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Dominion®

September 18, 2012

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

Serial No. 12-559
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
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By application dated December 12, 2011 (Reference 1), Dominion Energy Kewaunee, Inc. (DEK) requested approval of the inspection plan for reactor vessel internal (RVI) components at Kewaunee Power Station (KPS) pursuant to the provisions of Renewed Operating License DPR-43. This inspection plan was submitted in order to fulfill certain requirements of KPS 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. The inspection plan was supplemented on June 28 (Reference 2) and August 30, 2012 (Reference 3).

Subsequently, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the inspection plan (Reference 4). The NRC questions were discussed with NRC staff to obtain clarification, during a telephone conference on August 6, 2012. The DEK responses to the NRC RAI questions are provided in Attachment 1 to this letter.

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

Very truly yours,

for LNHartz

Leslie N. Hartz
Vice President – Nuclear Support Services

Attachment:

1. Response to Request for Additional Information, Kewaunee Power Station Reactor Vessel Internals Inspection Plan

A047

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. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request, Supplement and Response to Request for Additional Information," dated June 28, 2012.
3. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request, Supplement and Response to Request for Additional Information," dated August 30, 2012.
4. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "RE: ME7727 - Kewaunee - Draft Request for Additional Information Re: RVI components Inspection Plan - RAII-Cher-018 to -020," dated August 1, 2012.

Commitments made by this letter: NONE

cc: Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
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Kewaunee Power Station

ATTACHMENT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION KEWAUNEE POWER STATION REACTOR VESSEL INTERNALS INSPECTION PLAN

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION KEWAUNEE POWER STATION REACTOR VESSEL INTERNALS INSPECTION PLAN

On August 1, 2012, the NRC transmitted to Dominion Energy Kewaunee, (DEK) a request for additional information (RAI) (Reference 1) 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 August 6, 2012.

The RAI questions are provided below, followed by the DEK response.

NRC Question ME7727-RAII-EVIB-Cher-018-2012-08-01
(Follow-up Question to ME7727-RAII-EVIB-Cher-008-2012-05-09)

The licensee, in a letter dated June 28, 2012, provided a response to the NRC staff's question ME7727-RAII-EVIB-Cher-008-2012-05-09, which requested how the licensee determined to inspect six out of a total population of 36 CRGT guide cards. Furthermore, the NRC staff requested that the licensee provide a brief summary of its methodology used in selecting the inspection sample for the CRGT guide cards. The licensee responded with the following aspects in its selection of the number of CRGT cards for inspection:

(a) most susceptible areas to experience aging degradation,

(b) high stress areas, and,

(c) plant-specific operating experience.

In its response, the licensee stated that 20% of the active CRG tubes with active drive rods (totaling 29) were inspected, and wear was observed in guide cards in all 29 CRGTs. The licensee stated that it used the guidance provided by Westinghouse topical report WCAP-17562-P, Revision 0, "Westinghouse Pressurized Water Reactor Internals Guide Tube Guide Card Wear Criteria." According to this report, the operation of the control tubes is not impaired by wear in excess of 85% of the effective slot width opening for up to three adjacent guide plates in guide tube.

The NRC staff did not review this report. However, the licensee provided a summary of the guide card hole locations that are projected to experience wear up to 85% within 20 effective full power years (EFPY).

After reviewing the information, the NRC staff determined that the licensee did not provide the following information, which would demonstrate that compliance with the WCAP-17562-P, Revision 0 guidelines is adequate in effectively managing the aging degradation in guide cards.

Therefore, the NRC staff requests that the licensee:

1) describe how the criteria for maximum allowed wear was established;
2) provide an explanation for the meaning of the numerical values listed under columns -*Constant Volumetric* and *Operation Curve* in Table B of the response to the NRC staff's question- ME7727-RAII-EVIB-Cher-008-2012-05-09;
3) provide the methodology used to inspect the guide cards which includes removal of the drive rods from the upper internals and insert a comparator device down inside the CRGT and compare each guide card the existing conditions to the requirements of the WCAP-17562-P, Revision 0; and,
4) confirm that WCAP-17562-P, Revision 0 is not included in the current design basis.

Response:

RAI Part 1)

The criteria for maximum allowed wear was established by Westinghouse Electric Company (Westinghouse). The amount of wear that may be tolerated is related to insertability of the Control Rods within the Guide Tubes. Information pertaining to the study is contained in Westinghouse Proprietary Calculation Note CN-RIDA-09-103, Revision 0, "Guide Card Wear RCCA Control Rod Buckling Analysis," January 4, 2010. Information from CN-RIDA-09-103 was documented in Westinghouse Letter LTR-RIDA-09-234, Revision 0, "Evaluation of RCCA Guide Card Wear and Rodlet Jamming Analysis for Potential Issue PI-09-16," dated January 5, 2010. Acceptable levels of wear are documented in the following two Westinghouse WCAP reports.

- WCAP-17562-P, Revision 0, January 2012, "Westinghouse Pressurized Water Reactor Internals Guide Tube Guide Card Wear Criteria."
- WCAP-17451-P, Revision 0-A, "Reactor Internals Guide Tube Card Wear - Westinghouse Domestic Fleet Operational Projection."

The number of allowable open holes in adjacent cards due to wear stated in WCAP-17562-P, Revision 0, is the same as that stated in NSAL-10-1, "Rod Control Rod Assembly Guide Card Wear."

Use of an alternative justification that allows wear through a ligament in one or more guide cards is documented in WCAP-17096-NP, Revision 2, "Reactor Internals

Acceptance Criteria Methodology and Data Requirements," Appendix E, "Acceptance Criteria Methodology and Data Requirements for Westinghouse Components Included in MRP-227," for Component W-ID: 1, "Control Rod Guide Tube Assembly - Guide Plates (Cards)."

RAI Part 2)

The term "Constant Volumetric" refers to the methodology used to predict the wear progression at a guide card hole location. This methodology is documented in WCAP-17020-P, Revision 0, September 2009, "Point Beach Unit 1 Upper Internal Guide Tube - Guide Card Wear Evaluation." The values listed in the "Constant Volumetric" column indicate the additional EFPY of operation, after the time of inspection, needed to reach the wear criteria limit of W_3. W_3 refers to 85% of the rodlet diameter.

The term "Operation Curve" refers to an alternative methodology used to predict the wear progression at a guide card hole location. The basis for this methodology is contained in WCAP-17451-P, Revision 0-A. The values in this column indicate the additional EFPY of operation, after the time of inspection, to reach wear criteria limit of W_3. W_3 again refers to 85% of the rodlet diameter.

Wear projections using both methodologies were performed as part of the KPS inspection because the PWR owners group had not approved WCAP-17451-P, Revision 0-A at the time that the guide card wear evaluations were performed at KPS.

RAI Part 3)

The methodology used to inspect the guide cards relies on a fixture with geometry similar to the guide card used for calibration of the camera which is used for the inspection. The fixture contains characters similar to those on a Character Resolution Card, as outlined in ASME Section XI, IWA-2322, that are precisely known and have been validated by measurement at an independent laboratory. After removal of the drive rods, the camera is inserted into the guide tube.

A VT-3 inspection is performed and photographs are taken at each of the four critical hole locations on each of the nine guide cards located in the guide tube and at the top of the continuous section. All the guide cards and continuous section are video recorded.

The Westinghouse guide card wear measurement tool has an accuracy of +/- 0.01 inch, which satisfies the criteria noted in section 4.2 of WCAP-17562-P. A scaling factor is developed, related to the calibrated fixture parameters, and applied to the ligament measurements for each guide card hole. The resulting data (after the scaling factor is applied) is used for guide card hole wear projections.

The minimum number of guide tubes that require inspection at KPS per WCAP-17562-P is 22 (for allowing more than one open hole in adjacent guide cards). Since all 29 guide tubes were inspected at KPS, the use of the alternate criteria permitting more than one open hole in adjacent guide cards is acceptable. The technical basis for establishing that a minimum number of 22 guide tubes be inspected is to ensure a 95% probability of finding either at least two of three guide tubes with outlier wear or at least one of two guide tubes with outlier wear.

During the spring 2012 refueling outage, NRC inspectors from Region III observed DEK perform a portion of the KPS reactor vessel internals program. The NRC inspectors specifically observed inspection of the reactor vessel guide cards. As documented in NRC Inspection Report 05000305/2012-007 (Reference 10), the inspectors had no concerns with the observed activity.

RAI Part 4)

WCAP-17562-P, Revision 0, "Westinghouse Pressurized Water Reactor Internals Guide Tube Guide Card Wear Criteria" is not included in the current KPS design basis. WCAP-17562-P establishes guide card wear criteria for Westinghouse plants in response to Westinghouse Safety Advisory Letter NSAL-10-1, "Rod Control Rod Assembly Guide Card Wear." This new wear criteria was developed by Westinghouse (the original equipment manufacturer (OEM)) for management of guide card wear to ensure safe operation of the plant through insertibility of the control rods.

During the spring 2012 refueling outage inspections, guide card wear results were evaluated against the criteria in WCAP-17562-P. The evaluation confirmed that the criteria in WCAP-17562-P are not currently needed for assessing KPS guide card wear because the guide tubes meet all wear criteria projected over the next 10-year interval. Thus, even though the methodology in WCAP-17562-P was used in the wear assessment, existing wear does not exceed the threshold that would require use of the alternative justification in WCAP-17562-P for wear through ligaments in more than one guide card. Therefore, WCAP-17562-P is not included in the current KPS design basis.

<u>NRC Question ME7727-RAII-EVIB-Cher-019-2012-08-01</u>
(Follow-up Question to ME7727-RAII-EVIB-Cher-001-2012-05-09 - Action Item 1 of the NRC staff's SE)

The licensee in a letter dated June 28, 2012, provided a response to the NRC staff's question ME7727-RAII-EVIB-Cher-001-2012-05-09. The NRC staff reviewed the response and decided that the licensee did not adequately address the following issue:

A portion of NRC staff's question ME7727-RAII-EVIB-Cher-001-2012-05-09 reads as follows:

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:

Describe the process used to verify that the RVI components at KPS are bounded by the assumptions regarding the variable (i.e., 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.

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:

(i) 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:

1. *Lower Core Plate*
2. *Core Barrel Flange*
3. *Barrel-Former Bolts*
4. *Upper Core Barrel Welds*
5. *Lower Core Barrel Welds*
6. *Upper Core Plate Alignment Pins*

(1) The licensee did not provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for the aforementioned RVI components. An explanation is required as to how it was determined that the KPS values of neutron fluence, temperature, and, stress listed in Table A of the June 28, 2012, response to the subject (Cher-001) question fall within the same range as assumed for the "Typical Plant" of MRP-191.

(2) Was a neutron fluence analysis performed specifically for the KPS RVI? If so, provide the best estimate values for the sample components.

(3) Do the temperature ranges given for KPS represent fluid temperature only or the actual internal metal temperature of the components accounting for gamma heating effects?

Response:

RAI Part 1)

DEK has obtained confirmation from Westinghouse (the OEM) that the plant specific values for KPS fall with the same range as assumed for the "Typical Plant" of MRP-191 (Reference 6). As part of this confirmation, the OEM compared the reactor vessel internal components at KPS to the "Typical Plant" of MRP-191. The results of the comparison are documented in Westinghouse letter LTR-ARIDA-08-63, Revision 3 (Reference 4). The KPS plant assessment was performed by the same individuals at Westinghouse who developed MRP-191.

Correction to Table A

In the June 28, 2012 response to NRC Question ME7727-RAII-EVIB-Cher-001-2012-04-27 (Reference 2), DEK provided confirmation of certain information regarding plant specific reactor vessel (RV) internals components for KPS (in conjunction with Westinghouse Electric Company (Westinghouse)). A qualitative assessment of how the input parameters used in the FMECA and functionality analyses are bounded by the assumptions in MRP-227-A was provided in the response to that question. The response included a tabulation (Table A), which illustrated the input parameters for the typical Westinghouse PWR RV internals compared to those used for the KPS RV internals for the six sample components identified in the NRC question.

During preparation of the response to the followup question, DEK identified errors in the tabulated summary as follows:

- For RAI Items 2 and 6 (core barrel flange and upper core plate alignment pins), the table summary stated that the effective stress for these two items was greater than or equal to the threshold (for both the typical plant and for KPS). The table should have summarized that the effective stress for these two items was less than the threshold (for both the typical plant and for KPS).
- Several references to document MRP-191 were incorrectly listed as MRP-161.
- For RAI Items 4 and 5 (upper core barrel welds and lower core barrel welds), the table summary stated the estimated fluence range as $< 10^{20}$. This fluence range ($< 10^{20}$) is actually for the core barrel flange and outlet nozzles (which appears one row above the values for the core barrel in Table 4-6 of MRP-191). The estimated fluence range for the actual core barrel itself (per MRP-191, Table 4-6) is 1×10^{21} to 1×10^{22} (n/cm^2, E > 1 MeV). As documented in Westinghouse Letter WPS-11-55 (Reference 7), Table 1-6 and Table 1-7, a fluence analysis of the KPS core barrel projected the fluence values to be 5.72×10^{21} n/cm^2 on the ID surface and 2.74×10^{21} n/cm^2 (E > 1.0 MeV) on the OD surface, respectively, at 51.8 EFPY. These projected fluence values correlate favorably with the estimated fluence

range for the core barrel (1×10^{21} to 1×10^{22} [n/cm^2, E > 1 MeV]) that is listed in MRP-191.

- For RAI Item 5, the screening input parameter for the temperature of the typical plant's lower core barrel welds was listed as T-hot (same parameter as for the temperature of the upper core barrel welds). Whereas T-hot is the correct temperature parameter for the upper core barrel welds, the correct screening input temperature parameter for the lower core barrel welds is T-cold.

For ease of review, the entire table is provided below with the corrected parameters (as Revision 1 to Table A). The corrections are marked by revision bars.

As noted in the NRC question, DEK did not provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for the aforementioned RV internals components. With the exception of fluence values for Items 4 and 5 (discussed above), no analysis was performed to project these values for other RV internals components. The subject inspection plan will serve to suitably monitor these components.

Table A (Revision 1)
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress[5]		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-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$	Same as MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) $<10^{20}$ bounds KPS	Reference MRP-191 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 (Revision 1)
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress[5]		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS
3	Barrel-Former Bolts	Reference MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 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) 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. 316 SS or 347 SS	Reference WCAP-13266 Rev.1, Proprietary Class 2 347 SS
4	Upper Core Barrel Welds	Reference MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 1×10^{21} to 1×10^{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^{21} to 1×10^{22} bounds KPS[7]	Reference MRP-191 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 (Revision 1)
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress[5]		Materials	
		Typical[1] Plant	KPS[2]	Typical Plant	KPS[3]	Typical Plant	KPS	Typical Plant	KPS
5	Lower Core Barrel Welds	Reference MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants Estimated Fluence Range (n/cm2, E > 1 MeV) 1×10^{21} to 1×10^{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^{21} to 1×10^{22} bounds KPS[7]	Reference MRP-191 Table 4-6, Screening Input Parameters for Westinghouse-Designed Plants. T-cold	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-191 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-191 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-191 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 (Revision 1)
Review of sample components

RAI Item	Description	Parameters							
		Neutron Fluence		Temperature		Stress[5]		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 are depicted in MRP-191 Table 3-1, Stress Corrosion Cracking (SCC) Screening Criteria for PWR Internals Materials; 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. Stress threshold is 30 ksi.

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

7) Westinghouse Letter WPS-11-55, Table 1-6 and Table 1-7, list KPS projected core barrel fluence values on the ID surface (5.72×10^{21} (E > 1.0 MeV)[n/cm2]) and OD surface (2.74×10^{21} (E > 1.0 MeV)[n/cm2]), respectively, at 51.8 EFPY.

RAI Part 2)

Neutron fluence analysis was performed specifically for certain KPS RVI components. However, the analysis was only performed for a limited number of RVI components.

As discussed in the response to Part 1 of this question, Westinghouse had developed (and maintains) a fluence model of the KPS reactor vessel. The primary purpose of the model is to facilitate assessments for heatup/cooldown curves and pressurized thermal shock (PTS). The KPS plant specific model was not used to compare individual fluence values for the sample components in response to NRC staff's question ME7727-RAII-EVIB-Cher-001-2012-05-09. Instead, Westinghouse and DEK have verified that KPS falls within previous analysis assumptions. Specifically, the KPS reactor vessel was operated as a base loaded unit and is fully bounded by the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core loading patterns.

Westinghouse performed a plant specific assessment of the maximum fast neutron exposure (E > 1.0 MeV) at the inner and outer surfaces of the KPS core barrel. This assessment only analyzed the maximum neutron integrated exposure that would occur at the region of the core barrel exposed to the maximum neutron flux. The analysis projected that the inside and outside surface of the core barrel would reach a peak exposure of 5.72E+21 and 2.74E+21 (E > 1.0 MeV)[n/cm^2], respectively, at 51.8 EFPY. The end of the renewed operating license is projected to occur at 52.1 EFPY.

RAI Part 3)

The temperature ranges given for KPS RVI components represent the actual internal metal temperature of the components, accounting for gamma heating effects, as discussed in MRP-191, Table 4-6, "Screening Input Parameters for Westinghouse-Designed Plants," Footnote (a): Temperature rise due to gamma heating was considered for components in Fluence Regions 5 and 6 (1×10^{22} n/cm^2 and greater); these temperatures are indicated as > 608°F. MRP-191, Table A-1, "Results of Parameter Screening and Interviews with Analysts – Westinghouse Reactor Internals," provides a list of the components in Fluence Regions 5 and 6.

NRC Question ME7727-RAII-EVIB-Cher-020-2012-08-01
(Follow-up Question to ME7727-RAII-EVIB-Cher-013-2012-05-09)

In its response dated June 28, 2012, the licensee stated that in lieu of revising the inspection plan (addressed in AMP KLR-1309A), it will revise the RVI inspection plan that is included in Tables 1 and 2 of the December 12, 2011 submittal. The NRC staff does not agree with this disposition because the AMP KLR-1309A will not be consistent with the plant-specific application of MRP-227-A and this inconsistency can cause confusion among the inspectors.

(1) Therefore, the NRC staff requests that the licensee submit a corrected version of the AMP KLR-1309A.

Response:

The Tables in the December 12, 2011 submittal (Reference 3) were extracted from the inspection plan addressed in AMP KLR-1309A. These Tables are, in essence, the pending changes for AMP KLR-1309A. DEK is awaiting final NRC approval of the inspection plan prior to formally revising AMP KLR-1309A.

References:

1. Email from Karl D. Feintuch (NRC) to Jack Gadzala (DEK) et al, "RE: ME7727 - Kewaunee - Draft Request for Additional Information Re: RVI components Inspection Plan - RAII-Cher-018 to -020," dated August 1, 2012.
2. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request, Supplement and Response to Request for Additional Information," dated June 28, 2012.
3. Letter from J. Alan Price (DEK) to Document Control Desk (NRC), "Reactor Vessel Internals Inspection Plan Review Request," dated December 12, 2011.
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. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175) – EPRI Report 1012081, 2005.
6. 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).
7. Westinghouse Electric Company (Westinghouse) letter WPS-11-55, "Dominion Energy Kewaunee, Kewaunee Power Station, Kewaunee Surveillance Capsule A Location and Withdrawal Schedule Evaluation," dated December 15, 2011.
8. 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.

9. 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.

10. Letter from Ann Marie Stone (NRC) to David A. Heacock (Dominion), "Kewaunee Power Station – NRC Post Approval Site Inspection for License Renewal Inspection Report 05000305/2012-007," dated May 16, 2012.