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

July 30, 2013

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 – REQUEST FOR ADDITIONAL INFORMATION, ROUND 3, RE: ADOPTION OF NATIONAL FIRE PROTECTION ASSOCIATION STANDARD NFPA 805 (TAC NO. ME7046)

Dear Mr. Heflin:

By application dated August 29, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112420020), to the U.S. Nuclear Regulatory Commission (NRC), as supplemented by letters dated November 9, 2011 (ADAMS Accession No. ML113140044), April 17, 2012 (ADAMS Accession No. ML12108A239), July 12, 2012 (ADAMS Accession No. ML12194A624), and February 19, 2013 (ADAMS Accession No. ML13051A449), Union Electric Company (dba Ameren Missouri, the licensee) submitted a license amendment request to transition the fire protection licensing basis at the Callaway Plant, Unit 1, from Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.48(b), [Appendix R], to 10 CFR 50.48(c), "National Fire Protection Association Standard NFPA 805."

The NRC staff has determined that additional information, as requested in the enclosure, is needed to complete its review. Please provide a response to the questions within 30 days of the date of this letter. Review of your application is ongoing and additional questions may be forthcoming. If circumstances result in the need to revise the requested response date, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

CFLyon

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

Docket No. 50-483

Enclosure:
As stated

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT

NATIONAL FIRE PROTECTION ASSOCIATION STANDARD NFPA 805

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

By application dated August 29, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112420020), to the U.S. Nuclear Regulatory Commission (NRC), as supplemented by letters dated November 9, 2011 (ADAMS Accession No. ML113140044), April 17, 2012 (ADAMS Accession No. ML12108A239), July 12, 2012 (ADAMS Accession No. ML12194A624), and February 19, 2013 (ADAMS Accession No. ML13051A449), Union Electric Company (dba Ameren Missouri, the licensee) submitted a license amendment request (LAR) to transition the fire protection licensing basis at the Callaway Plant, Unit 1, from Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.48(b), [Appendix R], to 10 CFR 50.48(c), "National Fire Protection Association Standard NFPA 805."

The NRC staff has determined that additional information, as requested below, is needed to complete its review. With the request for additional information (RAI) responses, please provide an updated Table S to reflect any new commitments based on the answers to the RAI questions below.

Fire Protection Engineering RAI 18

The compliance statement for LAR Table B-1, Element 3.4.1 (c) [On-Site Fire-Fighting Capability] is "complies".

Please describe how the requirements of NFPA 805 Section 3.4.1(c) are met, specifically, "the brigade leader and at least two brigade members shall have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria."

An approach acceptable to the NRC staff for meeting this training and knowledge requirement is provided in Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," October 2009 (ADAMS Accession No. ML09258055), Section 1.6.4.1, Qualifications, which states, in part, that

> The brigade leader and at least two brigade members should have sufficient training in or knowledge of plant systems to understand the effects of fire and fire suppressants on safe-shutdown capability. The brigade leader should be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant systems.

Enclosure

Fire Protection Engineering RAI 19

Does the site have any storm drains in the yard area that discharge to the unrestricted area? If the answer is yes, then please confirm that you have completed a liquid release evaluation.

Probabilistic Risk Assessment Questions

Background: The NFPA-805 standard incorporated by reference into 10 CFR 50.48(c) states that the probabilistic risk assessment (PRA) approach, methods, and data shall be acceptable to the NRC. RG 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," October 2009 (ADAMS Accession No. ML092730314), identifies NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," September 2005 (Volumes 1 and 2 at ADAMS Accession Nos. ML052580075 and ML052580118, respectively), as documenting a methodology for conducting a fire PRA and endorses, with exceptions and clarifications, Nuclear Energy Institute (NEI) 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," April 2008 (ADAMS Accession No. ML081130188), as providing methods acceptable to the staff for adopting a fire protection program consistent with NFPA-805. RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014), describes a peer review process utilizing an associated American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) standard as one acceptable approach for determining the technical adequacy of the PRA. In its letter to NEI dated July 12, 2006, the NRC established the ongoing frequently asked question (FAQ) process where official agency positions regarding acceptable methods can be documented until they can be included in revisions to RG 1.205 or NEI 04-02.

NFPA-805 also requires that any change in public health risk that results from transition to an NFPA-805 based program from the plant's current fire protection program, and all future changes to the NFPA-805 based program, be acceptable to the NRC. RG 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (ADAMS Accession No. ML023240437), provides quantitative guidelines on core damage frequency and large early release frequency, and identifies acceptable changes to these frequencies that result from proposed changes to the plant's licensing basis. RG 1.174 also describes a general framework to determine the acceptability of risk-informed changes.

Probabilistic Risk Assessment RAI 36, Timing for Post-Fire Human Failure Events

The licensee reported relatively small error probabilities for some rapid actions. Acceptable methodologies for human error probability estimates generally assign large error probabilities for rapidly required responses. The licensee's response to RAI 07-B and RAI 35 discussed three specific cases where the time "margins" for completion of critical tasks were very short (approximately 1 minute or less). These were dispositioned via sensitivity evaluations where each human error probability (HEP) was assigned a value of 1.0 (totally unsuccessful). The reported increases in core damage frequency (CDF), large early release frequency (LERF), delta-CDF, and delta-LERF ranged from about 3 percent to about 15 percent, remaining below the numerical acceptance thresholds in RG 1.174, as cited in RG 1.205.

Please indicate whether these sensitivity evaluations (using HEP = 1.0) will be incorporated directly into the PRA or if alternative analyses (e.g., demonstrating more substantial time margins to support the original values) will be performed for the PRA. If it is the latter, please describe these analyses and provide the results.

Probabilistic Risk Assessment RAI 37, Use of Fractional Influence Factors for Transient Fires

NUREG/CR-6850 provides a method for apportioning transient fire frequencies among fire areas. FAQ 12-0064[1] provides an alternative method. In the responses to RAI 08-A and RAI 32, the licensee reported a "deviation from NUREG/CR-6850 (EPRI 1011989)" where fractional (<1.0) influence factors were assumed for certain transient fire scenarios. A special weighting factor of 0.05 was used for Maintenance in hot work prohibited zones and a factor of 0.1 was used for Storage in transient combustible free zones. Except for the reactor coolant pump (RCP) room in Containment, the minimum value for occupancy was 1.0; thus, the combined weighting factors were always greater than 1.0. The licensee indicated that these fractional values were always combined with at least a weight of 1.0 for the Occupancy influence factor. Therefore, the analyses as performed already constituted a sensitivity evaluation. However, with the publication of FAQ 12-0064, the accepted method now restricts use of fractional values to specific cases, which are not currently represented in the licensee analysis. Also, the RCP room was assigned a 0.0 for all three factors. Personnel occupancy and maintenance work does not occur in this area during power operation, due to health and safety concerns. The final transient frequency is 0.0, a value that is inconsistent with the ASME/ANS PRA Standard, although the impact on risk is negligible.

Please re-evaluate all cases, and provide the results, where fractional values were employed, including the RCP room where a total influence factor of zero was assigned, in accordance with FAQ 12-0064, or provide justification for not using FAQ 12-0064. Please indicate what method the licensee intends to use in its PRA when estimating the change in risk associated with post-transition changes to the fire protection program. If the method is not consistent with one of the two acceptable methods, please provide a justification for the proposed method.

Probabilistic Risk Assessment RAI 38, Low Ignition Frequency for Bus Duct Fires

The fire frequencies contained in NUREG/CR-6850 are average frequencies that have been compiled from industry wide experience and were developed to be generically applicable to each individual plant. The licensee's response to RAI 08-B cited plant-specific presence of "considerably fewer iso-phase bus ducts than a typical plant," to reduce the generic bus duct fire frequency by a factor of five. The licensee also provided the results of a sensitivity study without this reduction, which showed about 10 percent increases in CDF and LERF for the

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[1] Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association 805 Frequently Asked Question 12-0064 on Hot Work/Transient Fire Frequency Influence Factors," dated January 17, 2013 (ADAMS Accession No. ML12346A488).

ignition frequency bin, but less than 1 percent increases in total fire CDF and LERF. There was no change in the corresponding change in risk values.

Please indicate the values the licensee intends to use in its PRA when estimating the change in risk associated with post-transition changes to the fire protection program. If the licensee proposes to use the reduced value, please evaluate whether there is any equipment that might be more common at Callaway Plant than at an average plant such that the associated frequencies should be increased, and justify why it is acceptable for the licensee to modify frequencies that have been developed to be nominally applicable to all plants.

Probabilistic Risk Assessment RAI 39, Credit for Control Power Transformers (CPTs) for AC [Alternating Current] Circuit Failure Probabilities

Based on recent developments from cable fire tests, consensus between the nuclear industry and NRC is that the current credit for reducing "hot short" probabilities when CPTs are present now appears unverifiable. Volume 1, "Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," of NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," published in October 2012 (ADAMS Accession No. ML12313A105), states, in part, that

> Ultimately, the PIRT panel concluded that CPT size alone, nor indeed the mere presence of a CPT as the powering device, is not a predictable and repeatable circuit design parameter that reliably yields fewer spurious operations.

The licensee's application credited the presence of CPTs to reduce the hot short probability. In response to RAI 09-A, the licensee provided the results of a sensitivity study without taking credit for the presence of a CPT (nominally a reduction in hot short probability by a factor of 2). The licensee reported increases in CDF, LERF, delta-CDF, and delta-LERF ranging from about 30 percent to nearly 100 percent.

Please indicate whether the sensitivity analysis will be incorporated directly into the PRA, or if the recently published guidance based on the updated spurious actuation probabilities and durations will be used. Please indicate the values the licensee intends to use in its PRA when estimating the change in risk associated with post-transition changes to the fire protection program. If the factor of 2 will be maintained, please provide a justification for why the NUREG/CR-7150 conclusion is not applicable to Callaway Plant.

Probabilistic Risk Assessment RAI 40, Fire Growth Time to Peak Heat Release Rate for Trash Fires

FAQ 08-0052[2], in Supplement 1 to NUREG/CR-6850 (EPRI 1011989), suggested the 8-minute (min) growth time for common trash fires contained within receptacles. The licensee used a 10-min growth time. In response to RAI 10, the licensee provided the results of a sensitivity

[2] Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0052 Transient Fires - Growth Rates and Control Room Non-Suppression," dated August 4, 2009 (ADAMS Accession No. ML092120501).

evaluation using an 8-min fire growth time as the basis instead of 10 min indicating only about 0.1 percent increase in the CDF for control room fires. In the RAI response, the licensee also cited a re-evaluation of the data supporting the FAQ to support its use of 10 min, which the NRC staff did not accept as adequate justification for deviation from the FAQ. The FAQ recommendation is based on Tests 5 and 7 through 9 of NUREG/CR-4860 within plastic or metal receptacles. These are based on Tests 7 through 9 of NUREG/CR-4860, "Flaw Density Examinations of a Clad Boiling Water Reactor Pressure Vessel Segment," February 1988 (which is the reference cited by the licensee as its basis for assuming a 10-min growth time), and the National Institute of Standards and Technology (NIST) and Lawrence Berkeley National Laboratory (LBL) tests. The NRC staff questioned the reason for the licensee's inclusion of Tests 3 and 4 from this reference in its basis, since these two tests were previously discounted when the FAQ was developed, and the licensee provided no new justification for including the tests.

Please indicate what values the licensee intends to use in its PRA when estimating the change in risk associated with post-transition changes to the fire protection program. If the licensee proposes to use the 10-min factor, please provide additional justification.

Probabilistic Risk Assessment RAI 41, Uncertainty Analysis for Ignition Frequencies beyond FAQ 08-0048

FAQ 08-0048[3], in Supplement 1 to NUREG/CR-6850 (EPRI 1011989), requires limited sensitivity analyses for selected ignition frequency bins while the PRA standard requires uncertainty analyses for key assumptions. The licensee's application did not include any uncertainty analyses. In response to RAI 09-B, the licensee performed a sensitivity study of all ignition frequency bins by applying a multiplication factor that ratioed the 95th percentile frequency to the mean frequency for each bin in Supplement 1 to NUREG/CR-6850 (EPRI 1011989).

Please confirm that this uncertainty/sensitivity evaluation will be used as the uncertainty analysis regarding fire ignition frequencies for the PRA when the licensee estimates the change in risk associated with post-transition changes to the fire protection program.

Probabilistic Risk Assessment RAI 42, Effect of Internal Events PRA Update of Common Cause Failures (CCFs) on Fire PRA

The results of the most recent focused-scope peer review of the latest update to the internal events PRA indicated that the some CCFs were not modeled and that other CCF probabilities should be updated. The licensee's application stated that some CCFs were updated/added, but not in the fire PRA. In response to RAI 01-C and RAI 33, the licensee performed a sensitivity evaluation on CDF (CDF is the limiting risk metric at Callaway Plant, because LERF is always lower than 10 percent of CDF) using updated CCF probabilities from the internal events PRA in the fire PRA. The sensitivity analysis was conducted in two parts. The first part evaluated the

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[3] Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question 08 0048 Revised Fire Ignition Frequencies," dated September 1, 2009 (ADAMS Accession No. ML092190457).

CDF increase for those basic events already modeled in the fire PRA. For all the CCF events that have a direct match between the internal events and fire PRAs, the net change in both CDF and delta-CDF was negative. The second part was to evaluate the CDF increase for those basic events that are not modeled in the fire PRA. The only set of CCF events in the current internal events PRA which are not included in the fire PRA are the CCF combinations of the non-safety auxiliary feed water (NSAFP) pump and the safety-related motor-driven auxiliary feedwater (AFW) pumps. For this sensitivity study, a bounding risk approach employed surrogate events (NSAFP test and maintenance events), assuming the non-safety auxiliary feedwater pump is failed (basic event probability is set to 1.0). The results indicate that the increase in CDF remains under the RG 1.205 acceptance value of 1E-5/yr.

When the licensee estimates the change in risk associated with post-transition changes to the fire protection program, please indicate whether (1) the sensitivity analysis will be incorporated directly into the PRA, or (2) the PRA will be updated to include the latest CCFs. If the licensee proposes a different alternative, please justify why the most recent available data and most comprehensive modeling is not needed to support self-approval.

Probabilistic Risk Assessment RAI 43, Longer than Expected Time Available to Isolate Reactor Coolant System (RCS) Injection

The licensee credits about 36 min as available to isolate the RCS injection flow to avoid challenging the PORV during pressurizer overfill. This differs from the Callaway Final Safety Analysis Report (FSAR), Section 15.5.1.2, which states that the pressurizer becomes water solid following a spurious Safety Injection signal within 9 min, even if the operator terminates normal charging pump flow at 6 min. In its response to RAI-12, the licensee cited plant-specific calculations as the basis to justify the 36-min time frame and described the scenario in detail, providing the results of the Modular Accident Analysis Program (MAAP) analysis, which yields 36 min. This was further compared to the RETRAN analysis used for the FSAR estimate of about 9 min. The MAAP analysis, which is appropriate for PRA, involves best estimates, whereas the RETRAN analysis involves the much more conservative design basis. The key driver among the different parameter assumptions yielding the large difference in available time is the nominal flow rate into the RCS. In RETRAN, this is conservatively assumed to be 346 gallons per minute (gpm) for 6 min, followed by 299 gpm afterward. In MAAP, a supposedly more realistic 126 gpm flow rate is assumed throughout. However, each centrifugal charging pump is capable of discharging about 125 gpm into the RCS at full reactor pressure (based on the pump curve provided in the FSAR). The MAAP run appears to only assume that one centrifugal charging pump spuriously started at time zero and that the operators trip the normal charging pump early (5 – 6 min). The RETRAN calculation reflects what would be expected: for the first 6 min, the injection flow rate is based on the normal charging pump (about 47 gpm) plus two centrifugal charging pumps (about 299 gpm). After 6 min, the injection flow rate is based on the two centrifugal charging pumps (about 299 gpm). The RETRAN model appears optimistic in the flow rate (it credits about 150 gpm per pump versus an expected 125-130 gpm per pump). Regarding the apparently low injection flow rate in MAAP, doubling the injection flow (as would be expected for two high-head safety injection pumps) reduces the time in half, which means that only 18 min rather than the reported 36 min would be available. In addition, the licensee performed a sensitivity analysis taking no credit for an operator recovery action, which indicated increases in CDF, LERF, delta-CDF, and delta-LERF <10 percent.

When the licensee estimates the change in risk associated with post-transition changes to the fire protection program, please confirm that the sensitivity analysis will be used as the basis for the evaluation supporting the PRA, including any changes to the PRA, as needed, to reflect the sensitivity assumptions and results. Otherwise, please explain the apparent discrepancy between the RETRAN and MAAP results discussed above.

July 30, 2013

Mr. Adam C. Heflin
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Chief Nuclear Officer
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Sincerely,
/ra/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
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Docket No. 50-483

Enclosure:
As stated

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