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CHAPTER 19.0

LICENSE RENEWAL SUPPLEMENT

19.0 INTRODUCTION

This chapter provides the information in the Final Safety Analysis Report (FSAR) as required by 10 CFR 54.21(d) for the Callaway Plant License Renewal Application. Section 19.1 of this chapter contains summary descriptions of the programs used to manage the effects of aging during the period of extended operation. Section 19.2 contains summary descriptions of programs used for management of time-limited aging analyses (TLAAs) during the period of extended operation. Section 19.3 contains evaluation summaries of TLAAs for the period of extended operation. Section 19.4 contains summary descriptions of actions committed to by Ameren Missouri in its License Renewal Application to the NRC. Included in Section 19.4, Table 19.4-1, “License Renewal Commitments,” are commitments for license renewal and an associated implementation schedule for when Ameren Missouri plans to complete the commitments. Unless noted otherwise, the following implementation schedule applies for new programs, enhanced programs, and specific activities to be completed prior to the period of extended operation (PEO). The PEO begins at midnight October 18, 2024.

a. Ameren Missouri shall implement those new programs and enhancements to existing programs no later than 6 months prior to PEO.

b. Ameren Missouri shall complete those inspection and testing activities by the 6-month date prior to PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.

c. Ameren Missouri shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

These summary descriptions of aging management programs, time-limited aging analyses, and license renewal commitments are incorporated in the Callaway Plant FSAR in accordance with 10 CFR 50.71(e).

License Conditions are included in Section 2.C.(17) of Renewed Facility Operating License No. NPF-30.

This chapter meets the requirements of License Condition Section 2.C.(17)(a) which states:

> “The information in the Final Safety Analysis Report (FSAR) supplement, submitted pursuant to 10 CFR 54.21(d), is henceforth part of the FSAR which will be updated in accordance with 10 CFR 50.71(e). As such, the licensee may make changes to the programs and activities described in the

FSAR supplement, without prior Commission approval, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section."

Reference: NUREG-2172, "Safety Evaluation Report Related to the License Renewal of Callaway Plant, Unit 1," issued March 2015

19.1 SUMMARY DESCRIPTIONS OF AGING MANAGEMENT PROGRAMS

The integrated plant assessment and evaluation of time-limited aging analyses identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. Sections 19.1 and 19.2 describe the aging management programs and their implementation activities.

Quality Assurance for Aging Management Programs

Three elements common to all aging management programs discussed in Sections 19.1 and 19.2 are corrective actions, confirmation process, and administrative controls. These elements are included in the Callaway Plant QA Program, which implements the requirements of 10 CFR 50, Appendix B. The Callaway Plant QA Program is applicable to safety-related systems, structures and components that are subject to aging management review activities for license renewal. These three elements are also applied to the nonsafety-related systems, structures and components subject to aging management activities.

Consideration of Operating Experience in Aging Management Programs (AMPs)

Operating Experience from plant-specific and industry sources is captured and systematically reviewed on an on-going basis in accordance with the quality assurance program which meets the requirements of 10 CFR Part 50, Appendix B and the operating experience program, which meets the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff." The operating experience program interfaces with and relies on active participation in the Institute of Nuclear Power Operations' operating experience program, as endorsed by the NRC. In accordance with these programs, all incoming operating experience items are screened to determine whether they may involve age-related degradation or aging management impacts. Items so identified are further evaluated and applicable AMPs may be enhanced or new AMPs may be developed, as appropriate, if it is determined that the effects of aging may not be adequately managed. Training on age-related degradation and aging management is provided to those personnel responsible for implementing the AMPs and who are likely to submit, screen, assign, evaluate, or otherwise process plant-specific and industry operating experience. Plant-specific operating experience associated with aging management and age-related degradation is reported to the industry in accordance with guidelines established in the operating experience program.

19.1.1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program manages cracking, loss of fracture toughness, and loss of material. The program

consists of periodic volumetric, surface, and/or visual examinations and leakage testing of ASME Class 1, 2, and 3 pressure-retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting for assessment, signs of degradation, and corrective actions. Callaway inspections meet ASME Section XI requirements. Callaway will use the ASME Code Section XI edition and addenda consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

The thickness of the reactor vessel wall indications in the reactor vessel lower head will be determined by a) obtaining surface profile data of the indications and surrounding cladding using an ultrasonic examination from the inside of the reactor vessel, b) using an ultrasonic examination from the outside of the reactor vessel, or c) using remote mechanical gages inside the reactor vessel.

19.1.2 WATER CHEMISTRY

The Water Chemistry program manages loss of material, cracking, reduction of heat transfer, and wall thinning in components exposed to a treated water environment. The Water Chemistry program is used to control water chemistry for impurities that accelerate corrosion. The program is a mitigation program that relies on monitoring and control of primary and secondary water chemistry to keep peak levels of various contaminants below system-specific limits based on EPRI guidelines. The Water Chemistry program is based on EPRI primary and secondary water chemistry guidelines.

The One-Time Inspection program (19.1.18) verifies the effectiveness of the Water Chemistry program.

19.1.3 REACTOR HEAD CLOSURE STUD BOLTING

The Reactor Head Closure Stud Bolting program manages cracking and loss of material by conducting ASME Section XI inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, and washers.

The Reactor Head Closure Stud Bolting program includes periodic visual and volumetric examinations of reactor vessel flange stud hole threads, reactor head closure studs, nuts, and washers and performs visual inspection of the reactor vessel flange during primary system leakage tests. The program is implemented through station procedures consistent with the examination and inspection requirements specified in ASME Section XI, Subsection IWB, Table IWB-2500-1. Callaway will use the ASME Code edition consistent with the provisions of 10 CFR 50.55a during the period of extended operation.

The program includes preventive measures as recommended in Regulatory Guide 1.65, “Materials and Inspections for Reactor Vessel Closure Studs” to use stable lubricants and NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Plants,” to use bolting material for closure studs that has an actual yield strength less than 150 kilo-pounds per square inch.

19.1.4 BORIC ACID CORROSION

The Boric Acid Corrosion program manages loss of material and increased resistance of connection due to borated water or reactor coolant leakage. The program includes provisions to identify leakage through inspection and examination. When leakage is identified, an inspection is performed that includes identification of the leakage path, visual inspections of adjacent structures, components and supports, and cleaning of the leakage. When it is determined that an evaluation is necessary, it is performed in a timely manner. If the evaluation identifies aging effects, corrective action will be taken. Monitoring is provided by tracking and trending of existing and repaired leaks and the establishment of a component-based visual history of leakage. The principal industry guidance document used is WCAP-15988-NP, “Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors.” The program relies in part on implementation of recommendations of NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.” Additionally, the program includes examinations conducted during ISI pressure tests performed in accordance with ASME Section XI requirements.

The effects of boric acid corrosion on reactor coolant pressure boundary materials in the vicinity of nickel-alloy components are managed by the Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program (19.1.5).

19.1.5 CRACKING OF NICKEL-ALLOY COMPONENTS AND LOSS OF MATERIAL DUE TO BORIC ACID-INDUCED CORROSION IN REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

The Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program manages cracking of nickel-alloy components and associated welds in reactor coolant pressure boundary components. This program also manages loss of material due to boric acid-induced corrosion in susceptible, safety-related components in the vicinity of nickel-alloy reactor coolant pressure boundary components.

The program provides inspection requirements for the reactor pressure vessel, pressurizer, and reactor coolant pressure boundary piping components if they contain primary water stress corrosion cracking susceptible materials designated alloys 600/82/182. The program also includes inspection requirements for the reactor pressure vessel upper head.

19.1.6 PWR VESSEL INTERNALS

The PWR Vessel Internals program relies on implementation of the guidance included in Electric Power Research Institute (EPRI) 1022863 (MRP-227-A), “PWR Internals Inspection and Evaluation Guideline” and EPRI 1016609 (MRP-228), “Inspection

Standard for PWR Internals" to manage the aging effects of reactor vessel internal (RVI) components.

This program is used to manage (a) various forms of cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) changes in dimensions due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

19.1.7 FLOW-ACCELERATED CORROSION

The Flow-Accelerated Corrosion (FAC) program manages aging effects of wall thinning on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phases). The program implements the EPRI guidelines in NSAC-202L-R3 to detect, measure, monitor, predict, and mitigate component wall thinning.

Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to FAC are used to predict the amount of wall thinning. Program activities include analyses to determine critical locations, baseline inspections to determine the extent of thinning at these critical locations, and follow-up inspections to confirm the predictions. Inspections are performed using ultrasonic, visual or other approved testing techniques capable of detecting wall thinning. Repairs and replacements are performed as necessary.

19.1.8 BOLTING INTEGRITY

The Bolting Integrity program manages cracking, loss of material and loss of preload for pressure retaining bolting. The program includes periodic inspection of closure bolting for pressure-retaining components consistent with recommendations as delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants" and EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," Volume 1 and 2 with the exceptions noted in NUREG-1339. The Bolting Integrity program also includes activities for preload control, material selection and control, and use of lubricants/sealants as delineated in EPRI TR-104213, "Bolted Joint Maintenance and Application Guide."

The ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD program supplements the Bolting Integrity program by providing the requirements for inservice inspection of ASME Class 1, 2, and 3 safety-related pressure retaining bolting. The integrity of non-ASME Class 1, 2, 3 system and component bolted joints is evaluated by detection of visible leakage during maintenance or routine observation such as system walkdowns.

A sample of submerged bolting heads in raw water and waste water environments is visually inspected every four refueling outages (six years) when the pumps are dewatered. In addition, when submerged raw water and waste water pump casings are disassembled during maintenance activities, the bolting threads will be opportunistically inspected. A sample of submerged bolting on the fuel oil storage tank transfer pumps is visually inspected every 10 years when the pumps are disassembled during maintenance activities. The sample for submerged bolting will be 20% of the population with a maximum of 25 for each environment. The inspection of submerged bolting focuses on the bounding or lead components most susceptible to aging due to time in service and severity of operating conditions.

Safety-related and nonsafety-related structural bolting is managed by the following programs:

a. ASME Section XI, Subsection IWE program (19.1.26) provides the requirements for inspection of structural bolting.

b. ASME Section XI, Subsection IWF program (19.1.28) provides the requirements for inservice inspection of safety-related component support bolting.

c. Structures Monitoring program (19.1.31) monitors the condition of structures and structural supports that are within the scope of license renewal.

d. RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program (19.1.32) provides the requirements for inspection of water control structures associated with emergency cooling water systems.

e. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems program (19.1.12) provides the requirements for inspection of handling systems within the scope of license renewal.

Reactor pressure vessel head closure studs are managed by the Reactor Head Closure Stud Bolting program (19.1.3).

Inspection activities for bolting in buried and underground applications is performed in conjunction with inspection activities for the Buried and Underground Piping and Tanks (19.1.25) program due to the restricted accessibility to these locations.

19.1.9 STEAM GENERATORS

The Steam Generators program manages cracking, loss of material, reduction of heat transfer, and wall thinning of the steam generator tubes, plugs, sleeves and secondary side steam generator internal components. The program provides preventive measures

in the form of predictive assessment, tube plugging, foreign material exclusion, foreign object search, secondary side cleaning and maintenance, and maintaining the chemistry. The program detects degradation through nondestructive examinations, visual inspection, and in situ pressure testing. Assessments are used to verify that the steam generator performance criteria defined in the Callaway Technical Specifications have been met over the last operating interval and ensure that the criteria will be met over the next operating interval.

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

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

<u>Option 1: Inspection</u>

Perform a one-time inspection of each steam generator to assess the condition of the divider plate welds. The examination technique(s) will be capable of detecting PWSCC in the divider plate assemblies and the associated welds.

OR

<u>Option 2: Analysis</u>

Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator reactor coolant system (RCS) pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval.

OR

<u>Option 3: Industry/NRC Studies</u>

If results of industry and NRC studies and operating experience document that potential failure of the steam generator RCS pressure boundary due to PWSCC cracking of steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion.

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

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

Option 1: Inspection

Perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If weld cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators.

OR

Option 2: Analysis

Perform an analytical evaluation of the steam generator tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC, or redefining the reactor coolant pressure boundary of the tubes, where the steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.

19.1.10 OPEN-CYCLE COOLING WATER SYSTEM

The Open-Cycle Cooling Water System program manages loss of material, wall thinning, reduction of heat transfer, cracking, blistering, change in color, and hardening and loss of strength for components within the scope of license renewal and exposed to the raw water of the essential service water system and heat exchangers and other components in other systems serviced by the essential service water system. The program also manages loss of coating integrity on components with an internal coating.

The program is consistent with commitments as established in responses to NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Components” and includes:

a. surveillance and control of biofouling,

b. tests to verify heat transfer,

c. routine inspection and maintenance program,

d. system walkdown inspection, and

e. review of maintenance, operating, and training practices and procedures.

The Open-Cycle Cooling Water System program includes the essential service water system that transfers heat from the safety-related structures, systems and components to the ultimate heat sink as defined in NRC Generic Letter 89-13. Periodic heat transfer testing or inspection and cleaning of heat exchangers with a heat transfer intended function is performed in accordance with commitments to NRC Generic Letter 89-13 to verify heat transfer capabilities.

Visual inspections are performed on all accessible internal surface coatings of the component cooling water heat exchangers, Class 1E electrical equipment air conditioners, control room air conditioners, and essential service water self-cleaning strainers. For coated surfaces determined to not meet the acceptance criteria and that will not be repaired or replaced, physical testing is performed where physically possible (i.e., sufficient room to conduct testing). The test consists of destructive or nondestructive adhesion testing using ASTM International Standards endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the staff associated with a particular standard.

19.1.11 CLOSED TREATED WATER SYSTEMS

The Closed Treated Water Systems program manages loss of material, cracking, and reduction of heat transfer for components within the scope of license renewal in the closed-cycle cooling water systems.

The Closed Treated Water Systems program is a preventive program that relies on water treatment, including the use of corrosion inhibitors to modify the chemistry of the water and chemical testing to ensure that water chemistry is maintained within acceptable guidelines. The program also conducts periodic inspections to determine the presence or extent of corrosion, fouling, and/or cracking.

19.1.12 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems program manages loss of material and loss of preload for bolting for all cranes, trolley and hoist structural components, fuel handling equipment and applicable rails within the scope of license renewal. Visual inspections will manage loss of material due to corrosion of structural members and bolting, loss of materials due to wear of rails, and loss of preload for bolted connections.

Crane inspections are performed in accordance with the ASME B30 standards. Inspections are performed at a frequency that meets the requirements of the ASME B30

series. For cranes that are infrequently in service, such as the containment polar crane, periodic inspections are performed once every refueling cycle just prior to use.

19.1.13 FIRE PROTECTION

The Fire Protection program manages loss of material for fire rated doors, fire dampers and the Halon system, concrete cracking, spalling, and loss of material for fire barrier walls, ceilings and floors and increased hardness, shrinkage, and loss of strength of fire barrier penetration seals.

Periodic visual inspections of fire barrier penetration seals, fire dampers, fire barrier walls, ceilings and floors are performed to ensure that they can perform their intended functions. Visual inspections and functional tests are performed on fire-rated doors. The program also includes periodic visual inspection and functional testing of the Halon system.

19.1.14 FIRE WATER SYSTEM

The Fire Water System program manages loss of material and flow blockage for water-based fire protection systems. This program manages aging effects through the use of flow testing and visual inspections performed consistent with provisions of the 2011 Edition of National Fire Protection Association (NFPA) 25 noted in Table 19.1-1. Unless noted in Table 19.1-1, flow testing and visual inspections are performed at intervals specified in the 2011 Edition of NFPA 25. Testing or replacement of sprinklers that have been in place for 50 years is performed in accordance with the 2011 Edition of NFPA 25.

In addition to NFPA codes and standards, portions of the water-based fire protection system that are: (a) normally dry but periodically subjected to flow (e.g., dry-pipe or preaction sprinkler system components) and (b) cannot be drained or allow water to collect are to be subjected to augmented testing beyond that specified in NFPA 25, including (a) periodic full flow tests at the design pressure and flow rate or internal visual inspections and (b) volumetric wall-thickness examinations.

The water-based fire protection system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated.

Non-intrusive wall thickness examinations are performed on fire water piping to identify loss of material. Wall thickness examinations will be performed on fire water piping every three years. Each three year sample will include at least three locations for a total of 100 feet of above-ground fire water piping and be selected based on system susceptibility to corrosion or fouling and evidence of performance degradation during system flow testing or periodic flushes. The basis for the frequency is that three years is the frequency required by the FSAR for the yard fire loop flush and for the flow tests of the fire water loops.

Internal visual internal inspections are used when the internal surface of the piping is exposed during plant maintenance.

Samples are collected for microbiologically-influenced corrosion quarterly and when fire water piping and components are opened for maintenance or are accessible. Biofouling is prevented by periodically adding treatment chemicals such as an anti-scalant, a biopenetrant, and a biostat to the fire water system annually and when monitoring indicates they should be added. The MIC Index is trended to evaluate treatment effectiveness in specific locations.

Inspections of wetted normally dry piping segments that cannot be drained or that allow water to collect begin five years before the period of extended operation. The program's remaining inspections begin during the period of extended operation.

19.1.15 ABOVEGROUND METALLIC TANKS

The Aboveground Metallic Tanks program manages loss of material and cracking on the external surfaces of aboveground metallic tanks within the scope of license renewal that are supported on concrete. The program also manages cracking, blistering, and change in color of the acrylic/urethane insulation on the condensate storage tank (CST). The program applies to the CST and refueling water storage tank (RWST).

This program performs visual inspections during each refueling cycle to monitor for damage of the external surface insulation covering. A one-time inspection of 25 external surface locations of approximately one square foot in area will be inspected by removing insulation, with at least 10 locations near the base of the tank wall and at least two locations on the dome. Removal of the tank insulation permits visual and surface examinations of the tank external surfaces to confirm that environmental impacts are not present at sufficient levels to cause pitting corrosion, crevice corrosion or cracking.

One-time volumetric examinations are taken within five years prior to entering the period of extended operation from inside the emptied tanks to determine the thickness of the tank bottom. The entire tank bottom will be scanned to detect loss of material. Thickness measurements will be performed on regions of the tank bottom that indicate a loss of material below nominal plate thickness. The thickness measurements ensure significant loss of material is not occurring, so that the intended function of each tank is maintained during the period of extended operation. A soil sample is taken underneath the CST and RWST and analyzed to demonstrate that the soil is not corrosive. The soil sample is performed prior to entering the period of extended operation and during each ten-year period in the period of extended operation.

The chemical treatments of cooling tower water do not contain chemical compounds that could cause cracking, pitting, or crevice corrosion on the external surfaces of the tanks. Prior to entering the period of extended operation and during each ten year period in the period of extended operation, a soil surface sample near the CST and RWST will be performed. The soil surface sample and samples of residue on the top and sides of each

tank will be evaluated to ensure that chlorides or other aggressive cooling tower water treatment chemicals are not creating an aggressive environment that would degrade the CST, RWST, or their insulation jacketing.

A one-time inspection consistent with the One-Time Inspection program (19.1.18) will be performed on the inside surfaces of the CST and RWST shell, roof and bottom.

The Aboveground Metallic Tanks program is a new program that will be implemented within the five year period prior to the period of extended operation.

Industry and plant-specific operating experience will be evaluated in the development and implementation of this program.

19.1.16 FUEL OIL CHEMISTRY

The Fuel Oil Chemistry program manages loss of material on the internal surface of components in the emergency diesel engine fuel oil storage and transfer system, fire protection system, standby diesel generator engine system, alternate emergency power system, and EOF and TSC diesels security building system. The program also manages loss of coating integrity for the emergency diesel fuel oil storage tanks and day tanks. The program includes (a) surveillance and monitoring procedures for maintaining fuel oil quality by controlling contaminants in accordance with the Callaway Technical Specifications and ASTM Standards D1796-83 and D2276-78, (b) periodic draining of the emergency fuel oil system storage tanks and day tanks, (c) cleaning and visual inspection of internal surfaces of the emergency fuel oil system storage tanks and day tanks during periodic draining, (d) ultrasonic measurements of the emergency fuel oil system storage tank and fuel oil day tank bottom thickness if there are indications of reduced cross sectional thickness found during the visual inspection, (e) periodic volumetric examination of tank bottom from the external surface the diesel fire pump fuel oil day tank and security diesel generator fuel oil day tank where tank design prevents cleaning and inspection from the inside, (f) inspection of new fuel oil before introduction to storage tanks, and (g) monitoring the secondary containment of the alternate emergency power system fuel oil storage tanks for indications of leakage from the primay tank.

Visual inspections are performed on all accessible internal surface coatings of the emergency fuel oil storage tanks and day tanks. For coated surfaces determined to not meet the acceptance criteria and that will not be repaired or replaced, physical testing is performed where physically possible (i.e., sufficient room to conduct testing). The test consists of destructive or nondestructive adhesion testing using ASTM International Standards endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the staff associated with a particular standard.

The One-Time Inspection program (19.1.18) will be used to verify the effectiveness of the Fuel Oil Chemistry program.

19.1.17 REACTOR VESSEL SURVEILLANCE

The Reactor Vessel Surveillance program manages loss of fracture toughness. The surveillance capsules contain reactor vessel steel specimens of the limiting beltline material; and associated weld metal and weld heat affected zone metal. The program extends the scope of 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to provide sufficient material data and dosimetry for monitoring irradiation embrittlement at the end of the period of extended operation, and also determines the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux.

All capsules in the reactor vessel are removed and tested consistent with the test procedures and reporting requirements of ASTM E 185-82. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. Untested capsules placed in storage will be maintained for future insertion or testing.

Vessel fluence is determined by ex-vessel dosimetry since all capsules have been removed.

19.1.18 ONE-TIME INSPECTION

The One-Time Inspection program conducts one-time inspections of selected components in susceptible locations to verify the effectiveness of the Water Chemistry program (19.1.2), Fuel Oil Chemistry program (19.1.16), and Lubricating Oil Analysis program (19.1.24). The aging effects evaluated by the One-Time Inspection program are loss of material, cracking, and reduction of heat transfer. The One-Time Inspection program provides inspections that verify that unacceptable aging effects are not occurring.

The elements of the program include: (a) determination of the sample size based on 20% of the components in each material-environment group up to a maximum of 25 components, (b) identification of the inspection locations in each material-environment-aging effect group based on the potential for the aging effect to occur, (c) determination of the examination technique, including acceptance criteria that would be effective in detecting the aging effect for which the component is examined, (d) evaluation of aging effects and the need for follow-up examinations using the corrective action program.

The One-Time Inspection program is not used for structures or components with known aging effects or when a component is in a different environment in the period of extended operation than it experienced in the prior 40 years.

The One-Time Inspection program is a new program that will be implemented and completed within the 10-year period prior to the period of extended operation.

Industry and plant-specific operating experience will be evaluated in the development and implementation of this program.

19.1.19 SELECTIVE LEACHING

The Selective Leaching program manages loss of material due to selective leaching for gray cast iron and copper alloy with greater than 15% zinc components that are exposed to treated water, raw water, waste water, or groundwater environments and require aging management. The material of copper alloy greater than eight percent aluminum (aluminum-bronze) was not used in systems that require aging management at Callaway.

The Selective Leaching program includes a one-time visual inspection and other mechanical inspection techniques of selected components that may be susceptible to selective leaching. If these inspections detect selective leaching, then a follow-up evaluation is performed. The evaluation may require confirmation of selective leaching through a metallurgical evaluation. This is to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended functions for the period of extended operation. Buried gray cast iron fire protection valves will be inspected opportunistically. In addition, when any buried gray cast iron valves are removed from the fire protection system, then specimens from at least one of them will be sent to a laboratory for metallurgical testing to determine the extent, if any, of selective leaching of the valve. A minimum of two metallurgical tests will be performed within the five years prior to entering the period of extended operation.

The Selective Leaching program is a new program and inspections will be completed within the five-year period prior to the period of extended operation.

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.20 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping program manages cracking of ASME Code Class 1 piping less than four inches nominal pipe size (NPS 4) and greater than or equal to one inch nominal pipe size (NPS 1).

For ASME Code Class 1 small-bore piping, the Risk-informed (RI-ISI) ISI program requires volumetric examinations (by ultrasonic testing) on selected butt weld locations to detect cracking. Weld locations are selected based on the guidelines provided in EPRI TR-112657, “Revised Risk-Informed Inservice Inspection Evaluation Procedure.”

Ultrasonic examinations are conducted in accordance with ASME Section XI with acceptance criteria from paragraph IWB-3000 for butt welds.

The program will include a volumetric or opportunistic destructive examination of socket welds to identify potential cracking. Callaway has experienced one case of cracking, in 1995, of an ASME Code Class 1 small-bore piping butt weld resulting from cyclical loading which was mitigated with a design change to prevent recurrence. Eight small-bore Class 1 socket welds will be selected for examination, which represents 10% of the population. There are 80 Class 1 small-bore socket welds in the population of ASME Code Class 1 piping less than NPS 4 and greater than or equal to NPS 1 at Callaway. Alternatively, an opportunistic destructive examination may be used in lieu of volumetric examinations. An opportunistic destructive examination may be performed when a weld is removed from service for reasons other than inspection. Because more information can be obtained from a destructive examination than from a nondestructive examination, each weld destructively examined will be considered equivalent to having volumetrically examined two welds.

Socket welds that fall within the weld examination sample will be examined following ASME Section XI Code requirements. If a qualified volumetric examination procedure for socket welds endorsed by the industry or the NRC is available and incorporated into the ASME Section XI Code at the time of the small-bore inspections, then this will be used for the volumetric examinations. If no volumetric examination procedure for ASME Code Class 1 small-bore socket welds has been endorsed by the industry or the NRC and incorporated into ASME Section XI at the time Callaway performs inspections of small-bore piping, a plant procedure for volumetric examination of ASME Code Class 1 small-bore piping with socket welds will be used.

The program includes controls to implement an alternate plant-specific periodic inspection aging management program should evidence of ASME Class 1 small bore piping cracking caused by intergranular stress corrosion cracking or fatigue be confirmed by review of Callaway operating experience prior to the period of extended operation or by the examinations performed as part of this program.

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping program is a new program and inspections will be completed and evaluated within six years prior to the period of extended operation.

In conformance with 10 CFR 50.55a(g)(4)(ii), the ISI program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition of the ASME Code specified twelve 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 10-year period prior to the period of extended operation (fourth interval).

19.1.21 EXTERNAL SURFACES MONITORING OF MECHANICAL COMPONENTS

The External Surfaces Monitoring of Mechanical Components program manages loss of material and cracking for metallic components and cracking and changes in material properties for cement board components. The program also manages loss of material, cracking, and hardening and loss of strength for polymeric components. Periodic visual inspections of external surfaces conducted through engineering walkdowns will be used to identify loss of material and leakage. A sample of outdoor component surfaces that are insulated and a sample of indoor insulated components exposed to condensation (due to the in-scope component being operated below the dew point), are periodically inspected every 10 years during the period of extended operation. Periodic polymeric inspections will also include manual or physical manipulation in order to verify the absence of cracking, hardening, or loss of strength. Periodic monitoring of stainless steel components will also include visual inspection for cracking when exposed to an air environment containing halides.

The External Surfaces Monitoring of Mechanical Components program is a new program that will be implemented prior to the period of extended operation.

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.22 FLUX THIMBLE TUBE INSPECTION

The Flux Thimble Tube Inspection program manages loss of material by performing wall thickness eddy current testing of all flux thimble tubes that form part of the reactor coolant system pressure boundary. The pressure boundary includes the length of the tube inside the reactor vessel out to the seal fittings outside the reactor vessel. Eddy current inspection is performed on the portion of the tubes inside the reactor vessel. The Flux Thimble Tube Inspection program does not prevent loss of material but provides measures for inspection and evaluation to detect the loss of material prior to loss of intended function.

All flux thimble tubes are periodically inspected during refueling outages. Wall thickness measurements are trended and wear rates are calculated. The refueling outage for the next inspection is determined from the wear rate calculations. If the current measured wear exceeds the acceptance criteria or the predicted wear for a given flux thimble tube is projected to exceed the established acceptance criteria prior to the next refueling outage, corrective actions are taken to reposition, cap or replace the tube.

The Flux Thimble Tube Inspection program implements the recommendations of NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors.”

19.1.23 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program manages cracking, loss of material, hardening and loss of strength. The program also manages loss of coating integrity on components with an internal coating. The program inspects internal surfaces of metallic piping, piping components, piping elements, ducting, heat exchanger components, polymeric and elastomeric components, and other components that are exposed to plant indoor air, ventilation atmosphere, atmosphere/weather, condensation, borated water leakage, diesel exhaust, lubricating oil, fuel oil, and water system environment not managed by Open-Cycle Cooling Water System (19.1.10), Closed Treated Water System (19.1.11), Fire Water System (19.1.14), and Water Chemistry (19.1.2) programs.

Internal inspections are normally performed at opportunities where the internal surfaces are made accessible, such as periodic system and component surveillance activities or maintenance activities. Visual inspections of internal surfaces of plant components are performed by qualified personnel. For certain materials, such as polymers, visual inspections will be augmented by physical manipulation or pressurization to detect hardening, loss of strength, and cracking. The program includes inspections to detect material degradation that could result in a loss of component intended function.

At a minimum, in each 10-year period during the period of extended operation a representative sample of 20% of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 25 components per population is inspected. Where practical, the inspections focus on the bounding or lead components most susceptible to aging because of time in service, and severity of operating conditions. Opportunistic inspections continue in each period despite meeting the sampling limit.

Following a failure due to recurring internal corrosion, this program may be used if the failed material is replaced by one that is more corrosion-resistant in the environment of interest, or corrective actions have been taken to prevent recurrence of the recurring internal corrosion.

Visual inspections are performed on all accessible internal surface coatings of the service water self-cleaning strainers and a representative 73 one-foot axial length circumferential segments of service water piping from the circulating and service water pumphouse to the ESW system connection. For coated surfaces determined to not meet the acceptance criteria and that will not be repaired or replaced, physical testing is performed where physically possible (i.e., sufficient room to conduct testing). The test consists of destructive or nondestructive adhesion testing using ASTM International Standards endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the staff associated with a particular standard.

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program and will be implemented prior to the period of extended operation.

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.24 LUBRICATING OIL ANALYSIS

The Lubricating Oil Analysis program manages oil environments in order to prevent loss of material and reduction of heat transfer. The program does not manage component surfaces directly but maintains lubricating oil contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer.

The One-Time Inspection program (19.1.18) verifies the effectiveness of the Lubricating Oil Analysis program.

19.1.25 BURIED AND UNDERGROUND PIPING AND TANKS

The Buried and Underground Piping and Tanks program manages loss of material, cracking, blistering, and change of color of the external surfaces of buried and underground piping and tanks. The program augments other programs that manage the aging of internal surfaces of buried and underground piping and tanks. The materials managed by this program include steel, stainless steel, and high-density polyethylene. The program manages aging through preventive, mitigative, and inspection activities.

Preventive and mitigative actions include selection of component materials, external coatings for corrosion control, backfill quality control and the application of cathodic protection. The cathodic protection system is operated consistent with the guidance of National Association of Corrosion Engineers (NACE) SP 0169-2007 for piping, and NACE RP 0285-2002 for tanks. Trending of the cathodic protection system is performed to identify changes in the effectiveness of the system and to ensure that the rectifiers are available to protect buried components. An annual cathodic protection survey is performed consistent with NACE SP 0169-2007.

Soil samples will be conducted during the 10-year period prior to the period of extended operation and in each subsequent 10-year period during the period of extended operation. Soil samples will be performed in the vicinity of buried steel piping in which the cathodic protection system does not meet the following availability or effectiveness requirements:

- Cathodic protection has been operational (available) at least 85% of the time since either 10 years prior to the period of extended operation or since installation/refurbishment, whichever is shorter; or

- Cathodic protection has provided effective protection for buried piping as evidenced by meeting the acceptance criteria of -850 mV relative to a copper/copper sulfate electrode, instant off, at least 80% of the time since either 10 years prior to the period of extended operation or since installation/refurbishment, whichever is shorter.

Inspection activities include non-destructive evaluation of pipe or tank wall thickness, and visual inspection of the exterior, as permitted by opportunistic or directed excavations.

The Buried and Underground Piping and Tanks program is a new program that will be implemented within the 10-year period prior to entering the period of extended operation.

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.26 ASME SECTION XI, SUBSECTION IWE

The ASME Section XI, Subsection IWE program manages cracking, loss of material, loss of sealing, loss of preload, and loss of leak tightness by providing aging management of the steel liner of the concrete containment building, including the containment liner plate and its integral attachments, containment hatches and airlocks, and pressure-retaining bolting. IWE inspections are performed in order to identify and manage any containment liner aging effects that could result in loss of intended function. Acceptance criteria for components subject to Subsection IWE examination requirements are specified in Article IWE 3000. The Callaway containment inservice inspections program is consistent with the requirements of 2001 Edition of ASME Section XI, Subsection IWE (through the 2003 addenda), supplemented with the applicable requirements of 10 CFR 50.55a(b)(2)(ix). In conformance with 10 CFR 50.55a(g)(4)(ii), the Callaway containment inservice inspections 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.

19.1.27 ASME SECTION XI, SUBSECTION IWL

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

Inspections will be performed to identify and manage any aging effects of the containment concrete, post-tensioning tendons, tendon anchorages, and concrete surface around the anchorage that could result in loss of intended function. In conformance with 10 CFR 50.55a(g)(4)(ii), the ASME Section XI, Subsection IWL 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.

19.1.28 ASME SECTION XI, SUBSECTION IWF

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

19.1.29 10 CFR PART 50, APPENDIX J

The 10 CFR Part 50, Appendix J program manages cracking, loss of material, loss of leak tightness, loss of sealing, and loss of preload. The program monitors leakage rates through the containment pressure boundary, including the penetrations and access openings, in order to detect degradation of containment pressure boundary. Containment leak rate tests are performed in accordance with 10 CFR 50 Appendix J, “Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors (Option B);” NRC Regulatory Guide 1.163, “Performance-Based Containment Leak-Test Program,” NEI 94-01, “Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50 Appendix J;” and ANSI/ANS 56.8, “Containment System Leakage Testing Requirements.”

Containment leak rate tests are performed to assure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage limits specified in the Technical Specifications. Corrective actions are taken if leakage rates exceed established administrative limits for individual penetrations or the overall containment pressure boundary.

19.1.30 MASONRY WALLS

The Masonry Walls program manages cracking of masonry walls. The Masonry Walls program, administered as part of the Structures Monitoring program (19.1.31), 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 five years.

19.1.31 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 five years.

19.1.32 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 (19.1.31), 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 (19.1.31) 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 associated 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 five years.

19.1.33 PROTECTIVE COATING AND MONITORING AND MAINTENANCE PROGRAM

The Protective Coating Monitoring and Maintenance Program manages loss of coating integrity for Service Level 1 coatings inside containment so that the intended functions of post-accident safety systems that rely on water recycled through the containment sump/drain system are maintained consistent with the current licensing basis. The program includes a visual examination of all accessible Service Level 1 coatings inside containment, including those applied to the steel containment liner, structural steel, supports, penetrations, and concrete walls and floors. The program is consistent with the ASTM requirements, but Callaway is not committing to all the requirements noted in NRC Regulatory Guide 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” Revision 2.

19.1.34 INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages reduced insulation resistance to ensure that accessible electrical cables, connections and terminal blocks not subject to the environmental qualification (EQ) requirements of 10 CFR 50.49 and within the scope of license renewal are capable of performing their intended functions.

Non-EQ cables, connections and terminal blocks within the scope of license renewal in accessible areas with an adverse localized environment are inspected for embrittlement, melting, cracking, swelling, surface contamination, or discoloration that could indicate incipient conductor insulation aging from temperature, radiation, or moisture at least once every ten years.

19.1.35 INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

The Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits program manages reduced insulation resistance to ensure that cables and connections used in sensitive instrumentation circuits with high voltage low-level current signals within the ex-core neutron monitoring system are capable of performing their intended functions. All high voltage cables to radiation monitors within the scope of license renewal are managed by the Environmental Qualification (EQ) of Electric Components program (19.2.2).

This program provides reasonable assurance that the intended function of cables and connections used in instrumentation circuits with sensitive, low-level current signals that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are

exposed to adverse localized environments caused by temperature, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation. In most areas, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment for those areas.

Surveillances are used to manage the aging of the cable insulation and connections for ex-core neutron monitors so that instrumentation circuits perform their intended functions. When an instrumentation channel is found to be out of calibration during routine surveillances, troubleshooting is performed on the loop, including the instrumentation cable and connections. A review of surveillance results will be completed prior to the period of extended operation and every 10 years thereafter.

19.1.36 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 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.

19.1.37 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages increased resistance of connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation. As part of the Callaway predictive maintenance program, infrared thermography testing is being performed on non-EQ electrical cable connections, associated with active and passive components within the scope of license renewal. A sample of connections will be tested at least once prior to the period of extended operation using infrared thermography to confirm that there are no aging effects requiring management during the period of extended operation. The selected sample is based upon voltage level (medium and low voltage), circuit loading (high loading), connection type, and location (high temperature, high humidity, vibration, etc.).

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.38 MONITORING OF NEUTRON-ABSORBING MATERIALS OTHER THAN BORAFLEX

The Monitoring of Neutron-Absorbing Materials Other than Boraflex program manages reduction of neutron-absorbing capacity; change in dimensions, and loss of material to assure that aging of the Boral® neutron-absorbing material used in the spent fuel storage racks does not invalidate the criticality analysis of the spent fuel pool.

The program is a monitoring program which performs inspections and in-situ testing of the Boral® panels in the spent fuel pool. Testing includes areal density measurements of the boron-10 in the Boral® panels, and visual inspections of the Boral® panel sheaths look for geometry changes caused by bulging or swelling. The results are evaluated against acceptance criteria and previous inspections to determine whether corrective actions are required. If required, appropriate actions are taken to ensure the required five percent sub-criticality margin is maintained. Monitoring of the Boral® panels in the spent fuel pool will be performed on a ten-year frequency.

The Monitoring of Neutron-Absorbing Materials Other than Boraflex program is a new program that will be implemented prior to the period of extended operation.

Industry and plant specific operating experience will be evaluated in the development and implementation of this program.

19.1.39 METAL ENCLOSED BUS

The Metal Enclosed Bus program manages aging of in-scope non-segregated phase metal enclosed buses including bus connections, enclosures, and insulation and insulators. The internal surfaces of bus enclosure assemblies are inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. Bus insulation is inspected for signs of reduced insulation resistance due to thermal/thermoxidative degradation of organics/thermoplastics, radiation-induced oxidation, moisture/debris intrusion, or ohmic heating, as indicated by embrittlement, cracking, chipping, melting, discoloration, or swelling, which may indicate overheating or aging degradation. The internal bus insulating supports are inspected for structural integrity and signs of cracks. The external portions of a bus, including gaskets and sealants, are inspected for surface cracking, crazing, scuffing, dimensional change (e.g., "ballooning" and "necking"), shrinkage, discoloration, hardening and loss of strength due to elastomer degradation. The external surfaces are inspected for loss of material due to general, pitting, and crevice corrosion. A sample of the accessible bolted connections will be inspected by a visual inspection of insulating material.

Inspections of metal enclosed buses, including a sample of bolted connections, are performed on an interval not to exceed five years.

TABLE 19.1-1 FIRE WATER SYSTEM AGING MANAGEMENT

Component	NFPA 25 Section	Aging Management Performed
Sprinkler Systems: Sprinkler inspections	5.2.1.1	Sprinklers are inspected for signs of leakage, corrosion, and foreign material.
Sprinkler Systems: Sprinkler testing	5.3.1	Prior to 50 years in service, the Fire Water System program requires sprinkler heads to be replaced or have representative samples submitted for field-service testing by a recognized testing laboratory in accordance with NFPA 25. The program field-service tests additional representative samples every 10 years thereafter during the period of extended operation to ensure signs of aging are detected in a timely manner.
Standpipe and Hose Systems: Flow Tests	6.3.1	Flow testing is conducted at least every five years at the hydraulically most remote hose connections of each zone of an automatic standpipe system to verify the water supply provides the design pressure at the required flow.
Private Fire Service Mains: Underground and Exposed Piping	7.3.1	Underground and exposed piping is flow tested at flows representative of those during a fire to determine the internal condition of the piping at minimum three-year intervals.
Private Fire Service Mains: Hydrants	7.3.2	Hydrants are flow tested annually to ensure proper functioning.
Fire Pumps: Suction Screens	8.3.3.7	Not applicable. Callaway's fire protection pumps do not have suction screens.
Water Storage Tanks: Exterior Inspections	9.2.5.5	The exterior painted surface of the fire water storage tanks (FWSTs) is inspected annually for signs of degradation.
Water Storage Tanks: Interior inspections	9.2.6 9.2.7	The interior of each FWST is inspected every other refueling cycle for signs of aging. Testing of interior surfaces is performed for coating integrity and tank bottom integrity when FWSTs exhibit signs of interior pitting, corrosion, or coating failure.
Valves and System-Wide Testing: Main Drain Test	13.2.5	Main drain tests are performed on a representative sample of 20% of the main drains within the scope of License Renewal annually in order to check for potential flow blockage in system risers. During annual testing, one of the tests is performed in a radiologically controlled area.
Valves and System-Wide Testing: Deluge Valves	13.4.3.2.2 to 13.4.3.2.5	A full flow test using air or water is performed every refueling outage by trip testing each deluge valve to verify that spray/ sprinkler nozzles are unobstructed.

TABLE 19.1-1 (Sheet 2)

Component	NFPA 25 Section	Aging Management Performed
Water Spray Fixed Systems: Strainers	10.2.1.6, 10.2.1.7, 10.2.7	Spray system strainers are inspected and cleaned every refueling outage and after each system actuation. Callaway does not have main line strainers.
Water Spray Fixed Systems: Operation Test	10.3.4.3	A full flow test is performed every refueling cycle using air or water to verify that spray/sprinkler nozzles are unobstructed.
Foam Water Sprinkler Systems: Strainers	11.2.7.1	Not applicable. Callaway does not have a foam water sprinkler system.
Foam Water Sprinkler Systems: Operational Test Discharge Patterns	11.3.2.6	Not applicable. Callaway does not have a foam water sprinkler system.
Foam Water Sprinkler Systems: Storage tanks	Visual inspection for internal corrosion	Not applicable. Callaway does not have a foam water sprinkler system.
Obstruction Investigation: Obstruction, Internal Inspection of Piping	14.2 and 14.3	Wet pipe suppression systems are inspected every five years. For buildings containing multiple systems, half are inspected in the first five year interval, and the remaining half inspected in the next five year interval. If sufficient foreign material is found in any system in a building, then all systems in the building will be inspected. Dry pipe preaction systems will be inspected following actuation, prior to return to service. If sufficient foreign material is found to obstruct pipe or sprinklers, then an obstruction investigation is conducted. If the visual inspection detects surface irregularities that could indicate wall loss below nominal pipe wall thickness, then follow-up volumetric examinations will be performed.

19.2 SUMMARY DESCRIPTIONS OF TIME-LIMITED AGING ANALYSIS AGING MANAGEMENT PROGRAMS

19.2.1 FATIGUE MONITORING

The Fatigue Monitoring program manages fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The program ensures that actual plant experience remains bounded by the transients analyzed in the design calculations and fatigue crack growth analyses, or that corrective actions maintain the design and licensing basis. The Fatigue Monitoring program tracks the number of transient cycles and will track cumulative fatigue usage at monitored locations. The Fatigue Monitoring program tracks fatigue by one of the following methods:

1. The Cycle Counting (CC) monitoring method tracks transient event cycles affecting the location to ensure that the numbers of transient events analyzed by the fatigue analyses are not exceeded. This method does not calculate cumulative usage factors (CUFs).

2. The Cycle-Based Fatigue (CBF) monitoring method utilizes the CC results and stress intensity ranges generated with the ASME III methods that use three dimensional six component stress-tensor methods to perform CUF calculations for a given location. The fatigue accumulation is tracked to determine approach to the ASME allowable fatigue limit of 1.0.

3. The Stress-Based Fatigue (SBF) monitoring method computes a "real time" stress history for a given component from data collected from plant instruments to calculate transient pressure and temperature, and the corresponding stress history at the critical location in the component. The stress history is analyzed to identify stress cycles, and then a CUF is computed. The CUF will be calculated using a three dimensional, six component stress tensor method meeting ASME III NB-3200 requirements, or a method will be benchmarked consistent with the NRC Regulatory Issue Summary (RIS) 2008-30.

The program will also consider the effects of the reactor water environment for a set that includes the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant, plant-specific bounding EAF locations in the reactor coolant pressure boundary, and reactor vessel internals locations with fatigue usage calculations. F_{en} factors will be determined as described in Section 19.3.2.3.

If a cycle count or cumulative usage factor value increases to a program action limit, corrective actions include fatigue reanalysis, repair, or replacement. Action limits permit completion of corrective actions before the design limit is exceeded. The sentinel location analysis, when refined, will be revisited to confirm bounding Reactor Coolant Pressure Boundary Environmentally Assisted Fatigue susceptible sentinel locations are updated appropriately and remain consistent with the refined analysis.

19.2.2 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS

The Environmental Qualification (EQ) of Electrical Components program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. This program also manages the aging of mechanical EQ components. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished or replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

The Environmental Qualification (EQ) of Electrical Components program is consistent with the guidance of 10 CFR 50.49, NUREG-0588 Category I, and Regulatory Guide 1.89, “Qualification of Class 1E Equipment for Nuclear Power Plants,” Revision 0 for maintaining qualifications of equipment.

Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria and corrective actions (if acceptance criteria are not met).

19.2.3 CONCRETE CONTAINMENT TENDON PRESTRESS

The Concrete Containment Tendon Prestress program, which is part of the ASME Section XI, Subsection IWL Inservice Inspection program, manages loss of tendon prestress consistent with the requirements of 10 CFR 50.55a(b)(2)(viii)(B). The Concrete Containment Tendon Prestress program includes inspection procedures and acceptance criteria and prescribes specific corrective actions, including increased inspection scope, if inspection criteria are not met.

19.3 EVALUATION SUMMARIES OF TIME-LIMITED AGING ANALYSES

10 CFR 54.21(c) requires that an applicant for a renewed license identify time-limited aging analyses (TLAAs) and evaluate them for the period of extended operation. The following TLAAs have been identified and evaluated for Callaway.

19.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The following calculations of neutron fluence and of its embrittlement effects are TLAAs affected by the extended life of the plant:

- Neutron Fluence Values
- Charpy Upper-Shelf Energy (C_v USE)
- Pressurized Thermal Shock (PTS)
- Pressure-Temperature (P-T) Limits
- Low Temperature Overpressure Protection (LTOP)

The Reactor Vessel Surveillance program is described in Section 19.1.17.

19.3.1.1 Neutron Fluence Values

The End of Life - Extended (EOLE) fluence projections were revised to quantify those materials with an expected fluence greater than 1 x 10^{17} n/cm^2 (E>1.0 MeV) at the end of the period of extended operation using methodologies that follow the guidance of Regulatory Guide 1.190. Therefore this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.3.1.2 Charpy Upper-Shelf Energy (USE)

The projections demonstrated that the C_v USE in the limiting material will remain above the 10 CFR 50 Appendix G acceptance criteria of 50 ft-lbf. The C_v USE values were projected to the end of the period of extended operation. Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.3.1.3 Pressurized Thermal Shock

The revised projections of PTS reference temperature (RT_{PTS}) to the end of the 60-year licensed operating period meet the requirements of 10 CFR 50.61. Therefore, the evaluation of the pressurized thermal shock screening parameter is projected to the end of the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.3.1.4 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).

19.3.1.5 Low Temperature Overpressure Protection

The low temperature overpressure protection (cold overpressure mitigation system (COMS)) setpoints are established in the Callaway Plant Pressure and Temperature Limits Report and managed consistent with the P-T curves, which will be managed through the period of extended operation. Therefore the COMS setpoints will be managed through the period of extended operation. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2 METAL FATIGUE

ASME III Class 1 design specifications define a set of static and transient load conditions for which components are to be designed. The initial Callaway operating license was for 40 years. The Callaway design specifications state that the transient conditions are for a 40-year design life. However, the fatigue analyses are based on a specified number of occurrences of each transient rather than on the design or licensed life. The design number of occurrences of each transient for use in the fatigue analyses was specified to be larger than the number of occurrences now expected during the 40-year design life of the plant.

Operating experience at Callaway and at other similar units has demonstrated that the assumed frequencies of design transients, and therefore the number and severity of transient cycles assumed for a 40-year life were conservative, and that with few exceptions the design numbers are not expected to be exceeded during a 60-year life.

19.3.2.1 ASME Section III Class 1Fatigue Analysis of Vessels, Piping and Components

The following lists all vessels, pumps, and components subject to Class 1 analyses.

- Reactor Pressure Vessel, Nozzles, Head, Head Adapter Plugs, and Studs
- Control Rod Drive Mechanisms and Core Exit Thermocouple Nozzle Assembly
- Reactor Coolant Pumps
- Pressurizer and Pressurizer Nozzles
- Steam Generators, and Feedwater Nozzles
- ASME III Class 1 Valves
- ASME III Class 1 Piping and Piping Nozzles

The Fatigue Monitoring program described in Section 19.2.1 will track the numbers of events and transient severities. Therefore the analyses will be managed for the period of extended operation, and the TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Sections 19.3.2.1.1 through 19.3.2.1.4 include topics which required additional consideration when evaluating the ability of the Fatigue Monitoring program to manage fatigue and fatigue analyses which were not dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.1.1 Reactor Coolant Pump Thermal Barrier Flange

Even though the fatigue waiver conditions are satisfied for the pump, a cumulative usage factor was calculated as part of simplified elastic-plastic analyses for the thermal barrier flange at component cooling water connection. With the exception of the seasonal temperature change transient, the transients used in the fatigue analysis of the thermal barrier flange at the component cooling water connection will be tracked by the Fatigue Monitoring program, summarized in Section 19.2.1.

To account for the increase in usage caused by 20 additional years of operation associated with the seasonal temperature change transient in the RCP thermal barrier

flange fatigue analysis and to maintain the usage below the Code allowable of 1.0, the elevated CCW injection temperature transient will be limited to 75% of its design value, i.e. limited to 150 transients.

Therefore the fatigue analysis will be managed for the period of extended operation, and the TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.1.2 Pressurizer Insurge-Outsurge Transients

The thermal transients resulting from a reactor coolant insurge-outsurge during normal heatup and cooldown operations were not considered in the original design analyses of the pressurizer. The fatigue analyses have been revised to incorporate the effect of insurge-outsurge transients on the pressurizer lower head, surge nozzle, and heater well nozzles at plant specific conditions. The limiting locations for the pressurizer affected by the insurge-outsurge transient are managed as sentinel locations. Fatigue effects of components associated with the pressurizer insurge-outsurge transients including the effects of the reactor coolant environment on fatigue usage factors will be managed for the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.1.3 Steam Generator ASME Section III, Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses

Although the ASME classification for the secondary side of the replacement steam generators (RSGs) is specified to be Class 2, all pressure retaining parts of the steam generator, and thus both the primary and secondary pressure boundaries, are designed to satisfy the criteria specified in Section III of the ASME Code for Class 1 components.

The RSGs were installed during Refueling Outage 14 (Fall 2005). The Class 1 fatigue analyses of the RSG components used the design basis numbers of events assumed for a 40-year design life. The RSG design lives end in 2045 which extends beyond the period of extended operation. Therefore, the corresponding TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.2.1.4 NRC Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification

The analysis of thermal stratification effects at Callaway evaluated the hot-leg surge line nozzles and the pressurizer surge line. The Fatigue Monitoring program, summarized in Section 19.2.1, tracks events to ensure that a design basis number of events are not exceeded. The effects of fatigue on the pressurizer surge line will therefore be managed for the period of extended operation, and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.1.5 Replacement Reactor Vessel Closure Head, Control Rod Drive Mechanisms, and Core Exit Thermal Nozzle Assemblies

The replacement reactor vessel closure head, control rod drive mechanisms, and core exit thermal nozzle assemblies were replaced in Refueling Outage 20 (Fall 2014). The Class 1 fatigue analyses of the components used the design basis numbers of events assumed and cover a 45-year design life. The design lives end in 2059, which extends beyond the period of extended operation. Therefore, the corresponding TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.2.2 ASME Section III Subsection NG Fatigue Analysis of Reactor Vessel Internals

The reactor vessel internals analyses of record for Callaway were verified to continue to satisfy the ASME Subsection NG requirements. The fatigue analyses were performed using the 40-year design transients in FSAR Table 3.9(N)-1 SP. The Fatigue Monitoring program, summarized in Section 19.2.1, will track the number of events to manage the fatigue analysis of the reactor internals. Therefore, fatigue in the reactor vessel internals will be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.3 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

All of the locations specified in NUREG/CR-6260 for newer vintage Westinghouse plants will be monitored by the Fatigue Monitoring program, described in Section 19.2.1. If any of the analyzed CUF values for these locations exceeds the fatigue design limit, the analyses may be revised using actual plant transients experienced. Callaway has completed an evaluation to identify any additional plant-specific bounding EAF locations. The effects of the reactor coolant environment on fatigue usage factors in the NUREG/CR-6260 and plant-specific bounding EAF locations in the reactor coolant pressure boundary will be managed for the period of extended operation. The supporting environmental factors, F_{en}, calculations will be performed with NUREG/CR-6909 or NUREG/CR-6583 for carbon and low alloy steels, NUREG/CR-6909 or NUREG/CR-5704 for austenitic stainless steels, and NUREG/CR-6909 for nickel alloys. In addition, reactor vessel internals locations with fatigue usage calculations will be evaluated for the effects of the reactor water environment. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.2.4 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME Section III Class 2 and 3 Piping

The existing analyses of ANSI B31.1 or ASME Section III Class 2 and 3 piping for which the allowable range of secondary stresses depends on the number of assumed thermal cycles and that are within the scope of license renewal are valid for the period of

extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.2.5 Fatigue Design of Spent Fuel Pool Liner and Racks for Seismic Events

The spent fuel pool racks and liner were replaced in 1999 and were analyzed for fatigue effects of 1 safe-shutdown earthquake (SSE) and 20 operating basis earthquakes (OBE) using methods similar to those for ASME Section III Class 1 analyses. No OBE events have occurred in the operating history of the plant to date, so that the design basis number of events remains sufficient for the remainder of the original licensed operating period, plus the 20-year licensed operating period extension, and the replacement racks are therefore presently qualified for the number of these events now expected for the remainder of a 60-year life. Therefore the analyses are valid for the period of extended operation, and the TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.2.6 Fatigue Design and Analysis of Class 1E Electrical Raceway Support Angle Fittings for Seismic Events

The design of Class 1E electrical raceway included a fatigue evaluation of the effects of operating basis and safe shutdown earthquake loads (OBE and SSE loads). The analysis assumes a total of 750 cycles for the five OBE events plus 150 for the single SSE. No OBE events have occurred in the operating history of the plant to date. The design basis number of events therefore remains sufficient for the remainder of the original licensed operating period, plus the 20-year licensed operating period extension, and the Class 1E electrical raceway support angle fittings are therefore presently qualified for the number of these events now expected for the remainder of a 60-year life. Therefore, the analysis is valid for the period of extended operation, and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.2.7 Fatigue Analyses of Class 2 Heat Exchangers

Regenerative Heat Exchanger

Thermal fatigue analyses, in accordance with ASME III, NB-3222.4, were performed to qualify the regenerative heat exchanger to the Code design requirements. The fatigue analysis evaluated the tube side inlet and outlet tubesheets, the shell side nozzles, the tube side nozzles, and the cross shell juncture. The transients in the analysis will be counted by the Fatigue Monitoring program, described in Section 19.2.1. Therefore the fatigue analysis of the regenerative heat exchanger will be managed for the period of extended operation, and this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Letdown Heat Exchanger

The fatigue analysis of the letdown heat exchanger evaluated the flange, tubesheet, tube side nozzles, and studs. The transients significant to fatigue are monitored; or are not

projected to be approached during a 60-year plant life; or will not challenge the Code allowable CUF. Therefore these CUFs are projected through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

Letdown Reheat Heat Exchanger

The fatigue analysis of the letdown reheat heat exchanger evaluated the shell and tube side nozzles, the tubesheet, and the studs. The transients significant to fatigue are monitored; or are not projected to be approached during a 60-year plant life; or will not challenge the Code allowable CUF. Therefore these CUFs are projected through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

Residual Heat Removal Heat Exchangers

The Callaway residual heat removal (RHR) heat exchangers satisfied the fatigue waiver criteria of NB-3222.4(d). The transients in the waiver will be counted by the Fatigue Monitoring program, described in Section 19.2.1. Therefore the fatigue analysis of the RHR heat exchanger will be managed for the period of extended operation, and this TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Excess Letdown Heat Exchanger

The number of excess letdown initiation transient events anticipated for 60 years is less than the Code maximum allowable. Therefore the excess letdown heat exchanger CUF is projected through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.3.3 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

10 CFR 50.49 requires that certain electrical and instrument and control equipment, important to safety, located in harsh environments, be qualified to perform their safety-related functions in those harsh environments after the effects of in-service aging.

The Callaway Environmental Qualification (EQ) of Electric Components program is consistent with the guidance of NUREG-0588, Category I, and the requirements of 10 CFR 50.49. The program outlines the methodology for performing activities required to establish, maintain, and document the environmental qualification of electrical equipment important to safety. The current list of equipment requiring environmental qualification is maintained in accordance with plant procedures and the Equipment Qualification Management System (EQMS). Safety-related electrical equipment and components located in a harsh environment are qualified by test or combination of test and analysis in accordance with the requirements of 10 CFR 50.49 and NUREG-0588 Revision 1. Detailed qualification results for electrical equipment located in a harsh environment are maintained in the Equipment Qualification Data Package (EQDP).

The Environmental Qualification (EQ) of Electric Components program, summarized in Section 19.2.2, ensures that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. Aging effects addressed by the EQ program will therefore be managed for the period of extended operation, and the TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.4 CONCRETE CONTAINMENT TENDON PRESTRESS

Predicted Lower Limit

Predicted lower limit (PLL) force lines are incorporated in the Concrete Containment Tendon Prestress program to identify any abnormal degradation in tendon prestressing force. The calculations of predicted lower limit lines are consistent with NRC Regulatory Guide 1.35.1. Actual measured values for each tendon are compared to their respective PLL values, with acceptance criteria consistent with ASME Section XI, Subsection IWL requirements. Therefore, loss of tendon prestress will be managed through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

Regression Analysis

The trend lines are calculated by regression of individual tendon lift off data and are consistent with NRC Information Notice 99-10, “Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments,” Revision 1, Attachment 3. The regression analysis trend lines indicate lift-offs in excess of the MRV for at least 60 years; therefore the analysis is valid for the period of extended operation. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.5 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSES

The Callaway prestressed concrete containment vessel is poured against a steel membrane liner. No credit is taken for the liner for the pressure design of the containment vessel, but the liner and penetrations ensure the vessel is leak tight. These components are designed to ASME Section III requirements for metal containment components.

19.3.5.1 Design Cycles for the Main Steam Line and Feedwater Penetrations

Loading Condition IV

The analysis of Loading Condition IV (normal thermal gradient plus pipe rupture) for the main steam line and feedwater penetrations assumed 10 cycles of line ruptures which

will not be exceeded during the period of extended operation. Therefore the analysis will remain valid for the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Loading Condition V

The analysis of Loading Condition V (normal thermal gradient plus operating cycle) for the main steam line penetrations accounts for 500 startup-shutdown cycles which exceeds the number projected for 60 years, therefore the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.3.5.2 Fatigue Waiver Evaluations for the Access Hatches and Leak Chase Channels

Access Hatches

The number of pressure tests and startup and cooldown cycles assumed in the access hatch fatigue waiver evaluations will not be exceeded during the period of extended operation; therefore the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Leak Chase Channels

The numbers of pressure tests and thermal cycles assumed in the leak chase fatigue waiver evaluation will not be exceeded during the period of extended operation; therefore the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6 PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

19.3.6.1 Containment Polar Crane, Fuel Building Cask Handling Crane, Spent Fuel Pool Bridge Crane, and Refueling Machine CMAA 70 Load Cycle Limits

The design standard number of full-capacity lifts, even with a significant number of unforeseen lifts, far exceeds the number expected of each machine for a 60-year life. The lifting machine designs therefore remain valid for the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.2 In-Service Flaw Growth Analyses that Demonstrate Structural Stability for 40 Years

Cold Leg Elbow-to-Safe End Weld Flaw Indications

The fatigue crack growth analysis for the Cold Leg Elbow-to-Safe End Weld Flaw Indications assumes the design number of transients. The projected transient accumulations show that the numbers of transient cycles are expected to remain within the assumed numbers and therefore the analyses are valid through the period of

extended operation. This TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Pressurizer SWOL Fatigue Crack Growth Analysis

The pressurizer nozzle structural weld overlays, performed in 2007, depend on 40-year fatigue crack growth analyses, which will remain valid until 2047. The projected transient accumulations are expected to remain within the assumed numbers and therefore the analyses are valid through the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.3 Corrosion Analysis of the Reactor Vessel Cladding Indications

Two areas were identified during Refuel 13 (Spring 2004) and Refuel 15 (Spring 2007) where the reactor pressure vessel low-alloy steel has been left exposed to the reactor coolant. The evaluation demonstrated that reactor vessel design minimum wall thickness will be maintained after corrosion is considered. The evaluation considered a plant life of 40 years, which includes 20 years under the current license plus 20 years for the period of extended operation. Periodic inspection of the reactor vessel indications and reconciliation of the results with the corrosion analysis ensures the analytical bases of the analysis are maintained; therefore, the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.6.4 Reactor Vessel Underclad Cracking Analyses

Reactor Vessel Underclad Cracking been addressed in the Callaway vessel by weld cladding processes designed to avoid these defects, consistent with Regulatory Guide 1.43. In addition WCAP-15338-A found that the maximum flaw predicted by the crack growth analysis is less than the Section XI allowable flaw size. These WCAP-15338-A analyses assumed 1.5 times the numbers of cyclic and transient loads assumed for the original 40 year life and bound the numbers of cycles projected in 60 years. This TLAA is disposition in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.5 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis

Fatigue in the reactor coolant pump flywheels is supported by a fatigue crack growth analysis which demonstrates that 6,000 start-stop cycles (over an assumed 60 year life) will produce an acceptable extension of the crack. The evaluation is based on the 60-year operating period, therefore the TLAA extends to the end of the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.6 High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factors

The selection of ASME III, Class 1 piping High Energy Line Break (HELB) locations depends on usage factors, which will remain valid as long as the assumed numbers of cycles are not exceeded. The Fatigue Monitoring program, summarized in Appendix B, Section 19.2.1, ensures that the analytical bases of the HELB locations are maintained or that a HELB analysis for the new locations with a CUF greater than 0.1 is performed. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.6.7 Fatigue Crack Growth Assessment in Support of a Fracture Mechanics Analysis for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Piping Failures

Reactor Coolant Loops

The fatigue crack growth analysis associated with the leak-before-break analyses depend on design transient cycle assumptions, and will remain valid as long as the assumed numbers of cycles are not exceeded. The projected transient accumulations show that the numbers of transient cycles are expected to remain within the assumed numbers and therefore the analyses will remain valid for the period of extended operation. Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Accumulator Injection and Residual Heat Removal Lines

These analyses are based on assumed 40 year design transients. The projected transient accumulations are expected to remain within the assumed numbers and therefore the analyses will remain valid for the period of extended operation. Therefore, these TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.8 Replacement Class 3 Buried Piping

The replacement of buried Essential Service Water (ESW) piping with high-density polyethylene (HDPE) material began in 2008 with a service life of 40 years, which extends beyond the period of extended operation. Therefore the design of buried HDPE ESW piping will remain valid for the period of extended operation, and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.3.6.9 Replacement Steam Generator Tube Wear

The replacement steam generator tube wear analysis determined the maximum wear for a 45-year design life would remain below the maximum allowable wear of 40% of the tube wall thickness. The steam generator aging management program justifies the continued operation of all wear indications through the next scheduled inspection by establishing a wear rate based on measured data to ensure tube wear does not exceed

40% of the through-wall thickness. Therefore, this TLAA will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.6.10 Mechanical Environmental Qualification

The Mechanical Environmental Qualification (MEQ) program establishes qualified lives for safety-related mechanical components located in harsh environments in accordance with the provisions of Criterion 4 of Appendix A to 10 CFR Part 50. Mechanical Environmental Qualifications (MEQ) extend beyond 40 years and are TLAAs. The Environmental Qualification (EQ) of Electric Components program includes MEQ components. Therefore the aging effects on the MEQ components will be managed for the period of extended operation, and the TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.3.6.11 Break Exception for HDPE Piping

The break exception criterion described in FSAR-SP Section 3.6.2.1.2.4 and FSAR-SP Table 3.6-2, which requires the sum of the Service Level B Longitudinal Stress Equation and the Alternate Thermal Expansion or Contraction Evaluation to be less than 0.4 (1.2S + 1100), assumes an allowable stress that is based on 50 years. The replacement of buried Essential Service Water (ESW) piping with high-density polyethylene (HDPE) material began in 2008. This extends the exception beyond the period of extended operation; therefore, the TLAA will remain valid for the period of extended operation, and is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.4 LICENSE RENEWAL COMMITMENTS

Table 19.4-1, License Renewal Commitments, identifies actions committed to by Ameren Missouri for the Callaway Plant Unit 1 in its License Renewal Application.

TABLE 19.4-1 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
1	Procedures will be enhanced to apply the elements of corrective actions, confirmation process, and administrative controls of the Callaway Plant Quality Assurance Program to those nonsafety-related structures, systems, and components (SSCs) requiring aging management. (Completed LRA Amendment 23 dated April 26, 2013)	[19.1] B1.3	Completed	Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-05979 dated April 26, 2013
2	Enhance the station operating experience review process and Corrective Action Program (CAP) to perform reviews of plant-specific and industry operating experience to confirm the effectiveness of the license renewal aging management programs (AMPs), to determine the need for AMPs to be enhanced, or indicate the need to develop a new AMP. In order to provide additional assurance that internal and external operating experience related to aging management continues to be used effectively in the AMPs, Callaway will enhance its operating experience program to: • Explicitly require the review of operating experience for age-related degradation. (Completed LRA Amendment 18 dated December 19, 2012) • Establish criteria to define age-related degradation. In general, the criteria will be used to identify aging that is considered excessive relative to design, previous inspection experience, and inspection intervals. (Completed LRA Amendment 18 dated December 19, 2012) • Establish coding for use in identification, trending and communications of age-related degradation. This coding will assist plant personnel in ensuring that, in addition to addressing the specific issue, the adequacy of existing aging management programs is assessed. This could lead to AMP revisions or the establishment of new AMPs, as appropriate. (Completed LRA Amendment 18 dated December 19, 2012)	[19.1] B1.4	Completed	Letter ULNRC-05939 dated December 19, 2012 Letter ULNRC-05957 dated February 14, 2013 Letter ULNRC-05979 dated April 26, 2013 [Letter ULNRC-05903 dated September 6, 2012]

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TABLE 19.4-1 (Sheet 2)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
2(cont'd)	• Require communication of significant internal age-related degradation, associated with SSCs in the scope of license renewal, to the industry. Criteria will be established for determining when aging-related degradation is significant. (Completed LRA Amendment 18 dated December 19, 2012)			
	• Require review of external operating experience for information related to aging management, and evaluation of such information for potential improvements to Callaway aging management activities. License Renewal Interim Staff Guidance (LR-ISG) documents will be reviewed as part of this external operating experience information as they are issued on an ongoing basis, capturing new insights or addressing issues that emerge from license renewal reviews. (Completed LRA Amendment 21 dated February 14, 2013)			
	• Provide training to those responsible for screening, evaluating and communicating operating experience items related to aging-related degradation. This training will be commensurate with their role in the process. (Completed LRA Amendment 23 dated April 26, 2013)			
	• Explicitly require AMP activities, criteria, and evaluations integral to the elements of the AMPs be included in the operating experience evaluation. (Completed LRA Amendment 21 dated February 14, 2013)			
3	Enhance the Boric Acid Corrosion Program procedures to: include steel, copper alloy greater than 15% zinc, and aluminum as materials that are susceptible to boric acid corrosion. (Completed LRA Amendment 13 dated October 24, 2012)	[19.1.4] B2.1.4	Completed	Letter ULNRC-05920 dated October 24, 2012
	• Ensure that system engineers will observe for signs of boric acid residue when performing system walkdowns. (Completed LRA Amendment 13 dated October 24, 2012)			Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 3)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
3(cont'd)	• Specify that the corrective actions taken by the program will include a consideration to modify the present design or operating procedures to mitigate or prevent recurrence of aging effects caused by borated water leakage. Consideration will be given to modifications that (a) reduce the probability of primary coolant leaks at locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings or claddings. (Completed LRA Amendment 13 dated October 24, 2012)			
4	Implement the PWR Vessel Internals Program as described in LRA Section B2.1.6. As part of the implementation activities address the following Applicant/Licensee Action Items (A/LAI) of NRC MRP-227-A Safety Evaluation dated December 16, 2011. (Completed LRA Amendment 28 dated December 20, 2013) Applicant/Licensee Action Item (A/LAI) #1 Each applicant or licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of Materials Reliability Program (MRP)-227 is applicable to the facility. Each applicant or licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the failure modes, effects, and criticality analysis and functionality analyses for reactors of their design (i.e., Westinghouse, Combustion Engineering, or Babcock and Wilcox) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their reactor vessel internals (RVI) components or plant operating conditions, which result in different component inspection categories. The applicant or licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. (Completed LRA Amendment 28 dated December 20, 2013)	[19.1.6] B2.1.6	Completed	LRA Letter ULNRC-05950 dated January 24, 2013 Letter ULNRC-06050 dated October 17, 2013 Letter ULNRC-06057 dated December 20, 2013 Letter ULNRC-06080 dated February 14, 2014 Letter ULNRC-06106 dated March 28, 2014

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TABLE 19.4-1 (Sheet 4)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
5	Enhance the Bolting Integrity Program procedures to: • Reference NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants," and Electric Power Research Institute (EPRI) NP-5769 to meet the GALL Report recommendations. (Completed LRA Amendment 1 dated April 25, 2012) • Include bolting in the list of items to be inspected during walkdowns. (Completed LRA Amendment 15 dated November 8, 2012) • Include a visual inspection of a sample of submerged bolting heads in raw water and waste water environments every four refueling outages (6 years) when the pumps are dewatered. In addition, when submerged raw water and waste water pump casings are disassembled during maintenance activities, the bolting threads will be opportunistically inspected. A sample of submerged bolting on the fuel oil storage tank transfer pumps will be visually inspected every 10 years when the pumps are disassembled during maintenance activities. The sample for submerged bolting will be 20% of the population with a maximum of 25 for each environment. The inspection of submerged bolting will focus on the bounding or lead components most susceptible to aging due to time in service and severity of operating conditions.	[19.1.8] B2.1.8	Portions completed are as shown. Remaining portions to be implemented as follows: Completed no later than 6 months prior to the period of extended operation. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05856 dated April 25, 2012 Letter ULNRC-05928 dated November 8, 2012 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06121 dated June 5, 2014
6	Enhance the Open-Cycle Cooling Water System Program procedures to: • Include polymeric material inspection requirements, parameters monitored, and acceptance criteria. Examination of polymeric materials by the Open-Cycle Cooling Water System Program will be consistent with examinations described in the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program. • Include inspection and cleaning, if necessary, of the air-side of safety-related air-to-water heat exchangers cooled by the essential service water (ESW) system.	[19.1.10] B2.1.10	Completed no later than 6 months prior to the period of extended operation. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05892 dated August 21, 2012 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06057 dated December 20, 2013

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TABLE 19.4-1 (Sheet 5)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
6(cont'd)	• Prior to the period of extended operation, an inspection technique will be selected from available technologies to identify internal pipe wall degradation due to MIC for performance of a one-time inspection of a buried carbon steel piping segment that is representative of other accessible carbon steel ESW piping segments.			Letter ULNRC-06117 dated April 23, 2014 Letter ULNRC-06121 dated June 5, 2014
	• Procedures will be enhanced to perform periodic visual inspections on all accessible internal surface coatings of the component cooling water heat exchangers, Class IE electrical equipment air conditioners, control room air conditioners, and essential service water self-cleaning strainers. Baseline inspections will be conducted in the 10-year period prior to the period of extended operation on the accessible internal surfaces coatings of the in-scope components. Coatings are inspected every 6 years on an alternating train basis based on no observed degradation or cracking and flaking that has been evaluated as acceptable; and the component is not subject to turbulent flow. Baseline inspections may be used to demonstrate that long-term coatings are or are not subject to turbulent flow conditions that could result in mechanical damage to the coating. Coatings with blisters, peeling, delaminations, or rusting that has been determined not to require remediation are inspected on a 4-year frequency. For peeling, delaminations, and blisters determined to not meet the acceptance criteria and that will not be repaired or replaced, physical testing is performed where physically possible (i.e., sufficient room to conduct testing). Testing consists of destructive or nondestructive adhesion testing using ASTM International Standards endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." Monitoring and trending of coatings is based on a review of the previous two inspections' results (including repairs) with the current inspection results. The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the staff associated with a particular standard.			

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TABLE 19.4-1 (Sheet 6)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
6(cont'd)	Coating acceptance criteria are as follows: • Indications of peeling and delamination are not acceptable and the coatings are repaired or replaced. Blisters are evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54 including staff guidance associated with use of a particular standard. • Indications such as cracking, flaking, and rusting are to be evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54 including staff guidance associated with use of a particular standard. • Adhesion testing results meet or exceed the degree of adhesion recommended in engineering documents specific to the coating and substrate. Inspection results not meeting the acceptance criteria will be evaluated by a qualified coatings evaluator. Corrective actions will be determined using the Corrective Action Program.			

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TABLE 19.4-1 (Sheet 7)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
7	Enhance the Closed Treated Water Systems Program procedures to: • Include visual inspections of the surfaces of components with a closed treated water systems water environment. Representative samples of each combination of material and water treatment program will be visually inspected at least every 10 years or opportunistically when consistent with sample requirements. Sample locations will be selected based on the likelihood of corrosion and cracking. Inspections will be conducted and evaluated consistent with ASME Code inspections, industry standards, or a plant-specific inspection procedure by personnel qualified to detect aging. If adverse conditions are found, additional examinations will be performed. This periodic inspection will determine the extent of cracking, loss of material and fouling, and serves as a leading indicator of the condition of the interior of piping components otherwise inaccessible for visual inspection.	[19.1.11] B2.1.11	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05886 dated August 6, 2012 Letter ULNRC-05963 dated February 28, 2013
8	Enhance the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program procedures to: • Inspect crane structural members for loss of material due to corrosion and rail wear, and loss of preload due to loose or missing bolts and nuts. • Include performance of periodic inspections as defined in the appropriate ASME B30 series standard for all cranes, hoists, and equipment handling systems within the scope of license renewal. For handling systems that are infrequently in service, such as those only used during refueling outages, periodic inspections may be deferred until just prior to use. • Require evaluation of loss of material due to wear or corrosion and loss of bolting preload per the appropriate ASME B30 series standard. • Require repairs to cranes, hoists, and equipment handling systems per the appropriate ASME B30 series standard.	[19.1.12] B2.1.12	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 8)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
9	Enhance the Fire Protection Program procedures to: • Include visual inspections of the external surfaces of halon fire suppression system components for excessive loss of material due to corrosion. (Completed LRA Amendment 28 dated December 20, 2013) • Include trending of the performance of the halon system during testing. (Completed LRA Amendment 1 dated April 25, 2012)	[19.1.13] B2.1.13	Completed	Letter ULNRC-05856 dated April 25, 2012 Letter ULNRC-06057 dated December 20, 2013
10	Recoat the internal surface of fire water storage tanks. Enhance the Fire Water System Program procedures to: • Perform internal inspections on accessible exposed portions of fire water piping during plant maintenance activities. When visual inspections are used to detect loss of material, the inspection technique is capable of detecting surface irregularities that could indicate wall loss to below nominal pipe wall thickness due to corrosion and corrosion product deposition. Where such irregularities are detected, followup volumetric wall thickness examinations are performed. • Replace sprinkler heads prior to 50 years in service, or have a recognized testing laboratory field-service test a representative sample in accordance with National Fire Protection Association (NFPA) 25 and test additional samples every 10 years thereafter to ensure signs of aging are detected in a timely manner. • Review and evaluate trends in flow parameters recorded during the NFPA 25 fire water flow tests. • Perform annual hydrant flow testing in accordance with NFPA 25. • Perform annual hydrostatic testing of fire brigade hose.	[19.1.14] B2.1.14	Implementation is started 5 years before the period of extended operation. Recoat the internal surface of the fire water storage tanks and inspections of wetted segments that cannot be drained or that allow water to collect to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later. The program's remaining inspections begin during the period of extended operation.	Letter ULNRC-05923 dated October 31,2012 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06057 dated December 20, 2013 Letter ULNRC-05897 dated August 16, 2012 Letter ULNRC-06117 dated April 23, 2014 Letter ULNRC-06118 dated May 6, 2014 Letter ULNRC-06129 dated July 31, 2014

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TABLE 19.4-1 (Sheet 9)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
10(cont'd)	• Enhance the Fire Water System program to include nonintrusive pipe wall thickness examinations. Wall thickness measurements will be performed on fire water piping every 3 years. Each 3-year sample will include at least three locations for a total of 100 feet of aboveground fire water piping and will be selected based on system susceptibility to corrosion or fouling and evidence of performance degradation during system flow testing or periodic flushes. Pipe wall thickness examinations and internal inspections will be performed commencing after 2014 and throughout the period of extended operation. • Perform augmented tests and inspections of water-based fire protection system components that have been wetted but are normally dry. The augmented tests and inspections are conducted as follows on piping segments that cannot be drained or that allow water to collect: • In each 5-year interval, beginning 5 years prior to the period of extended operation, either conduct a flow test or flush sufficient to detect potential flow blockage, or conduct a visual inspection of 100% of the internal surface of piping segments that cannot be drained or allow water to collect. • A 100% baseline inspection will be performed prior to the period of extended operation. In each 5-year interval of the period of extended operation, 20% of the length of piping segments that cannot be drained or that allow water to collect is subject to volumetric wall thickness inspections. Measurement points will be obtained to the extent that each potential degraded condition can be identified (e.g., general corrosion, MIC). The 20% of piping that is inspected in each 5-year interval will be in different locations than previously inspected piping. If the results of a 100% internal visual inspection are acceptable, and the segment is not subsequently wetted, no further augmented tests or inspections will be performed.			

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TABLE 19.4-1 (Sheet 10)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
10(cont'd)	• Require the inspection of the interior of the fire water storage tanks to include checking for evidence of voids beneath the floor. (Completed LRA Amendment 38, July 31, 2014)			
	• Change the frequency of trip testing each deluge valve from every 3 years to every refueling outage.			
	• Change the frequency of tests of spray/sprinkler nozzle discharge patterns from every 3 years to every refueling outage.			
	• Perform the following additional inspections if pitting, corrosion, or coating failure is found during the inspection of the fire water storage tanks: (1) tank coatings are evaluated using an adhesion test consistent with ASTM D 3359, Standard Test Methods for Measuring Adhesion by Tape Test; (2) dry film thickness measurements are taken at random locations to determine the overall coating thickness; (3) nondestructive ultrasonic readings are taken to evaluate the wall thickness where there is evidence of pitting or corrosion; (4) interior surfaces are spot wet-sponge tested to detect pinholes, cracks, or other compromises in the coating; (5) tank bottoms are tested for metal loss on the underside by use of ultrasonic testing where there is evidence of pitting or corrosion; (6) bottom seams are vacuum-box tested in accordance with NFPA 22, Standard for Water Tanks for Private Fire Protection.			
	• Require the removal of foreign material if its presence is found during pipe inspections to obstruct pipe or sprinklers. In addition, the source of the material is determined and corrected.			
	• Perform main drain tests consistent with NFPA 25, Section 13.2.5, of a representative sample of 20% of the main drains within the scope of license renewal annually in order to check for potential flow blockage in system risers. During annual testing, one of the tests is performed in a radiologically controlled area.			

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TABLE 19.4-1 (Sheet 11)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source

10(cont'd)	• Inspect wet pipe suppression systems every 5 years consistent with NFPA 25, Section 14.2. For buildings containing multiple systems, half are inspected in the first 5-year interval and the remaining half inspected in the next 5-year interval. If sufficient foreign material is found in any system in a building, then all systems in the building will be inspected. The NFPA 25, Section 14.2, inspection of dry pipe preaction systems will be performed following actuation, prior to return to service. If sufficient foreign material is found to obstruct pipe or sprinklers, then an obstruction investigation is conducted per NFPA 25 Annex D. If the visual inspection detects surface irregularities that could indicate wall loss below nominal pipe wall thickness, then followup volumetric examinations will be performed.
	
	• Revise the implementation procedure and calculation for changing test and inspection frequencies associated with the NFPA 805 license amendment (Amendment 206) to note the following restrictions when changing license renewal Fire Water System program and Fire Protection program test and inspection frequencies.
	
	  • EPRI Report 1006756, Fire Protection Equipment Surveillance Optimization and Maintenance Guide will be used to adjust test and inspection frequencies.
	
	  • Data to be used in analyzing the potential for modifying test and inspection frequencies would not be obtained any earlier than 5 years prior to the period of extended operation.
	
	  • A minimum sample size consistent with EPRI Report 1006756 Section 11.2 will be used to modify test and inspection frequencies.
	
	• EPRI Report 1006756 would not be used to modify fire water storage tank inspections/tests, underground flow tests, and inspections of normally dry but periodically wetted piping that will not drain due to its configuration.			

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TABLE 19.4-1 (Sheet 12)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
11	Implement the Aboveground Metallic Tanks Program as described in LRA Section B2.1.15.	[19.1.15] B2.1.15	Implementation started within the 5-year period prior to the period of extended operation. Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05923 dated October 31, 2012 Letter ULNRC-05963 dated February 28, 2013
12	Remove the blisters in the coating, inspect the base metal for aging, and repair the coating in the train A emergency diesel generator fuel oil storage tank. Enhance the Fuel Oil Chemistry Program procedures to: • Include periodic removal of the water from the bottom of the emergency fuel oil system day tanks, diesel fire pump fuel oil day tanks, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks. • Include the addition of biocide to the diesel fire pump fuel oil day tank, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks, if periodic testing indicates biological activity or evidence of corrosion. • Include draining, cleaning, and inspection of the emergency fuel oil system day tanks within the 10-year period prior to the period of extended operation and at least once every 10 years after entering the period of extended operation. • Include a determination of water and sediment in the periodic sampling of the emergency fuel oil system day tanks, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks.	[19.1.16] B2.1.16	Completed no later than 6 months prior to the period of extended operation. Inspections and referenced coating repairs to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05923 dated October 31, 2012 Letter ULNRC-05950 dated January 24, 2013 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06117 dated April 23, 2014

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TABLE 19.4-1 (Sheet 13)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
12(cont'd)	• Include a determination of particulate concentrations in the periodic sampling of the emergency fuel oil system day tanks, diesel fire pump fuel oil day tanks, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks.			
	• Include a determination of microbial activity concentrations in the periodic sampling of the emergency fuel oil system storage tanks, emergency fuel oil system day tanks, diesel fire pump fuel oil day tanks, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks.			
	• Include new fuel oil receipt sampling for water and sediment prior to introduction into the security diesel generator fuel oil day tank, diesel fire pump fuel oil day tank, and alternate emergency power system fuel oil storage tanks.			
	• Perform a volumetric examination of the emergency fuel oil system storage tanks and day tanks after evidence of tank degradation is observed during the visual inspection within the 10-year period prior to the period of extended operation and at least once every 10 years after entering the period of extended operation.			
	• Perform a volumetric examination on the external surface of the diesel fire pump fuel oil day tanks and security diesel generator fuel oil day tank within the 10-year period prior to the period of extended operation and at least once every 10 years after entering the period of extended operation.			
	• Include at least quarterly trending for water, biological activity, and particulate concentrations on the emergency fuel oil system day tanks, diesel fire pump fuel oil day tanks, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks.			
	• Include immediate removal of accumulated water when discovered in the emergency fuel oil system day tank, diesel fire pump fuel oil day tank, security diesel generator fuel oil day tank, and alternate emergency power system fuel oil storage tanks.			

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TABLE 19.4-1 (Sheet 14)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
12(cont'd)	• Perform periodic visual inspections on all accessible internal surface coatings of the emergency fuel oil storage tanks and day tanks. Baseline inspections will be conducted in the 10-year period prior to the period of extended operation on the accessible internal surfaces coatings of the in-scope components. Coatings are inspected every 6 years on an alternating train basis based on no observed degradation or cracking and flaking that has been evaluated as acceptable. Coatings with blisters, peeling, delaminations, or rusting that has been determined not to require remediation are inspected on a 4-year frequency. For peeling, delaminations and blisters determined to not meet the acceptance criteria and that will not be repaired or replaced, physical testing is performed where physically possible (i.e., sufficient room to conduct testing). Testing consists of destructive or nondestructive adhesion testing using ASTM International Standards endorsed in RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Plants." Monitoring and trending of coatings is based on a review of the previous two inspections' results (including repairs) with the current inspection results. The training and qualification of individuals involved in coating inspections are conducted in accordance with ASTM International Standards endorsed in RG 1.54 including guidance from the staff associated with a particular standard. Coating acceptance criteria are as follows: • Indications of peeling and delamination are not acceptable and the coatings are repaired or replaced. • Blisters are evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54 including staff guidance associated with use of a particular standard.			

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TABLE 19.4-1 (Sheet 15)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
12(cont'd)	• Indications such as cracking, flaking, and rusting are to be evaluated by a coatings specialist qualified in accordance with an ASTM International standard endorsed in RG 1.54 including staff guidance associated with use of a particular standard. • Adhesion testing results meet or exceed the degree of adhesion recommended in engineering documents specific to the coating and substrate. Inspection results not meeting the acceptance criteria will be evaluated by a qualified coatings evaluator. Corrective actions will be determined using the Corrective Action Program.			
13	Enhance the Reactor Vessel Surveillance Program to: • Determine the vessel fluence by ex-vessel dosimetry, following withdrawal of the final capsule. (Completed LRA Amendment 28 dated December 20, 2013) • Require that pulled and tested surveillance capsules are placed in storage for future reconstitution or reinsertion unless given NRC approval to discard. (Completed LRA Amendment 28 dated December 20, 2013) • Specifically require the design change process to evaluate the impact of plant operation changes on reactor vessel embrittlement. (Completed LRA Amendment 14 dated October 31, 2012)	[19.1.17] B2.1.17 4.2	Completed	Letter ULNRC-05923 dated October 31, 2012 Letter ULNRC-06057 dated December 20, 2013

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TABLE 19.4-1 (Sheet 16)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
14	Implement the One-Time Inspection Program as described in LRA Section B2.1.18.	[19.1.18] B2.1.18	Implementation started within the 10-year period prior to the period of extended operation. Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
15	Implement the Selective Leaching Program as described in LRA Section B2.1.19.	[19.1.19] B2.1.19	Implementation started within the 5-year period prior to the period of extended operation. Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 17)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
16	Implement the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program as described in LRA Section B2.1.20.	[19.1.20] B2.1.20	Implementation started within the 6-year period prior to the period of extended operation. Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
17	Implement the External Surfaces Monitoring of Mechanical Components Program as described in LRA Section B2.1.21.	[19.1.21] B2.1.21	Implemented no later than 6 months prior to the period of extended operation and inspections to begin during the period of extended operation.	Letter ULNRC-06057 dated December 20, 2013
18	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program as described in LRA Section B2.1.23.	[19.1.23] B2.1.23	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
19	Enhance the Lubricating Oil Program procedures to: • Indicate that lubricating oil contaminants are maintained within acceptable limits, thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer. (Completed LRA Amendment 34 dated April 23, 2014)	[19.1.24] B2.1.24	Completed	Letter ULNRC-06117 dated April 23, 2014

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TABLE 19.4-1 (Sheet 18)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
19(cont'd)	• State the testing standards for water content and particle count. (Completed LRA Amendment 34 dated April 23, 2014) • State that phase separated water in any amount is not acceptable. (Completed LRA Amendment 34 dated April 23, 2014)			
20	Implement the Buried and Underground Piping and Tanks Program as described in LRA Section B2.1.25.	[19.1.25] B2.1.25	Implementation to be started within the 10-year period prior to the period of extended operation. Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
21	Enhance the ASME Section XI, Subsection IWE Program 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 TR 104213, and the additional recommendations of NUREG-1339. • Perform additional surface examinations of stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading for cracking, unless Appendix J testing is adequate to identify cracking.	[19.1.26] B2.1.26	Completed no later than 6 months prior to the period of extended operation. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 19)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
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, American Institute of Steel Construction specifications, and NUREG recommendations to prevent or mitigate degradation and failure of safety-related bolting due to stress corrosion cracking (SCC). 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.	[19.1.28] B2.1.28	Completed no later than 6 months prior to the period of extended operation.	Letter ULNRC-05891 dated August 9, 2012 Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 20)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
23	Enhance the Structures Monitoring Program procedures to: • Include the main access facility, the nitrogen storage tank foundation and pipe trench, the alternate emergency power system diesel generator enclosures, switch house, and transformer foundations, and the reinforced concrete structures under the turbine building and in the yard that provide a return flowpath for the circulating water system in the scope of the 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 SCC potential discussed in Section 2 of the 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 or enclosures, and associated anchorages, and complete a baseline inspection of these components.*	[19.1.31] B2.1.31	Completed no later than 6 months 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 were completed by December 31, 2012, and any corrective actions resulting from initial inspections will be completed no later than December 31, 2017. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05891 dated August 9, 2012 Letter ULNRC-05915 dated October 11, 2012 Letter ULNRC-05950 dated January 24, 2013 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06118 dated May 6, 2014
23(cont'd)	• Specify that groundwater is monitored for pH, chlorides, and sulfates, and every 5 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 inspector qualifications in accordance with American Concrete Institute (ACI) 349.3R-96.			

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TABLE 19.4-1 (Sheet 21)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
	• Quantify acceptance criteria and critical parameters for monitoring degradation, and to provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action in accordance with the three-tier quantitative evaluation criteria recommended in ACI 349.3R.			
	• 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.#			
24	Enhance the Protective Coating Monitoring and Maintenance Program procedures to: • Specify parameters monitored or inspected to include any visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage. • Specify inspection frequencies, personnel qualifications, inspection plans, inspection methods, and inspection equipment that meet the requirements of ASTM D 5163-08.	[19.1.33] B2.1.33	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
	• Specify a pre-inspection review of the previous two monitoring reports and, based on inspection report results, prioritize repair areas as either needing repair during the same outage or as postponed to future outages, but under surveillance in the interim period.			
24(cont'd)	• Specify characterization, documentation, and testing consistent with ASTM D 5163-08 Sections 10.2-10.4 and to specify an evaluation of the inspection reports by the responsible coating evaluation specialist who prepares a summary of findings and recommendations for future surveillance or repair.			

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TABLE 19.4-1 (Sheet 22)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
	• Specify that the inspection reports prioritize repair areas as either needing repair during the same outage or as postponed to future outages, but under surveillance in the interim period.			
25	Enhance the Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program procedures to: • Include all accessible in-scope cable in an adverse localized environment. • Ensure there are no unacceptable visual indications of surface anomalies. All unacceptable visual indications of cable jacket and connection insulation surface anomalies will be subject to an engineering evaluation.	[19.1.34] B2.1.34	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
26	Enhance the Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program procedures to: • Identify the scope of cables requiring aging management. • Require engineering review of surveillance results every 10 years. • Initiate corrective actions when surveillance results do not meet acceptance criteria, and to require an engineering evaluation be performed. When an unacceptable condition or situation is identified, a determination is also made as to whether the review of surveillance results or the cable testing frequency needs to be increased.	[19.1.35] B2.1.35	Completed no later than 6 months prior to the period of extended operation. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05856 dated April 25, 2012 Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 23)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
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 6 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.	[19.1.36] B2.1.36	Completed no later than 6 months prior to the period of extended operation. Inspections and testing to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05856 dated April 25, 2012 Letter ULNRC-05891 dated August 9, 2012 Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 24)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
28	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B2.1.37.	[19.1.37] B2.1.37	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
29	Implement the Monitoring of Neutron-Absorbing Materials Other than Boraflex Program as described in LRA Section B2.1.38.	[19.1.38] B2.1.38	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013
30	Implement the Metal Enclosed Bus Program as described in LRA Section B2.1.39. (Program implemented, LRA Amendment 38 dated July 31, 2014)	[19.1.39] B2.1.39	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06129 dated July 31, 2014

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TABLE 19.4-1 (Sheet 25)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
31	Enhance the Fatigue Monitoring Program procedures to: • Include fatigue usage calculations that consider the effects of the reactor water environment for a set of sample reactor coolant pressure boundary locations and reactor vessel internals locations with fatigue usage calculations. The reactor coolant pressure boundary set includes the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant and plant-specific bounding environmentally assisted fatigue (EAF) locations. • Ensure the scope includes the fatigue crack growth analyses, which support the leak-before-break analyses, ASME Code Section XI evaluations, and the high-energy line break (HELB) selection criterion remain valid by counting the transients used in the analyses. • Require the review of the temperature and pressure transient data from the operator logs and plant instrumentation to ensure actual transient severity is bounded by the design and to include environmental effects where applicable. If a transient occurs which exceeds the design transient definition the event is documented in the CAP and corrective actions are taken. • Include additional transients that contribute significantly to fatigue usage. These additional transients were identified by evaluation of ASME Code Section III fatigue and fatigue crack growth analyses. • Include additional locations which receive more detailed monitoring. These locations were identified by evaluation of ASME Section III fatigue analyses and the locations evaluated for effects of the reactor coolant environment. In addition, reactor vessel internals locations with fatigue usage calculations will be evaluated for the effects of the reactor water environment. The monitoring methods will be benchmarked consistent with the NRC Regulatory Issue Summary (RIS) 2008-30.	[19.2.1] 4.3.2.1 4.3.2.2 4.3.4 B3.1	Completed no later than 6 months prior to the period of extended operation.	Letter ULNRC-05874 dated June 5, 2012 Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06050 dated October 17, 2013 Letter ULNRC-05979 dated April 26, 2013 Letter ULNRC-05860 dated May 3, 2012

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TABLE 19.4-1 (Sheet 26)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
31(cont'd)	• Project the transient count and fatigue accumulation of monitored components into the future. • Include additional cycle count and fatigue usage action limits, which permit completion of corrective actions if the design limits are expected to be exceeded within the next three fuel cycles. The fatigue results associated with the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse plant and plant-specific bounding EAF locations will account for environmental effects on fatigue. The cycle count action limits for the hot leg surge nozzle will incorporate the 60-year cycle projections used in the hot leg surge nozzle EAF analysis. • Include appropriate corrective actions to be invoked if a component approaches a cycle count or CUF action limit or if an experienced transient exceeds the design transient definition. If an action limit is reached, corrective actions include fatigue reanalysis, repair, or replacement. When a cycle counting action limit is reached, action will be taken to ensure that the analytical bases of the HELB locations are maintained. Reanalysis of a fatigue crack growth analysis must be consistent with or reconciled to the originally submitted analysis and receive the same level of regulatory review as the original analysis. • Limit the number of the most severe RCP component cooling water transient, elevated CCW inlet temperature transients, to 75% of its design value (i.e., limited to 150) in order to accommodate the seasonal temperature change transient in the RCP thermal barrier flange fatigue analysis. (Moved here from Commitment No. 38 by LRA Amendment 23)			

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TABLE 19.4-1 (Sheet 27)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
31(cont'd)	• Include non-NUREG/CR-6260 locations with a U_{en} greater than 1.0 for further evaluation using the same methods as those used for NUREG/CR-6260 locations to remove conservatisms from the preliminary U_{en}. The results of these final analyses will be incorporated into the Fatigue Monitoring Program by either counting the transients assumed or incorporate the stress intensities into a CBF ability of the program. As an alternative, the Fatigue Monitoring Program will implement SBFs of certain locations in order to ensure the component does not exceed a U_{en} of 1.0. Any use of SBF will be implemented consistent with RIS 2008-30. • The sentinel location analysis, when refined, will be revisited to confirm bounding Reactor Coolant Pressure Boundary Environmentally Assisted Fatigue susceptible sentinel locations are updated appropriately and remain bounded consistent with the refined analysis.			
32	Enhance the Concrete Containment Tendon Prestress Program specification to: • Include random samples for the 40-, 45-, 50-, and 55-year surveillances. • Extend the predicted lower limit (PLL) lines for the vertical and hoop tendon groups to 60 years. • Specifically require the final report for each surveillance interval to plot the measured results against time and to include the PLL, Minimum Required Value, and trend lines. • Require a regression analysis consistent with the requirements of NRC Information Notice 99-10 Revision 1, Attachment 3.	[19.2.3] B3.3 4.5	Completed no later than 6 months prior to the period of extended operation.	Letter ULNRC-05963 dated February 28, 2013

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TABLE 19.4-1 (Sheet 28)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
33	As additional industry and plant-specific applicable operating experience becomes available, it will be evaluated and incorporated into each new program. [Completed LRA Section B2.1.6 with Amendment 28 dated December 20, 2013] [Completed LRA Section B2.1.39 with Amendment 38 dated July 31, 2014]	[19.1.6] [19.1.15] [19.1.18] [19.1.19] [19.1.20] [19.1.21] [19.1.23] [19.1.25] [19.1.37] [19.1.38] [19.1.39] [B2.1.6] B2.1.15 B2.1.18 B2.1.19 B2.1.20 B2.1.21 B2.1.23 B2.1.25 B2.1.37 B2.1.38 B2.1.39	Completed consistent with implementation schedule noted with each referenced AMP.	Letter ULNRC-05963 dated February 28, 2013 Letter ULNRC-06057 dated December 20, 2013 Letter ULNRC-06129 dated July 31, 2014
34	Ameren Missouri replacement steam generator divider plate assemblies are fabricated of Alloy 690. The divider plate to primary head and tubesheet junctions are welded with Alloy 152 weld materials. The tubesheet cladding is Alloy 182 and the primary head cladding is stainless steel. There is a concern regarding potential failure at the divider plate welds to primary head and tubesheet cladding and Ameren Missouri commits to perform one of the following three resolution options: <u>Option 1: Inspection</u> Perform a one-time inspection of each steam generator to assess the condition of the divider plate welds. The examination technique(s) will be capable of detecting primary water stress-corrosion cracking (PWSCC) in the divider plate assemblies and the associated welds. OR	[19.1.9] Section 3.1.2.2.11.1 Table 3.1.2-4	Option 1 completed between fall 2025 and fall 2029 when the replacement steam generators are in service for more than 20 years. Option 2 or Option 3 available for NRC review in the fall 2023.	Letter ULNRC-05920 dated October 24, 2012 Letter ULNRC-06057 dated December 20, 2013 Letter ULNRC-06080 dated February 14, 2014

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TABLE 19.4-1 (Sheet 29)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
34(cont'd)	Option 2: Analysis Perform an analytical evaluation of the steam generator divider plate welds in order to establish a technical basis which concludes that the steam generator RCS pressure boundary is adequately maintained with the presence of steam generator divider plate weld cracking. The analytical evaluation will be submitted to the NRC for review and approval. OR Option 3: Industry/NRC Studies If results of industry and NRC studies and operating experience document that potential failure of the steam generator RCS pressure boundary due to PWSCC cracking of steam generator divider plate welds is not a credible concern, this commitment will be revised to reflect that conclusion.			
35	The material of steam generator tubesheet cladding is Alloy 182. The tubes are made of Alloy 690 and are secured to the tubesheet by means of tube-to-tubesheet leaktight weld and tube expansion. There is a concern regarding potential failure of primary-to-secondary pressure boundary due to PWSCC cracking of tube-to-tubesheet welds. Ameren Missouri commits to perform one of the following two resolution options: Option 1: Inspection Perform a one-time inspection of a representative number of tube-to-tubesheet welds in each steam generator to determine if PWSCC cracking is present. The examination technique(s) will be capable of detecting PWSCC in the tube-to-tubesheet welds. If weld cracking is identified, the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program will be established to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators. OR	[19.1.9] Section 3.1.2.2.11.2 Table 3.1.2-4	Option 1 completed between fall 2025 and fall 2029 when the replacement steam generators are in service for more than 20 years. Option 2 available for NRC review in the fall 2023.	Letter ULNRC-05920 dated October 24, 2012 Letter ULNRC-06057 dated December 20, 2013 Letter ULNRC-06080 dated February 14, 2014

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TABLE 19.4-1 (Sheet 30)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
35(cont'd)	Option 2: Analysis Perform an analytical evaluation of the steam generator tube-to-tubesheet welds either determining that the welds are not susceptible to PWSCC or redefining the reactor coolant pressure boundary of the tubes, where the steam generator tube-to-tubesheet welds are not required to perform a reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC. The evaluation for determination that the welds are not susceptible to PWSCC and do not require inspection will be submitted to the NRC for review.			
36	Implement stress-based fatigue (SBF) or cycle-based fatigue (CBF) consistent with RIS 2008-30 to monitor the CUF of the limiting location out of the pressurizer lower head, surge nozzle, and heater penetrations to accommodate the insurge-outsurge transient. (Closed. LRA Amendment 11, dated October 11, 2012. The re-evaluation of insurge-outsurge analysis demonstrated that this type of detailed monitoring was not necessary.)	[19.2.1] 4.3.1 4.3.2.2 B3.1	Closed	Letter ULNRC-05915 dated October 11, 2012 Letter ULNRC-05963 dated February 28, 2013
37	Complete an evaluation to determine if there are any additional plant-specific bounding EAF locations. The supporting environmental factors, F(en), calculations will be performed with NUREG/CR-6909 or NUREG/CR-6583 for carbon and low-alloy steels, NUREG/CR-6909 or NUREG/CR-5704 for austenitic stainless steels, and NUREG/CR-6909 for nickel alloys. (Completed Amendment 2 dated May 3, 2012) In order to determine if the pressurizer contains a limiting EAF location, the fatigue analyses will be revised to incorporate the effect of insurge-outsurge transients on the pressurizer lower head, surge nozzle, and heater well nozzles at plant-specific conditions. (Completed Amendment 2 dated May 3, 2012) The pressurizer contains a limiting EAF location. The fatigue analyses will be revised to incorporate the effect of insurge-outsurge transients in the pressurizer lower head. (Completed LRA Amendment 11 dated October 11, 2012)	[19.2.1] 4.3.2.2 4.3.4	Completed	Letter ULNRC-05860 dated May 3, 2012 Letter ULNRC-05915 dated October 11, 2012 Letter ULNRC-05979 dated April 26, 2013

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TABLE 19.4-1 (Sheet 31)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
38	The number of the most severe RCP component cooling water (CCW) transient, elevated CCW inlet temperature transients, will be limited to 75% of its design value (i.e., limited to 150) to accommodate the seasonal temperature change transient in the RCP thermal barrier flange fatigue analysis. (Moved to Item #31 to be managed by the Fatigue Monitoring program - LRA Amendment 23 dated April 26, 2013)	See Commitment No. 31	Implemented as part of Commitment No. 31: Completed no later than 6 months prior to the period of extended operation.	Letter ULNRC-05979 dated April 26, 2013
39	NFPA 805 and LRA gap analysis: A gap analysis of LRA Tables 2.3.3-20 and 3.3.2-20 will be provided to identify differences between the existing and NFPA 805 post-transition changes. The results and the impacts of these gaps on the fire protection program described in LRA Tables 2.3.3-20 and 3.3.2-20 will be summarized, as the basis for transitioning to the NFPA 805 nuclear safety capabilities. The summary will also list the fire protection systems and components including structural fire barriers (e.g., fire walls and slabs, fire doors, fire barrier penetration seals, fire dampers, fire barrier coatings and wraps, equipment and personnel hatchways and plugs, metal siding) that will be added or removed based on the NFPA 805 transition in the scope of license renewal in accordance with 10 CFR 54.4(a) and whether they are subject to an AMR in accordance with 10 CFR 54.21(a)(1). (Completed Amendment 31 dated February 14, 2014)	[19.1.13] [19.1.14] B2.1.13 B2.1.14	Completed	Letter ULNRC-05877 dated July 2, 2012 Letter ULNRC-05946 dated January 10, 2013 Letter ULNRC-05971 dated March 20, 2013 Letter ULNRC-06080 dated February 14, 2014 Letter ULNRC-06114 dated April 15, 2014
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.	[19.1.27] B2.1.27	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to the period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-05891 dated August 9, 2012 Letter ULNRC-05963 dated February 28, 2013

Footnote: Text in this table was taken directly from NUREG-2172, “Safety Evaluation Report Related to the License Renewal of Callaway Plant, Unit 1,” issued March 2015.
Bracketed text represents:
· Supplemental Information
· FSAR Supplement Section Numbers

TABLE 19.4-1 (Sheet 32)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
41	To allow for monitoring of the condition of the threads on the RPV stud and flange hole threads, Ameren Missouri commits to remove RPV stud #18 through nondestructive or destructive means. If RPV stud hole repair is required following removal of RPV stud #18, the repair plan will include inspecting the RPV stud hole prior to the repair to assess the as-found condition and an inspection after the repair is complete to assess the results of the repair.	[19.1.3] B2.1.3	Implemented as part of proposed License Condition No. 3: Completed no later than 6 months prior to the period of extended operation or the refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-06032 dated August 29, 2013 Letter ULNRC-06057 dated December 20, 2013
42	It is noted that Ameren Missouri experienced problems with the reactor vessel head closure studs and stud holes early in plant life (1986-1992) and that multiple RPV stud holes required ASME Section XI repairs to remove damaged threads. To supplement the monitoring that is accomplished through regular volumetric inspections and to confirm that additional thread degradation is not occurring in the RPV stud holes, Ameren Missouri commits to perform a one-time inspection of select RPV stud holes using a method consistent with the Babcock and Wilcox laser inspection that was applied following stud hole repair in 1989 and 1992. RPV stud hole locations 2, 4, 5, 7, 9, and 53 have had more than one thread removed and will be inspected. If inspection of these RPV stud holes confirms that there was minimal or no additional degradation since the prior video inspection, then it is a reasonable conclusion that there will be minimal additional degradation in the period of extended operation. If additional degradation is observed in any of the repaired stud holes where more than one thread has been removed, the condition will be entered in the Corrective Action Program for evaluation and corrective action, and the remaining repaired RPV stud hole locations 13, 25, 39, and 54 will be inspected. The inspection is expected to confirm that further degradation is not occurring in the repaired stud holes and will provide a basis for the conclusion that acceptance criteria for thread engagement will continue to be met through the period of extended operation.	[19.1.3] B2.1.3	Implemented as part of proposed License Condition No. 3: Completed no later than 6 months prior to the period of extended operation or the refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-06032 dated August 29, 2013 Letter ULNRC-06057 dated December 20, 2013

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TABLE 19.4-1 (Sheet 33)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
43	The core design procedure will be modified to include a review for the following core design parameters to ensure that these limits are met in future core designs: • Active fuel - upper core plate distance >12.2 inches • Average core power density <124 watts/cm^3 • Heat generation figure of merit, F≤68 watts/cm^3 (Completed Amendment 31 dated February 14, 2014)	[19.1.6] B2.1.6	Completed	Letter ULNRC-06072 dated January 16, 2014 Letter ULNRC-06080 dated February 14, 2014 Letter ULNRC-06090 dated March 13, 2014
44	For all MRP-191 Table 4-4 components, as applicable to Callaway, Ameren Missouri commits to perform one or more of the following resolution options for the non-CASS RVI components: <u>Option 1: Replacement</u> RVI components determined to be subject to 20% or greater cold work and 30 ksi operating stress will be replaced. <u>Option 2: Inspection</u> For RVI components determined to be subject to 20% or greater cold work and 30 ksi operating stress, an augmented inspection program capable of detecting cracking will be developed. Minimum examination coverage criteria consistent with MRP-227-A Primary Inspection Category Components will apply. The augmented inspection program will be submitted to the NRC prior to performance of the inspection(s). <u>Option 3: Impact Evaluation</u> For RVI components determined to be subject to 20% or greater cold work and 30 ksi operating stress, an impact evaluation will be prepared to establish that the effects of aging are minimal and will not have an adverse impact on future plant operability or component intended function. The impact evaluation(s) will be submitted to the NRC.	[19.1.6] B2.1.6	Closed	Letter ULNRC-06079 dated February 5, 2014 Letter ULNRC-06106 dated March 28, 2014 Letter ULNRC-06117 dated April 23, 2014

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TABLE 19.4-1 (Sheet 34)

Item Number	Commitment	FSAR Supplement Section/ LRA Section	Implementation Schedule	Source
44(cont'd)	Option 4: Mitigation RVI components determined to be subject to 20% or greater cold work and 30 ksi operating stress will be mitigated of stress corrosion cracking (SCC) susceptibility. Note: Indeterminate components will be conservatively assumed to be subject to 20% or greater cold work and subject to 30 ksi operating stress. (Closed, evaluation (ULNRC-06106, March 28, 2014) concluded that the plant-specific material fabrication and design are consistent with the MRP-191 basis and the MRP-227-A aging management requirements as related to cold work are directly applicable to Callaway Unit 1. Therefore, Options 1 - 4 are not necessary.)			
45	Enhance the RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program procedures to include the concrete structures in the turbine building that provide a flowpath for the circulating water system in the scope of the program. (Moved to Item 23 to be managed by the Structures Monitoring program - LRA Amendment 35 dated May 6, 2014)	See Commitment No. 23	Implemented as part of Commitment No. 23: Completed no later than 6 months prior to the period of extended operation.	Letter ULNRC-06080 dated February 14, 2014 Letter ULNRC-06118 dated May 6, 2014
46	Enhance the ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD Program to perform periodic inspection of the reactor vessel cladding indications identified in FSAR Section 5.2.3.2.2 SP and reconcile the inspection results with the corrosion analysis to ensure the analytical basis of the analysis is maintained.	[19.1.1] 4.7.3	Completed no later than 6 months prior to the period of extended operation. Inspections to be completed no later than 6 months prior to period of extended operation or the end of the last refueling outage prior to the period of extended operation, whichever occurs later.	Letter ULNRC-06118 dated May 6, 2014

Footnote: Text in this table was taken directly from NUREG-2172, “Safety Evaluation Report Related to the License Renewal of Callaway Plant, Unit 1,” issued March 2015.
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