CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION

REQUEST FOR ADDITIONAL INFORMATION (RAI) Set #11 and Set #12 RESPONSES

Set #11 Responses

RAI 3.1.2.2.11-1

Background:

In LRA Section 3.1.2.2.11, potential cracking of the steam generator (SG) tube-to-tubesheet welds and divider plate assembly is discussed. The SG tubesheet cladding is Alloy 182, a material that is potentially susceptible to primary water stress corrosion cracking (PWSCC). The SG tubes are welded to the cladding (an autogenous weld) and the divider plate assembly is welded to the tubesheet cladding. As a result, Commitments Nos. 34 and 35 were made regarding the performance of analyses or inspections of the tube-to-tubesheet welds and the divider plate.

Requests:

a) Regarding the tube-to-tubesheet weld Commitment No. 35:

  i. Please discuss your plans to modify your commitment to indicate that the technical basis for the redefinition of the reactor coolant pressure boundary (RCPB) will be submitted to the staff for review and approval as part of the license amendment process prior to redefining the RCPB.

  ii. In addition, please discuss whether an analytical evaluation may be performed to assess whether the welds are susceptible to PWSCC. If it is determined that the welds are not susceptible to PWSCC, and the staff agrees with this determination, discuss your plans for using this as the basis for not performing inspections of the welds. Please discuss your plans to modify your commitment to reflect your response to this question.

  iii. Please confirm that the inspection technique(s) used to inspect the welds will be capable of detecting PWSCC. In addition, discuss your plans to modify your commitment to reflect that the inspection technique(s) will be capable of detecting PWSCC.

b) Regarding the divider plate Commitment No. 34:

  i. Please clarify the configuration of your divider plate assembly. In particular, clarify if the divider plate assembly (i.e., the stub runner) is welded to the carbon steel tubesheet or the tubesheet cladding. Clarify whether the divider plate is welded to the stainless steel channel head cladding or the low alloy carbon steel shell. In addition, clarify the weld material for the welds.

  ii. Discuss your plans for clarifying your commitment that the inspection technique used will be qualified to detect cracking in the divider plate assembly given that cracks have been observed outside the weld region (i.e., in the heat affected zone).

  iii. Clarify the frequency for this inspection, or if it is a one-time inspection.

  iv. Please discuss plans to remove the last two options in your commitment since, if such analyses become available, they could be submitted to the staff for review and if approved, may serve as a basis for revising the commitment. The staff notes that if additional analyses become available that demonstrate pressure boundary integrity is adequately maintained with divider plate weld cracking, or if studies indicate that failure of the pressure boundary is not a concern, then the commitment may be revised at that time.

c) Regarding both the divider plate and tube-to-tubesheet weld Commitments Nos. 34 and 35,

i. Please discuss your plans to include the tube-to-tubesheet weld and divider plate commitments in your Updated Final Safety Analysis Report (UFSAR) supplement.

ii. Please discuss your plans for revising the commitment to specify that an inspection of each SG, to assess the condition of the divider plate assembly and tube-to-tubesheet weld, will be performed during a specific time period (e.g., the inspections will be performed no earlier than three years prior to the period of extended operation and no later than two years after entering the period of extended operation). The actual timeframe chosen should consider the amount of operating time on the SG.

Callaway Response

a) Regarding the tube-to-tubesheet weld Commitment No. 35:

i) Amendment 13 in Enclosure 2 has revised option 2 of LRA Table A4-1, Item #35, to add "The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC."

ii) Amendment 13 in Enclosure 2 has revised option 2 of LRA Table A4-1, Item #35, to add the following in option 2:

- Perform an analytical evaluation of the steam generator tube-to-tubesheet welds to determine the susceptibility to PWSCC.
- Submit the analysis for NRC review if it is determined that the welds are not susceptible to PWSCC and do not require inspection.

iii) Amendment 13 in Enclosure 2 has revised option 1 to insert, "The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds."

b) Regarding the divider plate Commitment No. 34:

i) Based on Callaway steam generator design drawings, there is no stub runner at interface between the divider plate (called partition plate in Callaway steam generator design drawings) and the tubesheet of the steam generators. There is also a small rectangular Alloy 690 closure plate (also known as a filler plate on other steam generator designs) installed at the Alloy 690 divider plate to stainless steel clad primary head and Alloy 82/182 clad tube sheet junctions. The most stressed portions of the channel head in contact with the primary coolant are welded with Alloy 152. The most stressed areas are associated with the divider plate to primary head and tube sheet junctions. The Callaway steam generator design drawings show the following weld materials and buttering materials:

Divider plate to tubesheet cladding: Alloy 152 weld and buttering materials
Divider plate to closing plate: Alloy 152 weld materials
Divider plate to primary head cladding: Alloy 152 weld material
Closing plate to tubesheet cladding: Alloy 152 weld and buttering materials
Closing plate to tubesheet ring: Alloy 152 weld and stainless steel buttering
Closing plate to primary head cladding: Alloy 152 weld materials

ii) Amendment 13 in Enclosure 2 has revised option 1 of LRA Table A4-1, Item #34, to clarify the inspection will include the divider plate assembly and associated welds.

iii) Amendment 13 in Enclosure 2 has revised option 1 of LRA Table A4-1, Item #34, to specify a one-time inspection will be performed.

iv) Amendment 13 in Enclosure 2 has revised option 2 of LRA Table A4-1, Item #34, to submit the analytical evaluation of the divider plate welds to the NRC for review and approval. Option 3 of LRA Table A4-1, Item #34 currently includes a consideration for an NRC study confirmation that failure of the reactor coolant pressure boundary due to PWSCC cracking of steam generator plate welds is not a credible concern. Industry studies are in progress to evaluate the potential for divider plate crack growth and develop an industry-applied resolution to the concern through the EPRI Steam Generator Management Program (SGMP) Engineering and Regulatory Technical Advisory Group, which are scheduled to be complete in 2013.

c) Regarding both the divider plate and tube-to-tubesheet weld Commitments Nos. 34 and 35:

i) Amendment 13 in Enclosure 2 has revised LRA Section A1.9 to add the descriptions of LRA Table A4-1 Items #34 and #35. Amendment 13 in Enclosure 2 has also revised LRA Section 3.1.2.2.11 to be consistent with the changes to LRA Table A4-1 Items #34 and #35.

ii) Amendment 13 in Enclosure 2 has revised LRA Section A1.9 and LRA Table A4-1, Item #34 and #35 to reflect an implementation schedule between Fall 2025 and Fall 2029 when the RSGs have been in service for more than 20 years.

Corresponding Amendment Changes

Refer to the Enclosure 2 Summary Table "Amendment 13, LRA Changes from RAI Responses", for a description of LRA changes with this response.

RAI 3.1.2.4-1

Background:

During the staff's review of the aging management review (AMR) items in LRA Table 3.1.2-4 associated with SG tubes, the staff noted that heat transfer was listed as an intended function of the SG tubes, but reduction of heat transfer was not cited as an aging effect.

Request:

Discuss how the reduction of heat transfer of the SG tubes will be managed.

Callaway Response

LRA Table 3.1.2-4 has been revised by Amendment 13 in Enclosure 2, to include an AMR line that identifies the Steam Generators program (B2.1.9), for verifying the effectiveness of the Water Chemistry program (B2.1.2), in managing the aging effect of reduction of heat transfer due to fouling on the nickel alloy steam generator tube external surfaces in a secondary water environment. LRA Section 3.1.2.1.4, Appendix A1.9 and Appendix B2.1.9 have also been revised by Amendment 13 in Enclosure 2 to include an aging effect of reduction of heat transfer.

The aging effect of reduction of heat transfer due to fouling is not an applicable aging effect for the primary side of the steam generator tubes based on the decision of the Steam Generator Task Force (meeting summary ML110670317, 2-18-11). Callaway has not experienced reduction in heat transfer of the primary side of the nickel alloy tubing. In addition, Callaway is not aware of any industry operating experience that would suggest the aging effect of reduction of heat transfer due to fouling is applicable to the primary side of the nickel alloy steam generator tubes in a reactor coolant environment.

Corresponding Amendment Changes

Refer to the Enclosure 2 Summary Table "Amendment 13, LRA Changes from RAI Responses", for a description of LRA changes with this response.

CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION

REQUEST FOR ADDITIONAL INFORMATION (RAI) Set #11 and Set #12 RESPONSES

Set #12 Responses

RAI B2.1.6-4

Background:

Nuclear Energy Institute (NEI) Report No. NEI-95-10, Revision 6, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule,” provides NEI’s recommendations for formatting license renewal applications for U.S nuclear plants. NEI-95-10, Revision 6, states that Appendix C of LRAs represents an optional appendix that may be used for plant-specific information that is needed or required for the LRAs but that does not appropriately fit into any other of the sections that are recommended for LRAs. Callaway’s LRA states that LRA Appendix C was not used.

The Callaway LRA includes AMP B2.1.6, “PWR Vessel Internals,” which is based in part on the recommended guidelines for Westinghouse-designed RVI components in EPRI Report No. 1022863, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).” The EPRI Report was approved in a revised NRC safety evaluation (SE, Revision 1) dated December 16, 2011. This safety evaluation included appropriate applicant/licensee action items (A/LAIs) that were to be responded to for incoming LRAs.

Issue:

Per the NEI-95-10, Revision 6 guidelines, LRA Appendix C is the appropriate place for providing responses to A/LAIs; however, LRA Appendix C does not currently include Callaway’s responses to the A/LAIs on MRP-227-A.

Request:

Provide your basis for omitting responses to the applicable A/LAIs on MRP-227-A for Westinghouse-designed RVI components from the scope of the Callaway LRA. Alternatively, Callaway may amend its LRA to include the appropriate responses to those A/LAIs that have been issued on the MRP-227-A report and are applicable to the design of the Westinghouse-designed RVI components.

Callaway Response

Callaway will provide specific responses to the Applicant/Licensee Action Items (A/LAIs) identified in the December 16, 2011 NRC Safety Evaluation on the MRP-227-A methodology in LRA Appendix C. The LRA Appendix C supplement for the NRC Safety Evaluation A/LAIs will be provided consistent with the LRA Table A4-1 item #4 commitment to implement the PWR Vessel Internals program as described in LRA Section B2.1.6 within 24 months after the issuance of MRP-227-A. LRA Table A4-1 item #4 also includes the Callaway reactor vessel internals inspection plan noted in part (b) of A/LAI item 8 of the NRC Safety Evaluation on the MRP-227-A methodology. The Callaway reactor vessel internals inspection plan will be provided within 24 months after the issuance of MRP-227-A.

Summaries of the Callaway response to applicable December 16, 2011 NRC Safety Evaluation A/LAIs are provided below:

A/LAI #1. Applicability of Failure Modes, Effects, and Criticality Analyses (FMECA) and Functionality Analysis Assumptions

Applicability assumptions identified in MRP-227-A section 2.4 are identified in Callaway AMP XI.M16A basis document, Element 1.

A/LAI #2. PWR Vessel Internal Components Within the Scope of License Renewal

Callaway reactor vessel internals (RVI) AMRs in LRA Table 3.1.2-1 have been updated to be consistent with MRP-191 and MRP-227-A for Westinghouse reactor internals.

A/LAI #3. Evaluation of the Adequacy of Plant-Specific Existing Programs

Callaway RVI AMRs have been updated to address the cracking of the guide tube support pins (split pins) as Existing Program Components to be consistent with MRP-227-A.

A/LAI #4. B&W Core Support Structure Upper Flange Stress Relief

Not applicable to Callaway.

A/LAI #5. Application of Physical Measurements as Part of the I&E Guidelines for CE and Westinghouse RVI Components.

Physical measurements for the hold down spring are identified in Callaway AMP XI.M16A Element 6. Callaway will determine the acceptance criterion to be consistent with Table 5-3 of MRP-227-A.

A/LAI #6. Evaluation of Inaccessible and Non-inspectable B&W Components

Not applicable to Callaway with Westinghouse reactor internals.

A/LAI #7. Plant-Specific Evaluation of CASS Materials

Not applicable. Callaway reactor internals do not have CASS lower support column bodies addressed in this A/LAI.

A/LAI #8. Submittal of Information for Staff Review and Approval

(a) Callaway has provided AMP B2.1.6 for NRC audit that addresses the ten program elements of NUREG-1801, Rev 2, XI.M16A. (The basis document and LRA B2.1.6 have been updated to MRP-227-A and the December 16, 2011 NRC Safety Evaluation).

(b) Callaway will provide the Inspection Plan consistent with LRA Table A4-1 item #4 within 24 months after the issuance of MRP-227-A. The PWR Vessel Internals program (B2.1.6) is described in LRA Section A1.6 and LRA Section B2.1.6. The Inspection Plan will address plant specific action items and identify any deviations to MRP-227-A with justification.

(c) The Callaway FSAR Supplement described in LRA Section A1.6 includes an AMP consistent with MRP-227-A and NRC Safety Evaluation dated December 16, 2011. Reactor vessel internals TLAAs in the FSAR Supplement do not credit the PWR Vessel Internals program (B2.1.6).

(d) Technical Specification (TS) Changes - No TS changes are required as a result of implementation of the requirements of MRP-227-A.

(e) Reactor vessel internal TLAAs - Reactor vessel internals TLAAs are addressed in LRA Section 4.3.3 and do not credit the PWR Vessel Internals program (B2.1.6).

Corresponding Amendment Changes

No changes to the License Renewal Application (LRA) are needed as a result of this response.

RAI 3.1.2.1-3

Background:

LRA Table 3.1.2-1 provides a list of the AMR items for aging management of the Callaway RPV and RVI components. LRA page 3.1-69 provides the AMR items for managing loss of material and changes in dimension of the RVI baffle-former assembly. In these AMR items, the applicant credits the Water Chemistry Program (LRA AMP B2.1.2) to manage loss of material in the RVI baffle-former assembly and the PWR Vessel Internals Program (LRA AMP B2.1.6) to manage changes in dimension of the RVI baffle-former assembly.

Table 4.3-3 of MRP-227-A includes the EPRI MRP's recommended criteria for performing VT-3 visual inspections of Westinghouse-design baffle-to-former assembly components, including the baffle plates, former plates, and baffle-to-edge bolts.

Issue:

It is not evident to the staff which components are within the scope of the AMR items for the generic terminology "RVI baffle-former assembly." In addition, in Table 4-3 of MRP-227-A, the EPRI MRP identifies that the VT-3 examinations on the applicable RVI baffle-former assembly components looks for evidence of both changes in dimension that initiates by component distortion and cracking that initiates by an irradiation-assisted stress corrosion cracking mechanism. Although the LRA Table 3.1.2-1 does include appropriate AMR items on management of cracking for the Callaway baffle-to-edge bolts, it does not include any AMR items on management of cracking in the plant's baffle plates or former plates.

Request:

Clarify whether the scope of the AMR items for managing changes in dimension and loss of material in the "RVI baffle-former assembly" include baffle plates and former plates in the assembly. Provide the basis on why LRA Table 3.1.2-1 does not include any AMR items on management of cracking in the baffle plates and former plates of the baffle-former assembly.

Callaway Response

As clarified for RAI 3.1.2.1-1 part (a), the RVI baffle-former assembly includes the baffle plates and the former plates. The RVI baffle-former assembly component included on LRA Table 3.1.2-1 is consistent with the component description of GALL Rev.2, line item IV.B2.RP-270 for aging management of the changes in dimension in the RVI baffle-to-former assembly: baffle and former plates. The RVI baffle-former assembly component included on LRA Table 3.1.2-1 that is consistent with GALL Rev. 2 line item IV.B2.RP-24 is also applicable to the RVI baffle-to-former assembly: baffle and former plates.

Table 4-3 of MRP-227-A identifies that cracking due to irradiation-assisted stress corrosion cracking (IASCC) is an applicable aging effect to be managed within the scope of MRP-227-A for the RVI baffle-to-former assembly components. LRA Amendment 13 in Enclosure 2 has revised LRA Table 3.1.2-1 to include cracking due to IASCC of the RVI baffle-to-former assembly: baffle and former plates that is managed as a Primary Category Component without any Expansion Category Component Links by B2.1.6, PWR Vessel Internals program, consistent with MRP-227-A Table 4-3.

Corresponding Amendment Changes

Refer to the Enclosure 2 Summary Table "Amendment 13, LRA Changes from RAI Responses", for a description of LRA changes with this response.

RAI 3.1.2.1-4

Background:

License renewal application (LRA) Table 3.1.2-1 provides a list of the aging management review (AMR) items for aging management of the Callaway reactor pressure vessel (RPV) and reactor vessel internal (RVI) components. LRA pages 3.1-72 and 3.1-73 provide the AMR items for the control rod guide tube (CRGT) support pins (CRGT split pins). AMR items for the Callaway core barrel flange are provided on LRA page 3.1-74 and AMR items for the Callaway upper core plate are provided on LRA page 3.1-83. In the set of the AMR items for CRGT split pins, for the upper core plate, and upper core barrel flanges, Union Electric Company d/b/a Ameren Missouri (the applicant) credits its Water Chemistry Program (LRA AMP B2.1.2) to manage loss of material that might occur in the components during the period of extended operation. The LRA indicates that these components are all made from stainless steel materials and are exposed to the reactor coolant environment.

In LRA Amendment No. 1, the applicant added an AMR item on loss of material (due to wear) for the upper core plate and credited the PWR Vessel Internals Program to manage loss of material due to wear in the upper core plate by identifying them as “Expansion Category” components for the program. Electric Power Research Institute (EPRI) Report No. 1022863, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation - Guidelines (MRP-227-A),” identifies that the CRGT lower flange weld is the “Primary Category” link for upper core plate.

In LRA Amendment No. 1, the applicant also amended its AMR item on cracking for the core barrel flange to credit the Water Chemistry Program and PWR Vessel Internal Program to manage the aging effect and identified that the core barrel flange is an “Expansion Category” component for the applicant’s PWR Vessel Internals Program. MRP-227-A identifies that the CRGT lower flange weld is the “Primary Category” link for upper core plate.

Issue:

In Table 3-3 of EPRI Report No. 1022863, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A),” the EPRI MRP identifies that loss of material due to wear is an applicable aging effect for these components. Wear is a mechanical mechanism for inducing loss of material in a metallic component. Therefore, water chemistry control programs are not effective means for managing loss of material that could initiate in a metallic component as a consequence of wear because they are mitigative programs that are designed to control chemical species in the various plant coolant systems. Thus, the staff does not have sufficient information to conclude that wear would not be applicable to the CRGT split pins or how loss of material due to wear could be adequately managed solely through implementation of the Callaway Water Chemistry Program.

For the Westinghouse-designed core barrel flanges, the basis in MRP-227-A would have the applicant identify these components as ASME Section XI, “Existing Program” components, and not as “Expansion Category” components. If the core plates were actual ASME Section XI, Examination Category B-N-3 components for the current licensing basis (CLB), the MRP-227-A report would have the applicant perform ASME Section XI VT-3 visual “Existing Program” inspections of the core barrel flange using the existing ASME Section XI, Examination Category B-N-3 requirements in order to manage loss of material that might occur in the components as a result of wear. Under NUREG-1801, “Generic Aging Lessons Learned Report,” (GALL Report)

AMR item IV.B2.RP-24, the Water Chemistry Program (by itself) is only acceptable to manage loss of material that initiates by either a general corrosion or pitting corrosion mechanism.

Request:

a) Provide the basis as to why loss of material due to wear was not considered as an applicable aging effect and mechanism requiring management (AERM) for the CRGT split pins when this AERM is identified in the MRP-227-A report for these components. If loss of material due to wear is an applicable AERM for CRGT split pins, justify your basis for managing loss of material in the components using only the Callaway Water Chemistry Program. With regard to aging management of the CRGT split pins, specifically justify why the combination of the PWR Vessel Internals Program and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD Program had not been credited to manage potential loss of material due to wear in the split pins similar to the manner in which the applicant had credited this combination of AMPs to manage cracking in the components.

b) With regard to aging management of the core barrel flanges, clarify why the AMR item on cracking of the core barrel flanges in LRA Amendment No. 1 identifies these components as "Expansion Category" components and not as ASME Section XI "Existing Program" components and justify your basis on how the Water Chemistry Program, by itself, would be capable of managing loss of material in the core barrel flange(s) that initiates by a wear-induced mechanism (or similar mechanisms such as fretting or abrasion). Clarify whether the core barrel flange is defined in the Callaway CLB as an ASME Section XI, Examination Category B-N-3 core support structure component. If the core barrel flange is a B-N-3 component for the CLB, provide the basis for not crediting either the PWR Vessel Internals Program's ASME Section XI-based "Existing Program" Examination Category protocols or the applicant's ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD Program as the basis for managing both loss of material due to wear and cracking that may occur in the core barrel flanges. If the upper core flange is not defined in the CLB as an ASME Section XI, Examination Category B-N-3 core support structure component:

   i) identify the RVI "Primary Category" component or components that will be inspected for cracking and loss of material due to wear and will be used to determine whether the core barrel flange will need to be inspected for these aging effects,

   ii) identify the inspection methods and frequency that will be applied to both the "Primary Category" component links and potentially to the core barrel flange as an "Expansion Category" component for those components, and

   iii) justify the "Expansion Category" bases that will be applied to potential inspections of the core barrel flange.

Callaway Response

a) RVI Control Rod Guide Tube Support Pins

   One of the LRA Table 3.1.2-1 component lines for RVI control rod guide tube support pins (split pins) identifies the XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD program (B2.1.1) as managing cracking or loss of material due to wear in the split pins consistent with GALL line IV.B2.RP-382. Amendment 13 in Enclosure 2 has clarified LRA Table 3.1.2-1 to identify that the aging effects of cracking or loss of material due to wear applies for all aging evaluations consistent with GALL line IV.B2.RP-382.

In addition, Amendment 13 in Enclosure 2 has identified a different aging management program applicability associated with the aging evaluation of the CRGT split pins using GALL line IV.B2.RP-382. The aging management program was changed from the ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD program (B2.1.1) to PWR Vessel Internals (B2.1.6) (Existing Program Components – No Expansion Components).

b) RVI Core Barrel Flange

Consistent with MRP-227-A and the associated NRC Safety Evaluation, Expansion Category Component Links are not required for aging management of the RVI core barrel flange. The RVI core barrel flange will be managed as an ASME Section XI, Examination Category B-N-3 core support structure component. The LRA Table 3.1.2-1 component line for RVI core barrel flanges, that evaluated aging using GALL Line IV.B2.RP-282, has been revised as shown on Amendment 13 in Enclosure 2 to manage cracking consistent with GALL line IV.B2.RP-382. Amendment 13 in Enclosure 2 has identified a different aging management program applicability associated with the aging evaluation of cracking in the RVI core barrel flanges using GALL line IV.B2.RP-382. The aging management program was changed from the ASME Section XI Inservice Inspection, Subsections IWB, IWC, IWD program (B2.1.1) to PWR Vessel Internals program (B2.1.6) (Existing Program Components – No Expansion Components).

In addition Amendment 13 in Enclosure 2 has revised LRA Table 3.1.2-1 to identify that loss of material due to wear of the stainless steel core barrel flange is managed by the PWR Vessel Internals program (B2.1.6) (Existing Program Components – No Expansion Components)

Corresponding Amendment Changes

Refer to the Enclosure 2 Summary Table "Amendment 13, LRA Changes from RAI Responses”, for a description of LRA changes with this response.

RAI 3.1.2.1-5

Background:

In LRA Amendment No. 1, the applicant provided changes to the AMR items for the following RVI components at Callaway:

- core barrel girth and axial welds
- core barrel outlet nozzles and welds
- core support forging
- upper core plate.

Issue:

In the AMR item changes, the applicant either amended the LRA to correct the aging effect or mechanism consistent with those reported for the components in MRP-227-A or to propose aging management bases under the PWR Vessel Internals that were consistent with the recommended aging management bases as proposed in MRP-227-A report and endorsed by the staff for “Primary Category” or “Expansion Category” RVI components. However, the staff has noted that these components may fall into multiple inspection category bases if the components are also defined in the CLB as ASME Section XI, Examination Category B-N-3 core support structure components. Therefore, the staff seeks additional information on whether the core barrel girth welds, core barrel axial welds; core barrel outlet nozzles and welds; core support forging; and upper core plate are ASME Section XI, Examination Category B-N-3 components for the CLB.

Request:

Clarify whether the core barrel girth welds, core barrel axial welds, core barrel outlet nozzles and welds, core support forging, and upper core plate are ASME Section XI, Examination Category B-N-3 components for the CLB. If so, justify why the ASME Section XI, Subsection IWB, IWC, IWD, or the ASME Section XI “Existing Program” protocols of the PWR Vessel Internals Program have not been credited in addition to the appropriate “Primary Category” or “Expansion Category” bases that have been identified for the components.

Callaway Response

The core barrel girth welds, core barrel axial welds, core barrel outlet nozzles and welds, core support forging, and upper core plate are ASME Section XI, Examination Category B-N-3 components. Although they are categorized as either "Primary Category" components or "Expansion Category" components, the requirements of ASME XI, Examination Category B-N-3 are still in effect and may only be altered as allowed by 10 CFR 50.55a. The guidelines provided in MRP-227-A do not reduce, alter, or otherwise affect current ASME Section XI or Callaway inservice inspection commitments.

Corresponding Amendment Changes

No changes to the License Renewal Application (LRA) are needed as a result of this response.

RAI 3.1.2.2-1

Background:

LRA Section 3.1.2.2.9 provides the applicant's basis for assessing cracking in (a) the inaccessible areas of RVI components; or (b) the redundant components (such as bolting) in those redundant components that are inaccessible to inspection, if the inspections of the accessible areas of the components or redundant RVI components of the accessible redundant components reveal evidence of cracking in the components. LRA Section 3.1.2.2.10 provides the applicant's analogous assessment for loss of material due to wear, loss of fracture toughness, changes in dimension due to void swelling or distortion, or loss of preload in fastened RVI connections (e.g., bolted, pinned, or keyed connections). These LRA sections are based on the NRC's further evaluation "acceptance criteria" recommendations for assessing inaccessible RVI component areas or inaccessible redundant RVI components in SRP-LR Sections 3.1.2.2.9 and 3.1.2.2.10.

The applicant typically credits the MRP-defined "Primary Category" and "Expansion Category" inspections of its PWR Vessel Internals Program (LRA AMP B2.1.6) for performing condition monitoring-based aging management of those RVI components at Callaway that are defined as MRP-based "Primary Category" or "Expansion Category" components.

In LRA Amendment No. 1, the applicant stated the following with respect to the PWR Vessel Internals Program, as given in LRA Section B2.1.6:

> "PWR Vessel Internals (B2.1.6) examines one hundred percent of the accessible volume/area of each component for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g. bolting), the minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible). If defects are discovered during the examination, Callaway will enter the information into the corrective action program and evaluate whether the results of the examination ensure that the component (or set of components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP- 17096-NP."

Issue:

The aging management bases in LRA Sections 3.1.2.2.9 and 3.1.2.2.10 only discuss how the applicant will conform to the EPRI MRP's recommended inspection coverage bases for "Primary Category" and "Expansion Category" RVI components at Callaway. LRA Sections 3.1.2.2.9 and 3.1.2.2.10 do not discuss or explain how the applicant would evaluate relevant aging effects in those portions of the components that are inaccessible to the examinations, or for redundant components with areas inaccessible to the inspection techniques being performed (e.g., bolting).

In addition, the methodology in WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (ML101460156), was submitted by the EPRI MRP and Westinghouse Company for NRC review and approval. However, this report has not yet been approved by the staff and implementation of the methodology in

WCAP-17096-NP, if approved, may be subject to additional limitations or Applicant/Licensee Action Items that derive from the staff's conclusions on the acceptability of the report's methodology.

Request:

a) For those "Primary Category" or "Expansion Category" inspections that reveal evidence of aging, clarify when and how the evaluations of the inaccessible areas or inaccessible redundant components (such as inaccessible bolts) would be performed. Specifically, identify the types of structural or flaw tolerance evaluations that would be performed for the relevant aging effects that are assumed to occur in inaccessible RVI component areas or for redundant components in the inaccessible redundant RVI components. In addition, describe the specific acceptance criteria and relevant assumptions that will be used to initiate such structural or flaw tolerance evaluations of the inaccessible areas or inaccessible redundant components. With respect to these clarifications, provide separate inaccessible area/inaccessible component clarifications for the assessment of aging in non-redundant "Primary Category" RVI components, redundant "Primary Category" RVI components, non-redundant "Expansion Category" RVI components and redundant "Expansion Category" RVI components. In particular, for RVI "Expansion Category" components or redundant RVI "Expansion Category" components, the staff seeks clarification on whether it would only take detection of relevant aging effects in the "Primary Component" links to initiate such structural evaluations or flaw tolerance evaluations of the inaccessible "Expansion Category" component areas or inaccessible redundant "Expansion Category" components, or whether evidence of relevant aging effects would actually need to be detected in the Expansion Category components to initiate such inaccessible area or inaccessible component evaluations. Justify all bases made in response to this part of the RAI.

b) Justify your basis for relying on WCAP-17096-NP as part of the monitoring and evaluation bases for the PWR Vessel Internals Program when the methodology in WCAP-17096-NP has not yet been approved by the NRC for use and may be subject to additional limitations or Application/Licensee Action Items. If the report is eventually approved for use by the staff, clarify and justify how the applicant will address any limitations or Applicant/Licensee Action Items (or equivalent) that may be issued on the report, as identified in the NRC safety evaluation will be issued on the report's methodology. Clarify and justify how the evaluation of flaws or degraded areas will be evaluated in the RVI components at Callaway if WCAP-17096-NP is rejected for use by the NRC.

Callaway Response

(a) For those "Primary Category" or "Expansion Category" examinations that reveal evidence of aging, the corrective action process evaluation would consider the applicability of the aging in the inaccessible areas or redundant components. Examination acceptance criteria and expansion criteria will be consistent with MRP-227-A Table 5-3 for Westinghouse plants.

Examination acceptance criteria identify the visual examination relevant condition(s), or signal based level or relevance of an indication, that requires formal disposition for acceptability. Based on the identified condition, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or repair or replace the item. The evaluation would consider extent of the condition in inaccessible areas/components consistent with the associated accessible primary component or expansion component evaluation.

Expansion criteria are intended to form the basis for decisions about expanding the set of components selected for examination or other aging management activity, in order to determine whether the level of degradation represented by the detected conditions has extended to other components judged to be less affected by the degradation. The expansion criteria of MRP-227-A Table 5-3 would be used to determine when the aging detected in the primary component would require the additional examinations identified for the expansion component(s). Expansion criteria evaluations for accessible primary components will include consideration of inaccessible areas/components associated with the primary component in the evaluation.

Engineering evaluations including their specific acceptance criteria and relevant assumptions would be consistent with WCAP-17096-NP Revision 2.

(b) If WCAP-17096-NP is approved for use by the NRC with any limitations or Applicant/Licensee Action Items (A/LAIs) as identified in the NRC Safety Evaluation, the implementation of the PWR Vessel Internals Program will be supplemented consistent with the NRC Safety Evaluation/approval. If WCAP-17096-NP is rejected for use by the NRC, Callaway engineering evaluations for reactor vessel internals will be performed consistent with available industry guidance (e.g. EPRI-MRPs, PWR Owners Group Reports, etc.).

Corresponding Amendment Changes

No changes to the License Renewal Application (LRA) are needed as a result of this response.