Amendment 13, LRA Changes from RAI Responses

Enclosure 2 Summary Table

Affected LRA Section	LRA Page
Section 3.1.2.1.4	3.1-6
Section 3.1.2.2.11	3.1-12 and 3.1-13
Table 3.1-1	3.1-28
Table 3.1.2-1	3.1-70, 3.1-72 through 3.1-78, and 3.1-80 through 3.1-85
Table 3.1.2-4	3.1-120 and 3.1-121
Section A1.9	A-6
Table A4-1, item 3	A-36
Table A4-1, item 34	A-48
Table A4-1, item 35	A-48
Section B2.1.4	B-22, B-23, and B-24
Section B2.1.9	B-38, B-39, and B-40

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Section 3.1.2.1.4 (page 3.1-6) is revised to add the aging effect of Reduction of heat transfer (new text shown underlined)

3.1.2.1.4 Steam Generators

Materials

The materials of construction for the steam generator component types are:

- Carbon Steel
- Carbon Steel with Stainless Steel Cladding
- Nickel-Alloys
- Stainless Steel

Environment

The steam generator component types are exposed to the following environments:

- Borated Water Leakage
- Plant Indoor Air
- Reactor Coolant
- Secondary Water

Aging Effects Requiring Management

The following steam generator aging effects require management:

- Cracking
- Loss of material
- Loss of preload
- Reduction of heat transfer
- Wall thinning

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Section 3.1.2.2.11.1 and 3.1.2.2.11.2 (pages 3.1-12 and 3.1-13) are revised to be consistent with commitments #34 and #35 of Table A4-1 (deleted text shown in strikethrough and new text shown underlined):

3.1.2.2.11 Cracking due to Primary Water Stress Corrosion Cracking

3.1.2.2.11.1 Primary water stress corrosion cracking in steam generator divider plate assemblies

Callaway replacement steam generator divider plate assemblies are fabricated of Alloy 690. The divider plate to primary head and tubesheet junctions are welded with Alloy 152 weld materials. The tubesheet cladding is Alloy 182 and the primary head cladding is stainless steel. There is a concern regarding potential failure at the divider plate welds to primary head and tubesheet cladding and Callaway commits to perform one of the following three resolution options:

Option 1: Inspection
~~(1)~~ Perform ~~an~~ a one-time inspection of each steam generator to assess the condition of the divider plate welds. The examination technique(s) will be capable of detecting PWSCC in the divider plate assemblies and the associated welds.
OR
Option 2: Analysis
~~(2)~~ Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator reactor coolant system pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval.
OR
Option 3: Industry/NRC Studies
~~(3)~~ If results of industry and NRC studies and operating experience document that potential failure of the steam generator reactor coolant system pressure boundary due to PWSCC cracking of steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion.

3.1.2.2.11.2 Primary water stress corrosion cracking in steam generator nickel alloy tube-to-tubesheet welds

The material of steam generator tubesheet cladding is Alloy 182. The tubes are made of Alloy 690 and are secured to the tubesheet by means of tube to tubesheet leaktight weld and tube expansion. There is a concern regarding potential failure of primary-to-secondary pressure boundary due to primary water stress corrosion cracking (PWSCC) cracking of

tube-to-tubesheet welds. Callaway commits to perform one of the following two resolution options:

Option 1: Inspection
~~(1)~~ Perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If weld cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and ~~an ongoing~~ a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators.
OR
Option 2: Analysis
~~(2)~~ Perform an analytical evaluation of the steam generator tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC, or redefining the reactor coolant pressure boundary of the tubes, where the ~~in order to establish a technical basis which concludes that the structural integrity of the steam generator tube-to-tubesheet interface is adequately maintained with the presence of tube-to-tubesheet weld cracking. Establish a technical basis which concludes that the~~ steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.

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Revision to Table 3.1.1, Item 3.1.1.032 to address the exceptions.

Table 3.1.1 (pages 3.1-28) is revised as follows (deleted text shown in strikethrough and new text shown underlined):

Table 3.1-1 Summary of Aging Management Programs in Chapter IV of NUREG-1801 for Reactor Vessel, Internals, and Reactor Coolant System

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.032	Stainless steel, nickel alloy, or CASS reactor vessel internals, core support structure, exposed to reactor coolant and neutron flux	Cracking, or loss of material due to wear	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) for ASME components	No	~~Consistent with NUREG-1801.~~ Consistent with NUREG-1801 for all components except that different aging management programs are credited for the following. The aging of RVI CRGT support pins and RVI core barrel flange are managed by Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.6) (Existing Program Components - No Expansion Components).

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Revision to Table 3.1.2-1 to:
(a) Correct the aging effect of AMR lines using IV.B2.RP-382 to be consistent with GALL. (RAI 3.1.2.1-4, part (a))
(b) Add an AMR line of RVI Baffle-Former Assembly for the aging effect of Cracking due to IASCC using IV.B2.RP-294. Add Plant Specific Note #5. (RAI 3.1.2.1-3)
(c) Replace IV.B2.RP-282 with IV.B2.RP-382 for the aging effect of Cracking due to SCC of RVI Core Barrel Flange. Add Plant Specific Note #6. (RAI 3.1.2.1-4 (b))
(d) Add an AMR line for the aging effect of Loss of Material due to Wear of RVI Core Barrel Flange. (RAI 3.1.2.1-4 (b))

Table 3.1.2-1 (pages 3.1-70, 72 through 78, and 80 through 85) are revised as follows (deleted text shown in strikethrough and new text shown underlined):

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
RVI Baffle-Former Assembly	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.6) (Primary Components - No Expansion Components)	IV.B2.RP-294	3.1.1.053	E,5
RVI Control Rod Guide Tube Bolting	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
RVI Control Rod Guide Tube Support Pins	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1~~) PWR Vessel Internals (B2.1.6) (Existing Program Components - No Expansion Components)	IV.B2.RP-382	3.1.1.032	~~A~~ E,6
RVI Core Barrel	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Core Barrel Flanges	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.6) ~~(Expansion Components)~~ (Existing Program Components - No Expansion Components)	~~IV.B2.RP-282~~ IV.B2.RP-382	~~3.1.1.053~~ 3.1.1.032	~~A~~ E,6
RVI Core Barrel Flanges	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Loss of material	PWR Vessel Internals (B2.1.6) (Existing Program Components - No Expansion Components)	IV.B2.RP-345	3.1.1.059	A
RVI Core Barrel Outlet Nozzles	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Head/Vessel Alignment Pins	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
RVI Hold Down Spring	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI ICI Support Structure Bolting	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI ICI Support Structures (Exit Thermocouple)	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Irradiation Specimen Basket	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Lower Core Support-Clevis Insert Bolting	SS	Nickel Alloys	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Lower Core Support-Energy Absorber Assembly	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Neutron Shield Panel	SLD	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Radial Support Keys and Clevis Inserts	SS	Nickel Alloys	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
RVI Upper Core Support-Top Support Plate	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Upper Core Support-Upper Support Column	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A
RVI Upper Support Column Bolting	SS	Stainless Steel	Reactor Coolant (Ext)	Cracking or Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.032	A

Notes for Table 3.1.2-1:

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

F Material not in NUREG-1801 for this component.

Plant Specific Notes:

1 NUREG-1801 does not address cracking of the nickel-alloy reactor vessel flange leak monitoring tube with an internal environment of reactor coolant leakage. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) manages the cracking of the nickel-alloy reactor vessel flange leak monitoring tube with an internal environment of reactor coolant leakage.

2 The plant-specific aging management programs used to manage the cracking for stainless steel BMI guide tubes are ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) for Class 1 components and Water Chemistry (B2.1.2).

3 The MRP-227 Safety Evaluation section 4.1, Limitations and Conditions specifies that the core barrel flange welds need to be assigned as primary components for aging management of IASCC and neutron embrittlement. The core barrel cylinder girth (circumferential) welds component is managed as a primary component consistent with MRP-227-A, Table 4-3.

4 This reactor vessel internal expansion component is linked to the MRP-227-A primary component of CRGT lower flange weld, as specified in MRP-227 Safety Evaluation section 4.1, Limitations and Conditions.

5 Cracking due to IASCC of the RVI baffle plates and former plates is managed as a Primary Category Component without any Expansion Category Component links in accordance with MRP-227-A, Table 4-3. NUREG-1801, Rev. 2 does not address cracking due to IASCC of the RVI baffle plates and former plates.

6 The RVI component is managed as an Existing Program Component in accordance with MRP-227-A, Table 4-9. Therefore, IV.B2.RP-382 is used for addressing cracking or loss of material due to wear with an AMP for Existing Program Components.

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Revision to Table 3.1.2-4 to add an AMR line of SG Tubes for the aging effect of Reduction of heat transfer using a Non-GALL line. Add Plant Specific Note #5.

Table 3.1.2-4 (pages 3.1-120 and 121) are revised as follows (new text shown underlined):

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Steam Generators

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
SG Tubes	HT, PB	Nickel-Alloys	Secondary Water (Ext)	Reduction of Heat Transfer	Steam Generators (B2.1.9) and Water Chemistry (B2.1.2)	None	None	H, 5

Notes for Table 3.1.2-4:

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

H Aging Effect not in NUREG-1801 for this component, material and environment combination.

Plant Specific Notes:

1 The tubesheets in the replacement steam generators installed at Callaway are clad with Alloy 182. The tubes are secured to the tubesheet by means of tube-to-tubesheet leaktight weld and tube expansion. See Section 3.1.2.2.11.2 for the commitment to verify the integrity of the welds.

2. The divider plates in the replacement steam generators installed at Callaway are made of Alloy 690. The divider plate to primary head and tubesheet junctions are welded with Alloy 152 weld materials. See Section 3.1.2.2.11.1 for the commitment to verify the integrity of the welds.

3 Steam generator primary manway, secondary manway, handhole, and inspection ports are equipped with stainless steel gasket inserts.

4. This TLAA is applicable to the steam generator tube wear TLAA. Section 4.7.9 describes the evaluation of the TLAA for loss of material due to wear and fretting of steam generator tubes.

5. Reduction in heat transfer due to fouling is a potential aging effect/mechanism for steam generator tubes in secondary water. NUREG-1801, Rev. 2 does not address the aging effect of reduction in heat transfer for the combination of component, material, and environment.

A1.9 STEAM GENERATORS

The Steam Generators program manages cracking, loss of material, reduction of heat transfer and wall thinning of the steam generator tubes, plugs, sleeves and secondary side steam generator internal components. The program provides preventive measures in the form of predictive assessment, tube plugging, foreign material exclusion, foreign object search, secondary side cleaning and maintenance, and maintaining the chemistry. The program detects degradation through nondestructive examinations, visual inspection, and in situ pressure testing. Assessments are used to verify that the steam generator performance criteria defined in the Callaway Technical Specifications have been met over the last operating interval and ensure that the criteria will be met over the next operating interval.

NDE inspection and primary to secondary leak rate monitoring are conducted consistent with the requirements of Callaway Technical Specifications and NEI 97-06, *Steam Generator Program Guidelines*. The program ensures that performance criteria are maintained for operational leakage, accident induced leakage, and structural integrity as prescribed in the Callaway Technical Specifications.

There is a concern regarding potential failure at the divider plate welds to primary head and tubesheet cladding and Callaway commits to perform one of the following three resolution options between Fall 2025 and Fall 2029 when the RSGs are in service for more than 20 years:

Option 1: Inspection
Perform a one-time inspection of each steam generator to assess the condition of the divider plate welds. The examination technique(s) will be capable of detecting PWSCC in the divider plate assemblies and the associated welds.
OR
Option 2: Analysis
Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator reactor coolant system pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval.
OR
Option 3: Industry/NRC Studies
If results of industry and NRC studies and operating experience document that potential failure of the steam generator reactor coolant system pressure boundary due to PWSCC cracking of steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion.

There is a concern regarding potential failure of primary-to-secondary pressure boundary due to primary water stress corrosion cracking (PWSCC) cracking of tube-to-tubesheet welds.

Callaway commits to perform one of the following two resolution options between Fall 2025 and Fall 2029 when the RSGs are in service for more than 20 years:

Option 1: Inspection
Perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If weld cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators.
OR
Option 2: Analysis
Perform an analytical evaluation of the steam generator tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC, or redefining the reactor coolant pressure boundary of the tubes, where the steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
3	Enhance the Boric Acid Corrosion program procedures: • to include steel, copper alloy greater than 15 percent zinc, and aluminum as materials that are susceptible to boric acid corrosion (Completed LRA Amendment 13). • so that system engineers will observe for signs of boric acid residue when performing system walkdowns (Completed LRA Amendment 13). • to specify that the corrective actions taken by the program will include a consideration to modify the present design or operating procedures to mitigate or prevent recurrence of aging effects caused by borated water leakage. Consideration will be given to modifications that (a) reduce the probability of primary coolant leaks at locations where they may cause corrosion damage, and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings or claddings (Completed LRA Amendment 13).	B2.1.4	Prior to the period of extended operation

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
34	Callaway replacement steam generator divider plate assemblies are fabricated of Alloy 690. The divider plate to primary head and tubesheet junctions are welded with Alloy 152 weld materials. The tubesheet cladding is Alloy 182 and the primary head cladding is stainless steel. There is a concern regarding potential failure at the divider plate welds to primary head and tubesheet cladding and Callaway commits to perform one of the following three resolution options: Option 1: Inspection ~~(1)~~ Perform ~~an~~ a one-time inspection of each steam generator to assess the condition of the divider plate welds. The examination technique(s) will be capable of detecting PWSCC in the divider plate assemblies and the associated welds. OR Option 2: Analysis ~~(2)~~ Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator reactor coolant system pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval. OR Option 3: Industry/NRC Studies ~~(3)~~ If results of industry and NRC studies and operating experience document that potential failure of the steam generator reactor coolant system pressure boundary due to PWSCC cracking of steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion.	Section 3.1.2.2.11.1, Table 3.1.2-4	~~Prior to the period of extended operation~~ Between Fall 2025 and Fall 2029 when the replacement steam generators are in service for more than 20 years.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
35	The material of steam generator tubesheet cladding is Alloy 182. The tubes are made of Alloy 690 and are secured to the tubesheet by means of tube to tubesheet leaktight weld and tube expansion. There is a concern regarding potential failure of primary-to-secondary pressure boundary due to PWSCC cracking of tube-to-tubesheet welds. Callaway commits to perform one of the following two resolution options: Option 1: Inspection ~~(1)~~ Perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If weld cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and ~~an ongoing~~ a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators. OR Option 2: Analysis ~~(2)~~ Perform an analytical evaluation of the steam generator tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC, or redefining the reactor coolant pressure boundary of the tubes, where the ~~in order to establish a technical basis which concludes that the structural integrity of the steam generator tube-to-tubesheet interface is adequately maintained with the presence of tube-to-tubesheet weld cracking. Establish a technical basis which concludes that the~~ steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.	Section 3.1.2.2.11.2, Table 3.1.2-4	~~Prior to the period of extended operation~~ Between Fall 2025 and Fall 2029 when the replacement steam generators are in service for more than 20 years.

B2.1.4 Boric Acid Corrosion

Program Description

The Boric Acid Corrosion program manages loss of material and increased resistance of connection due to borated water or reactor coolant leakage. The program monitors mechanical, electrical, and structural components that are within the scope of license renewal and susceptible to boric acid corrosion. The program relies in part on implementation of recommendations of NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*. The principal industry guidance document used is WCAP-15988-NP, *Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors*. Additionally, the program includes examinations conducted during ISI pressure tests performed in accordance with ASME Section XI requirements.

The program includes provisions to identify leakage through inspection and examination. When leakage is identified, an inspection is performed that includes identification of the leakage path, visual inspections of adjacent structures, components and supports, and cleaning of the leakage. If it is determined that an evaluation is necessary, it is performed in a timely manner. If the evaluation identifies aging effects, corrective action will be taken. Monitoring is provided by tracking and trending of existing and repaired leaks and establishment of a component-based visual history of boric acid leakage.

The scope of monitoring and inspections of this program includes all components that contain borated water reactor coolant in proximity to structures and components within the scope of license renewal. The scope of the inspections, evaluations, assessments and corrective actions include all observed leakage sources and the affected structures, components and supports.

The effects of boric acid corrosion on reactor coolant pressure boundary materials in the vicinity of nickel alloy components are managed by the Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program (B2.1.5).

NUREG-1801 Consistency

The Boric Acid Corrosion program is an existing program that~~, following enhancement, will be~~ is consistent with NUREG-1801, Section XI.M10, *Boric Acid Corrosion*.

Exceptions to NUREG-1801

None

Enhancements

None.

~~Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:~~

~~*Scope of the Program (Element 1)*~~

~~Procedures will be enhanced to include steel, copper alloy greater than 15 percent zinc, and aluminum as materials that are susceptible to boric acid corrosion.~~

~~*Detection of Aging Effects (Element 4)*~~

~~Procedures will be enhanced so that system engineers will observe for signs of boric acid residue when performing system walkdowns.~~

~~*Corrective Actions (Element 7)*~~

~~Procedures will be enhanced to specify that the corrective actions taken by the program will include a consideration to modify the present design or operating procedures to mitigate or prevent recurrence of aging effects caused by borated water leakage. Consideration will be given to modifications that (a) reduce the probability of primary coolant leaks at locations where they may cause corrosion damage, and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings or claddings.~~

Operating Experience

The following discussion of operating experience provides objective evidence that the Boric Acid Corrosion program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation:

1. In Fall 2008, pre-outage preparation walkdowns revealed a boric acid leak in the pressurizer auxiliary spray line. Corrosion was found near structural supports (hangers) that had boric acid dripping on them, requiring piping replacement. Destructive examinations found stress corrosion cracking (SCC) below the boron and rust deposits. Additional inspections found several additional cracks under the hangers. Additional destructive examinations revealed axially oriented through-wall cracking that was OD-initiated, transgranular, and confined to beneath the hangers in-service. The OD-initiated SCC was promoted by the presence of chlorides, operating stresses due to pressure/temperature changes, and was found at crevices under hanger locations. The source of chlorides was most likely introduced during original construction. As a corrective action, the pressurizer cubicle and the seal table were added to the quarterly Operations walkdown of containment. Additional

inspections for ODSCC in the alternate charging, RCS excess letdown, and auxiliary pressurizer spray lines were carried out in Refuel 17 (Spring 2010). There were unacceptable indications at seven locations. One location required pipe replacement, and the other six locations were repaired.

2. Prior to Refuel 17 (Spring 2010), Callaway was tracking 108 borated water leaks and during the outage, another 40 leaks were identified. During the outage the Valve Team completed work on over 90 borated water leaks and, at the conclusion of the outage, 55 leaks remained. An evaluation of the 55 remaining leaks determined that they did not impact plant operation and there were no boric acid corrosion concerns.

3. At the end of Spring 2011, Callaway had an average age of open jobs on borated water leaks of approximately 10.5 months, down from an average age of two years and one month during 2008. The backlog for boric acid leak jobs was down 19 percent from fall of 2010. A low concern leak that was being tracked worsened into a steam leak. The leak was evaluated and corrective actions were taken to perform a letdown outage to immediately repair the leak.

4. Between Fall 2010 and Spring 2011, the Boric Acid Corrosion program was revised as follows. Screening criteria were revised to match the format and context of the fluid leakage management program. Requirements were added to assure that boric acid evaluations were performed in a timely manner, and a due date must be specified for the evaluation of susceptible material. Guidance was also provided on the use of trending indicators such as RCS leakage rate, containment cooler fouling, and containment air monitors. Along with the program modifications, Callaway provided training to personnel based on the recommended training points in WCAP-15988. The purpose of this training is so that plant personnel know they should be looking for signs of borated water leakage, and what they need to look for.

The Boric Acid Corrosion program includes provision to identify, inspect, examine, and evaluate leakage and corrosion of structures and components within the scope of license renewal, and to initiate corrective actions. Occurrences that would be identified under the Boric Acid Corrosion program will be evaluated to ensure there is no significant impact to the safe operation of the plant and corrective actions will be taken to prevent recurrence. Guidance for re-evaluation, repair or replacement is provided for locations where aging is found. There is confidence that the continued implementation of the Boric Acid Corrosion program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Boric Acid Corrosion program~~, following enhancement,~~ provides reasonable assurance that aging effects will be managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

B2.1.9 Steam Generators

Program Description

The Steam Generators program manages cracking, loss of material, reduction of heat transfer and wall thinning of the steam generators. This program is applicable to the steam generator tubes, plugs, sleeves, and secondary side steam generator internal components. Aging is managed through assessment of potential degradation mechanisms, inspections, tube integrity assessments, plugging and repairs, primary to secondary leakage monitoring, maintenance of secondary side component integrity, primary side and secondary side water chemistry, and foreign material exclusion. Callaway procedural guidance implements the performance criteria for tube integrity, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable tube repair methods, leakage monitoring requirements, operational leakage and accident induced leakage requirements of Callaway Technical Specifications.

The program reporting criteria, inspection scope and frequency, assessments, plugging criteria, and primary to secondary leak rate monitoring, and monitoring/controlling primary and secondary side water chemistry are consistent with the requirements of Callaway Technical Specifications, the Maintenance Rule (10 CFR 50.65), EPRI 1019038, *Steam Generator Integrity Assessment Guidelines*, EPRI 1013706, *PWR Steam Generator Examination Guidelines*, EPRI 1008219, *PWR Primary-to-Secondary Leak Guidelines*, EPRI 1014983, *Steam Generator In-Situ Pressure Test Guidelines*, EPRI 1014986, *PWR Primary Water Chemistry Guidelines*, and EPRI 1016555, *PWR Secondary Water Chemistry Guidelines*. The EPRI guidelines provide a generic industry program to implement the NEI 97-06, *Steam Generator Program Guidelines*, Revision 3.

The Steam Generators program includes preventive measures to mitigate aging related to corrosion phenomena through foreign material exclusion as a means to inhibit wear degradation. The Callaway Water Chemistry program (B2.1.2) also monitors and controls reactor water chemistry and secondary water chemistry for the steam generators consistent with EPRI guidelines applicable to reactor water chemistry and secondary water chemistry as a preventive measure.

The Steam Generators program detects flaws in tubing, plugs, and tube supports needed to maintain tube integrity. Nondestructive examination (NDE) techniques are used to inspect all tubing materials to identify tubes that may need to be removed from service or repaired in accordance with plant technical specifications. The program provides criteria for the qualification of personnel, specific techniques, and the associated acquisition and analysis of data, including procedures, probe selection, analysis protocols, and reporting criteria. Assessment of tube integrity and plugging or repair criteria of flawed tubes is in accordance with plant technical specifications and the

program implementing procedures. Plugs and tube supports with aging are evaluated for corrective actions in accordance with the Callaway Corrective Action Program and the Callaway Steam Generators program. Condition monitoring assessments are performed to determine whether structural and accident leakage criteria have been satisfied. Operational assessments are performed after inspections to verify that structural and leakage integrity will be maintained for the operating interval between inspections, which is selected in accordance with the technical specifications and NEI 97-06 guidelines. Comparison of the results of the condition monitoring assessment with the predictions of the previous operational assessment provides feedback for evaluation of the adequacy of the operational assessment and additional insights that can be incorporated into the next operational assessment.

The original Callaway steam generators were replaced in 2005. The replacement steam generators incorporate features designed to improve reliability and minimize aging. Industry experience and laboratory testing have shown the materials used in fabricating the new steam generators to be more resistant to aging effects than those in the original steam generators.

NUREG-1801 Consistency

The Steam Generators program is an existing program that is consistent with NUREG-1801, Section XI.M19, *Steam Generators*.

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

The following discussion of operating experience provides objective evidence that the Steam Generators program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation.

1. During Refuel 14 (Fall 2005), steam generators were replaced with AREVA designed steam generators with alloy 690 thermally treated tubes. Pre-service eddy current inspections found 77 small dings, four tubes with signals similar to outside diameter axial cracking, and 33 tubes with a spiral signal pattern. After analyzing the signals and the tubes containing indications, the tubes were found to have no detectable degradation. One tube was plugged due to manufacturing defects. Visual inspections of the SG secondary side were performed to identify any foreign objects that may have been left behind after uprighting and installation of the steam

2. generators. Several foreign objects were found during these inspections and removed prior to placing the steam generators in service.

3. During Refuel 15 (Spring 2007), the first in-service inspection of the new steam generators identified a total of 92 anti-vibration bar (AVB) wear indications, with the largest indication being a 14 percent through-wall flaw. As discussed in the Refuel 15 operational assessment, Callaway does not expect to exceed the structural integrity performance criteria for AVB wear prior to the next scheduled steam generator inspection in Refuel 18 (Fall 2011).

4. In the degradation assessment for Refuel 17 (Spring 2010), Callaway monitored other plants with AREVA RSGs, both domestically and internationally. Specifically, Doel 4, Tihange 3, Prairie Island 1, and Salem 2 have Westinghouse-style steam generators manufactured by AREVA in the same time period as the Callaway RSGs. All four plants are experiencing various amounts of AVB wear in the same general location as Callaway.

5. Plant chemistry has been good and corrosion transport has been significantly reduced since the replacement of the main condenser tubes in Refuel 13 (Spring 2004). Sludge lancing for all four replacement steam generators was performed for the first time in Refuel 18 (Fall 2011).

6. During Refuel 18 (Fall 2011), the in-service inspection of the new steam generators identified a total of 497 new AVB wear indications, with the largest indication being a 39 percent through-wall flaw. Also observed were 34 new tube support plate wear indications, with the largest indication being a 15 percent through-wall flaw. The Refuel 18 operational assessment provided reasonable assurance that the structural integrity performance criteria will be met during the next 3-cycle operating period.

The operating experience of the Steam Generators program did not show any adverse trend in inspection results. Occurrences that would be identified under the Steam Generators program will be evaluated to ensure there is no significant impact to the safe operation of the plant and adequate corrective actions will be taken to prevent recurrence. Appropriate guidance for re-evaluation, repair, or replacement will be provided for locations where aging is found. There is confidence that the continued implementation of the Steam Generators program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Steam Generators program provides reasonable assurance that aging effects will be managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.