Ameren
MISSOURI

Callaway Plant

April 8, 2014

ULNRC-06093

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

10 CFR 50.55a

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
PROPOSED ALTERNATIVE TO ASME CODE, SECTION XI REQUIREMENTS, WHICH EXTENDS REACTOR VESSEL INSERVICE INSPECTION FREQUENCY FROM 10 TO 20 YEARS (RELIEF REQUEST I3R-17)

Pursuant to 10 CFR 50.55a(a)(3)(i), Union Electric Company (Ameren Missouri) hereby requests NRC approval of the attached relief request I3R-17. The requested relief is intended for the third (i.e., the current) 10-year inservice inspection interval of Callaway's Inservice Inspection (ISI) Program. With regard to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, i.e., Section XI, "Rules and Inservice Inspection of Nuclear Power Plant Components," the Code Edition and Addenda applicable to Callaway's third 10-year ISI interval is the 1998 Edition with 2000 Addenda.

Per the attached relief request, Ameren Missouri proposes to implement an alternative requirement to the requirement of IWB-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. In lieu of this requirement, Ameren Missouri proposes not to perform the ASME Code required volumetric examination of the Callaway Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds for the third inservice inspection, currently scheduled for 2014, and instead proposes to perform the third ASME Code required volumetric examination of the Callaway Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds in the fourth inservice inspection interval in 2023 plus or minus one refueling outage.

Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs. Using the

methodology described in WCAP-16168-NP-A Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," the proposed alternative requirements to be implemented have been determined to provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i).

Supporting information and essential details, including justification, is provided in the attached relief request. With respect to this request, and as noted in Section 7, "Precedents," of the attached, the NRC has approved similar requests for other licensees.

For the subject examinations of this request, the third ISI interval has been extended to December 18, 2015 using the provisions of IWA-2430(d). Ameren Missouri requests NRC review and approval of the attached relief request by May 1, 2015.

This letter does not contain new commitments.

If there are any questions, please contact J. A. Doughty at 573-220-5145.

Sincerely,

Scott A. M

Scott A. Maglio,
Manager, Regulatory Affairs

JPK/nls

Attachment

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Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the Callaway Unit 1 reactor vessel (RV); specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the RV. These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV, Code Section XI.

Category B-A welds are defined as "Pressure Retaining Welds in Reactor Vessel"
Category B-D welds are defined as "Full Penetration Welded Nozzles in Vessels"

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-A	B1.50	Repair Welds
B-A	B1.51	Beltline Region
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1998 Edition through 2000 Addenda (Reference 1).

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor vessel pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. The Callaway Unit 1 third 10-year inservice inspection (ISI) interval is scheduled to end on December 18, 2014, but, for the subject examinations of this request, the third ISI interval has been extended to December 18, 2015 using the provisions of IWA-2430(d) in order to allow adequate evaluation of this request. The applicable Code for the fourth 10-year ISI interval will be selected in accordance with the requirements of 10 CFR 50.55a.

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4. Reason for Request

An alternative is requested from the requirement of IWB-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor vessel pressure-retaining Examination Category B-A and B-D welds be performed once each 10-year interval. Extension of the interval between examinations of Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Ameren Missouri proposes not to perform the ASME Code required volumetric examination of the Callaway Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds during the third inservice inspection interval, currently scheduled to end December 18, 2015 for the subject examinations. Ameren Missouri will perform the third ASME Code required volumetric examination of the Callaway Unit 1 reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds in the fourth inservice inspection interval in 2023 plus or minus one refueling outage. The proposed inspection date is consistent with the latest revised implementation plan OG-10-238 (Reference 2).

In accordance with 10 CFR 50.55a(a)(3)(i), an alternate inspection interval is requested on the basis that the current interval can be revised with negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to conduct this analysis is based on that defined in the study WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval" (Reference 4). This study focuses on risk assessments of materials within the beltline region of the RV wall. The results of the calculations for Callaway Unit 1 were compared to those obtained from the Westinghouse pilot plant evaluated in WCAP-16168-NP-A, Revision 3. Appendix A of the WCAP identifies the parameters to be compared. Demonstrating that the parameters for Callaway Unit 1 are bounded by the results of the Westinghouse pilot plant qualifies Callaway Unit 1 for an ISI interval extension. Table 1 below lists the critical parameters investigated in the WCAP and compares the results of the Westinghouse pilot plant to those of Callaway Unit 1. Tables 2 and 3 provide additional information that was requested by the NRC and included in Appendix A of Reference 4.

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Table 1: Critical Parameters for the Application of Bounding Analysis for Callaway Unit 1

Parameter	Pilot Plant Basis	Plant-Specific Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 5)	PTS Generalization Study (Reference 6)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 4)	3.98E-14 Events per year (Calculated per Reference 10)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 4)	Bounded by 7 heatup/cooldown cycles per year (Reference 10)	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer (Reference 10)	No

Table 2 below provides a summary of the latest reactor vessel inspection for Callaway Unit 1 and an evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on the Callaway Unit 1 reactor vessel.

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<table>
<tr><th colspan="2">Table 2: Additional Information Pertaining to Reactor Vessel Inspection for Callaway Unit 1</th></tr>
<tr><td>Inspection methodology:</td><td>The latest ISI was conducted in accordance with the ASME Code, Section XI and Section V, 1989 Edition and 1995 Edition, with the 1996 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Examinations of Category B-A and B-D welds were performed to ASME Section XI, Appendix VIII, 1995 Edition with the 1996 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future inservice inspections will be performed to ASME Section XI, Appendix VIII requirements.</td></tr>
<tr><td>Number of past inspections:</td><td>Two 10-year inservice inspections have been performed.</td></tr>
<tr><td>Number of indications found:</td><td>There were twenty-four indications identified in the beltline region during the most recent inservice inspection. These subsurface indications are located in the intermediate shell to lower shell circumferential weld (Item 20 in Table 3), intermediate shell longitudinal welds (Items 13-15 in Table 3) and a lower shell longitudinal weld (Item 17 in Table 3). The indications are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Thirteen indications are within the inner 1/10th or 1" of the reactor vessel thickness. These indications are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 7), since the number of flaws is less than the allowable number of flaws for each flaw size increment. A disposition of these flaws against the limits of the Alternate PTS Rule is shown in the tables below.

The following indications are located within the weld material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled maximum number of weld flaws</th><th rowspan="2">Number of weld flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0.075</td><td>0.475</td><td>163</td><td>7 (7/0)</td></tr>
<tr><td>0.125</td><td>0.475</td><td>89</td><td>6 (6/0)</td></tr>
<tr><td>0.175</td><td>0.475</td><td>23</td><td>4 (4/0)</td></tr>
<tr><td>0.225</td><td>0.475</td><td>9</td><td>1 (1/0)</td></tr>
<tr><td>0.275</td><td>0.475</td><td>4</td><td>1 (1/0)</td></tr>
<tr><td>0.325</td><td>0.475</td><td>3</td><td>1 (1/0)</td></tr>
<tr><td>0.375</td><td>0.475</td><td>2</td><td>1 (1/0)</td></tr>
</table>
The following indications are located within the plate material of the reactor vessel beltline.
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled maximum number of plate flaws</th><th rowspan="2">Number of plate flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_{MIN}</th><th>TWE_{MAX}</th></tr>
<tr><td>0.075</td><td>0.375</td><td>74</td><td>6 (4/2)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>29</td><td>6 (4/2)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>8</td><td>3 (2/1)</td></tr>
<tr><td>0.225</td><td>0.375</td><td>3</td><td>2 (2/0)</td></tr>
<tr><td>0.275</td><td>0.375</td><td>1</td><td>1 (1/0)</td></tr>
</table></td></tr>
<tr><td>Proposed inspection schedule for balance of plant life:</td><td>The third inservice inspection is scheduled for 2014. This inspection will be performed in 2023 plus or minus one refueling outage. The proposed inspection date is consistent with the latest revised implementation plan, OG-10-238 (Reference 2).</td></tr>
</table>

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Table 3 summarizes the inputs and outputs for the calculation of through-wall cracking frequency (TWCF).

Table 3: Details of TWCF Calculation for Callaway Unit 1 at 54 Effective Full-Power Years (EFPY)

Inputs

Reactor Coolant System Temperature, T_C [°F]:			N/A	Inter. & Lower Shell T_{wall} [inches]:				8.79
				Nozzle Shell T_{wall} [inches]:				10.91
No.	Region and Component Description	Material Heat No.	Cu[(1)] [wt%]	Ni[(1)] [wt%]	R.G. 1.99 Pos.[(1)]	CF[(1)] [°F]	$RT_{NDT(u)}$[(1)] [°F]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]
1	Nozzle Shell Plate R-2706-3	B8307-1	0.075	0.60	1.1	47.5	30	0.0726
2	Nozzle Shell Plate R-2706-1	C4202-1	0.045	0.615	1.1	28.5	20	0.0726
3	Nozzle Shell Plate R-2706-2	C4242-1	0.055	0.665	1.1	34.0	30	0.0726
4	Intermediate Shell Plate R-2707-2	C4383-1	0.06	0.61	1.1	37.0	10	2.94
5	Intermediate Shell Plate R-2707-3	C4383-2	0.06	0.62	1.1	37.0	-10	2.94
6	Intermediate Shell Plate R-2707-1	C4344-1	0.05	0.58	1.1	31.0	40	2.94
7	Lower Shell Plate R-2708-2	C4472-1	0.06	0.57	1.1	37.0	10	2.94
8	Lower Shell Plate R-2708-1	C4499-2	0.07	0.58	2.1	25.6	50	2.94
9	Lower Shell Plate R-2708-3	C4499-1	0.08	0.62	1.1	51.0	20	2.94
10	Nozzle Shell Long. Weld 101-122A	87000	0.045	0.13	1.1	38.7	-40	0.0726
		GABID	0.02	1.00	1.1	27.0	-50	0.0726
11	Nozzle Shell Long. Weld 101-122B	87000	0.045	0.13	1.1	38.7	-40	0.0726
		FAOED	0.03	0.98	1.1	41.0	-60	0.0726
12	Nozzle Shell Long. Weld 101-122C	87000	0.045	0.13	1.1	38.7	-40	0.0726
		EACAE	0.035	1.005	1.1	47.5	-80	0.0726
13	Inter. Shell Long. Weld 101-124A	90077	0.04	0.05	2.1	40.8	-60	2.94
14	Inter. Shell Long. Weld 101-124B	90077	0.04	0.05	2.1	40.8	-60	2.94
15	Inter. Shell Long. Weld 101-124C	90077	0.04	0.05	2.1	40.8	-60	2.94
16	Lower Shell Long. Weld 101-142A	90077	0.04	0.05	2.1	40.8	-60	2.94
17	Lower Shell Long. Weld 101-142B	90077	0.04	0.05	2.1	40.8	-60	2.94
18	Lower Shell Long. Weld 101-142C	90077	0.04	0.05	2.1	40.8	-60	2.94
19	Nozzle To Inter. Shell Circ. Weld 103-121	90211	0.028	0.0825	1.1	26.5	-60	0.0726
		IAOJE	0.04	1.005	1.1	54.0	-80	0.0726
		JAACE	0.025	0.98	1.1	34.0	-70	0.0726
20	Inter. To Lower Shell Circ. Weld 101-171	90077	0.04	0.05	2.1	40.8	-60	2.94

Outputs

Methodology Used to Calculate ΔT_{30}:			Regulatory Guide 1.99, Revision 2[(2)]			
	Controlling Material Region No. (From Above)	RT_{MAX-XX} [°R]	Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]	FF (Fluence Factor)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Limiting Axial Weld - AW	9	545.25	2.94	1.286	65.58	0.00E+00
Limiting Plate - PL	9	545.25	2.94	1.286	65.58	1.59E-14
Circumferential Weld - CW	9	545.25	2.94	1.286	65.58	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW})$:						3.98E-14

(1) Reference 8

(2) Reference 9

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6. Duration of Proposed Alternative

This request is applicable to the Callaway Unit 1 inservice inspection program for the third and fourth 10-year inspection intervals.

7. Precedents

- "Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Relief Request No. RR-40, Reactor Vessel Weld Examination Interval Extension (TAC Nos. ME1634, ME1635, and ME1636)," dated February 22, 2010 (ADAMS Accession Number ML100290415).
- "Safety Evaluation of Relief Requests to Extend the Inservice Inspection Interval for Reactor Vessel Examinations for Salem Nuclear Generating Station, Unit Nos. 1 and 2 (TAC Nos. ME1478, ME1479, ME1480 and ME1481)," dated February 22, 2010 (ADAMS Accession Number ML100491550).
- "Arkansas Nuclear One, Unit 2 – Request for Alternative ANO2-ISI-004, to Extend the Third 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examinations (TAC No. ME2508)," dated September 21, 2010 (ADAMS Accession Number ML102450654).
- "Joseph M. Farley Nuclear Plant, Unit 2 (Farley Unit 2) – Relief Request for Extension of the Reactor Vessel Inservice Inspection Date to the Year 2020 (Plus or Minus One Outage) (TAC No. ME3010)," dated July 12, 2010 (ADAMS Accession Number ML101750402).
- "Three Mile Island Nuclear Station, Unit 1 (TMI-1) – Request to Extend the Inservice Inspection Interval for Reactor Vessel Weld and Internal Examinations, Proposed Alternative Request Nos. RR-09-01 and RR-09-02 (TAC Nos. ME2483 and ME2484)," dated September 21, 2010 (ADAMS Accession Number ML102390018).
- "Surry Power Station Units 1 and 2 – Relief Implementing Extended Reactor Vessel Inspection Interval (TAC Nos. ME8573 and ME8574)," dated April 30, 2013 (ADAMS Accession Number ML13106A140).
- "McGuire Nuclear Station, Unit 2, Relief 10-MN-002 to Extend the Inservice Inspection Interval for Reactor Vessel Category B-A and B-D Welds (TAC Nos. ME7329 and ME 7330)," dated September 6, 2012 (ADAMS Accession Number ML12249A175).

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8. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda, American Society of Mechanical Engineers, New York.
2. PWROG Letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval.'" PA-MSC-0120," July 12, 2010 (ADAMS Accession Number ML11153A033).
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. Nuclear Regulatory Commission, November 2002.
4. Westinghouse Report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011 (ADAMS Accession Number ML113060207).
5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," U.S. Nuclear Regulatory Commission, March 2010.
6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," U.S. Nuclear Regulatory Commission, December 14, 2004 (ADAMS Accession Number ML042880482).
7. Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010 and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010.
8. Westinghouse Report WCAP-17168-NP, Revision 0, "Callaway Unit 1 Time-Limited Aging Analysis on Reactor Vessel Integrity," September 2010.
9. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
10. Calculation MCOE-CN-13-8, "Implementation of WCAP-16168-NP-A, Revision 3, for Callaway Unit 1."