Amendment 11, LRA Changes from RAI Responses

Enclosure 2 Summary Table

Affected LRA Section	LRA Page
Table 3.3.2-28	3.3-274 and 276
Table 3.4.2-3	3.4-43
Table 4.3-1	4.3-4
Table 4.3-2	4.3-6 and 7
Table 4.3-3	4.3-20
Section 4.3.2.2	4.3-25
Section 4.3.5	4.3-38, 39, and 40
Section 4.3.8	4.3-43
Section A3.2.1.1	A-25
Section A3.2.1.2	A-26
Table A4-1, Item 23	A-43
Table A4-1, Items 36 and 37	A-49
Section B2.1.31	B-104, 105, 106, and 107

Callaway
License Renewal Application
Amendment 11

Revision to include TLAA AMR lines for the Boron Recycle System, Condensate System, and Main Feedwater System.

Table 3.3.2-28 (page 3.3-274 and 3.3-276) and Table 3.4.2-3 (page 3.4-43) are revised as follows (new text shown underlined).

Table 3.3.2-28 Auxiliary Systems – Summary of Aging Management Evaluation – Miscellaneous Systems in scope ONLY for Criterion 10 CFR 54.4(a)(2)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Piping	LBS, SIA	Carbon Steel	Secondary Water (Int)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	VIII.D1.S-11	3.4.1.001	A
Piping	LBS, SIA	Stainless Steel	Treated Borated Water (Int)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	VII.E1.A-57	3.3.1.002	A

Table 3.4.2-3 Steam and Power Conversion System – Summary of Aging Management Evaluation – Main Feedwater System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
Piping	LBS, PB, SIA	Carbon Steel	Secondary Water (Int)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	VIII.D1.S-11	3.4.1.001	A

Table 4.3-1 Fatigue Monitored Locations and Management Methods

Component	Fatigue Monitoring Method	LRA Section
Accumulator SI Nozzle	CBF	4.3.2
Boron Injection Header Nozzles-Cold Leg (NUREG/CR 6260 locations)	CBF	4.3.4
RPV Inlet Nozzle (NUREG/CR-6260 locations)	CBF	4.3.4
RPV Outlet Nozzle (NUREG/CR-6260 locations)	CBF	4.3.4
RPV Lower Head (NUREG/CR-6260 locations)	CBF	4.3.4
Normal Charging Nozzle (NUREG/CR-6260 locations)	SBF	4.3.4
Alternate Charging Nozzle (NUREG/CR-6260 locations)	SBF	4.3.4
~~Pressurizer Lower Head~~	~~CBF/SBF(1)~~	~~4.3.2.2~~
~~Pressurizer Surge Nozzle~~	~~CBF/SBF(1)~~	~~4.3.2.2~~
~~Pressurizer Heater Well~~	~~CBF/SBF(1)~~	~~4.3.2.2~~
RHR Nozzle-Hot Leg Loops 1 and 4 (NUREG/CR-6260 locations)	CBF	4.3.4
Crossover Leg Loops 1 and 2 Drain Nozzles	CBF	4.3.2
Crossover Leg Loop 4 Excess Letdown Nozzles	CBF	4.3.2
Normal Letdown/Drain Line Loops 2 and 3	CBF	4.3.2
Pressurizer Safety and Relief Valve Piping	CBF	4.3.2

~~1 The method to monitor these locations has not been determined, but is a commitment as discussed in Section 4.3.2.2.~~

Table 4.3-2 Transient Accumulations and Projections

Transient Description	FSAR Design Cycles	Design Limiting Value[(1)]	Baseline (1983 – 2011)	Projected Events for 60 Years	Comments
Normal Condition Transients					
5a. Steady state fluctuations, Initial fluctuations	1.5 E5	1 E6	Not Counted (NC)	NC	The limiting value is from the reactor coolant loop (RCL) leak-before-break analysis described in Section 4.7.7. These fluctuations are assumed to occur only during the first 20 full power months of operation, therefore they do not need to be counted.
5b. Steady state fluctuations, Random fluctuations	3.0 E6		NC	NC	The limiting value is from the RCL leak-before-break analysis described in Section 4.7.7. ~~This number of cycles is beyond the endurance limit of the fatigue curve, therefore this transient does not need to be counted.~~ Transient does not result in the accumulation of fatigue usage.

Table 4.3-3 ASME Class 1 Fatigue Analyses Under the Fatigue Monitoring Program

Component	CUF
Pressurizer Components	
Spray Nozzle	0.411
Upper Head	0.928
Surge Nozzle	0.963(1) 0.034(2) (Dry) / 0.032(2) (Wet)
Safety and Relief Nozzle	0.169
Support Skirt and Flange	0.734(1) 0.7284(2,3) (Dry)
Lower Head	0.112(1) 0.098(2) (Dry) / 0.019(2) (Wet)
Heater Penetration (Heater Well)	0.128(1) 1.441(2) (Dry) / 0.562(2) (Wet) 0.8146(2,3) (Dry) / 0.0103(2,3) (Wet)
Seismic Support Lug	0.444
Shell at Support Lug	0.992
Trunnion Buildup	0.567
Instrument Nozzle	0.236(1) 0.746(2) (Dry) / 0.0103(2,3) (Wet)
Manway Bolt	0.915
Manway Pad	0.141
Valve Support Bracket	0.118
~~Immersion Heater~~	~~0.123~~

1 Original fatigue usage used as for common basis comparison in the EAF screening evaluation. It is provided in Table 4.3-7.
2 Includes the results from evaluation of revised insurge-outsurge transients and further refined analysis.
3 Based on an elastic-plastic analysis.

4.3.2.2 Pressurizer Insurge-Outsurge Transients

Westinghouse Nuclear Safety Advisory Letter NSAL 04-5 describes the thermal transients resulting from a reactor coolant insurge-outsurge during normal heatup and cooldown operations. The limiting CUF locations for Westinghouse NSSS Plants are at the heater penetrations and pressurizer surge nozzle. This type of transient was not considered in the original design analyses of the pressurizer because it was assumed that when a pressurizer insurge occurred, the screen covering the surge nozzle opening inside the pressurizer caused mixing of the colder hot leg and hotter pressurizer water. However, instead of mixing, surge line stratification data led to the realization that the cooler, denser, hot leg water flows underneath the pressurizer water, resulting in a moving stratified condition.

To mitigate pressurizer insurge-outsurge transients Callaway has used modified operating procedures (MOPs) since 1996. Sample fatigue analyses for typical Westinghouse plants using MOPs show that the expected maximum fatigue for the pressurizer and pressurizer components is expected to remain below the ASME Code limit of 1.0.

Guidance on Operations and Engineering actions for monitoring and evaluating fatigue if the pressurizer surge line ΔT limits are exceeded is incorporated into the Plant Heatup and Cooldown procedures. These actions include direction to initiate an engineering evaluation of the surge line accumulated fatigue if the surge line ΔT exceeds 80°F. 80°F ΔT is the threshold below which an insurge-outsurge event will not significantly contribute to fatigue usage.

~~In order to determine if the pressurizer contains a limiting environmentally-assisted fatigue (EAF) location, t~~The fatigue analyses ~~will be~~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. These revised insurge-outsurge analysis results demonstrate that the CUFs, including environmentally-assisted fatigue (EAF), are less than 1.0. 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).

~~For license renewal Callaway has committed to monitor the CUF of the limiting location out of the pressurizer lower head, pressurizer surge line nozzle, and heater well nozzles using fatigue monitoring software consistent with RIS 2008-30. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).~~

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

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

Piping in the scope of license renewal that is designed to ANSI B31.1 or ASME Section III Class 2 and 3 requires the application of a stress range reduction factor to the allowable stress range (expansion and displacement) to account for thermal cyclic conditions. If the number of equivalent full temperature cycles exceeds 7,000, a factor less than 1 must be used. These piping analyses would be TLAAs because they are part of the current licensing basis, are used to support safety determinations, and depend on an assumed number of thermal cycles that can be linked to plant life.

None of ANSI B31.1 or the ASME Section III Subsections NC and ND for Class 2 and 3 piping invokes fatigue analyses. If the number of full-range thermal cycles is expected to exceed 7,000, these codes require the application of a stress range reduction factor to the allowable stress range for expansion stresses (secondary stresses). The allowable secondary stress range is 1.0 S_A for 7,000 equivalent full-temperature thermal cycles or less and is reduced in steps to 0.5 S_A for greater than 100,000 cycles.

Temperature screening criteria were used to identify components that might be subject to significant thermal fatigue effects. Normal and upset operating temperatures less than 220°F in carbon steel components, or 270°F in stainless steel, will not produce significant thermal stresses, and will not therefore produce significant fatigue effects.

Piping systems that exceed the temperature screening criteria are subject to thermal fatigue effects and are therefore included in the aging management review (AMR) results presented in Chapter 3. A systematic survey of all plant ANSI B31.1 or ASME Section III Class 2 and 3 piping systems found that the systems that exceed the temperature screening criteria are:

- Main Steam Supply System
- Main Turbine System
- Condensate System
- Main Feedwater System
- Reactor Coolant System
- Chemical and Volume Control System
- Steam Generator Blowdown System
- Fuel Pool Cooling and Cleanup System
- Residual Heat Removal System
- High Pressure Coolant Injection System
- Auxiliary Feedwater System
- Containment Hydrogen Control System
- Containment Purge System
- Decontamination System
- Boron Recycle System
- Standby Diesel Generator Engine System
- Nuclear Sampling System

With the exception of reactor coolant sample lines described below, these piping and components in the scope of license renewal clearly do not operate in a cycling mode that would expose the piping to more than three thermal cycles per week, i.e. to more than 7,000 cycles in 60 years.

For most of these systems, the assumed thermal cycle count for the analyses depend on reactor operating cycles, and can therefore conservatively be approximated by the thermal cycles used in the ASME III Class 1 vessel and piping fatigue analyses. Thermal cycles likely to produce full-range thermal cycles in balance-of-plant Class 2, 3, and B31.1 piping, in a 40-year plant lifetime, are the 200 heatup cycles, 200 cooldown cycles, and 400 reactor trips.

Other events may contribute part-range cycles; however, even if the part–range cycles are assumed to be full-range cycles, the total number of design basis thermal events from FSAR Table 3.9(N)-1 SP is only 2,190. This discounts those transients whose design values are unrealistically overestimated (loading/unloading and feedwater cycling) and those transients applicable to load following operation (boron concentration equalization, and reduced temperature return to power). The total number of design basis thermal events actually expected in a 60-year life is about 1,537 including those overestimated transients previously excluded (loading/unloading and feedwater cycling). See Table 4.3-2, *Transient Accumulations and Projections*.

The total count of expected full-range thermal cycles for these systems is less than 1,600 for a 60-year plant life, which is a fraction of the 7,000 cycle threshold for which a stress range reduction factor is required in the applicable piping codes.

Reactor Coolant Sample Lines

A survey of plant piping systems found that some reactor coolant sample lines may be subject to more than 7,000 thermal cycles. Review of FSAR Table 9.3 3 SP and Callaway Chemistry Schedule identified that the only sample piping in the scope of license renewal that meets the temperature screening criteria, and could possibly exceed 7,000 cycles, is the RCS hot leg sample piping, which requires 3 samples per week. This potential was noted in the design calculations which reduced the allowable secondary stress range from 1.0 S_A to 0.9 S_A. This accounts for up to 14,000 thermal cycles or over 4 samples per week for 60 years.

~~Review of operating practice at Callaway indicates that RCS samples are taken weekly from the hot leg during operation. Therefore none of the lines associated with this sample location will exceed 7,000 cycles during the period of extended operation.~~

The existing analyses of ANSI B31.1 or ASME Section III Class 2 and 3 piping within the scope of license renewal for which the allowable range of secondary stresses depends on the number of assumed thermal cycles are valid for the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

4.3.8 Fatigue Analyses of Class 2 Heat Exchangers

Letdown Reheat Heat Exchanger

The fatigue analysis for the letdown reheat heat exchanger indicated a maximum CUF of 4.431 for the studs. This CUF is the result of the reanalysis to account for operation with a letdown flow of 140 gpm. The CUF is driven mainly by transient 7, “Letdown flow step increase and return to normal,” and transient 11, “Load follow boration.” These are load following transients and Callaway does not practice load following operation. The assumed number of these transients was dropped by an order of magnitude and the CUF dropped to about 0.503.

The fatigue analysis of the Callaway letdown reheat heat exchanger evaluated the shell and tube side nozzles and the tubesheet. These components have maximum CUFs of 0.054 and 0.47 respectively. These fatigue analyses included transients 1, 2, 4, 5, 6, 7, and 11. Transients 1 and 2 will be monitored by the Fatigue Monitoring program (B3.1). The remaining transients, transients 4, 5, 6, 7, and 11, are not monitored. Transients 4, 5, 7, and 11 assume 24,000 events and are load following events. As stated above, Callaway will not approach the limiting number of events during a 60-year plant life. For Transient 6, if the number of events is extended through the period of extended operation, then 3,000 events will be assumed to occur and the CUFs will increase to ~~0.57~~0.50 for the tubesheet and 0.0563 for the tube side nozzles. ~~The nozzles CUFs are not affected by this increase.~~

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).

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

A3.2.1.1 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~~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 A2.1.

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 percent 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).

A3.2.1.2 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 limiting cumulative usage factors (CUF) locations for Westinghouse NSSS plants are at the heater penetrations and pressurizer surge nozzle.~~ The fatigue analyses have been ~~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. ~~Callaway has committed to monitor the CUF of the limiting location out of the pressurizer lower head, pressurizer surge line nozzle, and heater well nozzles using fatigue monitoring software consistent with RIS 2008-30.~~ 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).

Table A4-1 License Renewal Commitments

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

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
36	Implement SBF or 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 Amendment 11, re-evaluation of insurge-outsurge analysis demonstrated that this type of detailed monitoring was not necessary.)	4.3.1 4.3.2.2 B3.1	Prior to the period of extended operation

Item #	Commitment	LRA Section	Implementation Schedule
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) In order to determine if the pressurizer contains a limiting EAF location, the fatigue analyses will be revised to incorporate the affect effect of insurge-outsurge transients on the pressurizer lower head, surge nozzle, and heater well nozzles at plant specific conditions. (Completed Amendment 2) Those non-NUREG/CR-6260 locations with an EAF CUF greater than 1.0 will be further evaluated using same methods as those used for NUREG/CR-6260 locations to remove conservatisms from the preliminary EAF CUF. 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 an EAF CUF of 1.0. Any use of SBF will be implemented in compliance with RIS 2008-30. 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 Amendment 11)	4.3.2.2 4.3.4	Prior to the period of extended operation

B2.1.31 Structures Monitoring

Program Description

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

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

Plant procedures, following enhancements, will specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, EPRI TR-104213, *Bolted Joint Maintenance & Application Guide*, and the additional recommendations of NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*.

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

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

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

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

NUREG-1801 Consistency

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

Exceptions to NUREG-1801

None

Enhancements

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

Scope of the Program (Element 1)

Procedures will be enhanced to include the main access facility into the scope of Structures Monitoring program.

Preventive Actions (Element 2)

Plant procedures will be enhanced to specify that whenever replacement of bolting is required, bolting material, installation torque or tension, and use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, EPRI NP-5067, EPRI TR-104213, and the additional recommendations of NUREG-1339.

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

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

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

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

Parameters Monitored or Inspected (Element 3)
Procedures will be enhanced to specify that structural bolts greater than one inch in diameter with actual measured yield strength greater than or equal to 150 ksi are evaluated for susceptibility to stress corrosion cracking, and, if necessary, visual inspections are supplemented with volumetric or surface examinations.

Detection of Aging Effects (Element 4)
Procedures will be enhanced to specify inspector qualifications in accordance with ACI349.3R-96.

Acceptance Criteria (Element 6)
Procedures will be enhanced to quantify acceptance criteria and critical parameters for monitoring degradation, and to provide guidance for identifying unacceptable conditions requiring further technical evaluation or corrective action in accordance with the three tier quantitative evaluation criteria recommended in ACI 349.3R.
Procedures will be enhanced to incorporate applicable industry codes, standards and guidelines for acceptance criteria.

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

Operating Experience

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

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

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

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

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

   The above examples provide objective evidence that the Structures Monitoring program is capable of both monitoring and detecting the aging effects associated with the program. Occurrences that would be identified under the Structures Monitoring program will be evaluated to ensure there is no significant impact to safe operation of the plant and corrective actions will be taken to prevent recurrence. Guidance for re-evaluation, repair, or replacement is provided for locations where aging is found. There is confidence that the continued implementation of the Structures Monitoring program will effectively identify aging prior to loss of intended function.

Conclusion

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