NRR-PMDAPEm Resource

From: Feintuch, Karl
Sent: Thursday, May 17, 2012 9:40 AM
To: Feintuch, Karl
Subject: FW: ME7727 - Kewaunee - Request for Additional Information Re: RVI components Inspection Plan
Attachments: ME7727 RAI set 2012-04-27.docx

Licensee has requested due date be changed to 6/8/2012. Reviewer has consented.

From: Feintuch, Karl
Sent: Monday, April 30, 2012 12:05 PM
To: Jack Gadzala; 'Craig D Sly'
Cc: Cheruvenki, Ganesh; Hiser, Allen; Purtscher, Patrick
Subject: ME7727 - Kewaunee - Request for Additional Information Re: RVI components Inspection Plan

By letter dated December 12, 2011, Dominion Energy Kewaunee, Inc. (DEK, the licensee), submitted an inspection plan for the reactor vessel internals (RVI) components at Kewaunee Power Station (KPS). In its e-mail submittal dated March 19, 2012, the licensee submitted the Technical Report KLR-1309A, "License Renewal Project, Aging Management Program, ASME Section XI, In-service Inspection, Subsection IWB, IWC, and IWD, Reactor Vessel Internals Inspection, Kewaunee Power Station," to the NRC staff for review. Pursuant to the license renewal commitments items 1 and 2 addressed in Chapter 15, Table 15.7.1, "License Renewal Commitments," of the Updated Final Safety Evaluation Report (UFSAR), the licensee requested that the NRC staff review and approve the subject inspection plan.

The NRC staff from the Vessel and Internals Integrity Branch (EVIB) has reviewed the inspection plan for the KPS's RVI components and requests additional information from DEK as described in the attachment.

We seek a clarification call to discuss these items, as needed, and to confirm that the stated "request by" dates are mutually agreed. With that confirmation the draft items would become firm for response

Please contact me with any questions.

Karl Feintuch
Project Manager
USNRC
301-415-3079

Hearing Identifier: NRR_PMDA
Email Number: 377

Mail Envelope Properties (26E42474DB238C408C94990815A02F097B32B81C6E)

Subject: FW: ME7727 - Kewaunee - Request for Additional Information Re: RVI components Inspection Plan
Sent Date: 5/17/2012 9:39:44 AM
Received Date: 5/17/2012 9:41:00 AM
From: Feintuch, Karl

Created By: Karl.Feintuch@nrc.gov

Recipients:
"Feintuch, Karl" <Karl.Feintuch@nrc.gov>
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DRAFT REQUEST FOR ADDITIONAL INFORMATION (RAI)
VERSION 2012-04-27
RELATED TO LICENSEE'S REACTOR VESSEL INTERNALS INSPECTION PLAN REVIEW REQUEST
KEWAUNEE POWER STATION (TAC NO. ME7727)
DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated December 12, 2011, Dominion Energy Kewaunee, Inc. (DEK, the licensee), submitted an inspection plan for the reactor vessel internals (RVI) components at Kewaunee Power Station (KPS). In its e-mail submittal dated March 19, 2012, the licensee submitted the Technical Report KLR-1309A, "License Renewal Project, Aging Management Program, ASME Section XI, In-service Inspection, Subsection IWB, IWC, and IWD, Reactor Vessel Internals Inspection, Kewaunee Power Station," to the NRC staff for review. Pursuant to the license renewal commitments items 1 and 2 addressed in Chapter 15, Table 15.7.1, "License Renewal Commitments," of the Updated Final Safety Evaluation Report (UFSAR), the licensee requested that the NRC staff review and approve the subject inspection plan.

The NRC staff from the Vessel and Internals Integrity Branch (EVIB) has reviewed the inspection plan for the KPS's RVI components and requests additional information from DEK as described below.

In the text that follows type fonts used for paragraphs (not just for headings or titles) differentiate the source of the text:

- ***Bold Italic font*** is used for text quoted from the NRC pertaining to document MRP-227 and MRP-227-A
- *Italics* (not bolded) is used for RAI responses pertaining to MRP-227.
- Standard font (not bolded and not italics) is used for content pertaining to ME7727 and this RAI.

2.0 SCOPE OF THIS REQUEST

This RAI accounts for items asked in earlier versions and earlier increments. It also resets the "request-by" date to May 25, 2012 for all requested items. Items earlier requested as Cher-001 and Cher-002 are withdrawn and replaced by Cher-015 and 016; Cher-003 through Cher-014 have not changed. The full text of the requests is included in this version.

The request consists of 14 requested items and 2 withdrawn items:
ME7277-RAII-EVIB-Cher-001-2012-04-27 – now withdrawn and replaced
ME7277-RAII-EVIB-Cher-002-2012-04-27 – now withdrawn and replaced

ME7727-RAII-EVIB-Cher-003-2012-05-25
ME7727-RAII-EVIB-Cher-004-2012-05-25
ME7727-RAII-EVIB-Cher-005-2012-05-25
ME7727-RAII-EVIB-Cher-006-2012-05-25
ME7727-RAII-EVIB-Cher-007-2012-05-25
ME7727-RAII-EVIB-Cher-008-2012-05-25
ME7727-RAII-EVIB-Cher-009-2012-05-25
ME7727-RAII-EVIB-Cher-010-2012-05-25
ME7727-RAII-EVIB-Cher-011-2012-05-25
ME7727-RAII-EVIB-Cher-012-2012-05-25
ME7727-RAII-EVIB-Cher-013-2012-05-25
ME7727-RAII-EVIB-Cher-014-2012-05-25
ME7277-RAII-EVIB-Cher-015-2012-05-25
ME7277-RAII-EVIB-Cher-016-2012-05-25

Within these tracking numbers:
EVIB = Vessel and Internals Integrity Branch
Cher = Reviewer Cheruvenki
00n = 001, 002 and corresponding to the RAI items (RAII) shown below.
2012-05-25 = May 25, 2012, which is subject to confirmation with DEK that the referenced items (RAII) are clear to enable response.

3.0 GENERAL BACKGROUND

The NRC staff approved MRP-227-A in ADAMS Accession No. ML111600498. Regarding Action Item 1, the NRC staff wrote:

> ***4.2 Plant-Specific Action Items***
>
> ***4.2.1 Applicability of FMECA and Functionality Analysis Assumptions***
>
> ***As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee should refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1.***

This action item was developed because in the following responses to the staff's RAIs, the MRP stated that -- These evaluations were based on representative configurations and operational histories, which were generally conservative, <u>but not necessarily bounding in every parameter.</u> [emphasis by NRC staff]

Consequently for TAC ME7727, the NRC staff issued "RAI Set 1 of 2 (Rev1)." That request consisted of two items:
ME7277-RAII-EVIB-Cher-001-2012-04-27, now withdrawn and replaced by ME7277-RAII-EVIB-Cher-015-2012-05-25, and
ME7277-RAII-EVIB-Cher-002-2012-04-27, now withdrawn and replaced by ME7277-RAII-EVIB-Cher-016-2012-05-25

4.0 REQUEST FOR INFORMATION ITEMS (RAII)

Subsequently, the NRC staff issued eleven more items designated Cher-003 through Cher-013 as "RAI Set 2 of 2." Next, Cher-014 was prepared. Finally, Cher-015 and Cher-016 were prepared to replace Cher-001 and 002.

The requested RAI items start with Cher-003:

RAI-3 (ME7727-RAII-EVIB-Cher-003-2012-05-25):

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.

Please provide the missing items.

RAI-4 (ME7727-RAII-EVIB-Cher-004-2012-05-25):

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.

RAI-5 (ME7727-RAII-EVIB-Cher-005-2012-05-25):

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,

(b) baffle-edge bolts,

(c) clevis insert bolts,

(d) flux thimble tubes,

(e) core barrel bolting, and

(f) thermal shields.

RAI-6 (ME7727-RAII-EVIB-Cher-006-2012-05-25):

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

RAI-7 (ME7727-RAII-EVIB-Cher-007-2012-05-25):

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.

RAI-8 (ME7727-RAII-EVIB-Cher-008-2012-05-25):

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:

(1) most susceptible areas to experience aging degradation,

(2) high stress areas, and,

(3) plant-specific operating experience.

RAI-9 (ME7727-RAII-EVIB-Cher-009-2012-05-25):

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.

RAI-10 (ME7727-RAII-EVIB-Cher-010-2012-05-25):

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.

RAI-11 (ME7727-RAII-EVIB-Cher-011-2012-05-25):

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.

RAI-12 (ME7727-RAII-EVIB-Cher-012-2012-05-25):

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.

RAI-13 (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 the MRP-227-A is not included
Upper Core Barrel Flange Weld	Note 4 in Table 4-3 of the MRP-227-A is not included
Upper and Lower Core Barrel Cylinder Girth Welds	These welds are not included (Reference Table 4-3 of the MRP-227-A report)
Lower Core Barrel Flange Weld	These welds are not included (Reference Table 4-3 of the MRP-227-A report)
Baffle-Edge Bolts	Note 3 in Table 4-3 of the 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 the MRP-227-A)
Upper Core Plate	This component is not included (Reference Table 4-6 of the MRP-227-A report)
Lower Support Forging/Casting	These components are not included (Reference Table 4-6 of the 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 the 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 the MRP-227-A)
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 the 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 the MRP-227-A)

RAI-14 (ME7727-RAII-EVIB-Cher-014-2012-05-25)

RAI-14 addresses the emphasized text in the quoted “Section 4.2.1,” within Section 3.0 “GENERAL BACKGROUND” above.

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.

RAI-15 (ME7277-RAII-EVIB-Cher-015-2012-05-25)

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 document, not available to the public),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 document, not available to the public) 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) 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.

(2) 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) Provide a qualitative assessment regarding the differences between the plant-specific variables (neutron fluence, temperature, stress values, and materials) and the variables of a “representative” PWR vessel used in developing the MRP-227-A report, for those components listed in part a) or for those components that are either identified as “Expansion” or were scoped out in the FMECA.

(3) 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.

(4) 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.

(5) 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.

RAI-16 (ME7277-RAII-EVIB-Cher-016-2012-05-25)

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.

5.0 DETAILED BACKGROUND REGARDING PLANT SPECIFC APPLICABILITY

ADAMS Accession Numbers ML12017A191 and ML12017A192 contain the RAI items for MRP-227 from which the NRC staff performed its safety evaluation. Among all RAI items the following selected RAI items and Responses relate to plant-specific applicability:

> ***RAI 2-11:***
> ***Following on to RAI-10, additional aspects of the TR MRP-227 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The NRC staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.***
>
> ***RAI 2-11 Response by Pressurized Water Reactor (PWR) Nuclear Steam Supply System (NSSS) Vendors (all Responses that follow are from the same group of vendors):***
> *With respect to the first potential action item, guidance on regulatory submittals is outside the scope of MRP-227; however the MRP is willing to work with the NRC to help as requested. With respect to the second potential action item, Section 2.4 of MRP-227 states explicitly that*
>
> > *"The guidelines are intended to serve as the primary basis for owner preparation of a reactor internals AMP in accordance with the*

requirement cited in Section 7. It is beyond the scope of the guidelines, however, to ensure the satisfaction of every plant-specific license renewal or power uprate commitment. Plant-specific commitments remain the responsibility of the owner."

Licensee action items for a typical plant will relate to guidelines development assumptions. Section 2.4 of MRP-227 contains a list of the assumptions that need to be verified as applicable by individual plant owners. These assumptions are cited here for completeness.

"The guidelines are based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the U.S. domestic fleet of PWRs. The functionality analyses and supporting aging management 21 *strategies in MRP-231 [13] and MRP-232 [14] provide the basis for these guidelines. These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. [emphasis added by NRC staff]*
General assumptions used in the analysis include:

- *30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation;*
- *base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and*
- *no design changes beyond those identified in general industry guidance or recommended by the original vendors*

These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified. Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines."

Therefore, all of the important assumptions underlying the MRP-227 methodology were aggregated into Section 2.4 of the document [referring to MRP-227].

RAI 2-18:
Clarify how plant-specific differences were considered within the FMECA. Discuss whether any additional plant-specific analyses are required, either as a supplement to TR MRP-227 or as identified plant-specific action items, in order to assure that FMECA analysis supporting the TR MRP-227 program is applicable to a given facility.

RAI 2-18 Response:
During the development of the FMECA, a conservative approach was taken. Differences in internals design or operation were considered and factored into the

expert panel's qualitative assessments for determining potential susceptibility of degradation and reduced capacity to perform intended functions. Where critical to final categorization of components, these considerations are captured in the text of the documents. MRP-190 for Babcock & Wilcox (B&W) design and MRP-191 for Combustion
Engineering and Westinghouse designs provided a detailed discussion of how design variances were considered. The failure modes, effects, and criticality analysis (FMECA) approach used was not a quantitative probabilistic risk analysis, but rather a semiquantitative approach with expert elicitation. The experts were instructed to consider how design and operational differences affected the evaluation. MRP-190 Section 5 and MRP-191 Section 6 provide description of the categorization process using the FMECA for the designs considered in MRP-227.

The applicability of the evaluations in MRP-190 and MRP-191 to specific plants was noted in the text of the documents. In addition, Section 4 of MRP-190 stated that the assumptions "are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed units." While a similar statement was not contained in MRP-191, the intention is the same and the applicability of the outcome of the process to the operating fleet was documented in the listing of plants considered and process of evaluation especially noting the grouping approach to addressing the design variances. The inspection requirements included for individual components, noting specifically Combustion Engineering designs for bolted or welded configurations, were contained in MRP-227.

Section 1 of MRP-227 stated that the contents were applicable to the currently operating pressurized water reactors (PWRs) as of the date of publication. The demonstration of the applicability of MRP-227 to individual units was specified in Section 2.4. Design changes that may have occurred subsequent to May 2007 are managed by the plant configuration control process.

RAI 2-19:
Discuss how a licensee will demonstrate adherence to the reference core loading pattern on a unit-specific basis. Address plant-to-plant variability in neutron flux at various peripheral core locations. Confirm, based on significant operating experience, that "low leakage" core designs, when normalized by power density, have peripheral neutron fluxes that are consistently within the estimates for the generically studied plants.

RAI 2-19 Response:
The core loading patterns used in the MRP-227 reference documents were chosen to represent known operating practice, they are not intended to be used as a reference for plant-specific analysis. The intention of using the representative core loading patterns was not to bracket operation, but to perform an analysis that demonstrates both historic and current fuel management programs. The MRP-227 inspection recommendations based on these calculations are robust and do not require the utility to perform additional analysis of core loading patterns to qualify their applicability. The condition of the internals at the time of the first required inspections is dominated by the power distribution used to represent the first thirty years of full power operation. During this period the analysis assumed that the fresh fuel was loaded in the peripheral fuel assemblies. This "out-in" loading

pattern produced results in relatively high heat loadings and neutron fluences in the near core structure. In practice all plants in the United States abandoned fuel management based on the "out-in" loading prior to thirty years of operation. There are no current or planned fuel management programs that would result in more deleterious conditions than those assumed in this analysis during the first thirty years of operation. For this reason there is no reason to require any plant to perform an analysis to demonstrate adherence to the assumed core loading pattern prior to performing the first round of inspections. The timing and extent of the first round of MRP-227 examinations is governed by damage that has already been accumulated.

The representative power distributions used for the simulation of years 31 to 60 incorporate the effects of aggressive power uprate programs. Qualification of the core loading pattern is considered in the design analysis for the plant uprate. Although it is not possible to anticipate all possible future options, both current fuel management practice, which maximizes fuel utilization, and concerns about neutron damage in the reactor pressure vessel preclude return to the practice of loading fresh fuel in the periphery locations. It is unlikely that future core loading patterns would invalidate the assumptions of the analysis.

Although the shift from "out-in" core loading patterns to low-leakage patterns resulted in a sharp decrease in the peak temperature in the internals structure, the shift had minimal effect on the location of the peak temperature or the character of the peak damage. There is no reason to expect that changing the loading pattern would change the base inspection recommendations. The MRP-227 recommendations are based on reasonable assumptions about the effects of power uprates. In many cases power uprates can be accomplished without significantly increasing the heat or neutron loading to the internals. Return to the more aggressive core loading patterns could conceivably result in a decrease in the reinspection interval. However, there is no reason to anticipate any change of this scale.

MRP-227 is intended to be a living document. The MRP will monitor both inspection results and plant operating experience and make appropriate modifications. There is currently no need to require plants to demonstrate adherence to any reference core loading practice.

RAI 3-6:
Clarify whether or not the existing methodology in TR MRP-227, Rev. 0, can be applied to a PWR facility whose reactor core loading pattern operating history is not bounded by the assumptions in the report. If the methodology can be applied, justify why that is the case. If the methodology cannot be applied to these PWRs, identify what actions a licensee with a nonconforming PWR would have to take in order to develop a plant-specific AMP for its RVI components, which is consistent with the intent of TR MRP-227, Rev. 0. Identify whether license renewal applicants should demonstrate that their facility's reactor core loading pattern operating history is bounded by the assumptions in the report as part of the license renewal application (i.e., should be a license renewal applicant action item).

RAI 3-6 Response:
Similar concerns were expressed in RAI 2-19 from the 8/24/09 inquiries. Based on the previous discussion the following conclusions may be drawn:

(1) Section 2.4 of MRP-227 states: "The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified."

(2) Section 2.4 further states: "Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines.

(3) The strategies in MRP-227 do not assume that the core loading patterns used in the analysis are bounding.

(4) The inspection recommendations are robust and there is no reason to anticipate plant modifications that would impact the MRP-227 requirements.

(5) To apply MRP-227, the license renewal applicant needs to demonstrate that core loading patterns going forward are reasonably represented by the assumptions of the report.

(6) MRP-227 is a living document and the industry will monitor any trends in operating practice that might impact the MRP-227 recommendations.

RAI 3-8:
In Section 2.4 of TR MRP-227, Rev. 0, the MRP assumes that the design of a PWR plant applying the TR MRP-227, Rev. 0, methodology would not include any design changes beyond those identified in either general industry guidance or recommended by the original vendors. The NRC staff is aware that many of the licensees owning PWR facilities have been granted license amendments to implement measurement uncertainty recapture (MUR) power uprates, stretch power uprates, or extended power uprates for their facilities. However, it is not evident to the NRC staff whether any design changes associated with these type of power uprates would be within the scope of the MRP's term "design changes identified in general industry guidance or recommended by original vendors." Clarify whether design changes that will need to be implemented in order to receive NRC approval of a MUR, stretch, or extended power uprate, or that have been implemented as a result of receiving NRC approval of a power uprate, are within the scope this type of assumption.

RAI 3-8 Response:
The TR MRP-227, Rev. 0 recommendations were based on evaluations relevant to current plant operating experience at the time the report was issued. This operating experience includes measurement uncertainty uprates, stretch power uprates and extended power uprates. Therefore, all three uprate types are considered within the scope of "design changes identified in general industry guidance or recommended by original vendors." As it is not possible to anticipate all possible future plant uprates or modifications, MRP-227 clearly states that "Plant-specific commitments remain the responsibility of the owner." Average core power must be

increased to implement a plant uprate. This increase in power necessarily implies an increase in the average neutron flux in the core. However, for those reactor internals components that are subject to neutron radiation damage, the neutron exposure tends to be determined by power levels in the peripheral fuel assemblies, rather than the core average power. The original "out-in" core loading patterns used in many plants produced relatively flat core power distributions. These flat power distributions lead to higher neutron leakage at the edges of the core. In addition to causing high damage rates in the internals, this leakage results in increased neutron exposure of the reactor pressure vessel and relatively poor fuel utilization. Current core design practice utilizes a "low leakage" loading that tends to reduce power levels in the peripheral assemblies, which in turn reduces the neutron exposure of the reactor internals.

While plant uprates may lead to some increase in neutron exposure of the internals, these are increases of an exposure level already reduced by the core loading, and they are generally moderate compared to the overall increase in power level. Measurement uncertainty recapture uprates take advantage of improved power monitoring systems to allow the plant to increase power inputs by operating closer to the plant allowables. As these uprates remain within the original design basis of the plant, there is no reason to believe that a measurement uncertainty recapture uprate would fall outside the scope of the MRP-227 recommendations.

Stretch power uprates take advantage of excess margins that are buried in the plant operating limits. In many cases, a plants operating below the true plant capacity can demonstrate safe operation at higher power outputs by removing overly conservative limits. In most cases, the stretch power uprates result in power increases in an individual plant, but do not move the plant outside the envelope of fleet operating experience. Therefore, there is no reason to believe that a stretch power uprate would fall outside the scope of the MRP-227 recommendations.

Extended power uprates produce the largest increases in plant power. An extended power uprate may rely on both more detailed analysis of plant operation and upgrades to plant equipment. The experience base considered in support of the MRP-227 recommendations included plants with extended power uprates. The core power distributions used in the modeling of irradiation-induced aging of the core baffle and shroud structures included a typical extended power uprate. It is the responsibility of the plant owner to demonstrate that the changes in plant operation are consistent with the general assumptions of MRP-227.

The finite element-based aging analysis of the core baffle-formers and core shroud completed in support of the MRP-227 guidelines were never intended to provide bounding plant results. The recommendations are robust and not dependent on the details of the analysis. The assumption that the representative plant operated for thirty years with "out-in" core loading patterns has a large impact on the results. Most plants moved away from this aggressive core loading pattern much earlier in plant life. The effects of this conservative assumption about plant operating history are generally larger than any potential effect of plant uprate.

RAI 4-6:
Several previous RAIs (e.g., RAIs 11 and 18 (Set #2), and RAI 3-8) have questioned whether plant-specific analyses are required to demonstrate that each plant is appropriately represented by MRP-227 such that the proposed aging management programs (AMPs) are applicable. That is, confirmation that the plant's initial design and operating conditions fall within the scope of the MRP-227 evaluation, the plant complies with important assumptions underlying the MRP-227 analysis, and changes in plant design or operating conditions (e.g., resulting from power uprates) have been appropriately considered. Meeting these conditions is necessary to ensure that the plant-specific AMP inspection requirements (i.e., the primary inspection components, inspection type and periodicity) would not be different from the MRP-227 recommendations determined through more generic evaluation.

The responses to these various RAIs have indicated that a plant-specific analysis to demonstrate the applicability of MRP-227 guidance is not required because plant-specific differences have been considered by: (a) evaluating operating experience throughout the commercial fleet; (b) using a conservative "out-in" core loading pattern in the functionality analysis; and, (c) assessing several known plant-specific conditions in the FMECA. The responses also justify the representativeness of MRP-227 because (a) base load operational profiles (i.e., fixed power levels) are similar among plants, and (b) no design changes have been enacted by plants other than those identified in generic industry guidance or recommended by the original nuclear steam system supply vendors. However, given the variability in design and operational conditions that currently exists in PWR plants, the staff is not convinced that that the MRP-227 AMP requirements are necessarily appropriate for each plant. For instance, it is not clear that the "out-in" core loading pattern is conservative given that some degradation mechanisms do not initiate until low-leakage core conditions are imposed in the functionality model.

Therefore, the staff requests that guidance be developed that will allow individual licensees to assess the applicability of the MRP-227 method and results. This guidance should particularly focus on demonstrating the applicability of (a) the FMECA and functionality assessments, and (b) the recommended inspection category, inspection method and periodicity for each component. Specifically, this guidance should allow a licensee to determine if plant-specific differences in the RVI design or operating conditions (i.e., power uprate level) result in different component inspection categories (i.e., primary, expansion, existing, and no additional measures) than recommended within MRP-227. Alternatively, additional analysis or justification may be provided to demonstrate that the MRP-227 approach and results are generically applicable such that plant-specific differences in the RVI design or operating conditions do not result in different component inspection categories then recommended within MRP-227.

In that absence of adequate guidance, the staff will consider the need to implement limitations and conditions on the use of MRP-227 which would address plant-specific action items necessary to address this issue for each facility.

RAI 4-6 Response:
The starting point for the response to this RAI is from Section 2.4 (Guidelines Applicability) of MRP-227, which have been cited in previous RAI responses – The last two paragraphs state that:

> *"These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.*
>
> *Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines."*

These two paragraphs are based on the review and assessment by vendors that: (1) even though power uprates were not specifically addressed in the representative plant component functionality analyses, all plant uprates and other plant modifications up until May 2007 were considered to be within the envelope of the representative plant analysis results; and (2) no inspection recommendation cited in MRP-231 and MRP-232 would have been altered by a change in the functionality assumption of an earlier conversion over to a low-leakage core loading pattern. The first of these vendor findings is covered by the last paragraph, which clearly states the action required by a plant that has sought a power uprate or has undergone a significant plant modification as of May 2007. No further guidance is needed on the power uprate or major plant modification issue. The second of the vendor findings is not intended to argue that degradation mechanisms only initiate during high-leakage core loading operation, or that degradation effects cannot worsen during low-leakage core loading operation. The finding is simply that the vendors have reviewed the functionality analysis results and have determined that the recommended inspection requirements would not be altered by a change in functionality analysis assumption to an earlier conversion from high-leakage to low-leakage core loading. This core loading assumption only has relevance for those components which were aged and assessed using the detailed irradiation analysis finite element analysis (FEA) model. The aging analyses were conducted to understand the complex interactions between active degradation mechanisms in highly irradiated components. These detailed modeling efforts were applied to the B&W and Westinghouse baffleformer-barrel structures, and a welded CE core shroud assembly. The intent of the irradiation aging analysis was to identify trends and limits in the component behavior.

The analysis was used to identify factors that could potentially cause component failure. Representative plant designs with relatively severe irradiation conditions were selected for the irradiated aging analysis. These conditions were chosen to test the capability of the structure and identify points of potential concern. While the results of the FEA work provided insights into where degradation would most be expected, neither the vendors nor other members of the core writing team pinpointed the recommended examination scope solely based on the results of

irradiation aging analysis. Instead, the irradiation aging analysis results were combined with engineering judgment and experience to provide examination recommendations. The only limited scope recommendations confirmed by the FEA results were for the CE welded designs where the most highly stressed and irradiated weld seams are specified. Thus, while another damage mechanism could play a more important role in the overall aging of the components when a more realistic core loading history is employed, in no case would the recommendations to detect that degradation change because:

- *the anticipated effect and the overall degradation hierarchy would not change and*
- *no component or component assemblies have inspection requirements directed at local effects predicted in the FEA results to the extent that a shift in degradation mechanism predominance would necessitate a change in location recommendations.*

An excellent example of this is provided by baffle-to-former bolts in B&W and Westinghouse plant designs, where the effects of irradiation-induced stress relaxation of bolt pre-load has been observed to reduce the potential susceptibility to IASCC for the baffle-to-former bolt with the highest radiation exposure, while a baffle-to-former bolt with somewhat lower radiation exposure (somewhat further from the core) would be more susceptible. This shift of susceptibility to baffle-to-former bolts further from the core does not lead to a condition where core barrel-to-former bolts are more susceptible to IASCC than baffle-to-former bolts and, since the examination recommendation is for examination of 100% of the accessible baffle-to-former bolts, no change in the examination recommendation is warranted. Therefore, the additional level of detail provided by the functionality analysis does identify complex structural interactions, particularly in bolted assemblies, but did not lead to recommendations for changing the scope of examinations.

While the MRP agrees with the staff that a wide variety of designs are addressed by the representative plants particularly the Westinghouse and CE designs, the plant designs selected do correspond more closely with plants with earlier implementation dates for the MRP-227 requirements.

In addition, as plants begin the implementation process for Issue Program (IP) guidance, such as implementation of MRP-227 guidance, the responsibility for reviewing and determining the applicability of the explicit assumptions given in Section 2.4 are well understood, as outlined in NEI 03-08, including either the need or the wish to use valid alternatives through the deviation process. Thus it would be a plant-specific action to confirm the guidance in MRP-227 remains applicable within the boundaries indicated in Section 2.4. The general framework for both the determination of applicability and the process for justifying deviations is described in industry documents and is further discussed in the response to RAI 4-7.