CALLAWAY PLANT UNIT 1
LICENSE RENEWAL APPLICATION

REQUEST FOR ADDITIONAL INFORMATION (RAI) Set #4 RESPONSES

RAI 2.1-1

Background:

Title 10, Section 54.4, "Scope," of the Code of Federal Regulations (CFR) states, in part:

a) Plant systems, structures and components within the scope of this part are –

1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions –

i. The integrity of the reactor coolant pressure boundary;

ii. The capability to shut down the reactor and maintain it in a safe shutdown condition;

or

iii. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

Issue:

During the on-site scoping and screening methodology audit, the staff determined that the applicant had used a plant equipment database, the Callaway Equipment List contained within the eB Director (CEL/eB Director), which provides design and quality classification for each component, as an information source used in identifying systems, structures, and components (SSCs) within the scope of license renewal. The CEL/eB Director contains a quality field "Q" used to identify safety-related SSCs included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). However, during the audit the staff determined that not all components identified as "Q" were included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

Request:

The staff requests that the applicant perform a review of this issue and provide a description of the process used to evaluate components identified as Q in the CEL/eB Director and the basis for not including components identified as Q within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). Indicate if the review concludes that use of the scoping methodology precluded the identification of SSCs that should have been included within the scope of license renewal in accordance with 10 CFR 54.4(a). Describe any additional scoping evaluations performed to address the 10 CFR 54.4(a) criteria. List any additional SSCs included within the scope of license renewal as a result of the review, and any structures and components (SCs) for which aging management reviews were performed. For SCs for which aging management reviews (AMRs) were performed, describe the aging management programs (AMPs), as applicable, to be credited for managing the identified aging effects.

Callaway Response

Various documentation sources were used to determine what components were within the scope of license renewal. One of the sources was the Callaway Equipment List (CEL), which is maintained in a database called EB Director. The CEL includes a field named Q-QUAL, which

is used to flag whether or not a component is safety-related. The basis for determining whether a component is flagged as safety-related is the definition of safety-related in FSAR Section 1.1.7. The FSAR definition is consistent with the scoping criteria of 10 CFR 54.4(a)(1). All safety-related components in the CEL have a Y flag in the Q-QUAL field to indicate that they are safety-related.

During the scoping process, all components were considered to be safety-related if they had a Y flag in the Q-QUAL field of the CEL. After review of plant basis documents, if a component with a Y flag in the CEL Q-QUAL field appeared not to have a safety-related function, an additional review was conducted. If there was a basis in plant documentation such as the CLB or an engineering evaluation that the component does not have a safety-related function, then the component was not included within the scope of license renewal based on the criteria of 10 CFR 54.4(a)(1). For cases where a component has a Y flag in the Q-QUAL field but was determined not to have a safety-related function, a corrective action document was initiated.

Since it is assumed during the initial scoping and screening that a component with a Y flag in the Q-QUAL field of the CEL has a safety-related function, all components with a Y flag were reviewed. Thus, this methodology did not preclude the identification of safety-related SSCs that should have been included within the scope of license renewal. No additional scoping evaluations were required. No SSCs were added to the scope of license renewal, and no additional AMRs were performed.

Corresponding Amendment Changes

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

RAI 2.1-2

Background:

Title 10, Section 54.4, "Scope," of the CFR states, in part:

a) Plant systems, structures and components within the scope of this part are –

1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions –

i. The integrity of the reactor coolant pressure boundary;

ii. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

iii. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

2) All nonsafety-related systems, structures and components whose failure could prevent satisfactory accomplishment of any of the functions identified in (a)(1)(i), (ii), or (iii) of this section.

Issue:

During the on-site scoping and screening methodology audit the staff reviewed the license renewal application (LRA), the 10 CFR 54.4(a)(2) implementing documents and license renewal drawings, and also performed plant walkdowns. LRA Section 2.4-4 does not indicate that the turbine building contains safety-related SSCs, however, the applicable license renewal drawings were marked to indicate that safety-related piping components are located within the turbine building. During discussion with the applicant, the staff determined that the applicant had identified safety-related portions of the main steam supply system, main feedwater system and steam generator blowdown system located within the turbine building. However the applicant had performed an evaluation and concluded that the safety-related portions of the systems located within the turbine building were not included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). In addition, the applicant had not included the nonsafety-related SSCs located within the turbine building and within the proximity of safety-related SSCs, whose failure could prevent satisfactory accomplishment of the intended functions of safety-related SSCs, within the scope of license renewal in accordance with 10 CFR 54.4(a)(2).

Request:

The staff requests that the applicant describe the process used to identify and evaluate safety-related piping components located within the turbine building, the results of the evaluation, and the basis for any determination to not include safety-related SSCs within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). In addition, describe the process used to identify and evaluate nonsafety-related SSCs located in the turbine building whose potential failure could impact safety-related SSCs, the results of the evaluation, and the basis for any determination to not include the nonsafety-related SSCs within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). The staff requests that the applicant perform a review of this issue and indicate if the review concludes that use of the scoping methodology precluded the identification of SSCs that should have been included within the scope of license renewal in

accordance with 10 CFR 54.4(a). Describe any additional scoping evaluations performed to address the 10 CFR 54.4(a) criteria. List any additional SSCs included within the scope of license renewal as a result of the review, and any SCs for which AMRs were performed. For SCs for which AMRs were performed, describe the AMPs, as applicable, to be credited for managing the identified aging effects.

Callaway Response

Various documentation sources were used to determine what components were within the scope of license renewal. One of the sources was the Callaway Equipment List (CEL), which is maintained in a database called EB Director. The CEL includes a field named Q-QUAL, which is used to flag whether or not a component is safety-related. The basis for determining whether a component is flagged as safety-related is the definition of safety-related in FSAR Section 1.1.7. The FSAR definition is consistent with the scoping criteria of 10 CFR 54.4(a)(1). All safety-related components in the CEL have a Y flag in the Q-QUAL field to indicate that they are safety-related.

During the scoping process, all components were considered to be safety-related if they had a Y flag in the Q-QUAL field of the CEL. Twelve piping segments were identified in the turbine building which had Y flags. These piping segments extended from the auxiliary building through the auxiliary building-turbine building wall and into the turbine building. They included four piping segments downstream of the four main steam isolation valves, four piping segments downstream of the blowdown isolation valves for each of the steam generators, and four piping segments upstream of the four main feedwater isolation valves. In addition, drain valves from three of the four blowdown lines also had Y flags in the Q-QUAL field. No other piping components in the turbine building have Y flags.

Since these piping segments had Y flags in the Q-QUAL field, they were initially considered to be within the scope of license renewal as safety-related components. From a review of FSAR Section 3.6.2.1.1e, High-Energy Piping in Containment Penetration Areas, and engineering evaluations, it was determined that the sections of the piping extending into the turbine building do not have a safety-related function. For the purpose of license renewal, new subcomponents were created representing the sections of piping within the turbine building. Since these subcomponents do not have a safety-related function, they are not within the scope of license renewal as safety-related components. The boundary drawings for the main steam system, steam generator blowdown system, and main feedwater system have been revised to indicate that the safety-related portion of these lines ends at the auxiliary building-turbine building wall. A corrective action document has been initiated to change the Y flags to N in the Q-QUAL fields for the blowdown drain valves, and a note will be added to the CEL entry for all 12 piping segments stating that the section of the lines in the turbine building is not safety-related.

The main feedwater and steam generator blowdown systems were screened differently than the main steam system. For the main feedwater and steam generator blowdown systems, credit was not taken for the equivalent anchors in the auxiliary building-turbine building wall. Therefore, the attached piping in the turbine building was put into the scope of license renewal as structural integrity attached.

The same approach was not taken for the main steam system because of the large number of components in the turbine building which would have to be included within the scope of license renewal for structural integrity attached. Therefore, credit was taken for the equivalent anchors at the auxiliary building-turbine building wall for the main steam system. Since the safety-

related portion of the main steam lines end at the auxiliary building-turbine building wall, the attached main steam piping in the turbine building was not put into the scope of license renewal.

Since it is assumed during the initial scoping and screening that a component with a Y flag in the Q-QUAL field of the CEL has a safety-related function, all components with a Y flag were reviewed. Thus, this methodology did not preclude the identification of safety-related SSCs that should have been included within the scope of license renewal. No additional scoping evaluations were required. No SSCs were added to the scope of license renewal, and no additional AMRs were performed.

Corresponding Amendment Changes

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

RAI 2.1-3

Background:

Title 10, Section 54.4, "Scope," of the CFR states, in part:

a) Plant systems, structures and components within the scope of this part are –

1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions –

   i. The integrity of the reactor coolant pressure boundary;

   ii. The capability to shut down the reactor and maintain it in a safe shutdown condition;

   or

   iii. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

2) All nonsafety-related systems, structures and components whose failure could prevent satisfactory accomplishment of any of the functions identified in (a)(1)(i), (ii), or (iii) of this section.

Issue:

During the on-site scoping and screening methodology audit the staff reviewed the LRA, the 10 CFR 54.4(a)(2) implementing documents and license renewal drawings, and also performed plant walkdowns. The staff determined that the applicant had identified safety-related electrical SSCs, located within the turbine building, that were not included within the scope of license renewal in accordance with 10 CFR 54.4(a)(1).

Request:

The staff requests that the applicant describe the process used to identify and evaluate the safety-related electrical SSCs, located within the turbine building, and the basis to not include the SSCs within the scope of license renewal in accordance with 10 CFR 54.4(a)(1). The staff requests that the applicant perform a review of this issue and indicate if the review concludes that use of the scoping methodology precluded the identification of SSCs that should have included within the scope of license renewal in accordance with 10 CFR 54.4(a). Describe any additional scoping evaluations performed to address the 10 CFR 54.4(a) criteria. List any additional SSCs included within the scope of license renewal as a result of the review, and any SCs for which AMRs were performed. For SCs for which AMRs were performed, describe the AMPs, as applicable, to be credited for managing the identified aging effects.

Callaway Response

Various documentation sources were used to determine what components were within the scope of license renewal. One of the sources was the Callaway Equipment List (CEL), which is maintained in a database called EB Director. The CEL includes a field named Q-QUAL, in which safety-related components are flagged. The basis for determining whether a component is flagged as safety-related is the definition of safety-related in FSAR Section 1.1.7. The FSAR definition is consistent with the scoping criteria of 10 CFR 54.4(a)(1). All safety-related

components in the CEL have a Y flag in the Q-QUAL field to indicate that they are safety-related.

During the scoping process, all components were considered to be safety-related if they had a Y flag in the Q-QUAL field of the CEL. Several electrical components (along with the associated cabling) in the turbine building had Y in the Q-QUAL field. These components included eight solenoid valves associated with the air operators of the main feedwater regulating valves, eight solenoid valves associated with the main feedwater bypass valves, two turbine first stage pressure transmitters, four turbine control valve hydraulic oil pressure transmitters, and four turbine stop valve position switches. No other electrical components in the turbine building were identified with Y in the Q-QUAL field of the CEL.

From a review of the FSAR and engineering evaluations, it was determined that these components do not have a safety-related function. The components were therefore not included within the scope of license renewal based on the criteria of 10 CFR 54.4(a)(1). For cases where a component has a Y flag in the Q-QUAL field but was determined not to have a safety-related function, a corrective action document was initiated.

Since it is assumed during the initial scoping and screening that a component with a Y flag in the Q-QUAL field of the CEL has a safety-related function, all components with a Y flag were reviewed. Thus, this methodology did not preclude the identification of safety-related SSCs that should have been included within the scope of license renewal. No additional scoping evaluations were required. No SSCs were added to the scope of license renewal, and no additional AMRs were performed.

Corresponding Amendment Changes

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

RAI 2.1-4

Background:

Title 10, Section 54.4, "Scope," of the CFR states, in part:

a) Plant systems, structures and components within the scope of this part are –

1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49 (b)(1)) to ensure the following functions –

   i. The integrity of the reactor coolant pressure boundary;

   ii. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

   iii. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

2) All nonsafety-related systems, structures and components whose failure could prevent satisfactory accomplishment of any of the functions identified in (a)(1)(i), (ii), or (iii) of this section.

Issue:

During the on-site scoping and screening methodology audit the staff reviewed the LRA, the 10 CFR 54.4(a)(2) implementing documents and license renewal drawings, and also performed plant walkdowns. The staff determined that LRA Section 2.1.2.2, "10 CFR 54.4(a)(2) – Nonsafety-Related Affecting Safety-Related," states that for nonsafety-related SSCs directly connected to safety-related SSCs, "equivalent anchors as defined in the Current Licensing Basis (CLB)," were not used because equivalent anchors are not defined in the Callaway CLB. However, during a review of the license renewal drawings, the staff noted examples where the drawing notes credited "equivalent anchors as defined in the CLB" as the termination point for attached piping included within the scope of license renewal in accordance with 10 CFR 54.4(a)(2).

Request:

The staff requests that the applicant discuss the use of equivalent anchors supporting nonsafety-related SSCs, connected to safety-related SSCs, included within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). The staff requests that the applicant perform a review of this issue and indicate if the review concludes that use of the scoping methodology precluded the identification of SSCs that should have been included within the scope of license renewal in accordance with 10 CFR 54.4(a). Describe any additional scoping evaluations performed to address the 10 CFR 54.4(a) criteria. List any additional SSCs included within the scope of license renewal as a result of the review, and any SCs for which AMRs were performed. For SCs for which AMRs were performed, describe the AMPs, as applicable, to be credited for managing the identified aging effects.

Callaway Response

Credit was taken in license renewal for equivalent anchors as defined in the CLB at the locations where the main steam piping passes through the auxiliary building-turbine building wall.

For this piping, the Callaway FSAR defines the portion of piping extending from the outside of the inboard isolation restraint to the outside of the outboard isolation restraint as a "no break zone". The inboard isolation restraint for a main steam line is the anchor at the containment penetration. The outboard isolation restraint is at the penetration in the auxiliary building-turbine building wall. The FSAR states that the maximum stress in the "no break zone" is acceptable when subjected to the combined loadings of internal pressure, deadweight, and postulated pipe break beyond the "no break zone".

The FSAR further states that isolation restraints protect an essential portion of a piping system from postulated leaks either upstream or downstream of the protected area. These restraints limit pipe motion in all directions.

Thus, the isolation restraints in the auxiliary building-turbine building wall for the main steam piping are equivalent anchors as defined in the CLB.

LRA Section 2.1.2.2 has been revised to delete the statement that equivalent anchors are not defined in the Callaway CLB.

Crediting the isolation restraints in the auxiliary building-turbine building wall for the main steam piping as equivalent anchors did not preclude the identification of any SSCs which should have been included within the scope of license renewal. No additional scoping evaluations were required. No SSCs were added to the scope of license renewal, and no additional AMRs were performed.

Corresponding Amendment Changes

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

RAI B2.1.26-1

Background:

LRA Section B2.1.26, states that the American Society for Mechanical Engineers (ASME) Section XI, Subsection IWE Program is an existing program that, following enhancement, will be consistent with NUREG 1801, "Generic Aging Lessons Learned (GALL) Report", AMP XI.S1, "ASME Section XI, Subsection IWE," Program Element 3, "parameters monitored or inspected," of the GALL Report AMP XI.S1 recommends the examination of "moisture barriers" for wear, damage, erosion, tear, surface cracks, or other defects that permit intrusion of moisture in the inaccessible areas of the pressure retaining surfaces of the metal containment shell or liner. Furthermore, ASME Section XI, Subsection IWE, Table-2500-1 "Examination Category E-A" requires 100 percent general visual examinations of moisture barriers (E1.30) during each examination period.

Issue:

During the audit, the staff reviewed Callaway Plant Aging Management Program Evaluation Report, "ASME Section XI, Subsection IWE-B2.1.26." The staff noted that Program Element 3, "parameters monitored or inspected," of this evaluation report does not include inspection of moisture barriers as part of the IWE program. In addition, during the audit, the staff did not find any reference to the moisture barriers in the IWE periodic examination inspection reports.

Request:

a) Confirm if moisture barriers, as shown in Figure IWE-2500-1 of the ASME Section XI, Subsection IWE Code, are installed in the Callaway containment.

b) If moisture barriers have been installed, provide justification for not including the examinations of "moisture barriers" during each examination period.

Callaway Response

a) and b) The Callaway Plant does not have a moisture barrier seal at the interface between the steel containment liner and the internal concrete structures. A fill slab is installed directly over the horizontal liner plate on the floor and directly against the vertical liner plate on the wall.

LRA Table 2.4-1, Table 3.5-1, Table 3.5.2-1, and Section 3.5.2.2.1.3.1 have been revised as shown on Amendment 6 in Enclosure 2 to delete reference to moisture barriers in the scope of the IWE inspections.

Corresponding Amendment Changes

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

RAI B2.1.27-1

Background:

GALL Report AMP XI.S1 “ASME Section XI, Subsection IWL,” Program Element 6, “acceptance criteria,” recommends acceptance criteria provided in IWL-2510, which references American Concrete Institue (ACI) 201.1R and ACI 349.3R for identification of concrete degradation. The LRA AMP basis document for Program Element 6, “acceptance criteria,” states that Callaway acceptance criteria for concrete degradation is in accordance with IWL-2510, and is consistent with ACI 201.1R and ACI 349.3R.

Issue:

During its onsite audit, staff reviewed plant procedures applicable to the applicant’s IWL program and could not find a reference to ACI 349.3R in any plant procedures. The staff also reviewed results of previous IWL examinations and noted that it appeared that the evaluation criteria specified in ACI 349.3R Chapter 5, “Evaluation Criteria,” are not used during the examinations of the concrete containment building.

Request:

State whether plant procedures reference ACI 349.3R, and if so, how the procedures incorporate the code. Also, provide clarification on whether the evaluation criteria specified in ACI 349.3R Chapter 5 are used during IWL examinations, or justification for not using those code provisions during the examinations of the concrete containment building.

Callaway Response

LRA Appendix B2.1.27 has been revised and LRA Table A4-1, Item 40 has been added as a new commitment as shown on Amendment 6 in Enclosure 2 to specify that acceptability of concrete surfaces is based on the evaluation criteria provided in ACI-349.3R. Basis document, XI.S2, ASME Section XI, Subsection IWL, has been revised to specify that acceptability of concrete surfaces is based on the evaluation criteria provided in ACI-349.3R.

Corresponding Amendment Changes

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

RAI B2.1.27-2

Background:

GALL Report AMP XI.S1 "ASME Section XI, Subsection IWL," Program Element 10, "operating experience," references NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," that described instances of cracked, spalled, and degraded concrete for reinforced and prestressed concrete containments. During the staff's onsite walkdown on May 2, 2012, mapped cracks in concrete were identified around the vertical tendon casings in the tendon gallery.

Issue:

The 25th Year Containment IWL Precision Surveillance Corporation (PSC) Inspection Report identified the referenced cracks as "hair line cracks," less than 0.040 inches in width. The second-tier evaluation criteria of ACI 349.3R is 0.04 inches for "passive cracks." It is not clear to the staff whether those cracks were determined to be "passive cracks," and whether the "Evaluation Criteria" of ACI 349.3R was used to evaluate the findings.

Request:

Provide justification whether the mapped cracks in concrete around the vertical tendon casings in the tendon gallery are "passive cracks," and provide the evaluation criteria used for accepting those conditions.

Callaway Response

ACI 349.3R defines passive cracks as those having an absence of recent growth and an absence of other degradation mechanisms at the crack. The 20th year IWL inspection report noted concrete cracks less than 0.010 inches surrounding the bearing plates. The 25th year IWL inspection report also noted concrete cracks less than 0.010 inches surrounding the bearing plates. There is no indication that these cracks have experienced any recent growth, and the inspections have not identified any other degradation mechanisms at the crack. Therefore, these cracks are passive.

The 25th year IWL inspection report contains the vendor's procedure that specifies the inspection criteria. This procedure specifies that visual examinations will identify concrete deterioration and distress as defined in ACI 201.1 and ACI 349.3R. The section of the 25th year inspection report that noted cracks less than 0.040 inches was the general visual examination performed on the exterior surface of the containment. Describing the cracks as "less than 0.040 inches" denotes that they do not exceed the second-tier limits provided in ACI 349.3R. These cracks were then inspected with a detailed visual examination and found to be less than 0.010 inches in width, and therefore, within the first-tier limits of ACI 349.3R and require no further evaluation.

Corresponding Amendment Changes

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

RAI B2.1.27-3

Background:

During its walkdown on May 2, 2012, the staff observed the containment at various elevations and noted that the containment vent duct installation is blocking a section of the containment exterior surface.

Issue:

ASME IWL requires visual inspection of the concrete containment. The staff could not determine how the area of concrete that is obstructed by the containment vent duct installation has been/will be visually inspected in accordance with the ASME Code.

Request:

Describe how the area of concrete containment that is obstructed by the containment vent duct has been/will be examined during the scheduled IWL inspections, to ensure that the effects of aging of the containment concrete are adequately managed.

Callaway Response

As specified in IWL-1220, portions of the concrete surface that are obstructed by adjacent structures, components, parts, or appurtenances are exempt from the examination requirements of IWL-2000. In accordance with 10CFR50.55a(b)(2)(viii)(E), the Callaway ASME Section XI, Subsection IWL program requires that if concrete inspections determine that conditions exist in accessible areas that could indicate the presence of or result in degradation to inaccessible areas, for each inaccessible area identified, the following information shall be provided in the Owners Activity Report:

a. Description of the type and estimated extent of degradation, and the conditions that led to the degradation.

b. Evaluation of each area, and the result of the evaluation.

c. Description of necessary corrective actions.

A review of Callaway ASME Section XI, Subsection IWL program inspection reports indicates that no conditions exist in accessible areas that could indicate the presence of degradation in inaccessible areas.

Corresponding Amendment Changes

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

RAI B2.1.28-1

Background:

LRA sections B2.1.28 and B2.1.31 state that the applicant's ASME Section XI, Subsection IWF and Structures Monitoring programs are existing programs that after enhancement will be consistent with the GALL Report AMPs XI.S3 and XI.S6, respectively. GALL Report AMP XI.S3 "ASME Section XI, Subsection IWF," and XI.S6, "Structures Monitoring Program," recommend volumetric examinations for high-strength (actual measured yield strength greater or equal to 150 ksi) bolting greater than 1 inch in diameter. The GALL Report also recommends that molybdenum disulfide should not be used as a lubricant.

Issue:

The LRA AMPs include enhancements to perform volumetric examinations on high strength bolting; however, the documents do not clearly state the number of high strength bolts within the scope of the program, or how the high strength bolts will be selected for inspection. The documents also do not discuss molybdenum disulfide lubricants and whether or not they have been used at Callaway.

Request:

a) Clearly explain whether or not high strength bolts are included within the scope of the IWF program and the Structures Monitoring program.

b) If high strength bolts are within scope of either program, explain how the high strength bolts will be selected for volumetric inspection (i.e., all in-scope high strength bolts or a sampling of in-scope high strength bolts). If a sampling approach will be used clearly justify the technical adequacy of the approach, including the sample size and the methodology.

c) State whether or not molybdenum disulfide lubricants have been used on high strength bolts within the scope of either program. If these lubricants have been used, explain what will be done to age manage bolts lubricated with molybdenum disulfide.

d) Explain how the IWF Program will address the GALL Report recommendation to refrain from using molybdenum disulfide as a lubricant.

Callaway Response

a) Callaway Engineering has performed a review of plant documentation to identify structural bolting greater than one inch diameter with an actual measured yield strength greater than or equal to 150 ksi. This review included drawings, plant specifications, vendor specifications, procurement documents, and key word searches of plant databases. Of the structural bolts greater than one inch diameter, none were found to have a specified minimum yield strength greater than or equal to 150 ksi. Additionally, Certified Material Test Reports related to the Steam Generator Replacement Project were reviewed for bolts with a specified minimum yield strength greater than 120 ksi to identify any that might have an actual measured yield strength of 150 ksi or greater. No documentation has been found that identified bolting with an actual measured yield strength greater than or equal to 150 ksi. Therefore, Callaway's conclusion is that there are no structural bolts within the scope of license renewal for which augmented volumetric examinations are required.

b) Not applicable - see response (a).

c) Callaway plant procedures exclude lubricants with molybdenum disulfide from the list of acceptable lubricants, and as described in (a) above, Callaway does not have any structural bolts that are susceptible to stress corrosion cracking (SCC). Also, there has been no plant operating experience at Callaway involving SCC of structural bolting. Therefore, augmented examinations to manage SCC in high-strength bolting are not required.

d) Callaway plant procedures exclude lubricants with molybdenum disulfide from the list of acceptable lubricants.

LRA Appendix B2.1.28, Appendix B2.1.31, Table A4-1 item 22, and Table A4-1 item 23 have been revised as shown on Amendment 6 in Enclosure 2 to delete the enhancements that would require volumetric examinations of high-strength bolts. LRA Table 3.5-1, Item 068 and Item 069 have been revised as shown on Amendment 6 in Enclosure 2 to specify that these lines are not applicable to Callaway. LRA Table 3.5.2-1, Table 3.5.2-2, Table 3.5.2-3, Table 3.5.2-4, Table 3.5.2-5, Table 3.5.2-7, Table 3.5.2-9, Table 3.5.2-10, and Table 3.5.2-12 have been revised as shown on Amendment 6 in Enclosure 2 to delete the lines associated with Table 1 Item 068 and Item 069.

Corresponding Amendment Changes

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

RAI B2.1.28-2

Background:

The GALL Report AMP XI.S3 Program Element 2, “preventive actions” states that if American Society for Testing and Materials (ASTM) A325, ASTM F1852, and/or ASTM A490 bolts are used, the preventive actions as discussed in Section 2 of the Research Council for Structural Connections (RCSC) “Specification for Structural Joints Using ASTM A325 or A490 Bolts” should be followed. An enhancement in the LRA states that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the applicable EPRI guidelines, ASTM standards, AISC specifications, and NUREG recommendations to prevent or mitigate degradation and failure of safety-related bolting due to stress corrosion cracking.

Issue:

It is not clear whether the applicant follows the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts.”

Request:

If ASTM A325, ASTM F1852, and/or ASTM A490 bolts are used, explain how the preventive actions discussed in Section 2 of “Specification for Structural Joints Using ASTM A325 or A490 Bolts” are addressed, or why they are unnecessary. If this document will be added as a reference for the program, include a specific reference to the document in the Final Safety Analysis Report (FSAR) Supplement program summary for the IWF program.

Callaway Response

LRA Table A4-1 Item 22, and Appendix B2.1.28 have been revised as shown on Amendment 6 in Enclosure 2 to specify that if ASTM A325, ASTM F1852, and/or ASTM A490 bolts are used, the preventive actions as discussed in Section 2 of the Research Council for Structural Connections “Specification for Structural Joints Using ASTM A325 or A490 Bolts” will be followed.

Corresponding Amendment Changes

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

RAI B2.1.28-3

Background:

The LRA states that the ASME Section XI, Subsection IWF program, with enhancements, is consistent with GALL Report AMP XI.S3. The GALL Report AMP XI.S3 states that the ASME Code, Section XI, Subsection IWF, constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3 and MC component supports for license renewal. ASME Section XI, Subsection IWF, states that to the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval.

Recent industry operating experience has revealed situations where supports within the IWF sample were degraded, but did not meet the IWF threshold for repair. The supports were reworked to as-new condition and remained in the IWF sample

The staff's concern with respect to aging management is that if ASME Code, Section XI, Subsection IWF supports that are part of the inspection sample are reworked to as-new condition, they are no longer typical of the other supports in the population. Subsequent ASME Code, Section XI, Subsection IWF inspections of the same sample would not represent the age-related degradation of the rest of the population.

Issue:

The applicant's LRA and associated basis documents provide no discussion of how this issue would be addressed, or how the IWF sample would be altered if a support within the original sample was reworked.

Request:

Describe how the LRA ASME Section XI, Subsection IWF program would address a situation where supports in the sample population are reworked or replaced even though they do not necessarily require corrective actions per the ASME Section XI, Subsection IWF acceptance criteria. Clearly explain how the sample population would continue to be representative of the overall population.

Callaway Response

The Callaway ASME Section XI, Subsection IWF program has not reworked or replaced any supports in the sample population that did not require corrective actions per the ASME Section XI, Subsection IWF acceptance criteria. If supports are reworked or replaced under the current IWF program, these supports will not be included in subsequent sample selections. If such a situation should be encountered in the future, sample selection and subsequent inspections will be performed in accordance with the applicable codes and commitments at that time.

Corresponding Amendment Changes

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

RAI B2.1.29-1

Background:

LRA Section B2.1.29 states that the 10 CFR Part 50 Appendix J AMP has implemented Option B for the 10 CFR Part 50 Appendix J leak rate tests (LRTs) and is consistent with the GALL Report, Revision 2, AMP XI.S4. The LRA further states that the 10 CFR Part 50 Appendix J program ensures that the structural integrity of the containment will be maintained to withstand the maximum calculated pressure in the event of a loss of coolant accident (LOCA). Measure of leakage rates across pressure containing or leakage limiting boundaries and inspections as implemented through the program provide for the detection of age-related pressure boundary degradation for the period of extended operation. Per the "scope of program," program element of the GALL Report AMP XI.S4, all containment boundary pressure-retaining components are subject to leak rate testing and inspections.

Issue:

Callaway Plant Unit 1 FSAR-SP, and "ESP-SM-01001, Containment Leakage Rate Testing Program," procedure indicate that a number of penetrations are excluded from local leak rate tests (LLRTs). In addition, the audited plant's operating experience database indicated that the applicant has substituted LLRTs in lieu of VT-2 inspections. It is not clear how the applicant will manage the aging effects for any components that are not included in its "scope of program," program element.

Request:

For those components (valves, penetrations, and other components) that have been excluded from the 10 CFR Part 50 Appendix J program, identify how aging effects will be managed during the period of extended operation. Indicate which AMPs will be used to manage the aging effects for each of the exempted/excluded components, or justify why an AMP is not necessary for the period of extended operation.

Callaway Response

Pressure-retaining components whose failure (loss of leak-tightness) could contribute to an increase in the overall integrated leakage rate of the containment system are subjected to Type A Integrated Leak Rate Testing (ILRT).

Containment penetrations that are provided with double seal closures and connections to allow for pressurization between the seals are subjected to Type B Local Leak Rate Testing (LLRT).

Containment isolation valves that meet the following criteria are subjected to Type C LLRT:

a. The penetrating system provides a direct connection between the inside and outside atmospheres of the containment under normal operation.

b. The system is isolated by containment isolation valves that close automatically to effect containment isolation in response to a CIS signal.

c. The system is not an engineered safety feature system consisting of a closed piping system outside of the containment.

As stated in FSAR-SP, Section 6.2.6.1.2, ILRT Test Method, "For penetrations that are exempt from Type B or C tests, the leakage testing requirement of Appendix J is accomplished by the Type A test." Therefore, the scope of the 10 CFR Part 50 Appendix J program includes all

pressure-retaining components of the containment structure, and all of these components will be age-managed under this program during the period of extended operation.

Corresponding Amendment Changes

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

RAI B2.1.30-1

Background:

Callaway LRA Section B2.1.30, "Masonry Walls" program, states that the program is an existing program that is consistent with GALL Report AMP XI.S5. Element 6, "acceptance criteria" of GALL Report AMP XI.S5, states that further evaluation is conducted if the extent of cracking and loss of material is sufficient to impact the intended function of the wall or invalidate its evaluation basis.

Issue:

During its audit, the staff reviewed Appendix D of the applicant's procedure ESP-ZZ-01013, "Maintenance Rule Structures Inspection," which includes quantitative acceptance criteria for Masonry Walls. However, it was unclear what the basis was for the acceptance criteria and how the criteria related to the recommended acceptance criteria in the GALL Report AMP.

Request:

Provide the basis for the acceptance criteria described in Appendix D of procedure ESP-ZZ-01013. In addition, explain how the criteria meet the recommendations in Element 6 of the GALL Report AMP XI.S5.

Callaway Response

The acceptance criteria for masonry described in the plant procedure are based on the criteria for concrete provided in ACI 349.3R, modified as necessary due to the nature of masonry construction. Plant procedures require that a Callaway Action Request is written when other than very minor degradation is noted on the inspection report. This is consistent with Element 6 of the GALL Report AMP XI.S5, which specifies that the extent of observed shrinkage and/or separation and cracking of masonry may not invalidate the evaluation basis or impact the wall's intended function and that a further evaluation be conducted if the extent of cracking and loss of material is sufficient to impact the intended function of the wall or invalidate its evaluation basis.

Corresponding Amendment Changes

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

RAI B2.1.31-1

Background:

LRA Sections B2.1.30, “Masonry Walls,” and B2.1.32, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” states that these programs are consistent with the GALL Report, Revision 2, AMPs XI.S5 and XI.S7, respectively. In addition, Callaway LRA Section B2.1.31, “Structures Monitoring” program, states that the program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. GALL Report AMPs XI.S5 “Masonry Walls,” XI.S6, “Structures Monitoring Program,” and XI.S7, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” recommend structures within scope of the programs be inspected on a frequency not to exceed five years.

Issue:

Program Element 4, “detection of aging effects,” of the basis documents of the associated LRA programs state that inspections are performed at intervals of not more than five years. Although this is stated in the basis documents, no discussion of inspection interval is provided in the LRA or the FSAR supplement program summary. This level of detail needs to be captured in the FSAR supplement to provide the staff assurance that the programs will be properly implemented during the period of extended operation.

Request:

Include a statement of the inspection interval in the FSAR supplement. If the interval is greater than five years, list the structures that will be inspected under the longer interval and provide a technical justification for the extended interval. This issue applies to the Masonry Wall, Structures Monitoring, and Regulatory Guide (RG) 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants programs.

Callaway Response

LRA Appendix A1.30, Appendix A1.31, Appendix A1.32, Appendix B2.1.30, Appendix B2.1.31, and Appendix B2.1.32 have been revised as shown on Amendment 6 in Enclosure 2 to specify that inspections of all structural components, including masonry walls and water-control structures, are performed at intervals of no more than 5 years. Basis documents, XI.S5, Masonry Walls, XI.S6, Structures Monitoring, and XI.S7, RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants, have been revised to specify that inspections of all structural components, including masonry walls and water-control structures, are performed at intervals of no more than 5 years.

Corresponding Amendment Changes

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

RAI B2.1.31-2

Background:

LRA Section B2.1.31, "Structures Monitoring" program, states that the Structures Monitoring program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. The GALL Report XI.S6, "Structures Monitoring Program," in "parameters monitored or inspected," element states that ACI 349.3R and American National Standards Institute (ANSI)/American Society of Civil Engineers (ASCE) 11 provide an acceptable basis for selection of parameters to be monitored or inspected for concrete and steel structural elements. In addition, GALL Report XI.S6, "detection of aging effects" element states that qualifications of inspection and evaluation personnel specified in ACI 349.3R are acceptable for license renewal. The GALL Report also states that applicants who are not committed to ACI 349.3R and elect to use plant-specific criteria for concrete structures should describe the criteria and provide a technical basis for deviations from those listed in ACI 349.3R.

Issue:

In Element 3 of the LRA Structures Monitoring program basis document it states that the inspection methods, inspection frequency, and inspector qualifications at Callaway are consistent with the guidance provided in ACI 349.3R-96. However, a review of the Callaway implementing procedure ESP-ZZ-01013, "Maintenance Rule Structures Inspection," indicates that the inspection methods, including walkdown examination guidance, and qualification requirements for inspectors are not consistent with ACI 349.3R.

Callaway procedure ESP-ZZ-01013, Section 4.0, states that an assigned engineer from the Civil/Structural Design Group will perform the engineering responsibilities for Maintenance Rule Structures Inspection Program. The assigned engineer will possess the experience and skills in civil/structural engineering, consistent with the requirements of the current civil/structural Position Guide and the Engineering Qualification Module. This is inconsistent with the requirements specified in ACI 349.3R which states that responsible-in-charge engineer should be a licensed professional engineer, knowledgeable in the design, evaluation, and in-service inspection of concrete structures and performance requirements of nuclear safety-related structures; or structural engineering graduate of an Accreditation Board for Engineering and Technology, Inc. accredited college or university with at least 10 years' experience in the design, construction, and inspection of concrete structures, and with knowledge of the performance requirements of nuclear safety-related structures and potential degradation processes.

ACI 349.3R recommends a three tier quantitative evaluation criteria for inspection of structures. However, the walkdown guidelines in the Callaway procedure ESP-ZZ-010013 require inspection based on a qualitative acceptance criteria.

Request:

Explain the reason for inconsistency in inspection methods and inspector qualifications as described in the LRA Section B2.1.31 and implementing procedure ESP-ZZ-01013 as identified above.

Callaway Response

LRA Table A4-1 Item 21, documents the plant commitment to enhance the Structures Monitoring program procedures to specify inspector qualifications in accordance with ACI 349.3R-96. Basis document, XI.S6, Structures Monitoring, has been revised so that the first sentence of Element 4 reads, "Plant procedures, following enhancements, will specify that inspection methods, inspection frequency, and inspector qualifications at Callaway are consistent with the guidance provided in ACI 349.3R-96."

LRA Table A4-1 Item 21, also documents the plant commitment to enhance the Structures Monitoring program procedures to quantify acceptance criteria and critical parameters for monitoring degradation, and to provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action. Basis document, XI.S6, Structures Monitoring, has been revised to add the following sentence to Element 6: "The Callaway SMP, following enhancement, will quantify acceptance criteria and critical parameters for monitoring degradation, and provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action."

Corresponding Amendment Changes

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

RAI B2.1.31-3

Background:

Callaway LRA Section B2.1.31, "Structures Monitoring" program, states that the Structures Monitoring program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. The GALL Report XI.S6, "Structures Monitoring Program," in the "acceptance criteria" element, states that acceptance criteria should be derived from design bases codes and standards that include ACI 349.3R, ACI 318, ANSI/ASCE 11, or the relevant American Institute of Steel Construction (AISC) specifications, as applicable, and consider industry and plant operating experience. The criteria are directed at the identification and evaluation of degradation that may affect the ability of the structure or component to perform its intended function. Applicants who are not committed to ACI 349.3R and elect to use plant-specific criteria for concrete structures should describe the criteria and provide a technical basis for deviations from those in ACI 349.3R.

Issue:

In Element 6 of the LRA Structures Monitoring program basis document it states that the Callaway Structures Monitoring program provides guidance for the determination of the performance criteria for SSCs included within the scope of Maintenance Rule (Reference: ESP-ZZ-01013, Appendix B). These guidelines were used to establish the inspection attributes for SSCs. Callaway's Structures Monitoring program uses "Acceptable," "Acceptable with Deficiencies," and "Unacceptable," to classify levels of aging effects for each inspection attribute. Element 6 of the program also states that the Structures Monitoring program will be enhanced to quantify acceptance criteria and critical parameters for monitoring degradation, and procedures will also be enhanced to incorporate applicable industry codes, standards and guidelines for acceptance criteria.

The staff reviewed the proposed enhancement described in Appendix D of the procedure ESP-ZZ-01013, and could not find the basis or reference to any industry standard for some of the acceptance criteria. In addition, Section 7.5 and Appendix D of the procedure ESP-ZZ-01013 do not appear to have consistent acceptance criteria.

Request:

Provide the basis for the acceptance criteria described in Appendix D of procedure ESP-ZZ-01013. In addition, explain the inconsistency between Structures Monitoring program acceptance criteria of "Acceptable," "Acceptable with Deficiencies," and "Unacceptable," as identified in Section 7.5, and quantitative acceptance criteria described in Appendix D of the procedure ESP-ZZ-01013.

Callaway Response

The acceptance criteria for the Structures Monitoring program are based on those provided in ACI 349.3R. For inspection of components and materials that are not specifically addressed in ACI 349.3R, the acceptance criteria are modified as necessary to account for the nature of the component being inspected. The markup of the Structures Monitoring procedure has been revised to align with the criteria of ACI 349.3R and to clarify the definitions of the Acceptance Criteria categories so that they also align with ACI 349.3R.

Corresponding Amendment Changes

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

RAI B2.1.31-4

Background:

LRA Section B2.1.31, "Structures Monitoring," program states that the Structures Monitoring program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. The GALL Report requires that operating experience involving the AMP, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the structure and component intended functions will be maintained during the period of extended operation.

Issue:

Appendix C of the implementing procedure ESP-ZZ-01013 for the Structures Monitoring program states that gaps filled with an elastomeric are deemed to have degraded when they no longer perform their intended function. Functionality is compromised if the elastomer is cracked, torn, brittle, not bonded to adjoining media, or shows signs of in-leakage. Furthermore, an elastomer filled gap is deemed to have been degraded if the gap is occupied by a foreign material and is considered to have failed if the as-found gap dimension is less than that prescribed by design.

During the audit, the staff reviewed Callaway Action Request (CAR) 200609470, and also observed during plant walkdowns that elastomers in the seismic gaps of the containment, turbine and auxiliary buildings have been degraded and are not functional. According to CAR 200609470, this condition has been identified in 2006 and has caused in-leakage of ground and rain water into the buildings. In addition, during the audit, the staff observed that in one area in the auxiliary building the elastomer has been replaced with a foreign material. Lack of flexible elastomer gap can cause seismic interaction between the buildings resulting in stresses not considered in the design that may affect structural integrity of the structures during the period of extended operation.

Request:

Provide a summary of the plans and schedules to replace the non-functional elastomers located in seismic gaps. This information is required to determine if the Structures Monitoring AMP is effective and being implemented to ensure that the aging of the containment, auxiliary, and turbine buildings concrete will be managed adequately during the period of extended operation.

Callaway Response

Seismic isolation gaps at the roof, base slab, and exterior walls are constructed with waterstops embedded in the concrete. The elastomeric compressible material that is visible at the edge of the joints is a separate component from the waterstops. FSAR-SP Figure 3.8-85 shows typical isolation joint details. The intended function of the seismic isolation gap compressible material is to shelter and protect the gap from debris intrusion so that seismic isolation is maintained.

Inspections for all accessible seismic gaps between buildings will be performed before December 31, 2012. An engineer familiar with the seismic design of the plant will evaluate the results of these inspections. Any corrective actions resulting from these inspections will be completed no later than December 31, 2017.

LRA Appendix B2.1.31 and Table A4-1, Item 23, have been revised as shown on Amendment 6 in Enclosure 2 to enhance the Structures Monitoring program to specify that degradation associated with seismic isolation gaps, obstructions of these gaps, or questionable material in these gaps, will be evaluated by an engineer familiar with the seismic design of the plant. The evaluation will consider the seismic isolation function in determining what corrective actions may be required.

Corresponding Amendment Changes

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

RAI B2.1.31-5

Background:

LRA Section B2.1.31, “Structures Monitoring” program, states that the program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. In the “program description,” of LRA Section B2.1.31, the applicant states that the Structures Monitoring program implements the requirements of 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”

Issue:

Paragraph 50.65(b) of 10 CFR states:

b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:

1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in [Sec.] 50.34(a)(1), [Sec.] 50.67(b)(2), or [Sec.] 100.11 of this chapter, as applicable.

2) Nonsafety related structures, systems, or components:

   i. That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

   ii. Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

   iii. Whose failure could cause a reactor scram or actuation of a safety-related system.

Callaway LRA Section B2.1.31, “Structures Monitoring” program, and Commitment No. 23 state that the program will be enhanced before the period of extended operation to include inspections of penetrations, transmission towers, electrical conduits, raceways, cable trays, electrical cabinets/enclosures, and associated anchorages. However, contrary to 10 CFR 50.65 requirements, Element 1, “scope of program,” of the Structures Monitoring program basis document as well as the implementing procedure, ESP-ZZ-01013 of the Structures Monitoring program has excluded penetrations, pipe and raceway supports, cable trays, anchor bolts, transmission towers, electrical conduits, and ultimate heat sink retention pond from the scope of existing AMP. None of these structures have been inspected until now, and the applicant has no plans to inspect them until the period of extended operation. Baseline inspection and trending of degradations in the excluded structures prior to the period of extended operation is necessary for appropriate aging management.

Request:

Provide summary description of plans and schedule for baseline inspection and trending of degradations of the structures noted above that have been excluded from Element 1, “scope of program,” of the Structures Monitoring program. In addition, provide technical basis for not including baseline inspection of penetrations, transmission towers, electrical conduits, raceways, cable trays, electrical cabinets/enclosures, and associated anchorages, and ultimate sink retention pond as a part of the Structures Monitoring program.

Callaway Response

LRA Table A4-1, Item 23 and Appendix B2.1.31 have been revised as shown on Amendment 6 in Enclosure 2 to specify that baseline inspections of penetrations, transmission towers, electrical conduits, raceways, cable trays, electrical cabinets/enclosures, and associated anchorages will be completed by December 31, 2017. These baseline inspections will be performed as part of the Structures Monitoring program, which has been enhanced to add these components to its scope. Previously, these components were considered to be part of their respective mechanical or electrical systems. Beginning with these baseline inspections and continuing through the period of extended operation, these components will be within the scope of the Structures Monitoring program for all inspections, monitoring, and trending that may be required. The ultimate heat sink retention pond is already within the scope of the Structures Monitoring program and will continue to be so.

Corresponding Amendment Changes

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

RAI B2.1.31-6

Background:

Callaway LRA Section B2.1.31, "Structures Monitoring" program, states that the program is an existing program that, following enhancement, will be consistent with GALL Report, Revision 2, AMP XI.S6. In the GALL Report AMP XI.S6, Program Element 4, "detection of aging effects," notes that the Structures Monitoring program addresses detection of aging effects for inaccessible, below-grade concrete structural elements, and for plants with non-aggressive ground water/soil (pH greater than 5.5, chlorides less than 500 ppm, or sulfates less than 1500 ppm). The program recommends: (a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas and (b) examining representative samples of the exposed portions of the below grade concrete, when excavated for any reason. The GALL Report also notes that for plants with aggressive ground water/soil (pH less than 5.5, chlorides greater than 500 ppm, or sulfates greater than 1500 ppm) and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

Issue:

The LRA states that groundwater is monitored for pH, chlorides, and sulfates every five years, and the results evaluated by engineering to assess the impact, if any, on below grade structures. The LRA further states that groundwater has been sampled quarterly since November 2009. With the exception of two monitoring wells, pH, chlorides, and sulfates concentrations have been within the prescribed limits for non-aggressive ground water/soil. The two wells have shown seasonal increases in chloride levels of up to 680 mg/L while the pH and sulfate concentrations have remained non-aggressive. The LRA further states that Callaway will continue to monitor the results from the groundwater samples and will perform an engineering evaluation to determine if any adverse aging effects have occurred in any inaccessible concrete structural elements. The LRA does not provide any details on how the aging of the inaccessible concrete elements will be managed during the period of extended operation due to the presence of high chloride concentrations.

Request:

a) Provide historical results, including seasonal variations, for groundwater chemistry (i.e., pH, sulfates, and chlorides) to demonstrate that the groundwater is either aggressive or non-aggressive.

b) Provide details on how the aging of the inaccessible concrete elements will be managed during the period of extended operation due to the presence of high chloride concentrations.

Callaway Response

a) Beginning in November 2009, 24 monthly groundwater samples were taken from 11 sampling wells around the Callaway site. Sulfates and pH levels were well within acceptable limits with no significant seasonal variations. Samples taken from one well in August and September of 2010 showed chloride levels of 520 ppm and 570 ppm. In March and April of 2011, samples taken from a different well had chloride levels of 660 ppm and

610 ppm. All other samples showed chloride to be below the 500 ppm threshold for classifying groundwater as aggressive.

b) An engineering evaluation was performed to assess the impact, if any, on existing below-grade structures due to the elevated levels of chloride that have been detected in the groundwater monitoring wells. This evaluation included a visual inspection of the interior surfaces of concrete walls below grade in the auxiliary building, communications corridor, and turbine building. In the auxiliary building, some water in-leakage was observed around a wall penetration, and there was moisture on one of the concrete expansion joints. There was no indication of in-leakage in the communications corridor. In the turbine building, there was indication of previous leakage through minor cracks and some minor leachate. There was no indication of concrete damage resulting from aggressive groundwater in any of these buildings. Previous modification work in 2008 and 2009 exposed the below grade exterior surface of the control building. There was no indication of degradation of the concrete on the exterior of the wall at that time. The conclusion of this evaluation was that there is no indication that Callaway has concrete or reinforcing steel degradation resulting from aggressive groundwater.

   Callaway will continue to monitor groundwater for pH, chlorides, and sulfates every five years through the period of extended operation, and the results will be evaluated by engineering to assess the impact, if any, on below grade structures. The Structures Monitoring program monitors the plant structures with scheduled visual examinations of the accessible structures and opportunistic examinations of inaccessible structures. These examinations will continue through the period of extended operation to confirm the absence of concrete degradation due to aggressive groundwater.

Corresponding Amendment Changes

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

RAI 3.5.2.11-1

Background:

LRA Section B2.1.31 states that The Structures Monitoring program is an existing program that, following enhancement, will be consistent with GALL Report AMP XI.S6, "Structures Monitoring Program." In GALL Report AMP XI.S6, Elements 3 and 4 state that for each structure/aging effect combination the specific parameters monitored or inspected are selected to ensure that the aging degradation leading to loss of intended function will be detected and quantified before there is a loss of intended function. For stainless steel submerged in water (water-standing), the GALL Report, Item III.A7.T.23, lists cracking and loss of material as possible aging effects and recommends a plant specific AMP is to be evaluated.

Issue:

LRA Table 3.5.2-11 lists stainless steel supports in a submerged environment and does not include cracking as an applicable aging effect. In addition, it states that the loss of material will be managed by the Structures Monitoring program.

Request:

a) Describe the type, condition, and temperature of water in which the stainless steel supports are submerged.

b) Explain why cracking is not an applicable aging effect for the submerged stainless steel supports or include an appropriate AMP to manage cracking in submerged stainless steel supports.

c) If an AMP is credited with managing the aging effect due to cracking, please provide a technical justification for the credited aging management technique (i.e., inspection method, frequency, and acceptance criteria).

d) Explain how the Structures Monitoring program will manage aging effects due to loss of material (i.e., inspection method, frequency, and acceptance criteria).

Callaway Response

a) The submerged stainless steel components listed in LRA Table 3.5.2-11 are embedded in the surface of the concrete in the ESWS pumphouse. They are exposed to the raw water that passes through the pumphouse from the retention pond at ambient temperatures.

b) The water in which these components are submerged is not heated above atmospheric temperatures, and it is refreshed on a regular basis. The environment associated with GALL line III.A7.T.23 is Water-Standing, which is defined in GALL Section IX.D as, "Water that is stagnant and unrefreshed, thus possibly resulting in increased ionic strength up to saturation." Therefore, GALL line III.A7.T.23 is not applicable, and Cracking is not an aging effect requiring management for these components.

c) Not Applicable - see response b).

d) These components are within the scope of license renewal and are subject to aging management under the Structures Monitoring program (SMP). The pumphouse bays are dewatered on a frequency of approximately every five years, timed to coincide with plant outages. While the unit is dewatered, a visual examination of the structural components is

performed. Any indication of loss of material, such as corrosion, is evaluated for inclusion into the plant corrective action program.

Corresponding Amendment Changes

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

RAI 3.5.2.11-2

Background:

LRA Section B2.1.31 states that the Structures Monitoring program is an existing program that, following enhancement, will be consistent with GALL Report AMP XI.S6, "Structures Monitoring Program." In GALL Report AMP XI.S6, program Elements 3 and 4 state that for each structure/aging effect combination the specific parameters monitored or inspected are selected to ensure that the aging degradation leading to loss of intended function will be detected and quantified before there is a loss of intended function.

Issue:

LRA Table 3.5.2-11 states that the aging of the fiberglass reinforced plastic exposed to atmosphere/weather environment will be managed in accordance with the Structures Monitoring program by monitoring for cracking, blistering, change in color. However, a review of LRA Section B2.1.31, "Structures Monitoring" program indicates that this program does not have any specific guidance for monitoring or acceptance criteria for fiberglass reinforced plastic material.

Request:

Explain what parameters will be monitored and acceptance criteria will be used for aging management of fiberglass reinforced plastic components exposed to atmosphere/weather environment.

Callaway Response

LRA Appendices A1.31 and B2.1.31 list cracking, blistering, and change in color as aging effects that are managed by the Structures Monitoring AMP. Basis document XI.S6, Structures Monitoring, Element 3 has been revised to specify that Fiberglass Reinforced Plastic will be inspected for cracking, blistering, and change in color. Plant procedure markups have been revised to specify that Fiberglass Reinforced Plastic will be inspected and any indication of cracking, blistering, or change in color will be evaluated for inclusion into the plant corrective action program.

Corresponding Amendment Changes

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

RAI 3.5.2.12-1

Background:

LRA Section B2.1.28 states that the ASME Section XI, Subsection IWF program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S3, "ASME Section XI, Subsection IWF." LRA Section B2.1.32 states that the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program is an existing program that is consistent with GALL Report AMP XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants." GALL Report AMP XI.S3,

"ASME Section XI, Subsection IWF," states that ASME Code, Section XI, Subsection IWF, constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3, and MC component supports for license renewal.

Issue:

LRA Table 3.5.2-12 states that the aging of the ASME Class 2 and 3 carbon steel supports that are submerged in water will be managed by GALL Report AMP XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," by monitoring for loss of material. The scope, frequency, and acceptance criteria as specified in GALL Report AMP XI.S3 for ASME Class 2 and 3 supports is different from that specified in GALL Report AMP XI.S7 for non ASME steel components monitored for loss of material due to corrosion. GALL Report AMP XI.S3 recommends that a sample of ASME Class 1, 2, and 3 component supports that are not exempt from examination and 100 percent of MC component supports be examined as specified in Table IWF-2500-1. The sample size examined for ASME Class 1, 2, and 3 component supports is as specified in Table IWF-2410-2. The detailed acceptance standards for ASME Class 1, 2, and 3 component supports are delineated in IWF-3400. GALL Report AMP XI.S7 recommends inspection of structures and components at a frequency of five years in accordance with ACI 349.3R.

Request:

Explain why LRA AMP B2.1.32 instead of LRA AMP B2.1.28 is being used to monitor ASME Class 2 and 3 supports that are submerged in water. This should include a discussion on how the scope, frequency, and acceptance criteria specified in ASME Section III, Subsection IWF can be implemented by LRA AMP B2.1.32 for ASME Class 2 and 3 carbon steel supports that are submerged in water.

Callaway Response

The ASME Class 2 and 3 carbon steel supports submerged in water were included in LRA Table 3.5.2-12 as supports for the ESW discharge pipe located in the UHS Cooling Tower sump. As shown in FSAR-SA Figure 3.8-9, this 36 inch diameter pipe is supported within the concrete wall and does not require an additional support structure inside the sump. Therefore, LRA Table 3.5.2-12 and LRA Section 3.5.2.1.12 have been revised as shown on Amendment 6 in Enclosure 2 to indicate the absence of submerged ASME Class 2 and 3 supports.

Corresponding Amendment Changes

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

RAI B2.1.36-1

Background:

Callaway LRA Section B2.1.36, “Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” states that the program is an existing program that, following enhancement, will be consistent with GALL Report AMP XI.E3. The GALL Report AMP XI.E3 “scope of program” program element recommends a voltage level of greater than or equal to 400V.

Issue:

The “scope of program” program element of basis document CW-AMP-B2.1.36, “Inaccessible Power Cable Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” does not indicate the voltage level of in-scope inaccessible power cable (e.g., greater than or equal to 400 Volts). Additionally, LRA FSAR Supplement A1.36 and Table A4-1, “License Renewal Commitments,” do not specify voltage level.

Request:

Explain why the “scope of program” program element for basis document CW-AMP-B2.1.36, “Inaccessible Power Cable Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” LRA FSAR Supplement A1.36, “Inaccessible Power Cables Not Subject to10 CFR 50.49 Environmental Qualification Requirements,” and Table A4-1, “License Renewal Commitments,” do not reference in-scope inaccessible power cable voltage level (greater than or equal to 400 Volts).

Callaway Response

LRA Table A4-1 item 27, LRA Appendix A1.36, and the Enhancement to Element 1 in LRA Appendix B2.1.36, have been revised as shown by LRA Amendment 6 in Enclosure 2 to indicate that in-scope non-EQ inaccessible power cables are greater than or equal to 400 volts. Basis document CW-AMP-B2.1.36 has also been revised to indicate that in-scope non-EQ inaccessible power cables are greater than or equal to 400 volts.

Corresponding Amendment Changes

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

RAI B2.1.36-2

Background:

Basis document CW-AMP-B2.1.36, “Inaccessible Power Cable Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” specifies the in-scope manholes under “scope of program” program element.

Issue:

Basis document CW-AMP-B2.1.36, “Inaccessible Power Cable Not Subject to 10 CFR 50.49 Environmental Qualification Requirements,” does not include in-scope manholes MH59-8A and MH59-8B. See Table 2 of draft procedure EDP-ZZ-07001.

Request:

Explain why in-scope manholes MH59-8A and MH59-8B are not included in basis document CW-AMP-B2.1.36, “scope of program” program element.

Callaway Response

Basis document CW-AMP B2.1.36, Element 3.1, Scope of Program, has been revised to include MH59-8A and MH59-8B.

Corresponding Amendment Changes

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

RAI B2.1.36-3

Background:

GALL Report AMP XI.E3 Program Element 2, “preventive actions,” recommends that the inspection should include direct observation that cables are not wetted or submerged, that cable/splices and support structures are intact, and that dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. In addition, operation of dewatering devices should be inspected and operation verified prior to any known or predicted heavy rain or flooding events. GALL Report AMP XI.E3 Program Element 4, “monitoring and trending,” includes trending actions as part of the AMP, although the ability to trend results is dependent on the specific tests or inspection. GALL Report AMP XI.E3 Program Element 6, “acceptance criteria,” recommends that the acceptance criteria for each test are defined by the specific type of test performed and the specific cable tested. GALL Report AMP XI.E3 Program Element 7, “corrective actions,” states that corrective actions are taken and an engineering evaluation is performed when the test or inspection criteria are not met.

Issue:

The enhancements listed in LRA Appendix A, Table A4-1 Commitment No. 27, and the AMP described in LRA FSAR Supplement A1.36, are not in agreement. Specifically, Table A4-1 includes enhancements that state the following but are not addressed in LRA FSAR Supplement A1.36.

a. Operation of dewatering devices will be inspected and operation verified prior to any known or predicted heavy rain or flooding events.

b. Compare test results to previous test results to evaluate for additional information on the rate of cable degradation.

c. Acceptance criteria for cable testing will be defined prior to each test.

d. Require an engineering evaluation when the test or inspection acceptance criteria are not met.

Request:

Explain why the above enhancements described in LRA Appendix A, Table A4-1, Commitment No. 27 are not included in Appendix A FSAR Supplement A1.36.

Callaway Response

LRA Appendix A1.36 has been revised as shown by LRA Amendment 6 in Enclosure 2 and basis document CW-AMP-B2.1.36 has been revised to include:

1. Test results will be compared to previous test results to evaluate for additional information on the rate of cable degradation.
2. Acceptance criteria for cable testing will be defined prior to each test.
3. An engineering evaluation is required when the test or inspection acceptance criteria are not met.

LRA Appendix A1.36 was revised by LRA Amendment 1 to include the following: "Dewatering devices will be inspected and operation verified prior to any known or predicted heavy rain or flooding events."

Corresponding Amendment Changes

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

RAI B2.1.36-4

Background:

GALL Report AMP XI.E3 Program Element 4, “detection of aging effects,” recommends that for power cables exposed to significant moisture, test frequencies are to be at least once every six years adjusted based on test results (including trending of degradation where applicable) and operating experience.

Issue:

Basis document CW-AMP-B2.1.36 “detection of aging effects” program element, procedure EDP-ZZ-07001, and LRA FSAR Supplement A1.36 do not state that test frequencies are adjusted based on test results as well as operating experience.

Request:

Explain why Basis Document CW-AMP-B2.1.36 Element 4, LRA FSAR Supplement A1.36, and procedure EDP-ZZ-07001 do not include revising test frequencies consistent with the recommendations in GALL Report AMP XI.E3 program element “detection of aging effects.”

Callaway Response

LRA Appendix A1.36, LRA Appendix B2.1.36, and LRA Table A4-1 item 27 have been revised as shown by LRA Amendment 6 in Enclosure 2 to read “The first test for license renewal will be completed prior to the period of extended operation with subsequent tests performed at least every six years thereafter and adjusted based on test results and operating experience.”

Basis document CW-AMP-B2.1.36 element 4, and draft procedure EDP-ZZ-07001 paragraphs 4.1.8 and 4.1.9 have been enhanced to state that the frequency of inspection and test is “…adjusted based on test results and operating experience….”

Corresponding Amendment Changes

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

RAI B2.1.36-5

Background:

The applicant's responses to GL 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," and the applicant's discussion of plant operating experience for AMP B2.1.36 states that operating experience with the existing program did not identify any cable failures attributed to submergence or water intrusion.

Issue:

During the audit the staff noted that CAR 201008001 states that some Callaway cable failures were attributed to water submergence.

Request:

Describe the cable failures at Callaway that have been attributed to submergence and describe any changes to the existing program as a result of operating experience gained from these failures.

Callaway Response

During the audit the Staff noted that Callaway Corrective Action Report CAR 201008001 stated that some Callaway cable failures were attributed to water submergence. A subsequent search for cable failures at Callaway confirmed that no cable failures have occurred that could be directly attributed to water submergence, and the statement in CAR 201008001 was inaccurate. The inaccuracy in CAR 201008001 was entered into the Callaway Corrective Action Program as an adverse condition. As corrective action, an independent document review was conducted to verify that there have been no cable failures at Callaway attributed to submergence, and the text in CAR 201008001 was revised to remove the inaccurate statement. The Callaway response to GL 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," and the discussion of plant operating experience for AMP B2.1.36 correctly state that operating experience with the existing program did not identify any cable failures attributed to submergence or water intrusion.

Corresponding Amendment Changes

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

RAI B2.1.36-6

Background:

Draft procedure EDP-ZZ-07001 Section 2.0, "Scope" states that Table 2 lists the Inaccessible Power Cables not subject to environmental qualification requirements and within the scope of license renewal aging management. Table 2 is entitled Underground Cable Requiring Aging Management.

Issue:

Table 2 lists additional cables not in-scope of basis document CW-AMP-B2.1.36 or LRA AMP B2.1.36. Additionally, in-scope cable (B2.1.36) CA-3331L2 is duplicated in Table 2 with different routing.

Request:

Reconcile Table 2 title and Section 2, "Scope," descriptions of included cable for aging management. Identify cables in-scope of the LRA (license renewal) and associated AMPs, including LRA AMP B2.1.36 (GALL Report AMP XI.E3) in Table 2. Provide clarification on duplicate cable listing for CA-3331L2 in Table 2.

Callaway Response

Procedure EDP-ZZ-07001 markup has been revised to include the following:

- Section 2.0 third paragraph states: "Table 2 lists underground cable requiring aging management."
- Added column to Table 2 enhancement to identify whether the cable will be managed by XI.E1 or XI.E3.
- Revised Table 2 to reflect Cable CA-3331L2 is routed from MH59-1A, through MH59-4, and to MH59-5

Corresponding Amendment Changes

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

RAI B2.1.36-7

Background:

GALL Report AMP XI.E3 Program Element 7, “corrective actions,” states that corrective actions are taken and an engineering evaluation is performed when the test or inspection acceptance criteria is not met.

Issue:

Draft procedure EDP-ZZ-07001 Section 4.2.6 states that appropriate corrective actions shall be taken if significant aging that results from adverse environments is identified or suspected.

The corrective action is not consistent with GALL Report AMP XI.E3 Program Element 7, Corrective Actions or basis document CW-AMP-B2.1.36.

Request:

Explain why the corrective actions as described in EDP-ZZ-07001, Section 4.2.6 are not consistent with GALL Report AMP XI.E3 Program Element 7, “corrective actions,” or basis document CW-AMP-B2.1.36.

Callaway Response

Paragraph 4.2.6 of procedure EDP-ZZ-07001 markup was redundant to paragraph 4.3.10 which is cited in basis document CW-AMP B2.1.36, Element 7, Corrective Actions. Procedure EDP-ZZ-07001 markup has been revised to remove the redundancy in paragraph 4.2.6. The corrective actions as described in the basis document remain in paragraph 4.3.10.

Corresponding Amendment Changes

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

RAI B2.1.37-1

Background:

GALL AMP XI.E6 acceptance criteria states that cable connections should not indicate abnormal temperature for the application when thermography is used. In the Callaway Plant Aging Management Program Evaluation Report for Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements AMP, under the same program element, the applicant states that the acceptance criteria for thermography testing will be based on the temperature rise above the reference temperature. The reference temperature will be ambient temperature or the baseline temperature data from the same type of connections being tested. The applicant referenced procedure EDP-ZZ-07001 Section 4.3.13. Section 4.3.13 of this procedure states that the acceptance criteria for the review of thermography result on the one-time inspection of cable connections will be based on the temperature rise above a reference temperature. The reference temperature will be ambient temperature or the baseline temperature data from the connections being tested.

Issue:

The procedure does not specify the acceptance criteria for thermography testing. The acceptance criteria for the program are used to perform an evaluation if the thermography test acceptance criteria are not met. The evaluation may include changes to one-time inspection, increased inspection frequency, and replacement or repair of the affected connections.

Request:

Describe specific acceptance criteria for thermography or reference a plant specific procedure that specifies the acceptance criteria for thermography testing.

Callaway Response

Procedure EDP-ZZ-07001 markup, paragraph 4.1.11, the first sentence has been revised to read: “A one-time inspection of a sample of cable connections shall be conducted using thermography per EDP-ZZ-01113."

EDP-ZZ-01113 specifies thermography inspection action levels for temperatures above a reference temperature. Element 6 AMP basis document B2.1.37 has been revised to identify the action levels section of EDP-ZZ-01113 for determining actions to be taken based on observed temperature rises recorded during the thermography inspection.

Corresponding Amendment Changes

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

RAI 3.5.2.2.1.9-1

Background:

SRP-LR Sections 3.5.3.2.1.9, 3.5.3.2.2.1.4 and 3.5.3.2.2.3.3 recommend further evaluation of programs to manage increase in porosity and permeability due to leaching of calcium hydroxide and carbonation in inaccessible areas. The SRP-LR further states that a plant-specific aging management program is not required, even if reinforced concrete is exposed to flowing water if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.

Issue:

The corresponding sections of the LRA state that concrete structures were designed, constructed, and inspected in accordance with ACI and ASTM standards, which provide for a good quality, dense, well-cured, and low permeability concrete. Therefore, the LRA states that the aging effect of leaching of calcium hydroxide and carbonation is not applicable, and a plant-specific program is not necessary. However, during the AMP audit, the staff observed indications of concrete leaching in the tendon gallery and reviewed CAR 200806266 which documented the leaching. The LRA does not discuss any operating experience related to leaching.

Request:

a. Provide a summary of observed leaching throughout the site.

b. Provide an AMP to manage increases in porosity and permeability due to leaching in inaccessible areas, or provide an evaluation that demonstrates that the observed leaching has no impact on the intended functions of affected structures. This evaluation should cover all affected concrete within the scope of license renewal.

Callaway Response

a) A review of plant corrective action documents verifies that Callaway has not experienced significant leaching through concrete structures within the scope of license renewal. There have been no areas identified where other than minor leaching has occurred. The water in-leaking that has occurred was primarily through degraded elastomeric penetration seals, which have been evaluated and dispositioned in the corrective action program.

b) An engineering evaluation of the leaching in the tendon gallery wall determined that there was no structural integrity concern. The white powdery substance was chemically analyzed and found to be composed of sodium and potassium carbonate with no calcium or iron present. The wall of the tendon gallery was visually inspected. There were no visible cracks in the concrete, and there were no signs of rust or corrosion. Callaway does include the tendon gallery in the scope of the Structures Monitoring program and will continue to inspect and monitor the walls for signs of leaching or other degradation. As noted in (a) above, Callaway has not experienced leaching that would hinder the structures from performing their intended functions. The Structures Monitoring program will continue to monitor concrete structures within the scope of license renewal to confirm the absence of leaching to ensure that the structures continue to perform their intended functions.

Corresponding Amendment Changes

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

RAI 4.2.4-1

LRA Section 4.2.4 indicates that the Callaway pressure-temperature (P-T) limit curves and adjusted nil-ductility reference temperature (RTNDT)values are established in a P-T Limits Report (PTLR), the contents of which are controlled in accordance with Technical Specification (TS) 5.6.6 requirements. The current Callaway PTLR (Revision 5) documents chemistry factor (CF) calculations based on the application of surveillance data from capsules "U", "Y", "V" and "X" using the methods in RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Position 2.1. Section 4.0 of the Callaway PTLR states that all of the measured ΔRTNDT data for the surveillance plate and weld materials are deemed credible based on the RG 1.99, Revision 2 surveillance data credibility assessment.

a) Please confirm whether surveillance capsule "X" was the last capsule pulled from the Callaway reactor vessel. If not, the CF calculation must be revised to include the additional surveillance data.

b) Table 5.0-2 of the Callaway PTLR lists the RG 1.99, Revision 2, Position 2.1 CF for the surveillance plate as 25 ºF and the adjusted CF for the surveillance weld as 39.9 ºF. However, LRA Table 4.2-4 of the Callaway LRA lists a Position 2.1 CF of 25.6 ºF for the surveillance plate and 40.8 ºF for the surveillance weld. Please explain this discrepancy.

c) Please identify the calendar year when the current 28 EFPY P-T limit curves are projected to expire.

LRA Section 4.2.4 states that the P-T limit curves, as established in the PTLR, will be maintained and updated as necessary by TS 3.4.3 and TS 5.6.6, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

Part 50, Appendix G, Paragraph IV.A of 10 CFR states that,

[T]he pressure-retaining components of the reactor coolant pressure boundary [RCPB] that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth in [paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]

Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel beltline (neutron fluence ≥ 1 x 1017 n/cm2, E > 1 MeV), as well as ferritic materials not in the reactor vessel beltline (neutron fluence < 1 x 1017 n/cm2, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all reactor coolant pressure boundary (RCPB) components must meet the ASME Code, Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all RCPB components specified in Section III, NB-2332(b).

The current Callaway PTLR and the PTLR methodology described in Topical Report WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and Reactor Coolant System Heatup and Cooldown Limit Curves," Revision 4, dated May 2004, address P-T limit curve calculations for only the reactor vessel beltline shell region.

P-T limit calculations for ferritic RCPB components that are not reactor vessel beltline shell materials may define P-T curves that are more limiting than those calculated for the reactor vessel beltline shell materials. This may be due to the following factors:

i. Reactor vessel nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the reactor

vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the RTNDT for these components is not as high as that of reactor vessel beltline shell materials that have simpler geometries.

ii. Ferritic RCPB components that are not part of the reactor vessel may have initial RTNDT values, which may define a more restrictive lowest operating temperature in the P -T limits than those for the reactor vessel beltline shell materials.

d) Describe how the P-T limit curves to be developed for use in the period of extended operation, and the methodology used to develop these curves, will consider all reactor vessel materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

Callaway Response

a) Capsule X was the last capsule pulled and tested.

b) The PTLR CFs were calculated using the results of the Capsule X analysis, WCAP-15400-NP.

The LRA CFs were calculated using the results of the WCAP-17168-NP. WCAP-17168-NP was performed to support license renewal and developed a new transport model which incorporates operational information for the period after Capsule X was pulled. LRA Section 4.2.1 has been revised, as shown in Amendment 6 to clarify the source of the fluence information provided.

The source of the discrepancy between the CFs is the fluence calculations that support each document. Fluence is a factor in the RG 1.99, Rev. 2, Position 2.1 calculation of CF and the CFs in the PTLR and the LRA use 2 different fluence models. The small differences in results (25ºF vs. 25.6ºF; and 39.9ºF vs 40.8ºF) can be attributed to the modeling difference. Both models meet the requirements of RG 1.190 with uncertainty falling within the 20% limit specified in Regulatory Guide 1.190 for fluences used to determine RT(PTS) and RT(NDT). The table below compares the two fluence models.

Capsule	Best Estimate Fluence (E > 1.0 MeV) ($x10^{19}$ n/cm^2)	PTLR Fluence Model (WCAP-15400-NP)		LRA Fluence Model (WCAP-17168-NP)		% Diff. (PTLR to WCAP)
		Fluence (E > 1.0 MeV) ($x10^{19}$ n/cm^2)	BE/Calc	Fluence (E > 1.0 MeV) ($x10^{19}$ n/cm^2)	BE/Calc	
U	0.328	0.331	0.9%	0.298	-9.2%	10.0%
Y	1.20	1.27	5.8%	1.114	-7.2%	12.3%
V	2.29	2.52	10.0%	2.25	-1.8%	10.7%
X	2.92	3.33	14.0%	2.99	2.4%	10.2%
Average	-	-	7.7%	-	-4.0%	-
STD	-	-	5.62%	-	5.3%	-

Both fluence models are discrete ordinates transport calculations using the BUGLE 96 cross-section library which is derived from ENDF/B VI. These methodologies have been approved by the US NRC and are described in detail in:

- WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” J. D. Andrachek et al., May 2004.
- WCAP-16083-NP-A, Revision 0, “Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,” S. L. Anderson, May 2006.

c) As of October 2011, Callaway has experienced 23 EFPY of operation. Assuming a capacity factor of 90%, Callaway will reach 28 EFPY in April 2017.

d) When the current P-T limit curves expire at 28 EFPY, new curves will be developed and submitted to the NRC in accordance with the requirements of 10 CFR 50 Appendix G. The P-T limit curves will consider the effects of neutron embrittlement on the adjusted reference temperature for beltline locations and the higher stresses in the inlet/outlet nozzle corner region. The P-T limit curves also will consider the effects of the ferritic RCPB components outside the beltline and extended-beltline locations when determining the lowest service temperature.

   LRA Section 4.2.4 and Appendix A3.1.4 "Pressure–Temperature (P-T) Limits” have been revised, as shown in Amendment 6, to describe how the P-T limit curves will be revised to be consistent with the requirements of 10 CFR Part 50, Appendix G during the period of extended operation.

Corresponding Amendment Changes

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

Amendment 6, LRA Changes from RAI Responses

Enclosure 2 Summary Table

Affected LRA Section	LRA Page
Section 2.1.2.2	2.1-6
Table 2.4-1	2.4-6
Section 3.5.2.1.12	3.5-17and 3.5-18
Section 3.5.2.2.1.3.1	3.5-19 and 3.5-20
Table 3.5-1	3.5-35 and 3.5-46
Table 3.5.2-1	3.5-55 and 3.5-61
Table 3.5.2-2	3.5-71
Table 3.5.2-3	3.5-81
Table 3.5.2-4	3.5-90
Table 3.5.2-5	3.5-97
Table 3.5.2-7	3.5-111
Table 3.5.2-9	3.5-127
Table 3.5.2-10	3.5-133
Table 3.5.2-12	3.5-154, 3.5-155, and 3.5-157
Section 4.2.1	4.2-2
Section 4.2.4	4.2-11
Section A1.30	A-15
Section A1.31	A-16
Section A1.32	A-16
Section A1.36	A-18
Section A3.1.4	A-24
Table A4-1, item 22	A-41
Table A4-1, item 23	A-43
Table A4-1, item 27	A-45
Table A4-1, item 40	A-49
Section B2.1.27	B-94 to B-96
Section B2.1.28	B-97 to B-99
Section B2.1.30	B-102 and B-103
Section B2.1.31	B-104 to B-107
Section B2.1.32	B-108 and B-109
Section B2.1.36	B-118 to B-120

Callaway Plant
License Renewal Application
Amendment 6

Revision to Section 2.1.2.2 to delete the statement that equivalent anchors are not defined in the Callaway CLB.

Section 2.1.2.2 (page 2.1-6) is revised as follows (deleted text is shown strikethrough)

2.1.2.2 10 CFR 54.4(a)(2) – Nonsafety-Related Affecting Safety-Related

10 CFR 54.4(a)(2) requires that plant SSCs within the scope of license renewal include all nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the safety-related functions identified for safety-related SSCs. The guidance provided in NEI 95-10, Appendix F was used to develop the methodology for scoping to the criterion of 10 CFR 54.4(a)(2).

The methodology includes identification of nonsafety-related SSCs that are connected to safety-related SSCs and nonsafety-related SSCs that could spatially interact with safety-related SSCs. Determination and identification of any other SSCs satisfying criterion 10 CFR 54.4(a)(2) was completed as described below based on review of applicable CLB documents, plant specific and industry operating experience, and by system and structure functional evaluations.

Nonsafety-Related SSCs Performing Safety-Related 10 CFR 54.4(a)(1) Functions

The FSAR and other current licensing basis documents were reviewed for nonsafety-related plant systems or structures, to determine whether nonsafety-related systems or structures were credited with performing a safety-related function. Callaway does not have nonsafety-related systems or structures credited in CLB documents that perform a safety-related function.

Nonsafety-Related SSCs Directly Connected to Safety-Related SSCs

Nonsafety-related SSCs that are directly connected to safety-related SSCs were included within the scope of license renewal to ensure structural integrity of the safety-related SSC up to the first seismic anchor or equivalent anchor past the safety/nonsafety interface.

Seismic anchors and equivalent anchors were identified following the guidance of NEI 95-10, Appendix F as discussed below:

- A seismic anchor that ensures that forces and moments are restrained in three orthogonal directions.
- An equivalent anchor that is defined in the CLB. ~~(Equivalent anchors are not defined in the Callaway CLB; therefore, this criterion was not used.)~~

Callaway Plant
License Renewal Application
Amendment 6

Revision to Table 2.4-1 to delete Caulking and Sealant with Flood Barrier, Shelter, Protection Functions.

Table 2.4-1 (page 2.4-6) is revised as follows (deleted text shown in strikethrough):

Table 2.4-1 Reactor Building

Component Type	Intended Function
~~Caulking and Sealant~~	~~Flood Barrier~~ ~~Shelter, Protection~~

Callaway Plant
License Renewal Application
Amendment 6

Revision to Section 3.5.2.1.12 to remove submerged (structural) as an environment and RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants for Aging Management Programs.

Revision to Section 3.5.2.1.12 to remove cracking as an aging effect requiring aging management.

Section 3.5.2.1.12, Supports (page 3.5-17 and 3.5-18) is revised as follows (deleted text shown in strikethrough):

3.5.2.1.12 Supports

Environment

The supports component types are exposed to the following environments:

- Atmosphere/ Weather (Structural)
- Borated Water Leakage
- Plant Indoor Air (Structural)
- ~~Submerged (Structural)~~

Aging Effects Requiring Management

The following supports aging effects require management:

- ~~Cracking~~
- Loss of material
- Loss of mechanical function
- Loss of mechanical function and fatigue
- Loss of preload
- Reduction in concrete anchor capacity

Aging Management Programs

The following aging management programs manage the aging effects for the supports component types:

- ASME Section XI, Subsection IWF (B2.1.28)
- Boric Acid Corrosion (B2.1.4)

- ~~RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B2.1.32)~~
- Structures Monitoring (B2.1.31)

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Revision to Section 3.5.2.2.1.3.1 to delete reference to moisture barrier.

Section 3.5.2.2.1.3.1 (pages 3.5-19 and 3.5-20) is revised as follows (deleted text shown in strikethrough):

3.5.2.2.1.3.1 Loss of material due to general, pitting, and crevice corrosion in steel elements of inaccessible areas

General, Pitting and Crevice Corrosion:

The ASME Section XI, Subsection IWE program (B2.1.26) manages aging of the steel liner of the concrete containment building. The 10 CFR Part 50, Appendix J program (B2.1.29) manages loss of leak tightness, loss of sealing, and leakage through containment to assure that allowable leakage rate limits specified in the Technical Specifications are not exceeded. As required by 10 CFR 50.55a (b)(2)(ix)(A), an evaluation of the acceptability of the inaccessible areas is completed whenever conditions are detected in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

Reinforced concrete structures at Callaway were designed, constructed, and inspected in accordance with ACI and ASTM standards, which provide for a good quality, dense, well-cured, and low permeability concrete. The mix proportions were established in accordance with ACI 301. The mix designs contain an air-entraining admixture capable of entraining three to six percent air in accordance with ASTM C260, and maximum water content was controlled by placing the concrete at specified slumps. (See FSAR Section 3.8.1.6.1 SP.) Procedural controls ensured quality throughout the batching, mixing, and placement processes. ~~The ASME Section XI, Subsection IWE program (B2.1.26) manages aging of the moisture barrier at the interface between the liner and the concrete.~~ The ASME Section XI, Subsection IWL program (B2.1.27) identifies and manages any cracks in the containment concrete that could potentially provide a pathway for water to reach inaccessible portions of the steel containment liner. Crack control was achieved through proper sizing, spacing, and distribution of reinforcing steel in accordance with ACI 318-71. (See FSAR Section 3.8.1.2.2 SP.) Procedural controls ensure that borated water spills are not common, and when detected are cleaned up in a timely manner. Therefore, a plant-specific aging management program to manage the effects of general, pitting and crevice corrosion is not required.

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Table 3.5-1 line 026 is revised for Loss of sealing of Moisture barriers to indicate line 026 is not applicable.

Table 3.5-1 (page 3.5-35) is revised as follows (deleted text shown in strikethrough and new text is shown underlined):

Table 3.5-1 Summary of Aging Management Programs in Chapters II and III of NUREG-1801 for Containments, Structures, and Component Supports (Continued)

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.026	Moisture barriers (caulking, flashing, and other sealants)	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	ASME Section XI, Subsection IWE (B2.1.26)	No	~~Consistent with NUREG-1801.~~ Not applicable. The Callaway design of the interface between the containment steel liner and the internal concrete structures does not include a moisture barrier, so the applicable NUREG-1801 line was not used.

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Table 3.5-1 lines 068 and 069 are revised for cracking of high-strength structural bolting to indicate lines 068 and 069 are not applicable.

Table 3.5-1 (page 3.5-46) is revised as follows (deleted text shown in strikethrough and new text is shown underlined):

Table 3.5-1 Summary of Aging Management Programs in Chapters II and III of NUREG-1801 for Containments, Structures, and Component Supports (Continued)

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.068	High-strength structural bolting	Cracking due to stress corrosion cracking	ASME Section XI, Subsection IWF (B2.1.28)	No	~~Consistent with NUREG-1801.~~ Not applicable. Callaway has no in-scope High-strength structural bolting susceptible to Cracking due to stress corrosion cracking so the applicable NUREG-1801 lines were not used.
3.5.1.069	High-strength structural bolting	Cracking due to stress corrosion cracking	Structures Monitoring (B2.1.31). Note: ASTM A 325, F 1852 and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No	~~Consistent with NUREG-1801.~~ Not applicable. Callaway has no in-scope High-strength structural bolting susceptible to Cracking due to stress corrosion cracking so the applicable NUREG-1801 lines were not used.

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Revision to Table 3.5.2-1 to delete Caulking and Sealant AMR line with an aging effect of Loss of sealing.

Table 3.5.2-1 (page 3.5-55) is revised as follows (deleted text shown in strikethrough):

Table 3.5.2-1 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation - Reactor Building (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~Caulking and Sealant~~	~~FLB, SH~~	~~Elastomer~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Loss of sealing~~	~~ASME Section XI, Subsection IWE (B2.1.26)~~	~~II.A3.CP-40~~	~~3.5.1.026~~	~~A~~

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Revision to Table 3.5.2-1 through Table 3.5.2-5, Table 3.5.2-7, Table 3.5.2-9, Table 3.5.2-10, and Table 3.5.2-12 to delete high strength structural bolting AMR lines with an aging effect of cracking

Table 3.5.2-1 through Table 3.5.2-5, Table 3.5.2-7, Table 3.5.2-9, Table 3.5.2-10, and Table 3.5.2-12 (pages 3.5-61, 3.5-71, 3.5-81, 3.5-90, 3.5-97, 3.5-111, 3.5-127, 3.5-133, 3.5-154, and 3.5-155) are revised as follows (deleted text shown in strikethrough):

Table 3.5.2-1 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation - Reactor Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A1.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-2 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation - Control Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A1.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-3 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Auxiliary Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A3.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-4 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Turbine Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A3.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-5 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Diesel Generator Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A3.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-7 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – In-Scope Tank Foundations and Structures

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A3.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-9 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Radwaste Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A3.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-10 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Fuel Building

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.A5.TP-300~~	~~3.5.1.069~~	~~A~~

Table 3.5.2-12 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Supports

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~ASME Section XI, Subsection IWF (B2.1.28)~~	~~III.B1.1.TP-41~~	~~3.5.1.068~~	~~A~~
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.B2.TP-300~~	~~3.5.1.069~~	~~A~~
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.B3.TP-300~~	~~3.5.1.069~~	~~A~~
~~High Strength Bolting~~	~~SS~~	~~High Strength Low Alloy Steel (Bolting)~~	~~Plant Indoor Air (Structural) (Ext)~~	~~Cracking~~	~~Structures Monitoring (B2.1.31)~~	~~III.B4.TP-300~~	~~3.5.1.069~~	~~A~~

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Revision to Table 3.5.2-12 to remove submerged ASME 2 and 3 carbon steel supports.

Table 3.5.2-12 Containments, Structures, and Component Supports (page 3.5-157) is revised as follows (deleted text shown in strikethrough):

Table 3.5.2-12 Containments, Structures, and Component Supports – Summary of Aging Management Evaluation – Supports

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~Supports ASME 2 and 3~~	~~SS~~	~~Carbon Steel~~	~~Submerged (Structural) (Ext)~~	~~Loss of material~~	~~RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B2.1.32)~~	~~III.A6.TP-221~~	~~3.5.1.083~~	~~C~~

4.2.1 Neutron Fluence Values

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. Neutron fluence projections are made in order to estimate the effect on these reactor vessel material properties (Section 4.2.2, *Charpy Upper-Shelf Energy*, and Section 4.2.3, *Pressurized Thermal Shock*).

Increased plant capacity factors prompted the increase in the lifetime capacity factor assumed for fluence estimates to 90 percent, and hence increased the assumed EOL effective full power years (EFPY) for the period of extended operation to 54 EFPY. This is consistent with the plant's average capacity factor between 2004 and 2009 of 88 percent, and the exposure of approximately 23 EFPY as of October 2011.

The fluence values for EOLE were projected based on the results of the Capsule X analysis, WCAP-15400-NP (Reference 1). The fluence values were revised for license renewal to incorporate operation since the Capsule X was pulled, WCAP 17168 NP (Reference 3). The revised fluences were determined with transport calculations using the DORT discrete ordinates code and the BUGLE 96 cross-section library which is derived from ENDF/B-VI. The neutron transport and dosimetry evaluation methodologies follow the guidance and meet the requirements of Regulatory Guide 1.190 and are consistent with Westinghouse WCAP-14040-NP-A.

The vessel peak neutron fluences are listed in Table 4.2-1, *Calculated EOL Neutron Fluence Projections at the Peak Location on the Reactor Vessel Clad/Base Metal Interface for Callaway Beltline Materials* and Table 4.2-2, *Calculated EOL Neutron Fluence Projections at the Peak Location on the Reactor Vessel Clad/Base Metal Interface for Callaway Extended Beltline Materials* for the beltline and extended beltline materials, respectively.

The EOLE fluence projections were revised to quantify expected fluence at the end of the period of extended operation. Therefore this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

Disposition: Projection, 10 CFR 54.21(c)(1)(ii)

4.2.4 Pressure-Temperature (P-T) Limits

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the material properties (adjusted reference temperature, ART), effects of fluence on material properties obtained from the reactor surveillance capsules, and methodology of Appendix G of ASME Boiler and Pressure Vessel Code, Section XI.

These methods depend on the limiting ART of the beltline material and cause the calculation of the P-T limit curves to be a TLAA. Withdrawal and testing of the surveillance coupons verifies that the limiting ART value used in the P-T limit curves bounds the aging of the reactor vessel material. ART values are listed in the PTLR (Reference **Error! Reference source not found.**).

The current P-T limit curves and the assumed ART values are valid up to 28 EFPY based on a clad/base metal interface fluence of 1.625 x 10^{19} n/cm^2 from the Capsule X surveillance results, WCAP-15400-NP (Reference **Error! Reference source not found.**). The current curves assume a ¼T ART of 128°F and a ¾T ART of 112°F. These reference temperatures are based on the aging of lower shell R2708-1. These latest P-T limit curves were generated using the methodologies from ASME Section XI Appendix G and WCAP-14040-NP-A.
Ameren Missouri will revise the P-T limits curves before reaching 28 EFPY. These curves are required to be maintained and updated as necessary by Technical Specifications 3.4.3 and 5.6.6. . The revised curves will consider:

- effects of neutron embrittlement on the adjusted reference temperature for reactor vessel locations expected to receive a fluence of greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV).
- the higher stresses in the inlet/outlet nozzle corner region.
- the ferritic reactor coolant pressure boundary components which receive a fluence of less than 1 x 10^{17} n/cm^2 (E>1.0 MeV) when determining the lowest service temperature.

Therefore the P-T limit curves will be managed, as required by its current license, through the period of extended operation. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

A1.30 MASONRY WALLS

The Masonry Walls program manages cracking of masonry walls. The Masonry Walls program, administered as part of the Structures Monitoring program (A1.31), is based on guidance provided in NRC Bulletin 80-11, *Masonry Wall Design* and NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to NRC IE Bulletin 80-11*. The Masonry Wall program contains inspection guidelines and lists attributes that cause aging of masonry walls, which are to be monitored during structural monitoring inspections, as well as establishes examination criteria, evaluation requirements, and acceptance criteria. The inspections of all structural components, including masonry walls and water-control structures, are performed at intervals no more than 5 years.

A1.31 STRUCTURES MONITORING

The Structures Monitoring program manages the following aging effects of structures and structural supports within the scope of license renewal:

- Concrete cracking and spalling
- Cracking
- Cracking and distortion
- Cracking, blistering, change in color
- Cracking, loss of material
- Cracking, loss of bond, and loss of material (spalling, scaling)
- Increase in porosity and permeability, cracking, loss of material (spalling, scaling)
- Increase in porosity and permeability, loss of strength
- Loss of material
- Loss of material (spalling, scaling) and cracking
- Loss of mechanical function
- Loss of preload
- Loss of sealing
- Reduction in concrete anchor capacity

The Structures Monitoring program implements the requirements of 10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, consistent with guidance of NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 2 and NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 2.

The Structures Monitoring program provides inspection guidelines for concrete elements, structural steel, roof systems, masonry walls and metal siding, including all masonry walls and water control structures within the scope of license renewal. The Structures Monitoring program also monitors settlement for each major structure and inspects non ASME mechanical and electrical supports. The inspections of all structural components, including masonry walls and water-control structures, are performed at intervals no more than 5 years.

A1.32 RG 1.127, INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program, which is implemented as part of the Structures Monitoring program (SMP), manages the following aging effects:

- Cracking; loss of bond; and loss of material (spalling, scaling)
- Increase in porosity and permeability; loss of strength
- Loss of material
- Loss of material (spalling, scaling) and cracking
- Loss of material; loss of form

The scope of this program also includes structural steel and structural bolting associated with water-control structures. SNUPPS-Callaway positions are compliant with that of the Regulatory Guide 1.127 with respect to the ultimate heat sink (UHS) retention pond. The Structures Monitoring program (A1.31) includes all water-control structures within the scope of Regulatory Guide 1.127. The UHS retention pond, the essential service water pumphouse, the ESW supply lines yard vault, the UHS cooling tower and the submerged discharge structures are the water-control structures within the scope for license renewal that are monitored by this program. The UHS retention pond and its associated structures receive periodic in-service inspections for assessment of their structural safety and operational adequacy every five years. Callaway performs algae treatment and riprap inspections along the UHS retention pond to ensure smooth operation of the essential service water pumps. Callaway maintains benchmarks for monitoring settlement in any of the Category 1 structures including the UHS cooling tower. The inspections of all structural components, including masonry walls and water-control structures, are performed at intervals no more than 5 years.

A1.36 INACCESSIBLE POWER CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages reduced insulation resistance leading to electrical failure of in-scope non-EQ inaccessible power cables (greater than or equal to 400 volts) exposed to wetting or submergence caused by significant moisture. Significant moisture is defined as periodic exposures to moisture that last more than a few days.

Manholes, pits, and duct banks that contain in-scope non-EQ inaccessible power cables will be inspected to confirm that cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. Collected water will be removed as required. This inspection and water removal will be performed based on actual plant experience with inspection frequency being at least annually and after event driven occurrences (such as heavy rain or flooding). Dewatering devices will be inspected and operation verified prior to any known or predicted heavy rain or flooding events. The first inspection for license renewal is to be completed prior to the period of extended operation.

In-scope non-EQ inaccessible power cables routed through manholes, pits, and duct banks are tested to provide an indication of the conductor insulation condition. Testing that is appropriate to the application at the time of the testing will be performed to detect deterioration of the insulation system due to wetting. Cable testing may be a mix of proven testing methods (such as dielectric loss [dissipation factor/power factor], AC voltage withstand, partial discharge, step voltage, time domain reflectometry, insulation resistance and polarization index, or line resonance analysis) that are state-of-the-art at the time of testing. The first test for license renewal will be completed prior to the period of extended operation ~~and~~with subsequent tests performed at least every six years thereafter and adjusted based on test results and operating experience.

Acceptance criteria for cable testing will be defined prior to each test. Test results will be compared to previous test results to evaluate for additional information on the rate of cable degradation. An engineering evaluation is required when the test or inspection acceptance criteria are not met.

A3.1.4 PRESSURE-TEMPERATURE (P-T) LIMITS

Appendix G of 10 CFR 50 requires that reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences be accomplished within established pressure-temperature (P-T) limits. These curves are required to be maintained and updated as necessary by Technical Specifications 3.4.3 and 5.6.6.

The current P-T limit curves and the adjusted reference temperature (ART) values are valid up to 28 EFPY. The revision necessary to extend the P-T curves beyond 28 EFPY will consider the following in accordance with the requirements of 10 CFR 50 Appendix G:

- effects of neutron embrittlement on the adjusted reference temperature for locations expected to receive a fluence of greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV).
- the higher stresses in the nozzle corner region of inlet/outlet nozzles.
- the ferritic reactor coolant pressure boundary components which receive a fluence of less than 1 x 10^{17} n/cm^2 (E>1.0 MeV) when determining the lowest service temperature.

Therefore the P-T limit curves will be managed, as required by its current license, through the period of extended operation. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
22	Enhance the ASME Section XI, Subsection IWF program procedures to • specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the applicable EPRI guidelines, ASTM standards, AISC specifications, and NUREG-recommendations to prevent or mitigate degradation and failure of safety-related bolting due to stress corrosion cracking. Specifically, if ASTM A325, ASTM F1852, and/or ASTM A490 bolts are used, the preventive actions as discussed in Section 2 of the Research Council for Structural Connections "Specification for Structural Joints Using ASTM A325 or A490 Bolts" will be followed, ~~and~~ • ~~specify that, in addition to VT-3 examination, high strength bolting (actual measured yield strength greater than or equal to 150 ksi) in sizes greater than 1 in. nominal diameter, shall receive a volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 to detect cracking.~~	B2.1.28	Prior to the period of extended operation

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
23	Enhance the Structures Monitoring program procedures to: • include the main access facility into the scope of Structures Monitoring program. • specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP 5769, EPRI NP 5067, EPRI TR 104213, and the additional recommendations of NUREG-1339. • specify the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of Research Council for Structural Connections publication Specification for Structural Joints Using ASTM A325 or A490 Bolts for ASTM A325, ASTM F1852, and/or ASTM A490 structural bolts. • specify inspections of penetrations, transmission towers, electrical conduits, raceways, cable trays, electrical cabinets/enclosures, and associated anchorages, and complete a baseline inspection of these components.* • specify that groundwater is monitored for pH, chlorides and sulfates, and every five years at least two samples are tested and the results are evaluated by engineering to assess the impact, if any, on below grade structures. • ~~specify that structural bolts greater than one in. in diameter with actual measured yield strength greater than or equal to 150 ksi are evaluated for susceptibility to stress corrosion cracking, and, if necessary, visual inspections are supplemented with volumetric or surface examinations.~~ • specify inspector qualifications in accordance with ACI349.3R-96. • quantify acceptance criteria and critical parameters for monitoring degradation, and to provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action. • incorporate applicable industry codes, standards and guidelines for acceptance criteria. • specify that degradation associated with seismic isolation gaps, obstructions of these gaps, or questionable material in these gaps, will be evaluated by an engineer familiar with the seismic design of the plant, and the evaluation will consider the seismic isolation function in determining what corrective actions may be required. #	B2.1.31	Prior to the period of extended operation with the exception of item indicated by *, which will be completed by December 31, 2017, and item indicated by #, for which initial inspections will be completed by December 31, 2012, and any corrective actions resulting from these inspections will be completed no later than December 31, 2017.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
27	Enhance the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program procedures to: • identify the power cables (greater than or equal to 400 volts), manholes, pits and duct banks that are within the scope of the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program. • include periodic inspection of manholes, pits and duct banks, to confirm cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e, sump pumps) and associated alarms operate properly. • Identify that inspections will be performed at least annually based on water accumulation over time and after event driven occurrences (e.g., heavy rain or flooding). In addition, operation of dewatering devices will be inspected and operation verified prior to any known or predicted heavy rain or flooding events. • ensure in-scope power cables are tested at least once every six years and adjusted based on test results and operating experience. • compare test results to previous test results to evaluate for additional information on the rate of cable degradation. • confirm cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. Acceptance criteria for cable testing will be defined prior to each test. • require an engineering evaluation when the test or inspection acceptance criteria are not met.	B2.1.36	Prior to the period of extended operation

Item #	Commitment	LRA Section	Implementation Schedule
40	Enhance the ASME Section XI, Subsection IWL program to specify that acceptability of concrete surfaces is based on the evaluation criteria provided in ACI-349.3R.	B2.1.27	Prior to the period of extended operation

B2.1.27 ASME Section XI, Subsection IWL

Program Description

The ASME Section XI, Subsection IWL program manages the following aging effects of the concrete containment building and post-tensioned system:

- Cracking
- Cracking, loss of bond, and loss of material (spalling, scaling)
- Increase in porosity and permeability, cracking, loss of material (spalling, scaling)
- Increase in porosity and permeability, loss of strength
- Loss of material
- Loss of material (spalling, scaling) and cracking

For the current inspection interval, Callaway performs IWL Containment Inservice Inspections (CISIs) in accordance with the 2001 Edition of ASME Section XI, Subsection IWL (including 2002 and 2003 addenda), supplemented with the applicable requirements of 10 CFR 50.55a(b)(2). This program is consistent with provisions in 10 CFR 50.55a that specify use of the ASME Code edition in effect 12 months prior to the start of the inspection interval. Callaway will use the ASME Code edition consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

The ASME Section XI, Subsection IWL inspections are performed in order to identify and manage containment concrete aging effects that could result in loss of intended function. Included in this inspection program are the concrete containment structure (includes all accessible areas of the concrete dome, cylinder walls, and buttresses) and the post-tensioning system (includes tendons, end anchorages, and concrete surfaces around the end anchorages). A summary of the containment concrete components at Callaway, the examinations required, and a detailed schedule of examinations for items subject to IWL inspections are provided in plant procedures. The primary inspection method is a visual examination, supplemented by testing. Tendon wires are tested for yield strength, ultimate tensile strength, and elongation. Tendon corrosion protection medium is analyzed for alkalinity, water content, and soluble ion concentrations. Any free water contained in the anchorage end cap and free water which drains from tendons during the examination is documented. Samples of the free water are also analyzed for pH. Prestressing forces are measured in selected sample tendons. Evaluation of prestressing forces is addressed in Concrete Containment Tendon Prestress program (B3.3). Acceptance criteria, corrective

actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found, are in accordance with ASME Section XI, Subsection IWL

Post-tensioning system repair/replacement activities and the augmented examination requirements following post-tensioning system repair/replacement activities are in accordance with ASME Section XI, Subsection IWL.

In conformance with 10 CFR 50.55a(g)(4)(ii), the Callaway CISI program will be updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified 12 months before the start of the inspection interval.

NUREG-1801 Consistency

The ASME Section XI, Subsection IWL program is an existing program that is consistent with NUREG-1801, Section XI.S2, *ASME Section XI, Subsection IWL*.

Exceptions to NUREG-1801

None

Enhancements
~~None~~Prior to the period of extended operation, the following enhancement will be implemented in the following program element:

Acceptance Criteria (Element 6)

Plant procedures will be enhanced to specify that acceptability of concrete surfaces is based on the evaluation criteria provided in ACI-349.3R.

Operating Experience

The following discussion of operating experience provides objective evidence that the ASME Section XI, Subsection IWL program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation:

1. The 15th year tendon surveillance began in May 1999 and was completed in June 1999. Based on the data gathered during the 1999 physical surveillance and visual inspection, the conclusion was reached that no abnormal degradation of the post tensioning system had occurred at the Callaway containment building.

2. The 20th year tendon surveillance began in July 2004 and was completed in September 2004. All tendons were resealed and regreased. One tendon accepted less grease than was removed, and one tendon accepted more than 10 percent of the tendon duct volume. Nonconformance reports were written to record these

   findings and these conditions were found to be acceptable by engineering evaluation, as allowed by ASME Section XI, Subsection IWL-3310. Based on these evaluations and the other data gathered during the 2004 physical surveillance and visual inspection, the conclusion was reached that no abnormal degradation of the post tensioning system had occurred at the Callaway containment building.

3. The 25th year tendon surveillance began in March 2010 and was completed in April 2010. Sample wires were removed from one tendon in each group for physical testing. The test results on one of the wire samples indicated elongation values under the minimum prescribed in Callaway specifications. A nonconformance report was written to record this finding, and this condition was found to be acceptable by engineering evaluation, as allowed by ASME Section XI, Subsection IWL-3310. Based on this evaluation and the other data gathered during the 2010 physical surveillance and visual inspection, the conclusion was reached that no abnormal degradation of the post tensioning system had occurred at the Callaway containment building.

The above examples provide objective evidence that the ASME Section XI, Subsection IWL program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the ASME Section XI, Subsection IWL program will be evaluated to ensure there is no significant impact to 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 ASME Section XI, Subsection IWL program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the ASME Section XI, Subsection IWL 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.

B2.1.28 ASME Section XI, Subsection IWF

Program Description

The ASME Section XI, Subsection IWF program manages loss of material, cracking, fatigue, loss of preload, and loss of mechanical function for supports of Classes 1, 2, and 3 piping and components. There are no Class MC supports at Callaway. The program conforms to Inspection Program B of ASME Section XI.

During the third inservice inspection interval (December 2004 to December 2014), Callaway is performing inspections of supports for Class 1, 2, and 3 piping and components in accordance with 1998 Edition with 2000 addenda of ASME Section XI. In conformance with 10 CFR 50.55a(g)(4)(ii), the Callaway ISI program is updated during each successive 120 month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified 12 months before the start of the inspection interval. Callaway will use the ASME Code edition consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

Supports for Class 1, 2, and 3 piping and components are selected for examination per the requirements of ASME Section XI, Subsection IWF.

Acceptance standards are specified in Article IWF 3400. Scope of the inspection for supports is based on class and total population as defined in Table IWF 2500-1. When a component support requires corrective measures in accordance with the provisions of IWF 3000, that support is reexamined during the next inspection period. When the reexaminations do not require additional corrective measures during the next inspection period, the inspection schedule reverts to the requirements of the original inspection program. Component support examinations that detect flaws or relevant conditions exceeding the acceptance criteria of IWF 3400 are extended to include additional examinations in accordance with IWF 2430. Callaway also implements, as needed, Code Case N-586 which prescribes alternative additional examination requirements of Class 1, 2, and 3 Piping, Components and Supports.

The ASME Section XI, Subsection IWF program provides a systematic method for periodic examination of supports for Class 1, 2, and 3 piping and components. The primary inspection method is visual examination. The instructions and acceptance criteria for the visual examinations are included in Callaway plant procedures.

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NUREG-1801 Consistency

The ASME Section XI, Subsection IWF program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S3, *ASME Section XI Subsection IWF*

Exceptions to NUREG-1801

None

Enhancements

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

Preventive Actions (Element 2)

Procedures will be enhanced to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the applicable EPRI guidelines, ASTM standards, AISC specifications, and NUREG recommendations to prevent or mitigate degradation and failure of safety-related bolting due to stress corrosion cracking. Specifically, if ASTM A325, ASTM F1852, and/or ASTM A490 bolts are used, the preventive actions as discussed in Section 2 of the Research Council for Structural Connections "Specification for Structural Joints Using ASTM A325 or A490 Bolts" will be followed.

~~*Parameters Monitored or Inspected (Element 3) and Detection of Aging Effects (Element 4)*~~

~~Procedures will be enhanced to specify that, in addition to VT-3 examination, high strength bolting (actual measured yield strength greater than or equal to 150 ksi) in sizes greater than 1 in. nominal diameter, shall receive a volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 to detect cracking.~~

Operating Experience

The following discussion of operating experience provides objective evidence that the ASME Section XI, Subsection IWF program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation:

1. Review of the Owner Activity Reports since 2000 indicates there have been no conditions found through IWF inspections which required repair, replacement, or engineering evaluation during this period. Callaway inspected 100% of the supports

required per examination category F-A for the past inspection interval and found no signs of aging.

The above review provides objective evidence that the ASME Section XI, Subsection IWF program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the ASME Section XI, Subsection IWF program will be evaluated to ensure there is no significant impact to 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 the continued implementation of the ASME Section XI, Subsection IWF program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the ASME Section XI, Subsection IWF 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.

B2.1.30 Masonry Walls

Program Description

The Masonry Walls program manages cracking of masonry walls. The Masonry Walls program, is integrated and administered as part of the Structures Monitoring program (B2.1.31) that implements structures monitoring requirements as specified by 10 CFR 50.65 (Maintenance Rule). In Seismic Category I structures, masonry walls are within scope of license renewal based on guidance provided in NRC Bulletin 80-11, *Masonry Wall Design* and NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to NRC IE Bulletin 80-11*. Some masonry walls in Non-Category I structures are within the scope of license renewal based on FSAR commitments to satisfy fire protection requirements. The guidance of NRC Bulletin 80-11 does not apply to these walls. Refer to the Fire Protection program (B2.1.13) for aging management of the masonry wall fire barriers intended function.

The Masonry Walls program contains inspection guidelines and lists attributes that cause aging of masonry walls, which are to be monitored during structural monitoring inspections, as well as establishes examination criteria, evaluation requirements, and acceptance criteria. The provisions of the program are consistent with the guidance provided in NRC Information Notice 87-67 for inspections and evaluation of masonry wall cracking in Category I structures not addressed in the evaluation basis in response to NRC Bulletin 80-11. The inspections of all structural components, include masonry walls and water-control structures, are performed at intervals of no more than 5 years.

Concrete masonry unit (CMU) walls in proximity to safety-related systems and equipment such that wall failure could adversely affect the safety-related systems or equipment are designed as reinforced CMU walls, and/or restrained with steel framing provided on both faces of the walls to prevent collapse of the units. Removable CMU walls, which are built with masonry or concrete units stacked without any grouting or reinforcing, are also restrained with steel framing on both faces of the wall.

NUREG-1801 Consistency

The Masonry Walls program is an existing program that is consistent with NUREG-1801, Section XI.S5, *Masonry Walls*.

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

The following discussion of operating experience provides objective evidence that the Masonry Walls program, which is included in the Structures Monitoring program, will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation:

1. At Callaway, aging effects identified during the structures monitoring inspections are documented in Callaway Action Requests and corrective actions are taken prior to any loss of intended functions. A review of the structures monitoring inspection reports from the last 10 years has indicated that in-scope concrete masonry unit walls are in good condition. Isolated instances of cracking masonry walls such as in the lube oil storage tank room of the turbine building have been noticed, but none severe enough to warrant corrective action at the time of the inspection. Such minor degradations are evaluated, recorded for trending purposes, and tagged for future inspections in the area.

The above example provides objective evidence that the Masonry Walls program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the Masonry Walls program will be evaluated to ensure there is no significant impact to 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 Masonry Walls program will effectively identify aging prior to loss of intended function. Guidance for re-evaluation, repair, or replacement is provided for locations where aging is found. There is confidence the continued implementation of the Masonry Walls program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Masonry Walls 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.

B2.1.31 Structures Monitoring

Program Description

The Structures Monitoring program (SMP) monitors the condition of structures and structural supports that are within the scope of license renewal to manage the following aging effects:

- Concrete cracking and spalling
- Cracking
- Cracking and distortion
- Cracking, blistering, change in color
- Cracking, loss of material
- Cracking, loss of bond, and loss of material (spalling, scaling)
- Increase in porosity and permeability, cracking, loss of material (spalling, scaling)
- Increase in porosity and permeability, loss of strength
- Loss of material
- Loss of material (spalling, scaling) and cracking
- Loss of mechanical function
- Loss of preload
- Loss of sealing
- Reduction in concrete anchor capacity

~~Structural bolts greater than one inch in diameter with actual measured yield strength greater than or equal to 150 ksi are evaluated for susceptibility to stress corrosion cracking, and, if necessary, visual inspections are supplemented with volumetric or surface examinations.~~ Plant procedures, following enhancements, will specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, EPRI TR-104213, *Bolted Joint Maintenance & Application Guide*, and the additional recommendations of NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*.

The SMP implements the requirements of 10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, consistent with guidance of NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 2 and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 2.

The SMP provides inspection guidelines and walk-down checklists for structural steel, roof systems, reinforced concrete, masonry walls and metal siding. Electrical duct banks and manholes, valve pits, access vaults, and structural supports are inspected as part of the SMP. Callaway is committed to NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants* and the scope of the SMP includes water-control structures. The scope of SMP also includes masonry walls Callaway has a settlement monitoring program that monitors settlement for each major structure utilizing geotechnical monitoring techniques. The inspections of all structural components, include masonry walls and water-control structures, are performed at intervals of no more than 5 years.

Groundwater is monitored for pH, chlorides and sulfates every five years, and the results are evaluated by engineering to assess the impact, if any, on below grade structures.

Callaway does not take credit for any coatings to manage the aging of structural components and coating degradation is used only as an indicator of the condition of underlying material.

NUREG-1801 Consistency

The Structures Monitoring program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S6, *Structures Monitoring*.

Exceptions to NUREG-1801

None

Enhancements

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 the main access facility into the scope of Structures Monitoring program.

Preventive Actions (Element 2)

Plant procedures will be enhanced to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants.

are in accordance with the guidelines of EPRI NP-5769, EPRI NP-5067, EPRI TR-104213, and the additional recommendations of NUREG-1339.

Plant procedures will be enhanced to specify the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of Research Council for Structural Connections publication Specification for Structural Joints Using ASTM A325 or A490 Bolts for ASTM A325, ASTM F1852, and/or ASTM A490 structural bolts.

Scope of the Program (Element 1) and Parameters Monitored or Inspected (Element 3)

Procedures will be enhanced to specify inspections of penetrations, transmission towers, electrical conduits, raceways, cable trays, electrical cabinets/enclosures, and associated anchorages, and to complete a baseline inspection of these components prior to December 31, 2017.

Procedures will be enhanced to specify that groundwater is monitored for pH, chlorides and sulfates, and every five years at least two samples are tested and the results are evaluated by engineering to assess the impact, if any, on below grade structures.

~~*Parameters Monitored or Inspected (Element 3)*~~

~~Procedures will be enhanced to specify that structural bolts greater than one inch in diameter with actual measured yield strength greater than or equal to 150 ksi are evaluated for susceptibility to stress corrosion cracking, and, if necessary, visual inspections are supplemented with volumetric or surface examinations.~~

Detection of Aging Effects (Element 4)

Procedures will be enhanced to specify inspector qualifications in accordance with ACI349.3R-96.

Acceptance Criteria (Element 6)

Procedures will be enhanced to quantify acceptance criteria and critical parameters for monitoring degradation, and to provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action.

Procedures will be enhanced to incorporate applicable industry codes, standards and guidelines for acceptance criteria.

Procedures will be enhanced to specify that degradation associated with seismic isolation gaps, obstructions of these gaps, or questionable material in these gaps, will be evaluated by an engineer familiar with the seismic design of the plant, and the evaluation will consider the seismic isolation function in determining what corrective actions may be required. Initial inspections will be completed by December 31, 2012, and any corrective actions resulting from these inspections will be completed no later than December 31, 2017.

Operating Experience

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

1. A review of the most recent structure inspection reports show minor instances of cracking in concrete, corrosion in structural steel, and elastomeric degradation in various building structures which have been evaluated per acceptance criteria and with corrective action taken as needed. The northeast corner of the 'A' emergency diesel generator fuel vault exterior exhibited some cracking in 2010, which is not severe enough to warrant corrective action at this time but is tracked for trending purposes. The most recent reactor building inspection report (2010) cites instances of corrosion of structural steel, supports, and cable trays due to condensation.

2. Fuel building structural inspection report (2002) identified an instance of cracking on the interior face of the exterior wall, with leachate observed coming through the crack. Engineering evaluation determined this leaking was not severe enough to warrant corrective action. Inspections performed in 2010 did not identify any further cracking or leaking of leachate in this area. Minor cracking on the exterior of fuel building plant south and west wall was identified and no water leakage, either active or inactive, was observed.

   Callaway performs continuous monitoring of the spent fuel pool liner leak chase channels. A standpipe with automatic drain controls is used to measure the fuel pool leak rate and periodic updates of the leak rate are provided by the plant computer. The observed leakage has been small and remained steady. The leakage rate is small at approximately 0.119 gal/day, and does not challenge makeup capability. The exterior spent fuel pool walls show no evidence of external leakage, thus indicating that the leakage is contained within the leak chase channels and that there is no effect upon the structural integrity of the spent fuel pool.

3. Groundwater has been sampled monthly since November, 2009. With exception of two monitoring wells, pH, chlorides and sulfate concentrations have been within the prescribed limits for non-aggressive ground water/soil. These two wells are located north of the turbine building and adjacent to plant roads. The wells' high chloride levels can be attributed to the use of winter road salts. These two well locations have shown seasonal increases in chloride levels of up to 680 mg/L while the pH and sulfate concentrations have remained non-aggressive. Callaway will continue to monitor the results from the groundwater samples and will perform an engineering evaluation to determine if any adverse aging effects have occurred in any inaccessible concrete structural elements.

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The above examples provide objective evidence that the Structures Monitoring program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the Structures Monitoring program will be evaluated to ensure there is no significant impact to 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 Structures Monitoring program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Structures Monitoring 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.

B2.1.32 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

Program Description

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program, which is implemented as part of the Structures Monitoring program (SMP), manages the following aging effects:

- Cracking; loss of bond; and loss of material (spalling, scaling)
- Increase in porosity and permeability; loss of strength
- Loss of material
- Loss of material (spalling, scaling) and cracking
- Loss of material; loss of form

The scope of this program also includes structural steel and structural bolting associated with water-control structures. SNUPPS-Callaway positions are compliant with that of Regulatory Guide 1.127 with respect to the ultimate heat sink (UHS) retention pond. The Structures Monitoring program (B2.1.31), which is in compliance with 10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, includes all water-control structures within the scope of RG 1.127. The UHS retention pond, the essential service water pumphouse, the ESW supply lines yard vault, the UHS cooling tower and the submerged discharge structures are the water-control structures within the scope for license renewal that are monitored by this program. The UHS retention pond and its associated structures receive periodic inservice inspections for assessment of their structural safety and operational adequacy every five years. Callaway performs algae treatment and riprap inspections along the UHS retention pond. Callaway maintains benchmarks for monitoring settlement in any of the Category 1 structures including the UHS cooling tower. The inspections of all structural components, including masonry walls and water-control structures, are performed at intervals of no more than five years.

NUREG-1801 Consistency

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program is an existing program that is consistent with NUREG-1801, Section XI.S7, RG 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*.

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

The following discussion of operating experience provides objective evidence that the RG 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation:

1. An instance of concrete delamination occurred in 2005. Open electrical boxes that were part of the abandoned lighting system in the 'A' UHS fan deck room allowed water to enter the embedded conduits located in the concrete wall. This water contributed to the corrosion growth on the conduit which eventually deteriorated enough to cause spalling on the plant north face of the wall separating 'A' and 'B' UHS fan deck rooms. The spalled area was patched with cement grout in 2006.

2. Similar spalling was noted on the south wall in the 'D' UHS cooling tower fan room which had an area approximately 1 ft by 1 ft where the concrete had popped out. The degradation appeared to be about two to three inches deep. The apparent cause was rainwater seeping through an abandoned electrical conduit. A job was initiated to repair both spalled areas in the "D" cooling tower fan room. To prevent recurrence of the concrete spalling, prior to installing the grout patch, a hole was drilled in the exposed part of the conduit to drain any water remaining in the abandoned conduits.

The above examples provide objective evidence that the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program will be evaluated to ensure there is no significant impact to 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 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants 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.

B2.1.36 Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Program Description

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages reduced insulation resistance of power cables (greater than or equal to 400 V) to minimize aging effects which could potentially lead to failure of the cable's insulation system. The program provides reasonable assurance that the intended functions of inaccessible or underground power cables (greater than or equal to 400 V) that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to wetting or submergence are maintained consistent with the current licensing basis through the period of extended operation.

Manholes, pits, and duct banks that contain in-scope non-EQ inaccessible power cables will be inspected for water collection to identify significant moisture. Significant moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable wetting or submergence in water). Any collected water is removed. The inspection includes direct observation that cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. This inspection and water removal is performed based on actual plant experience with the inspection frequency being at least annually and after event driven occurrences (such as heavy rain or flooding). In addition, operation of dewatering devices will be inspected prior to any known or predicted heavy rain or flooding events. The first inspection for license renewal is to be completed prior to the period of extended operation.

In-scope non-EQ inaccessible power cables (greater than or equal to 400V) routed through manholes, pits, or duct banks are tested to provide an indication of the conductor insulation condition. Testing that is a proven test and appropriate to the application at the time of the testing will be performed to detect deterioration of the insulation system due to wetting. Cable testing may be a mix of proven testing methods such as dielectric loss (dissipation factor/power factor), AC voltage withstand, partial discharge, step voltage, time domain reflectometry, insulation resistance and polarization index, line resonance analysis, or other testing that is state-of-the-art at the time the tests are performed. The first test for license renewal will be completed prior to the period of extended operation ~~and~~with subsequent tests performed at least every six years thereafter and adjusted based on test results and operating experience.

Appendix B
AGING MANAGEMENT PROGRAMS

NUREG-1801 Consistency

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.E3, *Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*

Exceptions to NUREG-1801

None

Enhancements

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

Scope of the Program (Element 1)

Procedures will be enhanced to identify the power cables (greater than or equal to 400 volts), manholes, pits, and duct banks that are within the scope of the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.

Preventive Actions (Element 2)

Procedures will be enhanced to include periodic inspection of manholes, pits, and duct banks, to confirm cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. Inspections will be performed at least annually based on water accumulation over time and after event driven occurrences (e.g., heavy rain or flooding). In addition, operation of dewatering devices will be inspected and operation verified prior to any known or predicted heavy rain or flooding events.

Parameters Monitored or Inspected (Element 3)

Procedures will be enhanced to confirm cables are not submerged or immersed in water, cables/splices and cable support structures are intact, dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly, and that power cable subject to significant moisture are tested periodically.

Detection of Aging Effects (Element 4)

Procedures will be enhanced to ensure in-scope power cables are tested at least once every six years and adjusted based on test results and operating experience..

Monitoring and Trending (Element 5)

Procedures will be enhanced to require comparing test results to previous test results to evaluate for additional information on the rate of cable degradation.

Acceptance Criteria (Element 6)

Procedures will be enhanced to confirm cables are not submerged or immersed in water, cables/splices and cable support structures are intact, and dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. Acceptance criteria for cable testing will be defined prior to each test..

Corrective Actions (Element 7)

Procedures will be enhanced to require an engineering evaluation when the test or inspection acceptance criteria are not met.

Operating Experience

The following discussion of operating experience provides objective evidence that the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation.

1. Callaway is performing an inspection of safety related manhole MH01 on a 6 month frequency. Water was found in the manhole in 2003 and 2009 below medium-voltage safety-related cables, in 2006 all cable was found immersed in water. The evaluation indicates the manhole cover seals were degraded and, since 2009, new seals for the manhole covers have been installed at each inspection of MH01. Inspection frequency was changed from 36 to 6 months. Foundation sealant, drain pipes, and flashing have been installed around MH01 in 2010. The as-found water level in MH01A has dropped from 30 to 15 inches and MH01B from 24 to 2 inches.

2. Cable testing is done periodically with cable insulation resistance testers and megger testing. No degradation of safety related cables has been noted. A new tan delta test for ESF and plant service transformer feeder cables has been developed and implemented.

3. A new PM has been written and implemented in 2011 to inspect for water level and pump out all in-scope manholes on a weekly basis.

4. A modification to install sump pumps in all in-scope manholes is in progress, five manholes are complete as of March 12, 2012.

Operating experience with the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program has not identified any cable failures as a result of submergence and the water management in manholes has been enhanced. Occurrences that would be identified under the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will be evaluated to ensure there is no significant impact to 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. Therefore, there is confidence that the continued implementation of the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements 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.