, categorization, FMECA process and functionality analyses if the plant-specific variables (the neutron fluence, temperature, stress values, plant-specific operating experience, and materials) are used. This evaluation should address whether additional aging mechanisms would become applicable to the component.

3) For any non-bounded components, determine if any changes to the inspection requirements of MRP-227-A are needed. Provide plant-specific inspection requirements or an alternate aging management program, as appropriate. If no changes to the inspection requirements are proposed, provide a justification for the adequacy of the existing MRP-227-A inspections for the unbounded components.

4) In its e-mail submittal dated March 19, 2012, the licensee submitted the Technical Report KLR-1309A, to the NRC staff for review. On page 13 of the KLR-1309A report, the licensee stated that as a part of design change, it installed flexure-less inserts. The NRC staff requests that the licensee provide the following information with regard to this design change.

   a) Reason for installing the flexure-less inserts, (b) Type of material used, and information regarding the material selection, (c) Operating experience with respect to any degradation (observed so far) of the flexure-less inserts, and (d) If the flexure-less inserts were installed after May 2007, provide an assessment of the impact of this installation on the recommendations of the RVI Inspection Plan. Provide plant-specific inspection requirements if necessary for the affected components.

Response:

RAI Part 1)

DEK has confirmed information regarding plant specific internals components for KPS (in conjunction with Westinghouse Electric Company (Westinghouse)). The results of the favorable comparison of the KPS specific components to those components used in the generic FMECA and functionality analysis (MRP-191 Table 4-4) are documented in Westinghouse letter WPS-08-28, Revision 2, dated December 22, 2008 (LTR-ARIDA-08-63, Revision 3).

The process used to verify that the RVI components at KPS are bounded by the assumptions regarding neutron fluence, temperature, stress values, and materials in the FMECA and functionality analyses is as follows:

1. Identification of Typical Westinghouse PWR Internals Components.
2. Identification of KPS PWR Internals Components.
3. Comparison of the Typical Westinghouse PWR Internals Components to the KPS PWR Internals Components. No atypical items were identified by this comparison.
4. The materials are identified in WPS-08-28, Revision 2 (LTR-ARIDA-08-63, Revision 3, dated December 5, 2008) and are consistent with those materials identified in MRP-191 and MRP-232.
5. KPS verifies through AMP-KLR-1309, Revision 3 that no modification to the KPS reactor internals have been made over the lifetime of the plant except for those specifically directed by the OEM. Therefore, the assumptions regarding the neutron fluence, temperature, stress values, and materials used in performing the generic FMECA and functionality analysis are applicable for KPS.
6. KPS used new fuel at the peripheral locations during Fuel Cycles 1 and 2. Once-burned fuel was used at peripheral locations during Fuel Cycles 3 through 15. In Fuel Cycle 16, KPS switched to use of a low leakage core design. KPS continued to use a low leakage core design for all subsequent fuel cycles. KPS has operated under base load conditions over the life of the plant. The FMECA and functionality analyses for MRP-227 were based on the assumption of thirty years of operation with high leakage core loading patterns followed by thirty years of low leakage core loading patterns. Therefore, KPS satisfies the assumptions in MRP regarding fluence.
7. KPS has operated under base load conditions over the lifetime of the plant. Therefore, KPS satisfies the assumption in MRP documents regarding operational parameters.
8. The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}. The design temperature for the KPS reactor vessel is 650 °F.
9. The Westinghouse design has been maintained by the owner over the lifetime of the plant and is effectively represented by the stresses assumed in MRP-191 and MRP-232.
10. Design changes to the KPS reactor vessel internals components have been limited to Westinghouse replacement of split pins, installation of flexure-less inserts, and replacement of the reactor vessel head.
11. MRP-191 and MRP-232 was used to organize, characterize, and rank the reactor vessel internals parts into the various groups for development of aging management strategies: Existing, Primary, Expansion, and No Additional Measures.

NRC requested that DEK qualitatively address how the input parameters used in the FMECA and functionality analyses are bounded by the assumptions in MRP-227-A. Table A (below) illustrates the input parameters for the typical Westinghouse PWR internals compared to those used for the KPS internals for the six sample components identified in the NRC question above.

Table A
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS
1	Lower Core Plate	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 1×10^{22} to 5×10^{22}	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 1×10^{22} to 5×10^{22} bounds KPS	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. > 608°F	Same as MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. > 608°F	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 304 SS	Reference LTR-ARIDA-08-63, Rev. 3, dated December 5, 2008. 304 SS
2	Core Barrel Flange	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$ bounds KPS	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T-hot	The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}.	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 304 SS	Reference LTR-ARIDA-08-63, Rev. 3, dated December 5, 2008. 304 SS

Table A
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS
3	Barrel-Former Bolts	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 5×10^{22}	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 5×10^{22} bounds KPS	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. > 608°F	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. > 608°F	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 316 SS or 347 SS	Reference WCAP-13266, Rev.1, Proprietary Class 2 347 SS
4	Upper Core Barrel Welds	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$ bounds KPS	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T-hot	The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}.	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 304 SS	Reference LTR-ARIDA-08-63 Rev. 3, dated December 5, 2008. 304 SS

Table A
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS
5	Lower Core Barrel Welds	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$ bounds KPS	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T-hot	The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}.	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 304 SS	Reference LTR-ARIDA-08-63 Rev. 3, dated December 5, 2008. 304 SS
6	Upper Core Plate Alignment Pins	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 7×10^{20} to 1×10^{21}	Same as MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 7×10^{20} to 1×10^{21} bounds KPS	Reference MRP-161, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T-hot	The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}.	Reference MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Same as MRP-191, Table A-1, Results of Parameter Screening and Interviews with Analysts—Westinghouse Reactor Internals Effective Stress ≥ Threshold	Reference MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. 304 SS	Reference LTR-ARIDA-08-63 Rev. 3, dated December 5, 2008. 304 SS

Table A
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS

Notes

1) Assumed 30 years of high leakage core loading followed by 30 years of low leakage core loading.

2) Fuel Cycles 1 and 2. Once-burned fuel was used at peripheral locations during Fuel Cycles 3 through 15. In Fuel Cycle 16, KPS switched to use of a low leakage core design. KPS continued to use a low leakage core design for all subsequent fuel cycles.

3) The KPS reactor vessel materials operate at temperatures between T_{cold} and T_{hot} that are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}. The design temperature for the KPS reactor vessel is 650 °F.

4) Criteria for material, temperature, and fluence are listed in MRP-191, Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants.

5) Criteria for stress is depicted in MRP-191, Table 3-2, Irradiation Assisted Stress Corrosion Cracking (IASCC) Screening Criteria and MRP-191, Figure 3-1, MRP-175 Screening Criteria for IASCC.

6) Criteria for SCC is listed in MRP-191, Table 3-1, Stress Corrosion Cracking (SCC) Screening Criteria for PWR Internals Materials.

RAI Part 2)

All of the subject RVI components at KPS are representative of assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A. WPS-08-28, Revision 2 (LTR-ARIDA-08-63, Revision 3) documents the materials and compares the KPS components to those in a list of typical Westinghouse PWR internals components. The KPS components, materials, and design are consistent with those identified as Typical Westinghouse PWR Internals Components. KLR-1309-A documents compliance to the fluence requirements and specific design changes to the reactor vessel internals implemented by the OEM over the lifetime of the plant, all of which are addressed by MRP-191, MRP-232, and MRP-227-A. The reactor vessel internals operate between T_{hot} and T_{cold} which are approximately not less than 525 °F for T_{cold} nor higher than 611 °F for T_{hot}. Because the design has been maintained over the lifetime of the plant and the materials are the same as those identified on MRP-191, MRP-232, and MRP-227-A the stress values are bounded by the assumptions in MRP-191, MRP-232, and MRP-227-A.

RAI Part 3)

The KPS reactor vessel internals components are bounded by the typical Westinghouse PWR internals components outlined in MRP-227-A and the applicable referenced documents, including MRP-191 and MRP-232. The KPS reactor vessel internals inspection program was written to comply with MRP-227, Revision 0 and the seven (7) topical report conditions and eight (8) licensee action items identified in the NRC SER issued on June 22, 2011. Together, the requirements outlined by MRP-227, Revision 0, along with the seven (7) topical report conditions and eight (8) licensee action items from the NRC SER issued on June 22, 2011, satisfy the requirements specified in MRP-227-A. No changes in the inspection requirements are being proposed at this time because the KPS inspection program complies with MRP-227-A as indicated above. The additional notes identified in NRC Questions ME7727-RAII-EVIB-Cher-004-2012-05-09, ME7727-RAII-EVIB-Cher-009-2012-05-09, ME7727-RAII-EVIB-Cher-010-2012-05-09, and ME7727-RAII-EVIB-Cher-013-2012-05-09 have been added to the inspection tables.

RAI Part 4)

Flexure-less inserts have been installed at KPS in response to cracking issues identified in the commercial nuclear power industry. The flow restrictors on the 29 RCCA guide tubes were replaced under a design change (DCR 2030) in 1989. The flow restrictors on the four (4) plutonium recycle port guide tubes were replaced under a design change (DCR 1844) in 1986. KPS did not experience cracking of the original flexures. The replacement flexure-less insert is fabricated from subparts:

- Latch housing (1) – 304 SS
- Latch retainer (1) - 304 SS
- Latches (3) – 316 SS
- Spring (1) – 718 inconel
- Dowel pins (3) – 316 SS
- Spring retainer (1) 304 SS

KPS has not experienced any problems with the replacement flexure-less inserts. The replacement flexure-less inserts were removed and reinstalled in 2004 to facilitate replacement of the split pins. The replacement flexure-less inserts were also removed and reinstalled for the 29 RCCA active guide tubes to support the guide card inspection activity performed during the spring 2012 refueling outage.

NRC Question ME7727-RAII-EVIB-Cher-002-2012-04-27

Applicant/Licensee Action Item 2, Section 3.2.5.2, from the NRC staff's final SE of MRP-227, Rev.1, requires the following:

> "Consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of license renewal (LR) for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals," and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227-A, as modified by this SE, when submitting its plant-specific AMP such that the effects of aging on the missing component(s) will be managed for the period of extended operation."

On page 5, license action item-2 of its submittal dated December 12, 2011, the licensee stated that Westinghouse classified various RVI components based on their susceptibility to aging degradation. The NRC staff requests that the licensee confirm that Westinghouse complied with the aforementioned requirement in its entirety while performing scoping and screening for the license renewal for the KPS.

Response:

WPS-08-28, Revision 2 (LTR-ARIDA-08-63, Revision 3) tabulates the components in the KPS reactor vessel internals and compares them to a list of the typical Westinghouse PWR internals components. LTR-ARIDA-08-63 used MRP-191 for

categorization, analysis, and aging management strategy development. In addition, scoping and screening of the reactor vessel internals was performed by DEK and documented in Technical Report KLR-1104, License Renewal Project Integrated Plant Assessment Report Reactor Vessel, Internals, and Reactor Coolant System Kewaunee Power Station. There are no reactor vessel internal components included in Technical Report KLR-1104 that are not covered by LTR-ARIDA-08-63.

NRC Question ME7727-RAII-EVIB-Cher-003-2012-05-09

Contrary to the requirements addressed in the MRP-227-A report, the licensee did not include the following components in its submittal dated December 12, 2011:

(a) Lower core barrel flange shall be placed under a "Primary" inspection category as addressed in Table 4-3 of the MRP-227-A report.

(b) Upper and lower core barrel cylinder axial welds shall be included under an "Expansion" inspection category as addressed in Table 4-6 of the MRP-227-A report.

Response:

The inspection plan was revised to include the lower core barrel flange weld (i.e. lower core barrel girth weld) under a "Primary" inspection category as addressed in Table 4-3 of the MRP-227-A report. The revised Reactor Vessel Internals Inspection Plan, Table 1, showing this change is provided in Attachment 2 to the letter transmitting this response.

The inspection plan was revised to include the upper and lower core barrel cylinder axial welds under an "Expansion" inspection category as addressed in Table 4-6 of the MRP-227-A report. The revised Reactor Vessel Internals Inspection Plan, Table 2, showing these changes is provided in Attachment 2 to the letter transmitting this response.

NRC Question ME7727-RAII-EVIB-Cher-004-2012-05-09

Condition 4 of the NRC staff's SE Revision 1, dated December 16, 2011 requires that the licensee shall include inspection coverage for the RVI components. The NRC staff noted that the licensee did not include the inspection coverage for the following RVI components - (i) the control rod guide tube (CRGT) lower flange welds; (ii) For core barrel baffle-former bolts under the "Primary" inspection category; and (iii) core barrel baffle-former bolts under the "Expansion" inspection category.

(a) For CRGT lower flange welds, in Table 1 on page 8 of the December 12, 2011, submittal, a footnote shall be added consistent with the note 2 in Table 4-3 of the MRP-227-A report.

(b) For core barrel baffle-former bolts in Table 1 on page 12 of the December 12, 2011, submittal, a footnote shall be added consistent with the note 3 in Table 4-3 of the MRP-227-A report.

(c) For core barrel baffle-former bolts in Table 2 on page 1 of the December 12, 2011, submittal, a footnote shall be added consistent with the note 2 in Table 4-6 of the MRP-227-A report.

Response:

(a) A footnote was modified in Table 1 for the CRGT lower flange welds, consistent with Note 2 in Table 4-3 of the MRP-227-A report. The footnote was modified to add the following statement: "A minimum of 75% of the total identified sample population must be examined."

(b) A footnote was added to Table 1 for the core barrel baffle-former bolts under the "Primary" inspection category, consistent with Note 3 in Table 4-3 of the MRP-227-A report. The footnote states: "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection."

(c) A footnote was added to Table 2 for the core barrel former baffle-former bolts under the "Expansion" inspection category, consistent with Note 2 in Table 4-6 of the MRP-227-A report. The footnote states: "A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)."

The revised tables are provided in Attachment 2 of the letter transmitting this response.

NRC Question ME7727-RAII-EVIB-Cher-005-2012-05-09

Condition 7 of the NRC staff's SE Revision 1, dated December 16, 2011 requires that the licensee shall include a summary of the operating experience related to the aging degradation in the RVI components. The NRC staff requests that the licensee provide information regarding the extent of aging degradation (if any) that occurred thus far in all of the RVI components specifically, include the operating history of the following components at KPS:

(a) Baffle-former bolts, baffle-edge bolts, clevis insert bolts, flux thimble tubes, core barrel bolting, and thermal shields.

Response:

On April 19, 2010, the PWROG conducted a reactor internals operating experience survey under project authorization PA-MSC-0568. The survey included baffle-former bolts, baffle-edge bolts, clevis insert bolts, flux thimble tubes, thermal shield flexures, and core barrel assembly barrel-former bolts. The survey did not include the thermal shield. Information included in the RAI response is the same as provided to the PWROG for the baffle-former bolts, baffle-edge bolts, clevis insert bolts, flux thimble tubes, and core barrel assembly barrel-former bolts.

KPS has inspected portions of the reactor vessel internals 10 times (during the following years): 1976, 1980, 1985, 1988, 1995, 1996, 1998, 2004, 2006, and 2009.

VT-3 inspection has been performed on a portion of the baffle-former bolts during refueling outages KR1, KR10, KR13, KR20, KR26, KR27, and KR29. No recordable indications were noted.

A review of the annual inservice inspection (ISI) reports was performed to ascertain whether the edge bolts have been inspected to VT-3 criteria. The data sheets did not use the term edge bolts. However, portions of the baffles were inspected, and the edge bolts are part of the baffles. Therefore, evidence exists that portions of the edge bolts were inspected to VT-3 criteria concurrent with inspection of the baffle-former bolts during refueling outages KR1, KR10, KR13, KR20, KR26, KR27, and KR29. No recordable indications were noted.

VT-3 inspections were performed on the clevis insert bolts during refueling outages KR10, KR20, and KR26. No recordable indications were found.

Eddy current inspections were performed on the flux thimble tubes during refueling outages KR19, KR20, KR21, KR23, and KR26. The flux thimble tubes were inspected under procedure ER-AA-NDE-ET-501. Wear was observed in seven (7) flux thimble tubes during refueling outage KR19. The thimble tubes were replaced in 1988 due to wear. A baseline inspection was performed in 1988. The thimble tubes were inspected again in 1993, 1994, 1995, 1996, 2000, 2004, and 2009. Between the 1993 and 1994 inspections, wear damage was observed on seven tubes. Tubes with the most severe damage were repositioned to move the wear scars away from the lower nozzle. Subsequent inspections in 1995, 1996, 2000, 2004, and 2009 indicated no significant wear progression and no additional repositioning was recommended.

VT-3 inspections were performed on a portion of the core barrel assembly barrel-former bolts during refueling outages KR1, KR10, KR13, KR 20, KR26, KR27, and KR29. No recordable indications were found.

VT-3 inspection has been performed on portions of the thermal shield during refueling outages KR10, KR20, and KR26. No recordable indications were found.

NRC Question ME7727-RAII-EVIB-Cher-006-2012-05-09

Historically, the following materials used in the PWR RVI components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A report. In this context, the NRC staff requests that the licensee confirm that these materials are not currently used in the RVI components at KPS.

(1) Nickel base alloys - Inconel 600; Weld Metals - Alloy 82 and 182 and Alloy X-750

(2) Alloy A-286 ASTM A 453 Grade 660, Condition A or B

(3) Stainless steel type 347 material (excluding baffle-former bolts)

(4) Precipitation hardened (PH) stainless steel materials - 17-4 and 15-5

(5) Type 431 stainless steel material

Response:

Materials used in fabrication of the reactor internals are listed in WPS-08-28, Revision 2 (LTR-ARIDA-08-63, Revision 3, dated December 5, 2008). DEK reviewed the contents of this document to determine if any of the listed materials are used in RVI components. The results are as follows.

(1) The following KPS reactor internals are fabricated from Nickel base alloys - Inconel 600; Weld Metals - Alloy 82 and 182 and Alloy X-750: Clevis Insert Lock Keys (Alloy 600), Clevis Inserts (Alloy 600), and Clevis Insert Bolts (Alloy X-750).

(2) No reactor internal components are listed in LTR-ARIDA-08-63, Revision 3 as being fabricated from alloy A-286, ASTM A 453, Grade 660, Condition A or B.

(3) No reactor internal components are listed in LTR-ARIDA-08-63, Revision 3 as being fabricated from stainless steel type 347 material (excluding baffle-former bolts and barrel-former bolts). At KPS, the baffle-former bolts and barrel-former bolts are fabricated from type 347 materials.

(4) No reactor internal components are listed in LTR-ARIDA-08-63, Revision 3 as being fabricated from precipitation hardened (PH) stainless steel materials - 17-4 and 15-5.

(5) No reactor internal components are listed in LTR-ARIDA-08-63, Revision 3 as being fabricated from type 431 stainless steel materials.

NRC Question ME7727-RAII-EVIB-Cher-007-2012-05-09

To verify that the licensee is in compliance with the implementation and control of the ten elements of the aging management program (AMP) addressed in GALL AMP XI.M16-A, "PWR Vessel Internals," the NRC staff requests that the licensee submit the KPS's AMP-KLR-1309A report, Revision 3, "License Renewal Project, Aging Management Program, ASME Section XI, In-service Inspection, Subsection IWB, IWC, and IWD, Reactor Vessel Internals Inspection, Kewaunee Power Station," effective date September 30, 2011 as part of this review.

Response:

KLR-1309A report, Revision 3, was submitted to the NRC by letter dated April 2, 2012.

NRC Question ME7727-RAII-EVIB-Cher-008-2012-05-09

The licensee is required to inspect 20% of CRGT guide card assemblies per Table 4-3 on page 4-26 of the MRP-227-A report. The NRC staff requests that the licensee provide an explanation how 6 out of 36 CRGT guide plate cards were selected. The explanation for the selection process should include the following aspects: (a) most susceptible areas to experience aging degradation, (b) high stress areas, and, (c) plant-specific operating experience.

Response:

There are a total of 37 control rod guide tubes at KPS (29 control rod guide tubes contain active drive rods; four (4) tubes are for plutonium recycle ports which do not contain rods; and four (4) tubes are for part length control rods which do not contain rods). Because the four (4) plutonium recycle locations and the four (4) part length locations do not contain rods, and are not subject to wear, they are not part of the population of guide tubes subject to VT-3 inspection.

During the spring 2012 refueling outage (KR32), KPS performed a VT-3 inspection of the guide cards in all 29 of the control rod guide tubes with active drive rods. Various amounts of wear were observed in guide cards for all 29 of the control rod guide tubes. Engineering analysis per the guidance in WCAP-17562-P, Revision 0, "Westinghouse Pressurized Water Reactor Internals Guide Tube Guide Card Wear Criteria," confirmed that operation of the control tubes is not impaired with wear in excess of 85% of the effective slot width opening (ESWO) for up to three (3) adjacent guide plates in a guide tube. Engineering analysis indicates that wear is not projected to reach an unacceptable limit of 85% ESWO in three (3) adjacent guide plates in any guide tube within 20 effective full-power years (EFPY). A pattern analysis was performed, which

confirms that most of the observable wear occurs in guide cards located in the lower portion of the guide tube. A summary of the guide card hole locations that are projected to wear to 85% ESWO within 20 EFPY of the current inspection time is provided in Table B below.

Table B
Guide Card Hole Locations – Projected wear to reach 85% within 20 EFPY

Guide Tube	Guide Card	Hole Location	EFPY to reach 85%	
			Constant Volumetric	Operation Curve (Aligned High Flow Case)
B08	9	E5	27.6	18
D10	7	E5	30.3	19.9
D10	7	E2	27.6	18
G11	9	B2	29.7	19.5
L06	7	B2	22.5	14.4
L06	8	B2	18.6	11.9
L06	9	B2	30.3	19.9

DEK plans one of the following options in 10 years: 1) select six (6) of the 29 active locations for inspection per MRP-227-A; or, 2) submit a deviation request to not inspect again in 10 years since no unacceptable wear was observed nor projected to occur within 20 EFPY from the date of the current inspection (2012). DEK anticipates that the next inspection would include the guide cards in guide tube assembly locations B08, D10, G11, and L06.

NRC Question ME7727-RAII-EVIB-Cher-009-2012-05-09

Editorial corrections—The NRC staff requests that licensee include the following revisions in the December 12, 2011, submittal

(a) Table 1, on page 9 in "Comments" column related to upper core barrel flange weld, last sentence should be revised to read—Expansion Link-"Lower Support Column Bodies." — Reference-Table 4-3 on page 4-26 of the MRP-227-A report.

(b) Table 1, on page 9 in "Comments" column related to core barrel girth weld, last sentence should be revised to read—Expansion Link-"Upper and Lower Core Barrel Cylinder Axial welds." --- Reference-Table 4-3 on page 4-26 of the MRP-227-A report.

(c) Table 1, on page 10 in "Comments" column related to core barrel girth welds, last sentence should be revised to read—Expansion Link-"Upper and Lower Core

Barrel Cylinder Axial welds." - Reference-Table 4-3 on page 4-26 of the MRP-227-A report.

(d) Table 1, on page 13 in "Parts Examined" column related to core barrel baffle-former assembly should be revised to include description of the components (baffle plates, baffle-edge bolts, and former plates) which is consistent with Table 4-3 on page 4-28 in the MRP-227-A report.

Response:

Table 1 on (originally numbered) pages 9, 10, and 13 has been revised as requested above. The revised Table 1 is provided in Attachment 2 to the letter transmitting this response (the data that appeared on page 13 of the previous version of Table 1 now appears on page 14).

NRC Question ME7727-RAII-EVIB-Cher-010-2012-05-09

In Table 1, on page 10 of the December 12, 2011, submittal, the licensee stated that Type 347 stainless steel baffle-former bolts are used at KPS. In Appendix A of the MRP-227-A report on page A-3, the MRP states that Type 347 stainless steel material is susceptible to irradiation-assisted stress corrosion cracking (IASCC). With respect to IASCC in these bolts, the NRC staff requests that the licensee address the following issues:

(a) The number of Type 347 bolts in the baffle-former bolt assembly that are classified under "Primary," and "Expansion," categories, and;

(b) The number of Type 347 bolts in the baffle-edge bolt assembly that are classified under " Primary," category, and;

(c) The percentage of Type 347 bolts in the aforementioned assemblies that were inspected thus far, at KPS and the results of the inspections.

Response:

Table 1, under the column entitled Equipment No., indicates there are 728 baffle-former bolts (Primary) and 688 baffle-edge bolts (Primary). Table 2, under the column Equipment No., indicates there are 344 barrel bolts (Expansion). These installed items are fabricated of type 347 material.

There are 688 baffle-edge bolts (Primary) fabricated of type 316 stainless steel installed at KPS. There are no type 347 stainless steel baffle-edge bolts installed at KPS.

MRP-227-A identifies ultrasonic examination (UT) as the required examination method for inspection of the baffle-former bolts (Primary) and barrel-former bolts (Expansion).

DEK has not performed an ultrasonic inspection of either the baffle bolts or barrel-former bolts at KPS.

Various portions of the baffle bolts or barrel-former bolts have been inspected to VT-3 criteria during refueling outages KR-1, KR-10, KR-13, KR-20, KR-26, KR-27, and KR-29. No recordable indications were found during these VT-3 inspections.

NRC Question ME7727-RAII-EVIB-Cher-011-2012-05-09

In Table 1, on page 10 of the December 12, 2011, submittal, the licensee stated that the flux thimble tubes are examined every 5 year interval. Provide an explanation for selecting this inspection frequency.

Response:

The KPS response to NRC Bulletin 88-09, dated November 7, 1988, stated that the flux thimble tubes would be inspected at five year intervals.

The thimble tubes were replaced in 1988 due to wear. A baseline inspection was performed in 1988. The thimble tubes were inspected again in 1993, 1994, 1995, 1996, 2000, 2004, and 2009. Between the 1993 and 1994 inspections, wear damage was observed on seven tubes. Tubes with the most severe damage were repositioned to move the wear scars away from the lower nozzle. The subsequent inspections in 1995, 1996, 2000, 2004, and 2009 indicated no significant wear progression and no additional repositioning was recommended.

Based on the inspection history, inspection of the flux thimble tubes at five year intervals remains appropriate.

NRC Question ME7727-RAII-EVIB-Cher-012-2012-05-09

On page 8 in Attachment 1 of the December 12, 2011 submittal, the licensee stated that cast austenitic stainless steel (CASS) materials that were classified under "No Additional Measures (NAM)" by the MRP will be evaluated for their susceptibility to thermal and neutron embrittlement. RVI components under NAM classification were included in Table 4 of the December 12, 2011, submittal. After the review of Table 4, the NRC staff requests that the licensee provide following information.

(a) Provide the time frame for performing evaluations of the CASS materials per the criteria (fluence values, stress and delta ferrite) addressed on page 8 in Attachment 1 of the December 12, 2011 submittal, and,

(b) If the evaluations result in the implementation of enhanced visual testing (EVT-1) as a part supplemental examination addressed in Table 4 of the December 12,

2011, submittal, the licensee shall provide the results of the examinations of the CASS materials.

Response:

DEK plans to complete the evaluations of the CASS materials prior to refueling outage KR34 or conduct the enhanced visual examination during refueling outage KR34. KR34 is projected to occur in spring 2015.

If evaluations of the CASS materials require implementation of enhanced visual testing (EVT-1), then the results of these examinations would be provided to the NRC.

NRC Question ME7727-RAII-EVIB-Cher-013-2012-05-09

In Tables 2 and 4 of the licensee's AMP KLR-1309A, Revision 3, the staff noted several inconsistencies between the inspection and evaluation (I&E) guidelines that are addressed in the licensee's AMP and the MRP-227-A report. The following table includes these inconsistencies for the various RVI components at KPS and the NRC staff requests that the licensee revise its AMP accordingly.

RVI Component -- AMP KLR-1309A, Revision 3	Inconsistencies with MRP-227-A Report I&E Guidelines
CRGT Lower Flange Weld	Note 2 in Table 4-3 of MRP-227-A is not included
Upper Core Barrel Flange Weld	Note 4 in Table 4-3 of MRP-227-A is not included
Upper and Lower Core Barrel Cylinder Girth Welds	These welds are not included (Reference Table 4-3 of MRP-227-A report)
Lower Core Barrel Flange Weld	These welds are not included (Reference Table 4-3 of MRP-227-A report)
Baffle-Edge Bolts	Note 3 in Table 4-3 of MRP-227-A is not included
Baffle-Former Bolts	Note 3 in Table 4-3 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)
Upper Core Plate	This component is not included (Reference Table 4-6 of MRP-227-A report)
Lower Support Forging/Casting	These components are not included (Reference Table 4-6 of MRP-227-A report)
Baffle-Former Bolts	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)
Lower Support Column Bolts	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)

RVI Component -- AMP KLR-1309A, Revision 3	Inconsistencies with MRP-227-A Report I&E Guidelines
Lower Support Column Bodies (non-cast)	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)
Lower Support Column Bodies (cast)	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)

Response:

AMP KLR-1309A, Revision 3, was intended to provide guidance for developing the Reactor Vessel Internals Inspection Plan. Rather than revising AMP KLR-1309A, the corrected information was placed directly into Reactor Vessel Internals Inspection Plan, Tables 1 and 2. Table C below provides a listing of the specific changes made in response to this RAI item.

Table C
Resolution of Inconsistencies

RVI Component -- AMP KLR-1309A, Revision 3	Inconsistencies with MRP-227-A Report I&E Guidelines	Resolution
CRGT Lower Flange Weld	Note 2 in Table 4-3 of MRP-227-A is not included	The following note has been added to Table 1, for the lower flange welds: "A minimum of 75% of the total identified sample population must be examined."
Upper Core Barrel Flange Weld	Note 4 in Table 4-3 of MRP-227-A is not included	The following note has been added to Table 1 for the upper core barrel flange weld: "A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit."
Upper and Lower Core Barrel Cylinder Girth Welds	These welds are not included (Reference Table 4-3 of MRP-227-A report)	DEK has verified that the upper core barrel flange to core barrel weld is included in Table 1.
Lower Core Barrel Flange Weld	These welds are not included (Reference Table 4-3 of MRP-227-A report)	There are 4 circumferential welds in the KPS core barrel: upper core barrel flange to core barrel weld, core barrel mid plane weld, core barrel lower mid plane weld and core barrel bottom weld. DEK has verified that all 4 circumferential welds are included in Table 1.
Baffle-Edge Bolts	Note 3 in Table 4-3 of MRP-227-A is not included	The following note has been added to Table 1 for the baffle-edge bolts: "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit".

Table C
Resolution of Inconsistencies

RVI Component -- AMP KLR-1309A, Revision 3	Inconsistencies with MRP-227-A Report I&E Guidelines	Resolution
Baffle-Former Bolts	Note 3 in Table 4-3 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)	The following note has been added to Table 1 for the baffle-former bolts: "A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit." KPS is adding a note that inspection of the baffle-former bolts is required at 10 year intervals.
Upper Core Plate	This component is not included (Reference Table 4-6 of MRP-227-A report)	The upper core plate was verified to be in KLR-1309A Rev 3. The upper core plate is also listed in Table 2.
Lower Support Forging/Casting	These components are not included (Reference Table 4-6 of MRP-227-A report)	The lower support forging was verified to be in both LTR-ARIDA-08-63 Rev. 3 dated December 5, 2008 & KLR-1309A Rev 3. The lower support forging/casting is also in Table 2.
Barrel -Former Bolts	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)	The following note has been added to Table 1 for the baffle-former bolts: "Subsequent examination is required every 10 year interval."
Lower Support Column Bolts	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)	The following note has been added to Table 2 for the lower support column bolts: "A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)." Re-inspection on a 10-year frequency is added to Table 2.
Lower Support Column Bodies (non-cast)	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)	The following note has been added to Table 2 for the Lower Support Column Bodies (non-cast): "A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)." Re-inspection on a 10-year frequency is added to Table 2.
Lower Support Column Bodies (cast)	Note 2 in Table 4-6 of the MRP-227-A is not included; Subsequent examination is required every 10 year interval (Table 4-3 of MRP-227-A)	The following note has been added to Table 2 for the Lower Support Column Bodies (cast): "A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions)." Re-inspection on a 10-year frequency is added to Table 2.

NRC Question ME7727-RAII-EVIB-Cher-014-2012-04-27

Based on its evaluation of MRP-227, the NRC staff believes each plant, including Kewaunee Power Station (KPS), needs to perform a plant-specific analysis to ensure that it is bounded by the MRP-227-A assumptions unless KPS can submit an evaluation derived from MRP-227-A that: (1) is specifically relevant to KPS; and (2) is bounded for applicable parameters.

As an alternative the licensee can propose for analysis: (1) other reactor vessel internals (RVI) components that can be analyzed to satisfy this requirement; or (2) another component for which it verifies that the stress/fluence values used by MRP-227-A for that component are bounding.

Response:

DEK has confirmed with Westinghouse that the input parameters used for the generic FMECA and functionality analysis summarized in MRP-191 and MRP-232 are representative for KPS. There are no components at KPS that are not bounded by assumptions regarding neutron fluence, temperature, stress, or material used in the development of MPR-227-A. Therefore, KPS is bounded by the MRP-227-A assumptions.

REFERENCES

1. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 - Kewaunee - Review of RVI Inspection Plan RAI Set 1 of 2 (Rev 1)," dated March 28, 2012.
2. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 (not 7277) - Kewaunee - Review of RVI Inspection Plan - (1) RAI Set 2 of 2, adding to RAI Set 1 of 2 (Rev 1); and (2) request for clarification conference call," dated April 9, 2012.
3. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "ME7727 Kewaunee Request for Additional Information (RAI) ME7727-RAII-EVIB-Cher-014-2012-04-27," dated April 11, 2012.
4. Westinghouse Electric Company (Westinghouse) letter WPS-08-28 Revision 2, "Dominion Energy Kewaunee, Kewaunee Power Station, KPS Reactor Vessel Internals Fabrication and Design Information, LTR-ARIDA-08-63, Revision 3, Summary Report for the Fabrication and Design Information for KPS Reactor Vessel Internals," dated December 22, 2008.
5. WCAP-17562-P, Revision 0, "Westinghouse Pressurized Water Reactor Internals Guide Tube Guide Card Wear Criteria."
6. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.
7. MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR [pressurized water reactor] Designs" (proprietary).
8. MRP-232, Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals" (proprietary).
9. EPRI Report 1016596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, Electric Power Research Institute (EPRI), Palo Alto, California.
10. Final Safety Evaluation (SE) of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," dated June 22, 2011.

ATTACHMENT 2

SUPPLEMENT TO
REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST

TABLES 1 AND 2 (REVISION 1)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

Table 1 (Revision 1)

Reactor Vessel Internals Inspection Plan

MRP-227

Westinghouse Plants Primary Components

(15 pages)

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

TABLE 1 (Revision 1)
KEWAUNEE POWER STATION
FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) GUIDE PLATE CARDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
CRGT	Guide Plate Cards	Attachment Figure 2	Position 2F		N							X		Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. 20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. A total of 29 locations. It is suggested that the population selected for initial inspection coincide with the inlet nozzle locations.
CRGT	Guide Plate Cards	Attachment Figure 2	Position 2H		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 3E		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 3G		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 3I		Y							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 4D		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 4J		Y							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) GUIDE PLATE CARDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Guide Plate Cards	Attachment Figure 2	Position 5C		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 5E		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 5G		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 5I		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 5K		Y							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 6B		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 6F		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 6H		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 6L		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 7C		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 7E		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 7G		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 7I		N							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) GUIDE PLATE CARDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Guide Plate Cards	Attachment Figure 2	Position 7K		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 8B		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 8F		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 8H		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 8L		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 9C		Y							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 9E		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 9G		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 9I		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 9K		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 10D		Y							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 10J		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 11E		Y							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) GUIDE PLATE CARDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Guide Plate Cards	Attachment Figure 2	Position 11G		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 11I		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 12F		N							X		Same as above
CRGT	Guide Plate Cards	Attachment Figure 2	Position 12H		N							X		Same as above

Category Notes:

1. **End of Original License is December 21, 2013. The examinations may be performed during Refueling Outages KR 32 (spring 2012), KR 33 (fall 2013) or KR 34 (spring 2015).**

TABLE 1 (Revision 1)
KEWAUNEE POWER STATION
FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) LOWER FLANGE WELDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
CRGT	Lower Flange Welds	Attachment Figure 4	Position 2F		Y							X		Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the of the license renewal period and subsequent examination on a ten year interval. 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal See Figure 4-21 of MRP-227. Expansion Link – Bottom-mounted (BMI) column bodies and Lower support column bodies (cast). Expansion Link – Upper Core Plate and Lower Support Forging per NRC SER TRC-1. A total of 37 locations.
CRGT	Lower Flange Welds	Attachment Figure 4	Position 2H		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 3E		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 3G		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 3I		Y							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) LOWER FLANGE WELDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Lower Flange Welds	Attachment Figure 4	Position 4D		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 4J		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 5C		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 5E		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 5G		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 5I		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 5K		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 6B		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 6F		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 6H		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 6L		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 7C		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 7E		N							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) LOWER FLANGE WELDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Lower Flange Welds	Attachment Figure 4	Position 7G		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 7I		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 7K		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 8B		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 8F		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 8H		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 8L		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 9C		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 9E		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 9G		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 9I		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 9K		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 10D		Y							X		Same as above

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CONTROL ROD DRIVE TUBES (CRGT) LOWER FLANGE WELDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
CRGT	Lower Flange Welds	Attachment Figure 4	Position 10J		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 11E		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 11G		N							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 11I		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 12F		Y							X		Same as above
CRGT	Lower Flange Welds	Attachment Figure 4	Position 12H		Y							X		Same as above

Category Notes:

1. **End of Original License is December 21, 2013. The examinations may be performed during Refueling Outages KR 32 (spring 2012), KR 33 (fall 2013) or KR 34 (spring 2015).**
2. **There are a total of 20 active CRGT's on the periphery.**
3. **It is anticipated that approximately 180 degrees or half the weld length is accessible on each periphery CRGT.**
4. **100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. A minimum of 75% of the total identified sample population must be examined.**

TABLE 1 (Revision 1)
KEWAUNEE POWER STATION
FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CORE BARREL ASSEMBLY - UPPER CORE BARREL FLANGE WELD

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Assembly	Upper Core Barrel Flange Weld	M-1199	Upper Core Barrel Flange to Core Barrel Weld		Y						X[3]	X[3]		Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. See Figure 4-22 of MRP-227. Expansion Link – Core Barrel Outlet Nozzles. Expansion Link – Lower Support Column Bodies per NRC SER TRC-1. Reference Table 4-3, page 4-26 of MRP-227-A
Core Barrel Assembly	NRC SER Upper Core Barrel Cylinder Girth Welds	M-1199	Core Barrel Mid Plane Weld		Y						X[3]	X[3]		Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. Expansion Link – Upper and Lower Core Barrel Cylinder Axial welds. Reference Table 4-3, page 4-26 of MRP-227-A.

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CORE BARREL ASSEMBLY - UPPER CORE BARREL FLANGE WELD

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
Core Barrel Assembly	Lower Core Barrel Cylinder Girth Welds	M-1199	Core Barrel Lower Mid Plane Weld		Y						X[3]	X[3]		Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. No expansion required. Expansion Link – Upper and Lower Core Barrel Cylinder Axial welds. Reference Table 4-3, page 4-26 of MRP-227-A.
Core Barrel Assembly	NRC SER Lower Core Barrel Flange Welds	M-1199	Core Barrel Lower Bottom Weld		N						X[3,4]	X[3,4]		Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. Non expansion required. Expansion Link – Upper and Lower Core Barrel Cylinder Axial welds. Reference Table 4-3, page 4-27 of MRP-227-A.

TABLE 1 (Revision 1)
KEWAUNEE POWER STATION
FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 Description TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS CORE BARREL ASSEMBLY - UPPER CORE BARREL FLANGE WELD

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		

Category Notes:

1. End of Original License is December 21, 2013. The examinations may be performed during Refueling Outages KR33 (fall 2013) or KR34 (spring 2015).
2. Reference NRC SER dated June 22, 2011, Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines".
3. Enhanced visual may be satisfied through eddy current examination if elected in lieu of EVT-1.
4. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
5. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS BAFFLE FORMER ASSEMBLY - BAFFLE EDGE BOLTS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Baffle-Former Assembly	Baffle-Edge Bolts	WCAP-13266R1 Figures 6.1, 6.2, 6.3	688 Edge Bolts 316 ss		Y							X		Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval. Bolts and locking devices on high fluence seams. 100% of components accessible from the core side. Reference Figure 4-23 from MRP-227. No expansion required.

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **The KPS Reactor Vessel is projected to reach 33 EFPY at End of Original License and 52.1 EFPY at End of Life Extension.**
3. **A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.**

TABLE 1 (Revision 1)
KEWAUNEE POWER STATION
FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS BAFFLE-FORMER ASSEMBLY BAFFLE-FORMER BOLTS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Baffle-Former Assembly	Baffle-Former Bolts	WCAP-13266R1 Figures 6.1, 6.2, 6.3	728 Baffle Former Bolts 347 ss		Y					X				Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a 10-year inspection frequency per NRC SER TRC-5. 100% of accessible bolts or as supported by plant-specific justification. Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. Reference Figures 4-23 and 4-24 of MRP-227. Expansion Link – Lower support column bolts and barrel-former bolts.

Category Notes:

1. **End of Original License is December 21, 2013. The KPS Reactor Vessel is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at Refueling Outage KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**
2. **Reference WCAP-15425 for KPS Minimum Baffle-Former Bolt Pattern.**
3. **Reference NRC SER dated June 22, 2011, Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines".**
4. **A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit. Subsequent inspection of the baffle-former bolts is required every 10 year interval.**

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS BAFFLE - FORMER ASSEMBLY – ASSEMBLY

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Baffle-Former Assembly	Assembly (baffle plates, baffle-edge bolts, and former plates)	WCAP-13266R1 Figures 6.1, 6.2, 6.3 Attachment Figure 4	Baffle Former		Y							X		Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval. Inspections are performed on the core side surface. Reference Figures 4-24, 4-25, 4-26, and 4-27 of MRP-227. No expansion required.

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **The KPS Reactor Vessel is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at Refueling Outage KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**

TABLE 1 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-3 WESTINGHOUSE PLANTS PRIMARY COMPONENTS THERMAL SHIELD ASSEMBLY – THERMAL SHIELD FLEXURES

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Thermal Shield Assembly	Thermal Shield Flexures	Attachment Figure 4	Thermal Shield Flexure 0°		Y							X		Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval. 100% of thermal shield flexures. Reference Figures 4-29 and 4-36 of MRP-227. No expansion required.
Core Barrel Thermal Shield Assembly	Thermal Shield Flexures	Attachment Figure 4	Thermal Shield Flexure 90°		Y							X		Same as above
Core Barrel Thermal Shield Assembly	Thermal Shield Flexures	Attachment Figure 4	Thermal Shield Flexure 180°		Y							X		Same as above
Core Barrel Thermal Shield Assembly	Thermal Shield Flexures	Attachment Figure 4	Thermal Shield Flexure 270°		Y							X		Same as above

Category Notes:

1. **End of Original License is December 21, 2013. The examinations may be performed during Refueling Outages KR33 (fall 2013) or KR34 (spring 2015).**
2. **VT-3 examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.**

Table 2 (Revision 1)

Reactor Vessel Internals Inspection Plan

MRP-227

Westinghouse Plants Expansion Components

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TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS CORE BARREL ASSEMBLY - BARREL-FORMER BOLTS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel-Former Assembly	Barrel-Former Bolts	WCAP-13266R1 Figures 6.1, 6.2, 6.3	344 Barrel-Former Bolts 347 ss		N					X				Primary Link- Baffle-Former Bolts. Volumetric (UT) examination, with initial examination dependent on results of baffle-former bolt examinations. Re-inspection is on a 10-year frequency. 100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. Reference Figure 4-23 of MRP-227. Expansion Link – Lower support column bolts and Barrel-former bolts.

Category Notes:

1. **End of Original License is December 21, 2013. The KPS RV is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**
2. **Reference WCAP-13266 Revision 1 for details. The barrel-former bolts are fabricated from type 347 stainless steel.**
3. **Examinations are scheduled per the Corrective Action Process if the number of indications on the Baffle-Former Bolts exceeds the threshold.**
4. **Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles.**
5. **Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.**
6. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
7. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS CORE BARREL ASSEMBLY – UPPER AND LOWER CORE BARREL CYLINDER AXIAL WELDS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Assembly	Upper and lower core barrel cylinder axial welds				N					X				Primary Link- Upper and lower core barrel cylinder girth welds. Re-inspection is on a 10-year frequency. Enhanced visual (EVT-1) examination, with initial examination dependent on the examination results for upper and lower core barrel cylinder girth welds. Re-inspection on a 10-year frequency. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. Reference Figure 4-34 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013. The KPS RV is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**
2. **Examinations are scheduled per the Corrective Action Process if a surface breaking indication is observed in the upper core barrel flange weld.**
3. **The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.**
4. **The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.**
5. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
6. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS LOWER SUPPORT ASSEMBLY- LOWER SUPPORT COLUMN BOLTS

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Lower Support Assembly	Lower Support Column Bolts	W Drawing 882D685			N					X				Primary Link- Baffle-Former Bolts. Volumetric (UT) examination, with initial examinations dependent on results of baffle-former bolt examinations. Re-inspection is on a 10-year frequency. 100% of accessible bolts or as supported by plant-specific justification. Reference Figures 4-32 and 4-33 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013. The KPS RV is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**
2. **Examinations are scheduled per the Corrective Action Process if the number of indications on the Baffle-former bolts exceeds the threshold.**
3. **Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.**
4. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
5. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS CORE BARREL ASSEMBLY - CORE BARREL FLANGE, CORE BARREL OUTLET NOZZLES, CORE BARREL SAFETY INJECTION NOZZLES AND LOWER CORE BARREL FLANGE WELD

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Core Barrel Assembly	Core Barrel Flange (1), Core Barrel Outlet Nozzles(2), Safety Injection Nozzles (2)				N							X		Primary Link- Upper Core Barrel Flange Weld. Enhanced visual (EVT-1) examination, with initial examination dependent on the examination results for upper core barrel flange. Re-inspection on a 10-year frequency. 100% of one side of the accessible surfaces of the selected weld and adjacent base metal. Reference Figure 4-34 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013. The KPS RV is projected to reach 33 EFPY at End of Original License, 34.5 EFPY at KR-34 (spring 2015), and 52.1 EFPY at End of Life Extension.**
2. **Examinations are scheduled per the Corrective Action Process if a surface breaking indication is observed in the upper core barrel flange weld.**
3. **The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage.**
4. **If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.**
5. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
6. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS LOWER SUPPORT ASSEMBLY - LOWER SUPPORT COLUMN BODIES (NON CAST)

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Lower Support Assembly	Lower Support Column Bodies (Non Cast)	W685J896			N							X		Primary Link- Upper Core Barrel Flange Weld. Enhanced visual (EVT-1) examination, with initial examination dependent on the examination results for upper core barrel flange. Re-inspection on a 10-year frequency. 100% of accessible surfaces. Reference Figure 4-34 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **Examinations are scheduled per the Corrective Action Process if a surface breaking indication is observed in the upper core barrel flange weld.**
3. **The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage.**
4. **If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.**
5. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
6. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS LOWER SUPPORT ASSEMBLY - LOWER SUPPORT COLUMN BODIES (CAST)

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Lower Support Assembly	Lower Support Column Bodies (Cast)	W685J896			N							X		Lower Support Column Bodies are not cast at Kewaunee Power Station. This expansion item [from the CRGT lower flange welds] is Not Applicable to KPS. Primary Link- Control Rod Guide Tube Lower Flanges. Visual (EVT-1) examination. 100% of accessible support columns. Reference Figure 4-34 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **Examinations are scheduled per the Corrective Action Process if a crack-like surface indication is observed.**
3. **Lower support column bodies (cast) are not applicable to Kewaunee Power Station. Confirmation of surface breaking indications in CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies within three fuel cycles following the initial observation. For cast lower support column bodies, the specific relevant condition is a detectable crack-like surface indication.**
4. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
5. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category MRP-227 **Description** TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS BOTTOM MOUNTED INSTRUMENTATION SYSTEM - BMI COLUMN BODIES

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Reactor Vessel Bottom Mounted Instrumentation System	Bottom-Mounted Instrumentation (BMI) Column Bodies (36)	685J896			N							X		Primary Link- Control rod guide tube lower flanges. Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval. 100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. Re-inspection on a 10-year frequency. Reference Figure 4-35 of MRP-227.

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **Examinations are scheduled per the Corrective Action Process if a detectable crack-like surface indication is detected in the CRGT lower flange welds.**
3. **Bottom-Mounted Instrumentation (BMI) column bodies. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. Confirmation of surface breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage.**
4. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
5. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**

TABLE 2 (Revision 1)

KEWAUNEE POWER STATION

FOURTH AND FIFTH INTERVAL ISI SCHEDULE

Examination Category – MRP-227 Description TABLE 4-6 WESTINGHOUSE PLANTS EXPANSION COMPONENTS Upper Internals & Lower Internals Assembly

Item No.	Parts Examined	ISI Drawing No.	Equipment No.	INT.	Examination Period					Examination Methods			Exemption, Code Case, or Relief Request	Comments
					Sch	1	2	3	EOI	Vol	Sur	Vis		
	Reactor Vessel Internals													
Lower Support Forging	Lower Support Forging	M-1199			N							X		Primary Link - Control Rod Guide Tube Lower Flange Welds. Enhanced visual (EVT-1) examination. Re-inspection on a 10-year frequency. 100% of accessible surfaces (Note 5). Ref. MRP-227, Figure 4-33.
Upper Core Plate	Upper Core Plate	M-1199			N							X		Primary Link - Control Rod Guide Tube Lower Flange Welds. Enhanced visual (EVT-1) examination. Re-inspection on a 10-year frequency. 100% of accessible surfaces (Note 5).

Category Notes:

1. **End of Original License is December 21, 2013.**
2. **Examinations are scheduled per the Corrective Action Process if a crack-like surface indication is observed.**
3. **Control Rod Guide Tube Flange Welds. Confirmation of surface breaking indications shall require EVT-1 examination of the lower support forging & upper core support plate within three fuel cycles following the initial observation. For the upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication.**
4. **If expansion is needed/invoked then subsequent examination is required every 10 year interval.**
5. **A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).**