UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 14, 2014

Mr. Adam C. Heflin
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO FINAL SAFETY ANALYSIS REPORT ASSOCIATED WITH SEISMIC DAMPING VALUES FOR THE INTEGRATED HEAD ASSEMBLY (TAC NO. MF0407)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 207 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Final Safety Analysis Report – Standard Plant (FSAR-SP) in response to your application dated December 20, 2012, as supplemented by letters dated June 6 and August 29, 2013.

The amendment revises a methodology in the licensing basis as described in the FSAR-SP to include damping values for the seismic design and analysis of the integrated head assembly that are consistent with the recommendations of NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, March 2007.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 207 to NPF-30
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Union Electric Company (UE, the licensee), dated December 20, 2012, as supplemented by letters dated June 6 and August 29, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan*

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 207 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance, and shall be implemented within 90 days of the date of issuance. In addition, the licensee shall include the revised information in the next Final Safety Analysis Report update submitted to the NRC in accordance with 10 CFR 50.71(e), as described in the licensee's application dated December 20, 2012, as supplemented by letters dated June 6 and August 29, 2013, and evaluated in the staff's safety evaluation enclosed with this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-30

Date of Issuance: January 14, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following page of the Facility Operating License No. NPF-30 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-3-	-3-

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 207 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3)**

Deleted per Amendment No. 169.

* Amendments 133, 134, & 135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO

FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated December 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13002A370), as supplemented by letters dated June 6 and August 29, 2013 (ADAMS Accession Nos. ML13158A009 and ML13242A241, respectively), Union Electric Company (dba Ameren Missouri; the licensee) requested changes to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1 (Callaway). The licensee proposed to revise a methodology in the licensing basis as described in the Final Safety Analysis Report - Standard Plant (FSAR-SP) to include damping values for the seismic design and analysis of the integrated head assembly that are consistent with the recommendations of U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, March 2007 (ADAMS Accession No. ML070260029).

The supplemental letters dated June 6 and August 29, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 4, 2013 (78 FR 14139).

The proposed change would revise the current licensing basis methodology of RG 1.61, Revision 0, October 1973 (ADAMS Accession No. ML003740213), as described in the FSAR-SP, to include damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of RG 1.61, Revision 1. The RG 1.61, Revision 1, Table 1 note allowing use of a "weighted average" for design-basis Safe Shutdown Earthquake (SSE) damping values applicable to steel structures of different connection types is also applied to determine the IHA design-basis Operating Basis Earthquake (OBE) damping values. The proposed damping values are to be used in conjunction with the response spectrum analysis of the IHA to qualify various structural components in the IHA and in developing the reaction loads from the IHA on the replacement reactor vessel closure head (RRVCH) and on the

containment cavity wall seismic embedments. The current licensing basis use of RG 1.61 Revision 0, is retained for all structural analyses that do not address the structural qualification of the IHA.

2.0 REGULATORY EVALUATION

The licensee requested a change to the Facility Operating License for Callaway, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit." Pursuant to 10 CFR 50.92, in determining whether an amendment to a license, construction permit, or early site permit will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses, construction permits, or early site permits to the extent applicable and appropriate. These considerations are as follows.

Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," General Design Criterion 2 (GDC 2), "Design bases for protection against natural phenomena," requires that seismic Category I nuclear power plant structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without losing the capability to perform their safety functions. Such SSCs must also be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation and postulated accidents.

Appendix S to 10 CFR Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants," specifies the requirements for the implementation of GDC 2 with respect to earthquakes. The OBE and SSE are described in 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." These postulated seismic events are also discussed in NUREG-0830, "Safety Evaluation Report Related to the Operation of Callaway Plant, Unit No. 1," October 1981, as supplemented.

The IHA consists of safety-related, seismic Category I and nonsafety-related, non-seismic Category I components. These components are evaluated using the acceptance criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Subsection NF Class 1 Component Supports, 2001 Edition through 2003 Addenda. The licensee requested NRC approval to revise the FSAR-SP Sections 3.7(B), 3.7(N), and Appendix 3A, to use new critical damping values of 4.5 percent for the OBE, and 6.25 percent for the SSE, in the IHA seismic analysis. These proposed damping values are based on the recommendations in RG 1.61, Revision 1, Tables 1 and 2, using a weighted average for "Welded Steel or Bolted Steel with Friction Connections" and "Bolted Steel with Bearing Connections." RG 1.61 provides damping values acceptable to the NRC staff for the seismic design of nuclear power plants. However, it does not specifically address the damping values applicable to each and every type of component. Section 3.7.1 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," notes that damping values in accordance with those addressed in RG 1.61 are acceptable.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Request for Damping Values

The licensee requested approval for the use of the calculated weighted average and conservatively adjusted damping values of 4.5 percent for OBE, and 6.25 percent for both SSE and loss-of-coolant accident (LOCA), for the structural dynamic analysis of the IHA. The licensee also requested the use of the weighted average approach for OBE.

The licensee provided a detailed description of the new IHA to be installed during the installation of the Replacement Reactor Vessel Closure Head (RRVCH). Although the IHA is a new structure that does not have an existing equivalent, it incorporates the functions of the former control rod drive mechanism (CRDM) seismic support structure, the CRDM ventilation cooling system, and the vessel head lift rig.

Revision 0 of RG 1.61, issued in October 1973, addresses some broad categories of structures and components, such as piping, welded and bolted steel structures, and concrete structures, and provides recommended damping values that are acceptable to the NRC. The original damping values in RG 1.61, Revision 0, were based on limited data, and information available in 1973. Since that time, the NRC and industry have been involved in various studies, research work, and testing to predict and estimate damping values of SSCs. As a result, the NRC updated and revised the damping values to reflect more realistic values in 2007. Revision 1 of RG 1.61, issued in March 2007, contained these revised or increased damping values and includes damping values for some additional components such as electrical distribution systems, heating, ventilation, and air conditioning (HVAC) duct systems, and mechanical and electrical components. The current licensing basis for damping values for the SSCs at Callaway is based on RG 1.61, Revision 0. The NRC staff notes that the licensee proposes to retain the damping values described in RG 1.61, Revision 0 for all structural analyses, except for the IHA. The damping values in RG 1.61, Revision 1 would be utilized by the licensee for the IHA analysis. The damping values in Revisions 0 and 1 are acceptable to the NRC staff for use in elastic design of nuclear power plants.

The IHA is a replacement structure for the existing reactor vessel head service structure. The IHA is an assembly that consists of seismic Category I (safety-related) as well as seismic Category II/I components. The IHA is a vertically standing structure bolted to three support pads on the RRVCH and pinned to three lift lugs on the RRVCH. The IHA design integrates all the removable upper reactor vessel head components into one removable structure. The IHA is primarily a bolted steel structure with some welded connections consisting of various components designed to provide cooling for the CRDMs, to provide radiation shielding for workers performing activities near the RRVCH, to provide seismic support for the CRDMs and other IHA components, and to facilitate the lifting of the IHA and the RRVCH during refueling outages. The IHA is a four-story high, approximately 43-foot-tall steel structure consisting of more than 10,000 parts assembled together by bolted and welded connections. There are four tie rod restraints that provide lateral support for the IHA. These tie rods have pinned connections at both ends, namely on the IHA ring beam side as well as the reactor cavity wall side. This allows for the transfer of loads from the IHA and the CRDMs to the reactor cavity concrete wall. These tie rods have a slight inclined orientation, and therefore provide some vertical restraint in addition to dominantly horizontal restraining effect.

As provided in Table 6 of Attachment 2 of the licensee's letter dated December 20, 2012, the materials for the IHA components are carbon, low alloy, and stainless steels, in accordance with the American Society for Testing and Materials (ASTM) specifications, and ASME SA specifications. In Table 1 of Attachment 3 to Enclosure 1 of the licensee's letter dated December 20, 2012, the licensee included a detailed table that contains the number of connections and the connection types, namely the bolted (bearing or pinned) type or welded type. For the IHA, the licensee listed different connections that transfer loads during a seismic event which amount to a conservative total of 315 connections of which 48 are of the welded type and 267 are of the bolted bearing type. All bolted connections in the IHA are bearing connections and not friction-type connections.

RG 1.61, Revision 1 differentiates between a welded steel or bolted steel with friction connections and a bolted steel with bearing connection based on the differences in their energy absorbing capabilities. According to RG 1.61, Revision 1, the damping values for welded steel or bolted steel with friction connections are 3.0 percent for the OBE and 4.0 percent for the SSE. The damping values for bolted steel with bearing connections are 5.0 percent for the OBE and 7.0 percent for the SSE.

The computed weighted average critical damping value for the SSE for the Callaway IHA is 6.54 percent [=(4 percent x 48 + 7 percent x 267) / (48+267)] based on the RG 1.61, Revision 1 methodology and utilizing the actual number of bolted and welded joints. The computed weighted average critical damping value for the OBE for the Callaway IHA is 4.7 percent [= (3 percent x 48 + 5 percent x 267) / (48+267]] based on the actual number of bolted and welded joints. The bolted and welded connections are critical for load transfer and energy dissipation during a seismic event. Table 1 of RG 1.61, Revision 1 has a note stating that "for steel structures with a combination of different connection types, use the lowest specified damping value, or as an alternative, use a 'weighted average' damping value based on the number of each type present in the structure," for the SSE damping values. The licensee chose to use the alternative which is the weighted average damping value. The use of this alternative is acceptable to the NRC staff because it is based on the weighted average, considering the actual number of bolted and welded connections. This provides a realistic damping computation and is permitted by Revision 1 of RG 1.61. However, the same note regarding the weighted average alternative is not specifically listed in RG 1.61, Revision 1, Table 2, for the OBE damping values. The licensee requested the application of the SSE table's note to the OBE table. The LOCA and seismic analysis of the IHA is based on linear elastic methods using three directional acceleration response spectra. A summary of damping values and analysis methods used for horizontal and vertical direction excitations from OBE, SSE, and LOCA events, are provided in Table A, below:

Table A: IHA Seismic and LOCA Load Analysis Method & Damping Values Summary

	OBE		SSE		LOCA	
	Horizontal	**Vertical**	**Horizontal**	**Vertical**	**Horizontal**	**Vertical**
Analysis Method	USM Response Spectra	USM Response Spectra	USM Response Spectra	USM Response Spectra	Response Spectra	Response Spectra
Damping	4.5%	4.5%	6.25%	6.25%	6.25%	6.25%

Notes for Table A:

- Uniform Support Motion (USM) (Enveloped Spectra: Reactor Building Internal Structure Spectra at Elevation 2047'-6" and RRVCH spectra at the rod attachments to cavity walls and ring beam attachments to the RRVCH) was used as input to the response spectra analysis (East-West, North-South, and Vertical).

- LOCA response spectra inputs were applied where the IHA lift rods and the bottom ring beam are attached to the RRVCH (RRVCH response spectra for LOCA are based on the envelope of response spectra associated with auxiliary line breaks, surge line break, residual heat removal (RHR) line break, and accumulator line break). Callaway was approved for leak-before-break, therefore; there is no need to postulate breaks in main reactor coolant loop.

3.2 <u>NRC Staff Evaluation of Damping Values</u>

Based on a review of the summary information provided in Table A, the NRC staff determined that no inelastic analysis method was used. The damping values provided in RG 1.61 are for the elastic dynamic analysis and design of SSCs. Although the note regarding weighted average in RG 1.61 does not specifically address OBE, the staff determined that the licensee's approach was reasonable and acceptable for the following reasons: (i) inelastic analysis methods were not used; (ii) the number of connections of welded type and bolted bearing type were properly and realistically accounted for in the weighted average calculation for damping; and (iii) the composite effect of the bolted and welded connections on damping is present for OBE as well as SSE. Utilizing the weighted average approach for OBE and SSE, the licensee computed damping values of 4.7 percent for OBE, and 6.54 percent for SSE, but used slightly lower but conservative values of 4.5 percent for OBE and 6.25 percent for SSE in the analysis of the IHA.

The NRC staff determined that the use of damping values for steel structures as described in RG 1.61, Revision 1, is acceptable for the IHA, because the IHA is primarily a tall steel structure with different connection types. The NRC staff concludes that the calculated damping values of 4.7 percent (4.5 percent used in analysis) for OBE, and 6.54 percent (6.25 percent used in analysis) for SSE for the analysis of the Callaway's IHA are reasonable and acceptable because they are based on a weighted average approach, as described in RG 1.61, Revision 1. The NRC staff also concludes that the licensee has appropriately considered the number and type of

bolted bearing and welded connection types in the IHA steel structure in its weighted average damping calculation.

3.3 Licensee's Proposed Request Regarding the IHA Analysis

The licensee's submittal included IHA analysis details, and discussion on boundary conditions, and structural modeling. The licensee also included a summary of the finite element analysis results.

The IHA is a vertically standing steel structure bolted to three support lugs on the RRVCH and pinned to three lift lugs on the reactor vessel head. In addition to these six attachment points on the RRVCH, there are four seismic tie rods pinned to the IHA seismic ring beam at the refuel floor elevation. On the cavity wall side, these tie rods are pinned to wall lugs that are an integral part of plates that are bolted to the containment wall. All four seismic tie rod connections on both ends of the tie rods are pin connections. A finite element model consisting of shell elements, beam elements, and mass elements was created by the licensee representing the IHA. The licensee used the appropriate weight and mass distribution along with proper boundary conditions and performed a structural dynamic finite element analysis of the IHA using ANSYS computer program. The computed critical damping values summarized in Table A above were incorporated into the analysis.

There is a net increase in overall weight as well as a change in mass distribution of the upper portion of the reactor vessel head due to the added weight of the IHA compared to the current lift rig and platform that the IHA is replacing. The approximate weight of the IHA is 103,842 pounds, while that of the support structure supported by the original reactor vessel closure head is 30,000 pounds. The impact of the additional mass is accounted for in the IHA analysis, and the resulting loads are considered in the qualification of reactor vessel support, IHA interface locations with support and lift lugs, and tie rod connections to the reactor cavity wall.

The IHA was evaluated for the following seismic events: OBE and SSE using weighted average damping values described previously. The IHA was also evaluated for LOCA, which is a faulted load. The SSE and LOCA loads are combined by square root sum of squares (SRSS). The results from the seismic analyses, that is the loads, stresses, and displacements, were combined using the appropriate load combinations with those from the other applicable loads, such as deadweight and pressure, to determine the total loads, stresses, and displacements for the various components of the IHA. The resulting IHA loads and stresses were evaluated by the licensee for acceptance using the ASME Code, Section III, Division 1 - Subsection NF, Component Supports, 2001 Edition through 2003 Addenda, and stress ratios were computed. The stress ratio is the calculated load or stress divided by the allowable value. A value of a stress ratio equal to 1.0 represents an acceptable load or a stress with the required margin per the applicable code. Stress ratios below 1.0 represent additional margin beyond that required by the applicable code. Callaway's Code of record is the ASME Code, Section III, Subsection NF (class 1), 1977 Edition through summer 1977 Addenda and the code utilized in the IHA analysis is the ASME Code, Section III, Subsection NF, 2001 Edition through 2003 Addenda. The NRC staff notes that the licensee performed a code reconciliation between Callaway's Code of record and the code utilized in the IHA analysis to the extent required by

ASME Code, Section XI (1988 Edition through 2003 Addenda) governing Callaway's Repair/Replacement Program.

The licensee evaluated the components of the IHA using the acceptance criteria in Section III of the ASME Code, Subsection NF, Component Supports. The components with the highest stress ratios for all five groups of components are listed in Table B.

Table B: Maximum Stress Ratio for IHA Components
(Summary extracted for the critical components with maximum stress ratio from Tables 1-5 of Enclosure 1, Attachment 2 of the application dated December 20, 2012.)

Component Description	Controlling Load Combination	Stress Ratio	Comment
(Safety Related Seismic Category I Linear Components) Duct Support in lower, mid, and upper shroud	DL+P+/- SRSS (SSE + LOCA)	0.92	Acceptable
(Safety Related Seismic Category I Plate Components) Support Bracket connecting Monorail & Walkway to Column	DL+P+ML	0.87	Acceptable
(Non-Safety Related Non-Seismic Category I Linear Components) Stiffener Plate at Core exit thermocouples door in lower duct	DL+P+/- SSE	0.98	Acceptable
(Non-Safety Related Non-Seismic Category I Plate Components) Upper Shroud lower panel	DL+P+/- SSE	0.82	Acceptable
(Connections between NF Components and safety-related Seismic Category I Components) Connection of Messenger wire support ring tube assembly to Support columns	DL+P+T+/- SRSS (SSE, LOCA)	0.94	Acceptable

Notes for Table B:
DL: Dead Load
P: Pressure Load
SSE: Safe Shutdown Earthquake Load
ML: Maintenance Load (Live load on Walkways during maintenance activities)
LOCA: Loss of Coolant Accident Load
SRSS: Square-Root-of-the-Sum-of-the-Squares Load Combination Method
Stress Ratio = Computed Stress / Allowable Stress (Acceptable when less than or equal to 1.0)

The licensee combined the seismic results from the IHA analysis with other applicable stresses to obtain the total stress for each applicable load combination. The combined stresses in the IHA were determined at each critical location. Since each stress ratio determined in this manner is less than or equal to 1.0, the licensee proposes that it has adequately demonstrated, by finite element analysis of the IHA using damping values consistent with RG 1.61, Revision 1,

that the various IHA components meet the acceptance criteria of ASME Code, Section III, Subsection NF.

Therefore, the structural dynamic analysis performed by the licensee confirmed the IHA's ability to function under a postulated seismic disturbance, combined with other applicable loadings, while maintaining the resulting stresses under the ASME Code, Section III allowable values.

3.4 NRC Staff Evaluation of the IHA Analysis

The NRC staff reviewed the information provided in the application regarding the impacts of the increased mass of the IHA and concludes that the results are acceptable because the licensee properly considered the effect of the increased mass in the IHA analysis, and the impact on reactor vessel (RV) support, tie rod connections, and IHA interfaces.

The NRC staff reviewed the summary of results of the IHA analysis, as summarized in Table B, and concludes that the structural components of the IHA are acceptable because the stress ratios are less than 1.0. This indicates that they meet the applicable ASME Code, Subsection NF acceptance criteria limits with margin, for all five groups of components as listed in Table B. The staff also concludes that the licensee evaluated the stresses in Table B in accordance with the applicable ASME Subsection NF rules of the ASME Code, Section III, 2001 Edition through 2003 Addenda, and demonstrated their acceptability. Therefore, authorizing this request for the use of RG 1.61, Revision 1, weighted average damping values for steel structures with different connection types for the IHA analysis is consistent with applicable Commission regulations and will not adversely impact the health and safety of the public.

Based on the above, the NRC staff determined that compliance with the requirements of ASME Code, Section III, Subsection NF, along with the use of the weighted average damping values of 4.5 percent for OBE and 6.25 percent for SSE and LOCA in the structural design and analysis qualification of the IHA, provides reasonable assurance of maintaining an acceptable level of quality and safety. Therefore, the use of the RG 1.61, Revision 1 damping values, for the Callaway IHA is acceptable for OBE, SSE, and LOCA events along with the other loads in the applicable load combinations.

3.5 Licensee's Proposed Request Regarding the CRDM System

The original application did not discuss any changes to the CRDM system as a result of the RRVCH. Based on a review of the licensee's responses to the NRC staff's requests for additional information (RAIs), the licensee noted that the CRDM and its supports were changed because the replacement CRDM housing was a two-piece design compared to the original CRDM housing which was a three-piece design. Also, the number of tie rods restraining the CRDM seismic support structure was reduced from six to four. As a result of these changes to the CRDM design and support structure, a reanalysis of the ASME Code Class 1 CRDM system was performed by the licensee. The damping values used in the CRDM reanalysis were 5 percent for OBE and SSE, and 4 percent for LOCA. These values did not change from the original CRDM Class 1 analysis. A summary of the results from the replacement CRDM analysis for governing maximum stress locations are provided in Table C below:

Table C: Callaway Replacement CRDMs: Damping and ASME Code Stress Evaluation
(Summary extracted for the critical components with maximum ratio from Table 2 of Enclosure 1 of the licensee's letter dated August 29, 2013)

	OBE	**SSE**	**LOCA**
% Critical Damping	5%	5%	4%

	Location		**Actual /Allowable**	**Comment**
Design	CRDM Latch Housing	Pm	14.32 ksi/16.2 ksi = 0.884 < 1	Acceptable
Design	CRDM Nozzle at J-Groove Weld	Pl+Pb	30.73 ksi/34.9 ksi= 0.880 < 1	Acceptable
Normal & Upset	CRDM Nozzle at J-Groove Weld	P+Q	69.63 ksi/ 69.9 ksi=0.996 < 1	Acceptable
Normal & Upset	CRDM Nozzle at J-Groove Weld	CUF	0.618 /1.0 =0.618 < 1	Acceptable
Faulted	CRDM Latch Housing	Pm	25.83 ksi/37.9 ksi = 0.682 < 1	Acceptable
Faulted	CRDM Nozzle between Head & DMW	Pl+Pb	73.96 ksi/83.8 ksi= 0.882< 1	Acceptable

Notes for Table C:

Faulted includes SSE and LOCA
Pm: Primary Membrane Stress Intensity
Pl+Pb: Primary Membrane+ Primary Bending Stress Intensity
P+Q: Primary+ Secondary Stress Intensity Range
CUF: Cumulative Fatigue Usage Factor
DMW: Dissimilar Metal Weld

3.6 NRC Staff Evaluation of CRDM System

The NRC staff's review of the CRDM reanalysis determined that the stresses and fatigue cumulative usage factor for the ASME Code Class 1 load combinations met the applicable ASME Code limits. Therefore, the staff concludes that the design and support changes made to the CRDM system are acceptable because the reanalysis of the CRDM system shows that the applicable ASME Code, Section III Class 1 acceptance criteria are met.

3.7 NRC Staff Conclusion

The NRC staff reviewed the licensee's supporting technical information and the available margins from the results of the IHA structural dynamic analysis, as provided in the application, and the responses to the RAIs pertaining to the critical damping values for the IHA. The NRC staff determined that the proposed critical damping values for the IHA analysis are in accordance with RG 1.61, Revision 1, and the ASME Code, Section III, Subsection NF, and provide an acceptable level of quality and safety. The licensee's proposed damping values for the IHA analysis provide reasonable assurance that the IHA, as designed and constructed, will

perform its intended safety functions as required by the applicable Commission regulations. Therefore, the NRC staff finds that the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Based on this finding, the NRC staff concludes that the requested critical damping values for the IHA, based on the RG 1.61, Revision 1 weighted average approach for steel structures with different connection types, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding published in *Federal Register* on March 4, 2013 (78 FR 14139). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Basavaraju, NRR/DE/EMCB

Date: January 14, 2014

January 14, 2014

Mr. Adam C. Heflin
Senior Vice President and
Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, MO 65251

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO FINAL SAFETY ANALYSIS REPORT ASSOCIATED WITH SEISMIC DAMPING VALUES FOR THE INTEGRATED HEAD ASSEMBLY (TAC NO. MF0407)

Dear Mr. Heflin:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 207 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Final Safety Analysis Report – Standard Plant (FSAR-SP) in response to your application dated December 20, 2012, as supplemented by letters dated June 6 and August 29, 2013.

The amendment revises a methodology in the licensing basis as described in the FSAR-SP to include damping values for the seismic design and analysis of the integrated head assembly that are consistent with the recommendations of NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, March 2007.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. Amendment No. 207 to NPF-30
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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ADAMS Accession No. ML13358A005 *memo dated **Previously concurred

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DE/EMCB/BC
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NAME	JWachutka NLO w/c**	MMarkley	FLyon
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