ENCLOSURE 1

AMP Changes for
Callaway Plant Unit 1
License Renewal Application
Amendment No. 3

Changes Related to Aging Management Programs (AMPs)

AMP XI.M19 (B2.1.9) Steam Generators

Affected LRA Sections
A1.9, "Steam Generators"
B2.1.9, "Steam Generators"

Reason for Change
Delete the reference to Regulatory Guide 1.121. The applicable criteria are incorporated in Callaway Technical Specifications, Section 5.5.9. Revised operating experience to include refueling outage 18 (Fall 2011).

AMP XI.M27 (B2.1.14) Fire Water System

Affected LRA Section
B2.1.14, "Fire Water System"

Reason for Change
Incorporated editorial correction to be consistent with LRA Appendix A1.14 that was included in LRA Amendment 1.

AMP XI.M31 (B2.1.17) Reactor Vessel Surveillance

Affected LRA Sections
B2.1.17, "Reactor Vessel Surveillance"

Reason for Change
Identify capsule W, X and/or Z in the program description and operating experience sections.

AMP XI.M38 (B2.1.23) Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

Affected LRA Sections
A1.23, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"
B2.1.23, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"

Reason for Change
Clarify AMP scope and provide justification for component sample size and sample selection criteria.

AMP X.M1 (B3.1) Fatigue Monitoring

Affected LRA Sections
4.3.1.1 Fatigue Monitoring Methods
A2.1, “Fatigue Monitoring”
Table A4-1 Item #31
B3.1, “Fatigue Monitoring”

Reason for Change
Revise LRA Appendix A to describe the fatigue monitoring methods that will be used. The enhancement to element 3 is also clarified.

Callaway LRA Amendment 3
Affected Pages

LRA Section	AMP	Page Nos
4.3.1.1	X.M1	4.3-2
A1.9	XI.M19	A-6
A1.23	XI.M38	A-13
A2.1	X.M1	A-21
Table A4-1	X.M1	A-46
B2.1.9	XI.M19	B-38 to B-40
B2.1.14	XI.M27	B-53 to B-56
B2.1.17	XI.M31	B-64 to B-65
B2.1.23	XI.M38	B-81 to B83
B3.1	X.M1	B127-B130

4.3.1.1 Fatigue Monitoring Methods

The Fatigue Monitoring program (B3.1) will include both manual cycle counting of certain transients along with automatic cycle counting of selected transients utilizing monitoring software. The program also monitors transient pressure and thermal conditions to calculate the actual fatigue usage for specified fatigue critical locations. Monitored locations will include locations that were identified by the evaluation of ASME Section III fatigue analyses, the NUREG/CR-6260 sample locations for a newer vintage Westinghouse Plant, and plant- specific bounding environmentally assisted fatigue (EAF) locations. The program also accounts for the effects of the reactor coolant environment on fatigue usage where applicable. See Section 4.3.4, *Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190).*

The program will track the occurrences of plant transients listed in Table 4.3-2, *Transient Accumulations and Projections*, using cycle counting, and monitors the cumulative usage factors (CUFs) at the components listed in Table 4.3-1, *Fatigue Monitored Locations and Management Methods* using either cycle-based fatigue or stress-based fatigue.

1. The Cycle Counting monitoring method: This monitoring method tracks transient event cycles affecting the location (e.g. plant heatup and plant cooldown) to ensure that the numbers of transient events analyzed by the design calculations are not exceeded.

2. The Cycle-based Fatigue (CBF) monitoring method: This monitoring method utilizes the cycle counting results and stress intensity ranges generated with the ASME III methods that use three dimensional six component stress-tensor methods~~six stress tensors~~ to perform cumulative usage factor (CUF) calculations for a given location. The fatigue accumulation is tracked to determine if the ASME allowable fatigue limit of 1.0 is approached. CBF monitoring is consistent with ASME Section III, which requires six component stress tensors; and Regulatory Issue Summary (RIS) 2008-30.

3. The Stress-Based Fatigue (SBF) monitoring method: This 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~~Ameren Missouri will benchmark the method in line~~ with RIS 2008-30. The benchmark for the charging nozzles has been completed.

Charging Nozzle SBF Benchmarking

RIS 2008-30 discussed NRC staff concerns about the use of a single stress term used in SBF algorithms. The RIS does not imply that all six stress components must be used. The RIS only requires that it be demonstrated that the simplification of the use of less than the six-stress tensors produces a conservative result. Callaway performed a benchmark to demonstrate that the charging nozzle SBF algorithm produces a conservative CUF as

A1.9 STEAM GENERATORS

The Steam Generators program manages cracking, loss of material, and wall thinning of the steam generator tubes, plugs, sleeves and secondary side steam generator internal components ~~tube supports~~. 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. ~~Tube structural integrity limits consistent with Regulatory Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes* are applied.~~

A1.23 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. ~~of the~~ The program inspects internal surfaces of metallic piping, piping components, ducting, polymeric components, and other components that are ~~not inspected by other aging management programs. The program inspects a representative sample of components with internal surfaces~~ exposed to plant indoor air, ventilation atmosphere, atmosphere/weather, condensation, borated water leakage, diesel exhaust, lubricating oil, and water system environment not managed by Open-Cycle Cooling Water System (A1.10), Closed Treated Water System (A1.11), Fire Water System (A1.14), and Water Chemistry (A1.2) 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.

If work opportunities are not sufficient to allow inspection of a representative sample of components, supplemental inspections are also performed. A representative sample size is 20 percent of the accessible and inaccessible component population (defined as components having the same material and environment combination) up to a maximum of 25 components. The locations and intervals for supplemental inspections are based on assessments of the likelihood of significant aging effects, derived from current industry and plant-specific operating experience.

Components having the same material-environment combination with repetitive failures due to aging require a plant-specific program, unless the component material has been replaced by a material of more corrosion resistance for the environment of interest.

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.

A2.1 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, and plant-specific bounding EAF locations. 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.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
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 system locations. The set includes the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant and plant-specific bounding EAF locations. • ensure the scope includes the fatigue crack growth analyses, which support the leakbefore-break analyses, ASME Section XI evaluations, and the HELB break 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 Corrective Action Program and corrective actions are taken. • include additional transients that contribute significantly to fatigue usage. These additional transients were identified by evaluation of ASME Section III fatigue and fatigue crack growth analyses. • include additional locations which receive more detailed monitoring. These locations were~~are~~ identified by evaluation of ASME Section III fatigue analyses and the locations evaluated for effects of the reactor coolant environment. The monitoring methods will be benchmarked consistent with the NRC RIS 2008-30. • 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 3 fuel cycles. The fatigue results associated with the NUREG/CR-6260 sample locations for a newer vintage Westinghouse plant and plant-specific bounding environmental-assisted fatigue 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 use in the hot leg surge nozzle EAF analysis.	B3.1	Prior to the period of extended operation

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
	• 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. Re-analysis 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.		

B2.1.9 Steam Generators

Program Description

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

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

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

The Steam Generators program detects flaws in tubing, plugs, and tube supports needed to maintain tube integrity. Nondestructive examination (NDE) techniques are used to inspect all tubing materials to identify tubes that may need to be removed from service or repaired in accordance with plant technical specifications. The program provides criteria for the qualification of personnel, specific techniques, and the associated acquisition and analysis of data, including procedures, probe selection, analysis protocols, and reporting criteria. Assessment of tube integrity and plugging or repair criteria of flawed tubes is in accordance with plant technical specifications and the program implementing procedures. ~~Tube structural integrity limits consistent with Regulatory Guide 1.121, *Bases for Plugging Degraded PWR Steam Generator Tubes*, are applied as detailed in Callaway Technical Specifications, Section 5.5.9.~~ Plugs and tube supports with aging are evaluated for corrective actions in accordance with the Callaway Corrective Action Program and the Callaway Steam Generators program. Condition monitoring assessments are performed to determine whether structural and accident leakage criteria have been satisfied. Operational assessments are performed after inspections to verify that structural and leakage integrity will be maintained for the operating interval between inspections, which is selected in accordance with the technical specifications and NEI 97-06 guidelines. Comparison of the results of the condition monitoring assessment with the predictions of the previous operational assessment provides feedback for evaluation of the adequacy of the operational assessment and additional insights that can be incorporated into the next operational assessment.

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

NUREG-1801 Consistency

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

Exceptions to NUREG-1801

None

Enhancements

None

Operating Experience

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

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

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

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

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

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

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

Conclusion

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

B2.1.14 Fire Water System

Program Description

The Fire Water System program manages loss of material for water-based fire protection systems consisting of aboveground, buried and underground piping, fittings, valves, fire pump casings, sprinklers, nozzles, hydrants, hose stations, standpipes and water storage tanks. Periodic fire main and hydrant inspections and flushing, sprinkler inspections, functional test, and flow tests in accordance with National Fire Protection Association (NFPA) codes and standards ensure that the water-based fire protection systems are capable of performing their intended function. The fire protection system is maintained at the required normal operating pressure and monitored such that a loss of system pressure is immediately detected and corrective actions initiated.

The Fire Water System program performs a flow test of the system at least once every three years in accordance with plant procedures meeting the requirements of NFPA 25, including a yard fire loop flush and a flush of associated hydrants. A visual inspection of yard fire hydrants is performed annually.

The Fire Water System program conducts flow tests through each open head spray/sprinkler nozzle in accordance with NFPA 25, to verify water flow is unobstructed. The Fire Water System program requires replacement of sprinklers prior to 50 years in service, or the program tests a representative sample of the sprinklers and tests another representative sample every 10 years thereafter during the period of extended operation to ensure signs of aging are detected in a timely manner.

Pipe wall thickness examinations are performed on fire water piping using non-intrusive techniques. As an alternative to wall thickness ~~evaluations~~ examinations, internal inspections are performed on accessible exposed portions of fire water piping during plant maintenance activities. The inspections evaluate wall thickness measurements to ensure against catastrophic failure and the inner diameter of the piping as it applies to the design flow of the fire protection system. If a representative number of inspections have not been completed prior to the period of extended operation, Callaway will determine what additional inspections or examinations are required. The representative sample will be selected, based on system susceptibility to corrosion or fouling and evidence of performance degradation during system flow testing or periodic flushes. If material and environment conditions for above grade and below grade piping are similar, the results of the inspections of the internal surfaces of the above grade fire water piping can be extrapolated to evaluate the condition of the internal surfaces of the below grade fire water piping. If not, additional inspection activities will be performed to ensure that the intended function of below grade fire water piping will be maintained consistent with the current licensing basis. Pipe wall thickness examinations and/or internal inspections will be performed prior to the period of extended operation and at 10-year frequencies throughout the period of extended operation.

Functional tests are periodically performed on fire detectors to ensure that they are operable.

The fire water storage tank external surfaces are inspected and volumetric examinations of the tank bottom are performed as described in the Aboveground Metallic Tanks program (B2.1.15). External surfaces of buried fire main piping are evaluated as described in the Buried and Underground Piping and Tanks program (B2.1.25).

NUREG-1801 Consistency

The Fire Water System program is an existing program that, following enhancement, will be consistent, with exception to NUREG-1801, Section XI.M27, *Fire Water System*.

Exceptions to NUREG-1801

Program Element Affected:

Detection of Aging Effects (Element 4)

NUREG-1801 requires inspection of fire protection systems in accordance with the guidance of NFPA-25. Callaway performs power block hose station gasket inspections at least once every 18 months. The inspection interval is in accordance with the approved fire protection program, as described in FSAR Table 9.5.1-2 - SP, Section 5.4, rather than annually as specified by NFPA-25.

NUREG-1801 requires annual testing of fire hydrant hose. Callaway hydrostatically tests fire hoses at fire hose stations that are older than five years at least every three years. The testing interval is in accordance with the approved fire protection program, as described in FSAR Table 9.5.1-2 - SP, Section 5.6.

NUREG-1801 requires fire hydrant flow tests to be performed annually. Callaway performs a yard loop and hydrant flush at least once every three years, and a flow test of the system at least once every three years. The testing interval is in accordance with the approved fire protection program, as described in FSAR Table 9.5.1-2 - SP, Sections 2.4 and 2.7.

Enhancements

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

Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4, and Acceptance Criteria (Element 6)

The Fire Water System program will be enhanced to include non-intrusive pipe wall thickness examinations on fire water piping. As an alternative to wall thickness examinations, internal inspections will be performed on accessible exposed portions of fire water piping during plant maintenance activities. Pipe wall thickness examinations and/or internal inspections will be performed prior to the period of extended operation and at 10-year frequencies throughout the period of extended operation.

Detection of Aging Effects (Element 4)

The Fire Water System program will be enhanced to replace sprinkler heads prior to 50 years in service or test a representative sample and test every 10 years thereafter to ensure signs of degradation are detected in a timely manner.

Monitoring and Trending (Element 5)

The Fire Water System program will be enhanced to review and evaluate trends in flow parameters recorded during the NFPA 25 fire water flow tests.

Operating Experience

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

1. In 2005, during a surveillance test, 10 sprinkler heads had signs of corrosion or mechanical damage. Two of the sprinkler heads were replaced, and the other eight were cleaned. There have been no additional issues with the sprinkler heads since then.

2. In 2005, an alarm was triggered for fire protection loop jockey pump excessive run time and an investigation was initiated to identify the leak. The location of the leak was determined and promptly isolated from the main fire water loop. The isolation of the leak did not affect any required suppression systems. The leak was promptly repaired and the fire water piping was returned to service.

3. In 2006, a low C-factor lead to the fire water system being chemically cleaned, resulting in removal of approximately 8900 pounds of corrosion products. The cleaning was successful in keeping the system C-factor above 91.5 as required by plant procedure. During the chemical cleaning, five leaks developed, all of which were repaired. Since that time, two additional leaks have occurred. One was due to a cracked valve, and the cause of the other is still under investigation.

4. In 2008, during microbiological sampling of the fire water system, elevated levels of microbiologically influenced corrosion (MIC) were detected in stagnant portions of fire water pipe supplying fire water to hose stations. As a result, a new preventive maintenance task has been created to flush hose stations with a biocide.

5. In 2011, C-factor testing was performed on the main fire loop piping to check for restrictions due to corrosion and or biofouling. The testing results did not meet the acceptance criteria, indicating excessive pressure drop leading to reduced fire water flow. The testing results were called into question so with more accurate digital crystal gauges, the system was reevaluated and the results improved by 6% to 89.5, still less than the required acceptance criteria of 91.5. A functionality determination

concluded that provided compensatory measures were taken, the reduced cleanliness could be fully offset so the required fire water flow rate could be achieved and maintained. As a corrective action, the acceptance criteria in Calculation KC-005 Addendum 2 have been modified, and the test procedure updated accordingly. These revisions provide significant margin and consider the cleanliness trends, ensuring the fire water system is capable of performing its intended function.

The above examples provide objective evidence that the existing Fire Water System program includes activities that are capable of detecting aging effects, evaluating system leakage, and initiating corrective actions. Occurrences that would be identified under the Fire Water System 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 Fire Water System program will effectively identify aging prior to loss of intended function.

Conclusion

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

B2.1.17 Reactor Vessel Surveillance

Program Description

The Reactor Vessel Surveillance program manages loss of fracture toughness in accordance with ASTM E 185-73, and the requirements of 10 CFR 50 Appendix H. The surveillance capsules contain reactor vessel steel specimens of the limiting beltline material; and associated weld metal and weld heat affected zone metal. Current examination methods and report requirements are also controlled by commitment to ASTM E 185-82.

The last-tested surveillance capsule removed from the reactor vessel, Capsule X, was exposed to fluences equivalent to about 54 effective full power years (EFPY), 3.33×10^{19} neutrons/cm^2 based on the calculated fluence. This fluence~~, which~~ exceeds the 60-year peak reactor vessel wall neutron fluence. Capsule results are used to demonstrate compliance with Charpy upper-shelf energy requirements in 10 CFR 50 Appendix G and pressurized thermal shock screening criteria in 10 CFR 50.61, using the methodologies in Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials,* Revision 2. Capsule results are also used to revise pressure-temperature curves and project the end-of-life fluence.

Two standby capsules will be removed at exposures greater than those expected at the beltline wall at 60 years. ~~One capsule~~Capsule Z was removed at 71 EFPY of equivalent exposure and is stored in the spent fuel pool for reinsertion or testing as deemed appropriate. ~~The other capsule~~Capsule W will be removed at approximately 108 EFPY of equivalent exposure. This withdrawal schedule meets the ASTM E 185-82 criterion which states that capsules may be removed when the capsule neutron fluence is between one and two times the limiting fluence calculated for the vessel at the end of expected life. Changes to the capsule withdrawal schedule will be communicated and approved by the NRC as appropriate.

Following withdrawal of the final capsule, vessel fluence will be determined by ex-vessel dosimetry.

NUREG-1801 Consistency

The Reactor Vessel Surveillance program is an existing program that, following enhancement, will be consistent to NUREG-1801, Section XI.M31, *Reactor Vessel Surveillance*.

Exceptions to NUREG-1801

None

Enhancements

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

Detection of Aging Effects - Element 4

Following withdrawal of the final capsule, vessel fluence will be determined by ex-vessel dosimetry.

Testing specification will be enhanced to require that pulled and tested surveillance capsules are placed in storage for future reconstitution or reinsertion unless given NRC approval to discard.

Monitoring and Trending - Element 5

Procedures will be enhanced to specifically require the evaluation of the impact of plant operation changes on the extent of reactor vessel embrittlement (i.e., Charpy upper-shelf energy and pressurized thermal shock screening criteria, and the P-T limit curves, including the effect of lower cold leg temperature or higher fluence).

Operating Experience

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

1. The last-tested capsule specimens in Capsule X were exposed to fluences equivalent to approximately 54 EFPY, 3.33 x 10^{19} neutrons/cm^2 based on the calculated fluence, and satisfy the upper-shelf energy criterion and the pressurized thermal shock reference temperature screening criteria. The adjusted reference temperatures have been shown to be less than that used in the P-T limit curves, thereby demonstrating margin in the operating limits.

The operating experience of the Reactor Vessel Surveillance program did not identify an adverse trend in performance. Occurrences that would be identified under the Reactor Vessel Surveillance 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 Reactor Vessel Surveillance program will effectively identify aging prior to loss of intended function.

Conclusion

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

B2.1.23 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

Program Description

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program manages cracking, loss of material, hardening and loss of strength. The program inspects internal surfaces of metallic piping, piping components, ducting, polymeric components, and other components that are exposed to plant indoor air, ventilation atmosphere, atmosphere/weather, condensation, borated water leakage, diesel exhaust, lubricating oil, and any water system environment not managed by Open-Cycle Cooling Water System (B2.1.10), Closed Treated Water System (B2.1.11), Fire Water System (B2.1.14), and Water Chemistry (B2.1.2) programs.

Internal inspections are performed opportunistically whenever 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 of at least 10 percent of the accessible surface area or pressurization to detect hardening, loss of strength, and cracking. Volumetric evaluations are performed when appropriate for the component environment and material. Volumetric evaluations such as ultrasonic examinations are used to detect stress corrosion cracking of internal surfaces such as stainless steel components exposed to diesel exhaust.

If work opportunities are not sufficient to allow inspection of a representative sample of components, supplemental inspections are also performed. A representative sample size is 20 percent of the accessible and inaccessible component population (defined as components having the same material and environment combination) up to a maximum of 25 components. The locations and intervals for supplemental inspections are based on assessments of the likelihood of significant aging effects, derived from current industry and plant-specific operating experience.

Identified aging deficiencies are documented and evaluated by the Corrective Action Program. Acceptance criteria are established in the maintenance and surveillance procedures or are established during engineering evaluation of the degraded condition. If the inspection results are not acceptable, the condition is evaluated to determine whether the component intended function is affected, and a corrective action is implemented.

Components having the same material-environment combination with repetitive failures due to aging ~~loss of material from corrosion, and all similar component-material-environment combinations,~~ require a plant-specific program, unless the component material has been replaced by a material of more corrosion resistance for the environment of interest.

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

NUREG-1801 Consistency

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new program that, when implemented, will be consistent with exception to NUREG-1801, Section XI.M38, *Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components*.

Exceptions to NUREG-1801

<u>Program Elements Affected:</u>

Scope of Program (Element 1), Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), and Monitoring and Trending (Element 5)

NUREG-1801 requires a visual examination of the internal surface of components within the scope of this program. The diesel exhaust is not available for internal surface inspection, so a volumetric examination will be performed for this component. The volumetric examination is adequate for detecting loss of material (wall thinning) and cracking of piping and tubing.

Enhancements

None

Operating Experience

The following discussion of operating experience provides objective evidence that the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will be effective in ensuring that intended functions are maintained consistent with the current licensing basis for the period of extended operation.

1. The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will be a new program at Callaway. Internal surface monitoring through visual inspections conducted during maintenance activities and surveillance testing are already in effect in Callaway. The results of the inspections provide data for performance trending, are an input to work planning and prioritization process, and are communicated in the System Health Reports and System Performance Monitoring Indicators. Plant-specific operating experience since 2000 was reviewed to ensure that the operating experience discussed in the corresponding NUREG-1801 aging management program is bounding, i.e., that there is no unique plant-specific operating experience in addition to that described in NUREG-1801. The review also showed that the Plant Health and Performance Monitoring Program had been effective in maintaining the condition of component internal surfaces.

2. In 2007, during maintenance activities, the threaded tube end plugs on the “B” centrifugal charging pump room cooler were found to have a loss of material due to corrosion as introduced by wear and deformation to the plugs from the repeated assembly/disassembly and cleanings. None of the plugs were leaking. An evaluation determined that 125 plugs would be replaced, future inspections of the room cooler coils would include inspection of tube plugs for loss of material due to corrosion, and replacements would be determined on a case-by-case basis. Later in 2007, the 'A' containment spray pump room cooler was inspected. There was no noticeable damage to the plugs in this cooler. An additional corrective action was to ensure a continuous on-site availability of enough plugs to replace all the plugs in one room cooler.

Internal inspections conducted during maintenance activities and surveillance testing and the Plant Health and Performance Monitoring Program have been effective in maintaining the condition of component internal surfaces. Occurrences that would be identified under the Internal Surfaces in Miscellaneous Piping and Ducting Components 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 implementation of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will effectively identify aging prior to loss of intended function.

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

Conclusion

The implementation of the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program will provide 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.

B3.1 FATIGUE MONITORING

Program Description

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 thermal and pressure transient numbers and severities analyzed in the design calculations, or that corrective actions maintain the design and licensing basis.

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~~six stress-tensors~~ 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~~Ameren Missouri will benchmark the method~~ consistent with the NRC Regulatory Issue Summary RIS 2008-30.

The Fatigue Monitoring program requires periodic reviews of the plant instrumentation and operator logs to ensure that the fatigue critical thermal and pressure transients have not exceeded design transient severity or analyzed number, and to ensure that usage factors will not exceed the allowable value of 1.0 without corrective actions.

The Fatigue Monitoring program will be enhanced to include the effects of the reactor coolant environment on component fatigue life for a set of sample reactor coolant system locations. The set includes fatigue monitoring of the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant and plant-specific bounding environmentally assisted fatigue (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.

NUREG-1801 Consistency

The Fatigue Monitoring program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section X.M1, *Fatigue 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 fatigue usage calculations that consider the effects of the reactor water environment for a set of sample reactor coolant system locations. The set includes the NUREG/CR-6260 sample locations for a newer-vintage Westinghouse Plant, and plant-specific bounding EAF locations.

Procedures will be enhanced to ensure the fatigue crack growth analyses, which support the leak-before-break analyses, ASME Section XI evaluations, and the HELB break selection criterion remain valid by counting the transients used in the analyses.

Preventive Actions - Element 2

Procedures will be enhanced to 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 Corrective Action Program and corrective actions are taken.

Parameters Monitored or Inspected - Element 3

Procedures will be enhanced to include additional transients that contribute significantly to fatigue usage. These additional transients were identified by evaluation of ASME Section III fatigue and fatigue crack growth analyses.

Procedures will be enhanced to include additional locations which receive more detailed monitoring. These locations were~~are~~ identified by evaluation of ASME Section III fatigue analyses and the locations evaluated for effects of the reactor coolant environment. The monitoring methods will be benchmarked consistent with the NRC RIS 2008-30.

Monitoring and Trending - Element 5

Procedures will be enhanced to project the transient count and fatigue accumulation of monitored components into the future.

Acceptance Criteria - Element 6

Procedures will be enhanced to 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.

Corrective Actions - Element 7

Procedures will be enhanced to 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. Re-analysis 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.

Operating Experience

The following discussion of operating experience provides objective evidence that the Fatigue 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. In response to NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, Westinghouse performed a plant-specific evaluation of Callaway pressurizer surge line. It was concluded that thermal stratification does not affect the integrity of the pressurizer surge line. Callaway responses to NRC Bulletin 88-11 describe the inspections, analyses, and procedural revisions made to ensure that thermal stratification does not affect the integrity of the pressurizer surge line. There have been no signs of damage from surge line movement.

2. NRC Regulatory Issue Summary RIS 2008-30, *Fatigue Analysis Of Nuclear Power Plant Components* informed licensees of analysis methodology (Green's function) used to demonstrate compliance with the ASME Code fatigue acceptance criteria could be non-conservative if not correctly applied. Ameren Missouri is committed to using a three dimensional, six component stress tensor method meeting ASME III NB-3200 requirements, or benchmarking the chosen method. This benchmarking has been performed for the normal and alternate charging nozzle to order to implement SBF at that location. Any additional locations which will be monitored with SBF must meet the ASME III NB-3200 requirements or be benchmarked.

3 An error was identified in the previous SBF transfer function for the normal and alternate charging nozzles. The SBF transfer function incorrectly included thermal sleeves for the nozzles and therefore would calculate less fatigue than the nozzles

would actually accumulate. The extent of this condition is limited only to the normal and alternate charging nozzle SBF models. The transfer function has been updated to exclude thermal sleeves. The SBF transfer functions for the normal and alternate charging nozzles were also benchmarked in accordance with NRC RIS 2008-30.

4. The CVCS design specification identifies the nominal letdown flow of 75 gpm with maximum flow of 120 gpm. Callaway operated from 1993 to 2011 at the maximum letdown flow, but has returned to the nominal value of 75 gpm. The effects of this increased flow rate have been evaluated. To account for the increase in fatigue, Callaway reduced the assumed number of load following transients to be more consistent with its operation as a base load plant. Also, starting in Refuel Outage 17, Callaway has switched from the normal charging flow path to the alternate charging flow path in order to spread fatigue over the two paths.

The operating experience of the Fatigue Monitoring program did not identify an adverse trend in performance. Occurrences that would be identified under the Fatigue 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 Fatigue Monitoring program will effectively identify aging prior to loss of intended function.

Conclusion

The continued implementation of the Fatigue 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.