Ameren MISSOURI

Callaway Plant

May 3, 2012

ULNRC-05860

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 2.101
10 CFR 2.109(b)
10 CFR 50.4
10 CFR 50.30
10 CFR 51.53(c)
10 CFR 54

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
AMENDMENT 2 TO APPLICATION
FOR RENEWED OPERATING LICENSE

References: 1. ULNRC-05830 dated December 15, 2011
2. ULNRC-05856 dated April 25, 2012

By Reference 1, Union Electric Company (Ameren Missouri) submitted a license renewal application (LRA) for Callaway Plant Unit 1. Reference 2 transmitted Amendment 1 to the Callaway LRA, and the purpose of this letter is to provide Amendment 2 to the Callaway LRA.

The changes being made by Amendment 2 are contained in Enclosure 1 and Enclosure 2. Enclosure 1 identifies Callaway time-limited aging analyses (TLAA) changes associated with reactor vessel underclad cracking analysis, limiting locations for environmental assisted fatigue, and other TLAA changes. Enclosure 2 identifies Callaway aging management review (AMR) changes that are being made as follows:

- In LRA Table 3.2.2-5, correct the identified valve material to indicate "carbon steel" for the high pressure coolant injection stainless steel valve in an "atmosphere/weather" environment (external) and the associated internal steam environment AMR lines.

- In LRA Table 3.3.2-5, correct the identified environment from "plant indoor air" to "atmosphere/weather" for the ductile iron valve AMR line in the service water system. In addition, carbon steel piping in an "atmosphere/weather" environment for the service water system will also be added to the scope of license renewal. Reference Section 3.3.2.1.5 of the LRA.

It should be noted that changes to one commitment (Item #37) are reflected in Table A4-1 (within Enclosure 1).

If you have any questions on LRA Amendment 2, please contact me at (573) 823-9286 or Ms. Sarah Kovaleski at (314) 225-1134.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Executed on: May 3, 2012

Les H. Kanuckel

Les H. Kanuckel
Manager, Engineering Design

DS/SGK/nls

Enclosures: 1. Callaway Plant Unit 1 License Renewal Application Amendment No. 2 Time-Limited Aging Analyses Changes
2. AMR Changes for Callaway Plant Unit 1License Renewal Application Amendment No. 2

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ENCLOSURE 1

Callaway Plant Unit 1
License Renewal Application
Amendment No. 2
Time-Limited Aging Analyses Changes

- **Further Evaluation 3.1.2.2.5 Crack Growth due to Cyclic Loading**
- **Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals**
- **Table 4.1-1 List of TLAAs (TLAA Category 5 and 6)**
- **Table 4.1-2 Review of Analyses Listed in NUREG-1800 Tables 4.1-2 and 4.1-3**
- **Section 4.3.4 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)**
- **Section 4.6 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses**
- **Section 4.7.2 In-Service Flaw Analyses that Demonstrate Structural Integrity for 40 years**
- **Section 4.7.4 Reactor Vessel Underclad Cracking Analysis**
- **4.8 References**
- **Appendix A3.2.3 Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)**
- **Appendix A3.6.4 Reactor Vessel Underclad Cracking Analysis**
- **Table A4-1 Item 37**

Callaway LRA Amendment 2
Time Limited Aging Analysis Changes
Affected Pages

LRA Section	Page Nos
3.1.2.2.5	3.1-10
Table 3.1.2-1	3.1-61 and 66
Table 4.1-1	4.1-5
Table 4.1-2	4.1-6 and 7
4.3.4	4.3-31 – 37
4.6	4.6-1
4.7.2	4.7-4
4.7.4	4.7-8 and 9
4.8	4.8-2
A3.2.3	A-27
A3.6.4	A-34 and 35
Table A4-1	A-49

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Revision to further evaluation to revise disposition of TLAA for underclad cracking.

Section 3.1.2.2.5 (page 3.1-10) is revised as follows (deleted text shown with strike through):

3.1.2.2.5 Crack Growth due to Cyclic Loading

An analysis of crack growth of underclad flaws in reactor vessel forgings due to cyclic loading to qualify them for the current licensed operating period would be a TLAA. ~~This phenomenon has been addressed in the Callaway vessel by weld cladding processes designed to avoid these defects.~~

~~No underclad cracks have been detected or analyzed for the Callaway vessel, in the absence of which there are no TLAAs.~~ Section 4.7.4 describes the ~~absence of a~~ TLAA for underclad cracking.

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Revision to Table 3.1.2-1 to add additional TLAA lines for crack growth due to cyclic loading.

Table 3.1.2-1 (pages 3.1-61 and 66) is revised as follows (new text shown underlined):

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

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
RV Inlet and Outlet Nozzles	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Crack growth due to cyclic loading	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2.R-85	3.1.1.018	A
RV Upper, Intermediate, Lower Shell and Welds	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Crack growth due to cyclic loading	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.A2.R-85	3.1.1.018	A

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Revision to Table 4.1-1 to revise titles and dispositions.

Table 4.1-1 (page 4.1-5) is revised as follows (deleted text shown with strikethrough, new text underlined):

Table 4.1-1 List of TLAAs

TLAA Category	Description	Disposition Category(1)	Section
1.	**Reactor Vessel Neutron Embrittlement Analysis**	N/A	4.2
	Neutron Fluence Values	ii	4.2.1
	Charpy Upper-Shelf Energy	ii	4.2.2
	Pressurized Thermal Shock	ii	4.2.3
	Pressure-Temperature (P-T) Limits	iii	4.2.4
	Low Temperature Overpressure Protection	iii	4.2.5
2	**Metal Fatigue**	N/A	4.3
	Fatigue Monitoring Program	N/A	4.3.1
	ASME Section III Class I Fatigue Analysis of Vessels, Piping and Components	iii	4.3.2
	Reactor Coolant Pump Thermal Barrier Flange	iii	4.3.2.1
	Pressurizer Insurge-Outsurge Transients	iii	4.3.2.2
	Steam Generator ASME Section III Class 1, Class 2 Secondary Side, and Feedwater Nozzle Fatigue Analyses	i	4.3.2.3
	NRC Bulletin 88-11 Revised Fatigue Analysis of the Pressurizer Surge Line for Thermal Cycling and Stratification	iii	4.3.2.4
	ASME Section III Subsection NG Fatigue Analysis of Reactor Pressure Vessel Internals	iii	4.3.3
	Effects of the Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)	iii	4.3.4
	Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction Factor in ANSI B31.1 and ASME Section III Class 2 and 3 Piping	i	4.3.5

Table 4.1-1 List of TLAAs

TLAA Category	Description	Disposition Category[1]	Section
	Fatigue Design of Spent Fuel Pool Liner and Racks for Seismic Events	i	4.3.6
	Fatigue Design and Analysis of Class 1E Electrical Raceway Support Angle Fittings for Seismic Events	i	4.3.7
	Fatigue Analyses of Class 2 Heat Exchangers	ii, iii	4.3.8
3.	**Environmental Qualification (EQ) of Electric Equipment**	**iii**	**4.4**
4.	**Concrete Containment Tendon Prestress**	**i, ii**	**4.5**
5.	**Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses**	**N/A**	**4.6**
	Design Cycles for the Main Steam Line and Feedwater Penetrations	i, ii	4.6.1
	Fatigue Waiver Evaluations for the Access~~Equipment~~ Hatches and Leak Chase Channels	i	4.6.2
6.	**Other Plant-Specific Time-Limited Aging Analyses**	**N/A**	**4.7**
	Containment Polar Crane, Fuel Building Cask Handling Crane, Spent Fuel Pool Bridge Crane, and Refueling Machine CMAA 70 Load Cycle Limits	i	4.7.1
	In-service Flaw Analyses that Demonstrate Structural Integrity for 40 years	i	4.7.2
	Corrosion Analysis of the Reactor Vessel Cladding Indications	i	4.7.3
	~~Absence of a TLAA for~~ Reactor Vessel Underclad Cracking Analyses	~~N/A~~i	4.7.4
	Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis	i	4.7.5
	High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factors	iii	4.7.6
	Fatigue Crack Growth Assessment in Support of a Fracture Mechanics Analysis for the Leak-Before-Break (LBB) Elimination of Dynamic Effects of Piping Failures	i	4.7.7
	Replacement Class 3 Buried Piping	i	4.7.8
	Replacement Steam Generator Tube Wear	i	4.7.9

**Callaway Plant
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Revision to Table 4.1-2 to revise applicability to Callaway Plant.

Table 4.1-2 (pages 4.1-6 and 7) is revised as follows (deleted text shown with strikethrough, new text underlined):

NUREG-1800 Examples	Applicability to Callaway	Section
NUREG-1800, Table 4.1-2 – Potential TLAAs		
Reactor Vessel Neutron Embrittlement	Yes	4.2
Metal Fatigue	Yes	4.3
Environmental Qualification (EQ) of Electric Equipment	Yes	4.4
Concrete Containment Tendon Prestress	Yes	4.5
In-Service Local Metal Containment Corrosion Analyses	<u>No – No explicit basis based on plant life applies.</u>~~Yes~~	~~4.7.3~~ <u>-</u>
NUREG-1800, Table 4.1-3 – Additional Examples of Plant-Specific TLAAs		
Intergranular Separation in the Heat-Affected Zone (HAZ) of Reactor Vessel Low-Alloy Steel Under Austenitic SS Cladding	~~No – No HAZ analyses were identified within the CLB.~~ <u>Yes</u>	4.7.4
Low-Temperature Overpressure (LTOP) Analyses	Yes	4.2.5
Fatigue Analysis for the Main Steam Supply Lines to the Turbine-Driven Auxiliary Feedwater Pumps	Yes	4.3.5
Fatigue Analysis for the Reactor Coolant Pump Flywheel	Yes	4.7.5
Fatigue Analysis of Polar Crane	Yes	4.7.1
Flow-Induced Vibration Endurance Limit for the Reactor Vessel Internals	No-No explicit basis based on plant life applies.	4.3.3
Transient Cycle Count Assumptions for the Reactor Vessel Internals	Yes	4.3.3
Ductility Reduction of Fracture Toughness for the Reactor Vessel Internals	No-No explicit basis based on plant life applies.	4.3.3
Leak Before Break	Yes	4.7.7

NUREG-1800 Examples	Applicability to Callaway	Section
Fatigue Analysis for the Containment Liner Plate	No – No fatigue or cycle-based analysis supports design of the liner.	4.6.0
Containment Penetration Pressurization Cycles	Yes	4.6.2
Metal Corrosion Allowance	~~No-No explicit basis based on plant life applies.~~ Yes	4.7.3~~--~~
High-Energy Line-Break Postulation Based on Fatigue Cumulative Usage Factor	Yes	4.7.6
In-Service Flaw Growth Analyses that Demonstrate Structure Stability for 40 Years	Yes	4.7.2

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Revision to Section 4.3.4.

Section 4.3.4 (pages 4.3-31 through 37) is revised as follows (deleted text shown with strikethrough, new text underlined):

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

The NRC concluded that effects of the reactor coolant environment might need to be included in the calculated fatigue life of components, and opened three generic safety issues to address this question, all finally closed to a single Generic Safety Issue 190. Subsequent research and studies refined the methods, which no longer use the interim fatigue curves of NUREG/CR-5999 but calculate an environmental fatigue effect multiplier F_{en}, which depends on material type, temperature, strain rate, and dissolved oxygen; and for carbon and low-alloy steel, sulfur content.

NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* states that "The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review," noting the staff recommendation "...that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal."

The GSI-190 review requirements are therefore imposed by the Standard Review Plan and do not depend on the individual plant licensing basis. Callaway addressed GSI-190 review requirements by assessing the environmental effect on fatigue at the NUREG/CR-6260 locations for the newer-vintage Westinghouse Plant.

NUREG/CR-6260 identifies seven sample locations for newer vintage Westinghouse plants which need to consider the effects of reactor coolant environment on component fatigue life for license renewal:

1. Reactor Vessel Lower Head to Shell Juncture
2. Reactor Vessel Primary Coolant Inlet Nozzle
3. Reactor Vessel Primary Coolant Outlet Nozzle
4. Hot Leg Surge Nozzle
5. Charging Nozzles
6. Safety Injection Nozzles
7. Residual Heat Removal Line Inlet Transition

Table 4.3-6, *Summary of Fatigue Usage Factors at NUREG/CR 6260 Sample Locations* is a summary of environmentally-assisted fatigue of the NUREG/CR-6260 locations. The F_{en} relationships are calculated from NUREG/CR-6583 for carbon and low-alloy steels and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each of these locations.

The NUREG/CR-6260 locations in Table 4.3-6, *Summary of Fatigue Usage Factors at NUREG/CR 6260 Sample Locations* with an EAF CUF below 1.0, when using the design basis

CUF and the maximum F_{en}, require no further analysis. Three of the NUREG/CR-6260 locations, (1) RPV lower head to shell juncture, (2) RPV inlet nozzle, and (3) RPV outlet nozzles meet this criterion (Reference 4). All three locations are low alloy steel locations.

The maximum F_{en} for low alloy steel assumes the dissolved oxygen level to be less than 0.05 ppm, which corresponds to a low oxygen environment. This is consistent with the Callaway primary chemistry program, which maintains RCS~~C~~ hydrogen level at 25 to 50 cc/kg. A minimum hydrogen concentration will ensure the RCS is free of oxygen. Sulfur content is assumed to be at the maximum concentration in the NUREG.

The remaining NUREG/CR-6260 locations were reevaluated with a refined fatigue analysis using NB-3200 methods in a 3-D finite element analysis model using the design number of transients to reduce the CUF values. After reanalysis the RHR inlet transition was the only location to pass the EAF CUF criterion of 1.0 (Reference 5).

Two options are available to further reduce the EAF CUFs for the charging system nozzles, safety injection nozzles, and hot leg surge line nozzle: (1) calculate a strain rate dependent F_{en}; and (2) calculate CUF based on the 60 year projected numbers of transient events or both.

Revision of F_{en} Based on Strain-Rate

The strain-rate dependent F_{en} values are calculated for the significant load set pairs in the fatigue analyses. Load set pairs that produce no significant stress range or fatigue contribution were assigned the maximum F_{en} for the material. The integrated strain rate method described in MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application*, was used to calculate F_{en} values for individual load pairs that produce significant stress ranges. Dissolved oxygen of less than 0.05 ppm is assumed, which corresponds to a low oxygen environment. This is conservative since lower dissolved oxygen concentrations yield higher F_{en} values for stainless steel. Sulfur content is only applicable to low alloy steel locations.

Revision of CUF Based on 60-Year Projections of Transients

However, multiplying the revised CUF by the weighted average F_{en} value computed above still results in EAF CUFs greater than 1.0 for the charging system nozzles, safety injection nozzles, and hot leg surge line nozzle after conservatism has been removed. In order to demonstrate that monitoring fatigue in these locations is a sufficient form of aging management, the EAF CUF was calculated based on the numbers of transients projected to 60 years in Table 4.3-2, *Transient Accumulations and Projections*. If the transient is not projected, then the full number of design basis events is used. There were two transients that were not analyzed at the 60 year projection. The two exceptions are the "inadvertent safety injection" and "loss of power" events, which were analyzed ~~at~~ as the events to-date. The only EAF CUF calculation ~~significantly~~ that could be affected by the use of the "inadvertent safety injection" and "loss of power" transients to-date is associated with the safety injection nozzle, e.g. affected greater than the order of magnitude. This is addressed below.

The projected normal and alternate charging nozzles EAF CUFs are 0.57 and 0.53 based on SBF usage factors of 0.092 and 0.078, and F_{en} of 6.22 and 6.75 (Reference 6). The SBF usage factors were generated with computer software that was benchmarked against NB-3200 methods consistent with RIS 2008-30 as discussed in Section 4.3.1.1, *Fatigue Monitoring Methods*.

The projected safety injection nozzle EAF CUF is 0.74 based on the usage factor of 0.11 and F_{en} of 6.5 (Reference 7). Even though this location is analyzed for the numbers of "inadvertent safety injection" and "loss of power" events to-date, it is monitored with CBF; therefore EAF CUF will be updated as additional events occur.

The projected hot leg surge line nozzle EAF CUF is 0.765 based on the usage factor of 0.076 and F_{en} of 10.10 (Reference 8).

All of the locations specified in NUREG/CR-6260 for newer vintage Westinghouse plants listed in Table 4.3-6, *Summary of Fatigue Usage Factors at NUREG/CR 6260 Sample Locations* will be monitored by the Fatigue Monitoring program, described in Appendix B3.1. Most of the locations will be monitored using CBF or SBF. The hot leg surge nozzle will be monitored by incorporating the 60 year cycle projections into the cycle counting action limits to ensure that the results for the hot leg surge nozzle presented in this section are not exceeded. Therefore, 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).

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

Evaluation of Limiting Locations for Environmental Assisted Fatigue

In order to assure that the limiting plant-specific EAF locations are identified, Callaway performed a systematic review of all wetted, RCPB components with a Class 1 fatigue analysis [Ref. 17]. This was done either to show that the NUREG/CR-6260 locations are bounding or to incorporate EAF into the licensing basis for those more limiting components. The screening used EPRI Technical Report 1024995 "Environmentally-Assisted Fatigue Screening, Process and Technical Basis for Identifying EAF Limiting Locations," [Ref. 18]. ~~The first step in the screening was to apply the maximum F_{en} to all non-NUREG/CR-6260 locations using NUREG/CR-6583 for carbon and low alloy steels, NUREG/CR-5704 for austenitic stainless steels, and NUREG/CR-6909 for nickel alloys. For those locations with an EAF CUF less than 1.0, no further work is required. Those locations with an EAF CUF greater than 1.0 were categorized based on the strain-rate of the dominant transient. The strain-rate classification was determined with a qualitative assessment based on experience and not a quantitative stress analysis for the strain-rate classification. The strain-rate classification, an assumed low dissolved oxygen environment, the maximum fluid/metal temperature, and the maximum concentration for sulfur content were used to calculate the F_{en} value based on the same NUREGs used in the initial screening. The F_{en} and the design basis CUFs were used to identify the limiting EAF CUF for each material type in each system including those systems not considered in the NUREG/CR-6260 evaluation (e.g., steam generator primary side and pressurizer, pressurizer spray line, etc.). Table 4.3-7, *Preliminary Identification of Additional Sentinel Locations for EAF* identifies the locations, in addition to the NUREG/CR-6260 locations, which were determined to be candidate sentinel locations. The results presented in Table 4.3-7, *Preliminary Identification of Additional Sentinel Locations for EAF* are preliminary and do not represent the final list of bounding EAF locations. Prior to the period of extended operation Callaway will submit to the NRC for approval a finalized list of bounding EAF locations which will be monitored for EAF with the Fatigue Monitoring program. The supporting 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.~~

The CUF for wetted, RCPB locations were categorized based on the strain-rate of the dominant transient. The strain-rate classification was determined with a qualitative assessment based on experience and not a quantitative stress analysis. The estimated strain rate was used to calculate an estimated F_{en}. The estimated F_{en} value was also calculated assuming a low DO environment; the maximum fluid/metal temperature; and the maximum sulfur concentration, and is based on the methods in NUREG/CR-5704 for austenitic stainless steels, NUREG/CR-6583 for carbon and low alloy steels, and NUREG/CR-6909 for Ni-Cr-Fe steels. This estimated F_{en} was then averaged with the maximum F_{en} for that material type to calculate the average F_{en}. The average F_{en} and the design basis CUFs were used to calculate the estimated EAF CUF.

These estimated EAF CUFs were then organized according to their system, thermal zone, and material type. A thermal zone is defined as a collection of piping and/or vessel components which undergo essentially the same group of thermal and pressure transients during plant operations. The maximum EAF CUF for each thermal zone and material was selected as a sentinel location. In addition, if the next highest EAF CUF with the same thermal zone and material is within 50% of the maximum, additional locations were identified as sentinel locations. This initial list was reduced further using EPRI Technical Report 1024995 [Ref. 18].

- *One Thermal Zone can bound another Thermal Zone in a System:*
  Both the CUF and F_{en} values for one sentinel location in one thermal zone are each higher than the CUF and F_{en} values for the sentinel locations in other thermal zones.

- *One material in a Thermal Zone can bound other materials in the same Thermal Zone:*
  This circumstance could be achieved if within the same thermal zone, both the CUF and F_{en} values for one sentinel location composed of one material are each higher than the CUF and F_{en} values for the sentinel locations composed for all other materials.

- *One material in a Thermal Zone can bound other materials in another Thermal Zone:*
  This circumstance combines the guidelines of the two listed above and must satisfy both criteria listed.

- *A location with EAF CUF < 1.0 may be removed from the sentinel location list:*
  If the sentinel location EAF CUF for the projected number of design cycles is low (e.g., EAF CUF < 0.25), that sentinel location may be removed from the final list due to the small likelihood that it will be the leading sentinel location in a system. If, however, the sentinel location EAF CUF for the projected number of design cycles is fairly high (e.g., EAF CUF > 0.8), the possibility exists that it could remain the sentinel location for its group and should be included in the monitoring program that ensures that it does not exceed a value of 1.0.

Table 4.3-7 identifies the final locations, including the NUREG/CR-6260 locations, that will be used as sentinel locations during the period of extended operation to manage the EAF aging mechanism. Those non-NUREG/CR-6260 locations with an EAF CUF greater than 1.0 will be evaluated further using the same methods as those used to remove conservatisms for the NUREG/CR-6260 locations described above. 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 may 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. Therefore, the effects of the reactor coolant environment on the non-NUREG/CR-6260 locations will be managed for the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

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

Table 4.3-6 Summary of Fatigue Usage Factors at NUREG/CR 6260 Sample Locations

Location	Material	CUF	F_{en}	EAF CUF	CUF-F_{en} Basis
RPV Bottom Head to Shell Junction	SA 533, Grade B, Class 1, Low Alloy Steel	0.0070	2.45	0.01715	Design basis CUF NUREG/CR-6583 maximum F_{en}
RPV Inlet Nozzle	SA 508, Class 2, Low Alloy Steel	0.0795	2.45	0.195	Design basis CUF NUREG/CR-6583 maximum F_{en}
RPV Outlet Nozzle	SA 508, Class 2, Low Alloy Steel	0.1078	2.45	0.264	Design basis CUF NUREG/CR-6583 maximum F_{en}
Hot Leg Surge Line Nozzle	SA 182, Type 316, Stainless Steel	0.07572	10.097	0.7646	CUF re-evaluated with NB-3200 methods based on 60 year cycle projections, NUREG/CR-5704 strain-rate dependent F_{en}
Charging System Nozzle [Normal and Alternate]	SA 182 Type 316, Stainless Steel	0.0919 / 0.0782	6.22 / 6.75	0.5715 / 0.5273	CUF re-evaluated with SBF and 60 year cycle projections, NUREG/CR-5704 strain-rate dependent F_{en}
Safety Injection Nozzle [Boron Injection Header nozzles]	SA 182 Type 316, Stainless Steel	0.1135	6.495	0.7374	CUF re-evaluated with NB-3200 methods based on 60 year cycle projections, NUREG/CR-5704 strain-rate dependent F_{en}
Residual Heat Removal Inlet Nozzle [RHR nozzle-hot-leg]	SA 182 Type 316, Stainless Steel	0.0234	15.35	0.3591	CUF re-evaluated with NB-3200 methods based on design cycles, NUREG/CR-5704 maximum F_{en}

Table 4.3-7: Sentinel Locations for EAF Monitoring

System	Thermal Zone	Material	Component	NUREG /CR-6260	Design CUF	Avg. F_{en}	Est. EAF CUF
Reactor Pressure Vessel	RPV Nozzles	LAS	1. RPV Outlet Nozzle	Y	0.1078	2.455	0.265
			2. RPV Inlet Nozzle	Y	0.0795	2.455	0.195
	RPV Upper Head	SS	3. CETNA Upper Nozzle Housing	N	0.37	13.117	4.853
	RPV Bottom Head	LAS	4. Bottom Head-to-Shell Junction	Y	0.007	2.455	0.017
		Ni-Cr-Fe	5. Bottom Head Instrument Tubes	N	0.3184	4.093	1.303
Pressurizer	PZR Lower Head	SS	6. Pressurizer Heater Penetration	N	0.562	13.117	7.372
		LAS	7. Pressurizer Shell at Support Lug	N	0.992	2.455	2.435
			8. Pressurizer Surge Nozzle	N	0.963	2.455	2.364
			9. Pressurizer Lower Head/Support Skirt	N	0.734	2.455	1.802
	PZR Spray	SS	10. Pressurizer Spray Nozzle	N	0.411	9.013	3.704
	PZR SRV/PORV	SS	11. Safety and Relief Valve Piping	N	0.975	11.486	11.199
			12. Power Operated Relief Valve Solenoid	N	0.68	11.486	7.811
Surge Piping	Surge Line	SS	13. Hot Leg Surge Nozzle	Y	0.3	11.486	3.446

Table 4.3-7: Sentinel Locations for EAF Monitoring

System	Thermal Zone	Material	Component	NUREG /CR-6260	Design CUF	Avg. F_{en}	Est. EAF CUF
CVCS	Charging	SS	14. Normal Charging Nozzles, Loop 1	Y	0.90	7.240	6.516
			15. Alternate Charging Nozzles, Loop 4	Y	0.90	7.240	6.516
	Auxiliary Spray	SS	16. Auxiliary Spray Piping	N	0.72	5.970	4.298
RCS	RCS Cold Leg	SS	17. RCP Casing/Discharge Nozzle Junction	N	0.915	9.628	8.810
RHR	RHR Inlet (Suction)	SS	18. RHR Nozzles, Hot Leg Loops 1 & 4	Y	0.81	10.350	8.384
SI	BIT	SS	19. BIT Nozzles (All Loops)	Y	0.999	7.811	7.803
	Accumulator	SS	20. Accumulator Nozzles (All Loops)	N	0.95	7.811	7.420
	SIS	SS	21. Hot Leg SIS Nozzles, Loops 2 & 3	N	0.1	10.350	1.035
Steam Generator	Tubesheet	LAS	22. RSG Tubesheet (Continuous Region)	N	0.428	2.455	1.051

~~*Table 4.3-7 — Preliminary Identification of Additional Sentinel Locations for EAF*~~

~~System~~	~~Thermal Zone~~	~~Component~~	~~Recommended Candidate Sentinel Locations~~
~~Reactor Pressure Vessel~~	~~RPV Nozzle~~	~~RPV Inlet Nozzle~~	~~RPV Nozzle - 6260 - LAS~~
	~~RPV Nozzle~~	~~RPV Outlet Nozzle~~	~~RPV Nozzle - 6260 - LAS~~
	~~RPV Upper Head~~	~~RPV Core Exit Thermocouple Nozzle Assembly Upper Nozzle Housing~~	~~RPV Upper Head - SS~~
	~~RPV Upper Head~~	~~RPV Vessel Flange~~	~~RPV Upper Head - LAS~~
	~~RPV Bottom Head~~	~~RPV Bottom Head to Shell Juncture~~	~~RPV Bottom Head - 6260 - LAS~~

~~*Table 4.3-7 Preliminary Identification of Additional Sentinel Locations for EAF*~~

~~System~~	~~Thermal Zone~~	~~Component~~	~~Recommended Candidate Sentinel Locations~~
	~~RPV Bottom Head~~	~~RPV Bottom Head Instrument Tubes (pos. 2)~~	~~RPV Bottom Head - Ni-Cr-Fe~~
~~Pressurizer~~	~~Pressurizer Lower Head~~	~~Pressurizer Heater Penetration~~	~~Pressurizer Lower Head - SS~~
	~~Pressurizer Lower Head~~	~~Pressurizer Surge Nozzle~~	~~Pressurizer Lower Head - SS~~
	~~Pressurizer Lower Head~~	~~Pressurizer Shell at Support Lug~~	~~Pressurizer Lower Head - SS~~
	~~Pressurizer Lower Head~~	~~Pressurizer Lower Head/Support Skirt~~	~~Pressurizer Lower Head - LAS~~
	~~Pressurizer Upper Head~~	~~Pressurizer 6-inch and 3-inch Pressurizer Safety and Relief Valve Piping~~	~~Pressurizer Upper Head - SS~~
	~~Pressurizer Upper Head~~	~~Pressurizer 3-inch x 6-inch Power Operated Relief Valve Solenoid~~	~~Pressurizer Upper Head - SS~~
	~~Pressurizer Upper Head~~	~~Pressurizer Upper Head/Upper Shell~~	~~Pressurizer Upper Head - LAS~~
~~Surge Piping~~	~~Surge Line Piping~~	~~14-inch Hot Leg Surge Nozzle~~	~~Surge Line Piping - 6260 - SS~~
~~Spray Piping~~	~~Spray Line Piping~~	~~4-inch Spray Piping at Pressurizer Spray Nozzle~~	~~Spray Piping - SS~~
~~CVCS~~	~~Charging~~	~~CVCS 3-inch Cold Leg Loop 1 Normal Charging Nozzle~~	~~Charging - 6260 - SS~~
	~~Charging~~	~~CVCS 3-inch Cold Leg Loop 4 Alternate Charging Nozzle~~	~~Charging - 6260 - SS~~
	~~Letdown~~	~~CVCS 3-inch Normal Letdown, Crossover Loop 3~~	~~Letdown - SS~~
	~~Letdown~~	~~CVCS 2-inch Crossover Leg Loop 4 Excess Letdown Nozzle~~	~~Letdown - SS~~
	~~Auxiliary Spray~~	~~Auxiliary Spray Piping~~	~~Auxiliary Spray - SS~~
	~~Drain~~	~~CVCS Drain Line, Loop 2~~	~~Drain - SS~~
	~~Drain~~	~~CVCS Drain Line, Loop 3~~	~~Drain - SS~~
	~~Seal Water~~	~~CVCS 1-1/2-inch, 2-inch Seal Water Injection Loops 3 Piping~~	~~Seal Water - SS~~
~~RCS~~	~~RCS Cold Leg~~	~~RCS 2-inch Crossover Leg Loops 1, 2 Drain Nozzles~~	~~RCS Cold Leg - CS~~
	~~RCS Cold Leg~~	~~RCP Casing/Discharge Nozzle Junction~~	~~RCS Cold Leg - CS~~
	~~RCS Hot Leg~~	~~RCS Hot Leg Loops 1, 2, 3, 4~~	~~RCS Hot Leg - CS~~
~~RHR~~	~~RHR Inlet~~	~~RHR 12-inch Hot Leg Loops 1, 4 RHR Nozzles~~	~~RHR Inlet - 6260 - SS~~
~~SI~~	~~BIT~~	~~SI 3-inch Cold Leg (All Loops) Boron Injection Nozzle~~	~~BIT - 6260 - SS~~

~~*Table 4.3-7 Preliminary Identification of Additional Sentinel Locations for EAF*~~

~~System~~	~~Thermal Zone~~	~~Component~~	~~Recommended Candidate Sentinel Locations~~
	~~Accumulator~~	~~SI 10-inch Cold Leg (All Loops) Accumulator Nozzle~~	~~Accumulator - SS~~
~~Steam Generator~~	~~Primary Head~~	~~RSG Primary Manway Drain Tube~~	~~Primary Head - LAS~~
	~~Primary Head~~	~~RSG Primary Manway Cover~~	~~Primary Head - LAS~~
	~~Primary Head~~	~~RSG Primary Nozzle Drain Tube~~	~~Primary Head - LAS~~
	~~Tubesheet~~	~~RSG Tubesheet (Continuous Region)~~	~~Tubesheet - LAS~~

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Revision to Section 4.6.

Section 4.6 (page 4.6-1) is revised as follows (deleted text shown with strikethrough, new text underlined):

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

The Callaway prestressed concrete containment vessel is designed to Bechtel Topical Report BC-TOP-5-A, Revision 3. It is poured against a steel membrane liner designed to BC-TOP-1 Revision 1. 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.

The Callaway containment liner and ~~other~~ metal containment (MC) components, e.g. containment penetrations, were designed to stress limit criteria of BC-TOP-1 Revision 1 <u>(Part I and Part II respectively)</u>, independent of the number of load cycles, and require no fatigue analyses with the exception of the main steam and feedwater penetrations, the containment access hatches, and the leak chases<u>. For the MC containment penetrations, BC TOP-1, Part II provides guidance on how to satisfy the ASME Section III, Division I NE requirements.</u>

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Revision to Section 4.7.2.

Section 4.7.2 (page 4.7-4) is revised as follows (deleted text shown with strikethrough, new text underlined):

4.7.2 In-Service Flaw Analyses that Demonstrate Structural Integrity for 40 years

In-service flaw growth is identified in NUREG-1800 as a potential TLAA. Flaws of such size that they cannot be dispositioned through comparison with the Code tables must be analyzed. These analyses depend on a specified number of operating events or years, and thus may be TLAAs.

A search of the CLB did not identify any flaws evaluated for the remaining life of the plant other than those identified below.

- ***Cold Leg Elbow-to-Safe End Weld Flaw Indications***

During the Refuel 13 (Spring 2004), two flaw indications were identified in the cold leg elbow-to-safe end weld. The weld and base metal material for the subject weld is stainless steel. The safe end is forged stainless steel (SA-182 F316). The weld is a stainless steel weld (ER308). The elbow is statically cast stainless steel (SA-351 CF8A, which is the same as wrought Type 304).

<u>Flaw Indication #1:</u> The flaw was an embedded flaw that was found acceptable in accordance with IWB-3500 (Acceptance Standards), but was conservatively treated as inside diameter surface breaking flaw (Reference 10). The depth of flaw #1 is 0.49 in. including inspection uncertainty. This represents 21.1 percent of the local pipe wall. The flaw length is 4.75 in. (5.1 percent of circumference based on nominal diameter).

<u>Flaw Indication #2:</u> The flaw was an inside diameter surface breaking flaw that was found to be greater than the size allowed by IWB-3500 (Reference 10). The depth of flaw #2 was found to be 0.94 in. including inspection uncertainty. This represents 40.5 percent of the local pipe wall. The length of the flaw was determined to be 2.625 in. (2.8 percent of circumference based on nominal diameter).

The root cause evaluation determined that these flaws were formed during initial plant construction. Low cycle fatigue, such as that experienced during pressurization, heatup and cooldown, caused flaw #2 to break through. Subsequent volumetric examinations did not identify any ~~degradation~~ <u>propagation</u> in either of the flaws. These flaws will continue to be inspected through the ISI program at regular 10-year intervals after the two remaining followup inspections.

The continued operation with these flaws in place was justified in accordance with IWB-3640, which is supported by a flaw evaluation. The evaluation concluded that wide margin exists for both flaws, which allows further services throughout the remainder of the plant design life, as long as the same plant operation conditions as those considered in the analysis are maintained.

The evaluation of the two flaws considers two possible modes of failure. A fatigue crack growth analysis is used to demonstrate that the crack will satisfy the IWB-3640 requirements for the remainder of the plant life. A fracture mechanic analysis was also performed to predict crack instability.

Fracture Mechanics

Operation with a crack is acceptable with respect to unstable ductile tearing mechanism if the applied J-integral remains below the J_{Ic} fracture toughness. The only aging mechanism that affects the criteria is thermal aging. The forged safe end material is not subject to thermal aging. The gas tungsten arc welds are subject to thermal aging, but the effects are considered negligible. The fracture mechanics analysis does not consider aging effects and is not a TLAA, by 10 CFR 54.3(a), Criterion 2.

Fatigue Crack Growth

The analysis procedure involves postulating an initial flaw at start of life and predicting the flaw growth due to an imposed series of loading transients. The incremental growth is then added to the original crack size, and the analysis proceeds to the next cycle or transient. The procedure is continued in this manner until all of the analytical transients known to occur have been analyzed. The transients considered were distributed evenly over the plant design life, with the exception of the preoperational tests, which are considered first. The design numbers of transients assumed to occur over the plant life are consistent with those of FSAR Table 3.9(N)-1 SP. As long as the plant design basis numbers of transients are maintained the same as those considered in the analysis, regardless of whether the transients occur over a 40 or 60-year plant life, the analysis and conclusions will remain valid.

This fatigue growth analysis does not consider intergranular stress corrosion cracking as a credible aging mechanism. This is based on stress corrosion cracking having been observed to occur in stainless steel in operating BWR piping systems, but not in PWR plants due to hydrogen overpressure. While the RPV inlet and outlet nozzles are Alloy 600 and the nozzle-to-safe end welds are Alloy 182, the cracks were identified in the safe end-to-elbow region which is a gas tungsten-arc process, with a root pass TIG weld and does not contain any susceptible material. The analyses are only applicable in this region, and primary water stress corrosion cracking (PWSCC) was not considered a viable mechanism because expert and industry experience indicate stainless steels have been shown to be very resistant to PWSCC.

The projected transient accumulations in Table 4.3-2, *Transient Accumulations and Projections* 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).

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

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Revision to Section 4.7.4.

Section 4.7.4 (pages 4.7-8 and 9) is revised as follows (deleted text shown with strikethrough, new text underlined):

4.7.4 ~~Absence of a TLAA for~~ Reactor Vessel Underclad Cracking Analyses

NUREG-1800 identifies "Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding" as a potential TLAA. ~~No such cracks have been discovered, nor therefore, analyzed at Callaway. In the absence of any analyses no TLAAs exist.~~ This phenomenon has been addressed in the Callaway vessel by weld cladding processes designed to avoid these defects, consistent with Regulatory Guide 1.43. <u>In addition, WCAP-15338-A has demonstrated that the vessel integrity is maintained in the presence of underclad cracks. No such cracks have been discovered at Callaway.</u>

Regulatory Guide 1.43 states that underclad cracking has been reported only in forgings and plate material of SA-508 Class 2 when clad using "high-heat-input" processes such as the submerged-arc wide-strip and the submerged-arc 6-wire processes. Cracking was not observed in SA-508 Class 2 materials clad by "low-heat-input" processes controlled to minimize heating of the base metal. Further, cracking was not observed in clad SA-533 Grade B Class 1 plate material, regardless of the welding process used.

Callaway Vessel Material Subject to Underclad Cracking

The vessel shell and head plates are vacuum treated SA-533, Grade A, B, or C, Class 1 or 2 (Grade A or B, Class 1 for beltline plates). Only the vessel nozzles and flanges are SA-508 Class 2 forgings. The cladding is stainless steel weld metal, Analysis A-8; and Ni-Cr-Fe Weld Metal, F-Number 43.

~~The Callaway ISI program examines flanges under IWB Table-2500-1 Category B-A using Code Case N-623, and examines RV nozzles under Category B-D using Code Case N-648-1. A review of inservice inspection reports found no record of indications of underclad cracking in the RV nozzles or flanges.~~

Qualification of Clad Welding Processes to Avoid Underclad Cracking

Although the Callaway vessel contains these SA-508 forgings clad by high-heat-input processes, freedom from underclad cracking is assured by special evaluation of the procedure qualification for cladding applied on low alloy steel (SA-508, Class 2).

This special evaluation is documented in FSAR SP, Appendix 3A and determined that Callaway meets the requirement of Regulatory Guide 1.43 by requiring qualification of any "high heat input" processes, such as the submerged arc wide strip welding process and the submerged-arc 6-wire process used on ASME SA-508, Class 2, material, with a performance test as described in Regulatory Position C.2 of the guide. No qualifications are required by the regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material.

The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3. Stainless steel weld cladding of low-alloy steel components is not employed on components outside the NSSS.

Applicability of Westinghouse Owners Group Generic 60-Year Flaw Growth Analysis

Westinghouse prepared a topical report on underclad cracking, WCAP-15338-A [Ref. 19], which included fatigue crack growth analyses and ASME Section XI allowable flaw size evaluations for typical Westinghouse vessels, and found that the expected 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 demonstrated that these effects are acceptable for a 60 year life.

The NRC safety evaluation of this topical report determined that it might be incorporated by reference in a license renewal application, provided that the analysis is applicable to the applicant's plant. The licensee must demonstrate that the vessel will withstand growth of underclad cracks for a 60 year life by (1) verifying that the design cycles and transients assumed in WCAP-15338-A bound the cycles for 60 years of operation, and (2) providing a description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. ~~However, no underclad cracks have been discovered and this analysis is not invoked in the Callaway CLB, therefore it is not a TLAA by 10 CFR 54.3(a) criterion 6.~~

For Callaway (1) the numbers of transient cycles assumed in the WCAP-15338-A bound the projected cycles in 60 years presented in Table 4.3-2. (2) LRA Appendix A3, the FSAR supplement, provides a description of the evaluation of TLAAs for the period of extended operation.

In conclusion, at Callaway, the WCAP-15338-A addresses the aging mechanism of Underclad Cracking. WCAP-15338-A is a TLAA, which is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

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

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Revision to Section 4.8 to identify new references.

Section 4.8 (page 4.8-2) is revised as follows (deleted text shown with strike through, new text underlined):

4.8 REFERENCES

1. Westinghouse Report WCAP-15400-NP. Analysis of Capsule X from the Ameren UE Callaway Unit 1 Reactor Vessel Surveillance Program. Rev. 0. June 2000. Westinghouse Non-Proprietary Class 3.

2. Callaway PTLR. "Callaway Plant Pressure and Temperature Limits Report." Rev. 5. Released 11. December 2006.

3. Westinghouse Report. WCAP-17168-NP. Callaway Unit 1 Time-Limited Aging Analysis on Reactor Vessel Integrity. Rev. 0. September 2010. Westinghouse Non-Proprietary Class 3.

4. SIA Calculation 0900694.301. "Environmentally-Assisted Fatigue (EAF) for Callaway." Rev. 0. Structural Integrity Associates, Inc. San Jose, California. 19 August 2010.

5. SIA Calculation 0901271.315. "Residual Heat Removal (RHR) Inlet Nozzle Environmentally-Assisted Fatigue Analysis Calculation." Rev. 0. Structural Integrity Associates, Inc. San Jose, California. 11 August 2010.

6. SIA Calculation 0901271.332. "Charging Nozzle Environmentally-Assisted Fatigue (EAF) Analysis Using 60-Year of Operation Using Stress Based-Fatigue (SBF) Results from the Baseline Evaluation." Rev. 0. Structural Integrity Associates, Inc. San Jose, California. 27 October 2011.

7. SIA Calculation 0901271.331. "Safety Injection (BIT) Nozzle Environmentally-Assisted Fatigue (EAF) Analysis Using 60-Year Projected Numbers of Cycles." Rev 0. Structural Integrity Associates, Inc. San Jose, California. 16 September 2011. .

8. SIA Calculation 0901271.330. "Hot Leg Surge Nozzle Environmentally-Assisted Fatigue (EAF) Analysis Using 60-Year Projected Numbers of Cycles." Rev. 0. Structural Integrity Associates, Inc. San Jose, California. 15 September 2011.

9. Precision Surveillance Corporation Document No. CA-N1042-500. Final Report of the 25th Year IWL Inspection. Rev. 0. 16 September 2010. Supplemented by Callaway CAR 201009644.

10. Ameren Missouri Letter ULNRC-5100. "Docket Number 50-483, Union Electric Company Callaway Plant, Transmittal of Inservice Inspection Summary Report for Refuel 13, and WCAP-16280-P, 'Flaw Evaluation Handbook For Callaway Unit 1 Reactor Vessel Inlet

Nozzle Safe-End Weld Region,' May 2004." 13 December 2004. (ADAMS Accession No ML043650441).

11. Ameren Missouri Calculation BB-183. "Evaluation of Reactor Vessel Cladding Indication Inside Bottom Head During Refuel 13." Rev. 1.

12. Westinghouse Topical Report WCAP-15666-A. Extension of Reactor Coolant Pump Motor Flywheel Examination. Rev. 1. October 2003.

13. Ameren Missouri Letter ULNRC-05553. Graessle, Luke H. "Docket Number 50-483 Callaway Plant Unit 1 Union Electric Co. Facility Operating License NPF-30 Follow-Up Information Regarding 10 CFR 50.55a Request: Proposed Alternative to ASME Section XI Requirements for Replacement of Class 3 Buried Piping (TAC No. MD6792)." Fulton, MO. 9 October 2008. (ADAMS Accession No ML082900027).

14. Ameren Missouri Letter ULNRC-05542. Graessle, Luke H. "Docket Number 50-483 Callaway Plant Unit 1 Union Electric Co. Facility Operating License NPF-30 Additional Information Regarding 10 CFR 50.55a Request: Proposed Alternative to ASME Section XI Requirements for Replacement of Class 3 Buried Piping (TAC No MD6792)." Fulton, MO. 15 September 2008. (ADAMS Accession No ML082630798).

15. SIA Report FP-CALL-310. Benchmarking of Charging Nozzle Stress-Based Fatigue. Rev. 0. San Jose, California: Structural Integrity Associates. 22 June 2011.

16. SIA Report FP CALL 304. Baseline Analysis of Callaway Plant Cycles and Fatigue Usage – Startup through 1/31/2011. Rev. 1. San Jose, California: Structural Integrity Associates. 13 October 2011.

17. SIA Report FP-CALL-307. "Environmentally-Assisted Fatigue Screening." Rev. 2. San Jose, California: Structural Integrity Associates. 30 April 2012.

18. EPRI Technical Report 1024995. "Environmentally-Assisted Fatigue Screening, Process and Technical Basis for Identifying EAF Limiting Locations."

19. WOG Topical Report WCAP-15338-A. A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants. Westinghouse Electric Company LLC. October 2002.

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Revision to Section A3.2.3 to show completion of commitment.

Section A3.2.3 (page A-27) is revised as follows (deleted text shown with strike through, new text underlined):

A3.2.3 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 A2.1. 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 ~~will~~has completed an evaluation ~~for~~to identify any additional plant-specific bounding EAF locations ~~prior to the period of extended operations~~. 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. T~~herefore t~~he effects of the reactor coolant environment on fatigue usage factors in the ~~remaining~~NUREG/CR-6260 and plant-specific bounding EAF locations will be managed for the period of extended operation. These TLAAs are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

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Revision to Section A3.6.4, A3.6.5, A3.6.6, A3.6.7, A3.6.8. and A3.6.9 to incorporate new Section A3.6.4 and renumber remaining sections.

Sections A3.2.4, 5, 6, 7, 8, and 9 (pages A-34 and 35) is revised as follows (deleted text shown with strike through, new text underlined):

A3.6.4 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).

A3.6.~~4~~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).

A3.6.~~5~~6 High Energy Line Break Postulation Based on Fatigue Cumulative Usage Factors

The selection of ASME III, Class 1 piping 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 A2.1, 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).

A3.6.~~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).

A3.6.~~7~~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).

A3.6.~~8~~9 Replacement Steam Generator Tube Wear

The replacement steam generator tube wear analysis determined the maximum wear for a 45-year design life. The 45-year design life of the replacement steam generator tubes extends beyond the period of extended operation. Therefore, the design of the replacement steam generator tubes is valid through the period of extended operation and the TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

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Revision to Section A4, Table A4-1 to revise Fatigue Monitoring commitments.
Sections A4, Table A4-1, Item 37 (page A-49) is revised as follows (new shown text underlined):

A4 LICENSE RENEWAL COMMITMENTS

Table A4-1 identifies proposed actions committed to by Ameren Missouri for the Callaway Plant Unit 1 in its License Renewal Application. These and other actions are proposed regulatory commitments. This list will be revised, as necessary, in subsequent amendments to reflect changes resulting from NRC questions and Ameren Missouri responses. Ameren Missouri will utilize the commitment tracking system to track regulatory commitments.

Table A4-1 License Renewal Commitments

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. <u>(Completed Amendment 2)</u> 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. <u>(Completed Amendment 2)</u> <u>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.</u> <u>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.</u>	4.3.2.2 4.3.4	Prior to the period of extended operation

ENCLOSURE 2

AMR Changes for Callaway Plant Unit 1 License Renewal Application Amendment No. 2

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Revision to Table 3.2.2-5 to delete the aging evaluation lines of stainless steel valve with intended function of LBS.

Table 3.2.2-5 (pages 3.2-64 and 65) are revised as follows (deleted text shown in strikethrough):

Table 3.2.2-5 Engineered Safety Features – Summary of Aging Management Evaluation – High Pressure Coolant Injection System (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
~~Valve~~	~~LBS~~	~~Stainless Steel~~	~~Atmosphere/ Weather (Ext)~~	~~Cracking~~	~~External Surfaces Monitoring of Mechanical Components (B2.1.21)~~	~~V.D1.EP-103~~	~~3.2.1.007~~	~~A~~
~~Valve~~	~~LBS~~	~~Stainless Steel~~	~~Atmosphere/ Weather (Ext)~~	~~Loss of material~~	~~External Surfaces Monitoring of Mechanical Components (B2.1.21)~~	~~V.D1.EP-107~~	~~3.2.1.004~~	~~A~~
~~Valve~~	~~LBS~~	~~Stainless Steel~~	~~Steam (Int)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and One-Time Inspection (B2.1.18)~~	~~VIII.B1.SP-98~~	~~3.4.1.011~~	~~A~~
~~Valve~~	~~LBS~~	~~Stainless Steel~~	~~Steam (Int)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2) and One-Time Inspection (B2.1.18)~~	~~VIII.B1.SP-155~~	~~3.4.1.016~~	~~A~~

Callaway Plant
License Renewal Application
Amendment 2

Revision to Section 3.3.2.1.5 to add Atmosphere/Weather as an environment in the Service Water System.

Section 3.3.2.1.5 (page 3.3-8) is revised as follows (deleted text shown in strikethrough and new text shown underlined):

3.3.2.1.5 Service Water System

Environment

The service water system components are exposed to the following environments:

- Atmosphere/Weather
- Borated Water Leakage
- Buried
- Plant Indoor Air
- Raw Water

Callaway Plant
License Renewal Application
Amendment 2

Revision to Table 3.3.2-5 to add Carbon Steel Piping in Atmosphere/Weather and change the external environment of the Ductile Iron Valve to Atmosphere Weather.

Table 3.3.2-5 (page 3.3-100 and 3.3-103) is revised as follows (deleted text shown in strikethrough and new text shown underlined):

Table 3.3.2-5 Auxiliary Systems – Summary of Aging Management Evaluation – Service Water System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Item	Table 1 Item	Notes
<u>Piping</u>	<u>PB</u>	<u>Carbon Steel</u>	<u>Atmosphere/ Weather (Ext)</u>	<u>Loss of material</u>	<u>External Surfaces Monitoring of Mechanical Components (B2.1.21)</u>	<u>VII.I.A-78</u>	<u>3.3.1.078</u>	<u>A</u>
Valve	PB	Ductile Iron	~~Plant Indoor Air (Ext)~~<u>Atmosphere/ Weather (Ext)</u>	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.21)	~~VII.I.A-77~~<u>VII.I.A-78</u>	3.3.1.078	A