March 23, 2011

ULNRC-05762

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

10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSE AMENDMENT APPLICATION FOR A TECHNICAL SPECIFICATION CHANGE THAT WOULD RELOCATE SPECIFIC SURVEILLANCE FREQUENCIES TO A LICENSEE CONTROLLED PROGRAM (LDCN 10-0020) (TAC NO. ME4506)

Reference: AmerenUE Letter (ULNRC-05725) dated August 5, 2010

In letter ULNRC-05725, Union Electric Company (dba AmerenUE, now Ameren Missouri) submitted an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant. The proposed amendment would modify Callaway Plant Unit 1 Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program consistent with Nuclear Energy Institute (NEI) 04–10, "Risk-Informed Technical Specification Initiative 5B, Risk-Informed Method for Control of Surveillance Frequencies."

During the NRC Staff review, a request for additional information (RAI) was transmitted to Ameren Missouri. Attachment 1 of this letter identifies the RAI questions/requests provided by the NRC Staff and provides the Callaway Plant response.

The No Significant Hazards Consideration determination provided in the original submittal is not altered by the additional information provided in this letter.

If there are any questions regarding this letter or the attached information, please contact Mr. Scott Maglio at (573) 676-8719 or Mr. Roger Wink at (314) 225-1561.

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

Very truly yours,

Executed on: 3/23/2011

Scott A. M

Scott A. Maglio
Regulatory Affairs Manager

Attachment 1 – Request for Additional Information (RAI) with Callaway Plant Response

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
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REQUEST FOR ADDITIONAL INFORMATION
(RAI) WITH CALLAWAY PLANT RESPONSE

By letter dated August 5, 2010 (ML102250056), Union Electric Company (Ameren Missouri) proposed changes to adopt the Nuclear Regulatory Commission (NRC) approved TS Task Force (TSTF)-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b." By email dated September 16, 2010 (ML102590588), the NRC provided its acceptance of this amendment request.

Following NRC staff review of the application, the NRC staff identified a number of questions/requests which were transmitted to Ameren Missouri via a Request for Additional Information (RAI). Each RAI question/request is identified below. Immediately following each RAI identified question/request is the Callaway Plant response to the RAI question/request.

1. In the application, the licensee identified peer review findings from the 2000 industry peer review (Table 2 of Attachment 2 of its submittal) and findings from the 2006 Scientech assessment (Table 1 of Attachment 2 of its submittal). In the staff's assessment of the overall technical adequacy of the internal events probabilistic risk assessment (PRA) model, the staff noted that the existing model (designated Update 4) appears to have a number of significant omissions and deficiencies, specifically:

   - Application of an outdated and non-conservative RCP seal LOCA model
   - Inadequate treatment of LERF
   - Inadequate and non-conservative treatment of internal floods
   - Inadequate treatment of system dependencies
   - Inadequate treatment of ISLOCA

   The licensee further stated that an updated PRA model (Update 5) is currently being developed which will address all findings from the 2006 assessment. This model is scheduled to be completed in the second quarter of 2011. The licensee has stated that, until that model is available, the findings from these reviews will be considered and sensitivity analyses completed as necessary. The staff requests the following clarifications:

   a) **Confirm the remaining open findings from the 2000 industry peer review will be addressed by the updated model.**

      Response:

      The scope of the PRA model update effort includes addressing open findings from the 2000 industry (i.e., Westinghouse Owners' Group (WOG)) peer review. Therefore, the updated internal events PRA model (i.e., "PRA Update 5") will include any changes required to address open WOG Facts/Observations (F&Os).

b) **Some findings involve missing scope (for example, 2000 peer review finding L2-1 states that containment isolation failures and internal floods are not considered in large early release frequency (LERF) calculations). Clarify how sensitivity analyses address missing scope items.**

Response:

Core damage frequency is typically the limiting risk metric for the Callaway Plant. This is due to Callaway Plant's large-volume, robust containment building. The baseline LERF is dominated by Interfacing Systems Loss of Coolant Accidents (ISLOCA) and Steam Generator Tube Rupture (SGTR) events. It is anticipated that most Surveillance Test Interval (STI) extension evaluations will be limited by core damage risk, not large early release risk.

For missing scope items, separate PRA analyses can be defined and performed to estimate the associated risk due to a proposed STI extension. Using the case of the intentional omission of containment isolation from the current LERF model, for example, an analysis could be performed wherein the core damage cutset equation is logically combined (AND' d) with the containment isolation failure cutset equation in order to estimate containment isolation failure contribution to LERF risk. Any such separate, but necessary, additional PRA analyses would be defined, run and evaluated by the analyst performing the given STI extension risk assessment.

c) **Some findings were dispositioned by stating that the updated model will be used to evaluate surveillance interval changes:**
   i. **Is the updated model sufficiently complete to permit its use in the Surveillance Frequency Control Program?**
   ii. **If so, is it the intent that the revised model be applied for any required sensitivity studies required for this application?**
   iii. **Provide the basis for the implementation of this model being delayed until mid-2011 if it is needed to support this application?**

Response:

At the present time, the updated internal events PRA model is not sufficiently complete to permit its use in the Surveillance Frequency Control Program.

The internal events PRA model update/upgrade project is a large-scope effort that is revisiting each of the internal events PRA elements addressed in the ASME/ANS PRA Standard. The project was initiated in 2008, and is anticipated to be completed approximately mid-2011. The updated model will become the Callaway Plant internal events PRA model of record, and will be used for PRA applications, including the Surveillance Frequency Control Program, upon its completion.

As part of the PRA update/upgrade project, the models for certain "stand alone" risk contributors, such as ISLOCA, have been upgraded. Other models, such as that for internal flooding, are nearing completion. To the extent that these models are available, or become available, they would be used to evaluate STI changes for which the risk contributor represented by the model (e.g., ISLOCA) is deemed relevant.

Plant modifications requiring PRA model updates are processed as interim updates to Callaway Plant PRA model Update 4. One such interim update, i.e., PRA Update 4b, has recently been completed to reflect the installation of an alternate electrical supply source that can be used to power safety related busses under station blackout scenarios. It is noted that the PRA model review findings, addressed in Tables 1 and 2 of the license amendment request, pertain to PRA Update 4 and the interim updates to PRA Update 4, including PRA Updates 4a and 4b. All interim updates will be incorporated into Update 5 and will be subject to industry peer review in 2011.

2. **In Attachment 2 Table 1, several items were dispositioned by sensitivity studies which show a minor impact on results. However, for findings AS-4 and SY-1, more detail is needed for the staff to conclude that the sensitivity studies bound the specific findings. The staff requests further specification on exactly how the model was varied in the sensitivity cases, and how this bounds the deficiency being evaluated.**

Response:

Finding AS-4: Scientech finding AS-4 from Attachment 2, Table 1 of the license amendment request states, "The RCP seal LOCA model needs to be updated to reflect the latest WOG model, which is approved by NRC." The RCP seal LOCA model used in the current Callaway Plant internal events PRA model of record is based on WCAP-10541, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power," Revision 2. For comparison of these seal LOCA models, the probabilities of the minimum and maximum per-RCP seal LOCA leakage rates are provided in the table below.

Leakage Rate (per RCP)	WOG2000 Leakage Rate Probability*	Current Callaway Internal Events PRA Leakage Rate Probability*
21 gpm	0.790	0.894
480 gpm	2.5E-3	3.13E-3

*Values shown are with RCS depressurization.

The sensitivity cited in Attachment 2, Table 1 of the license amendment request, in which the baseline Callaway Plant CDF increased by only 1.5 percent when RCP seal LOCA-related parameters were increased, involved increasing the RCP seal LOCA-related core uncovery probabilities in the core damage cutset equation by 50 percent. The intent of this sensitivity was to show that adverse RCP seal LOCA-related parameters in the model could be increased appreciably and this would result in only a minor change in CDF.

Finding SY-1: As noted in Table 1 of Attachment 2 of the license amendment request, the scope of finding SY-1 was that (1) the dependency of Main Feedwater on instrument air (IA) needs to be included in the model and (2) the applicability of data used for undeveloped events for loss of IA and failure of actuation signals needs to be verified. The sensitivity analysis that was performed to address this finding involved the following:

- An updated probability for the instrument air system undeveloped event was determined based on data reported in NUREG CR/6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." This updated probability was input into the Callaway Plant core damage cutset equation as the probability of the loss of instrument air undeveloped basic event. The updated probability was also added to the operator error event representing failure to re-establish feedwater flow using the Main Feedwater System should auxiliary feedwater fail. This latter change emulates the dependency of Main Feedwater on Instrument Air. Note that Main Feedwater is not credited in the Callaway Plant PRA model in scenarios (e.g., following a LOOP event) in which Instrument Air would be unavailable.
- Actuation signal failure probabilities resident in the core damage cutset equation were increased by 10 percent. This item, and using the updated instrument air system failure probability per the bullet above, was intended to address item (2) of the finding.

When the above probability changes were made in the core damage cutset equation, the baseline Callaway Plant CDF increased by only 0.59 percent, indicating that this finding would not preclude implementation of a surveillance frequency control program.

3. **In Attachment 2 Table 2, finding IE-7 states that the interfacing systems loss-of-coolant accident (ISLOCA) evaluation does not consider scenarios without containment bypass. The licensee has stated that it disagrees with this finding. Provide the complete peer review discussion of this item for the staff to better understand the scope of the issue and the disposition.**

Response:

Following is the relevant text from finding IE-7 of the WOG PRA Peer Review:

"1. Interfacing system LOCA locations are limited to only those scenarios where containment may be bypassed. There are several lines where ISLOCAs can occur and lead to a loss of coolant inside containment that is not covered by the other LOCAs inside containment (the frequency for which is based on pipe ruptures rather than interfacing valve failures) and which could lead to a LOCA inside containment with loss of mitigating system function. Additionally, the assumption of having greater than three valves as a barrier may neglect the consequence of the line failure leading to loss of multiple trains of equipment."

The above-quoted finding excerpt primarily questions whether there could be ISLOCA locations inside containment that could lead to loss of mitigating system function. This comment was deemed not to be valid for the following reasons:

1. Various documented ISLOCA definitions, e.g., as provided in SR IE-A2, item (d) of the ASME/ANS PRA Standard, either explicitly state, or imply, that ISLOCAs occur outside containment.

2. For any LOCA location inside the Callaway Plant containment, the water exiting the break will drain to one of the two safety-related containment sumps and then be available for Emergency Core Cooling System (ECCS) recirculation.

Item 1 of this F&O also states that line failures leading to the loss of multiple trains of equipment may have been neglected in the original ISLOCA analysis. In the currently available ISLOCA analysis update, the only equipment trains explicitly credited are associated with high-head ECCS injection. The event tree success criterion for a Residual Heat Removal (RHR) leak or rupture is any one of the Centrifugal Charging Pumps (CCPs) or Safety Injection Pumps (SIPs) (i.e., any one of the four pumps) in ECCS injection phase. Since an RHR system leak or rupture would not affect the functionality of the CCPs or SIPs, this success criterion is valid. The success criterion for an SI leak or rupture is any one of the two CCPs in the ECCS injection phase. This success criterion accounts for the fact that the SI leak or rupture may preclude functionality of one or both SI trains for ECCS. Therefore, no action is required in response to this second sub-comment of Comment 1 of this finding.

4. **In Attachment 2 Table 1, finding IE-8 identifies that recovery events do not have sufficient analysis or data, but there is no discussion of the significance of this finding. The staff requests further discussion of these events in more detail, identifying their probabilities, basis, and importance to the PRA results, in order to justify the adequacy of the existing PRA model.**

<u>Response</u>:

Finding IE-8 states, "The Callaway Plant PRA credits repair of hardware faults in the recovery of the loss of CCW and loss of SWS initiating events. The repair events, which include repair of CCF of pumps and valves lack sufficient analysis or data. Crediting repair of components is not acceptable unless the probability of repair is justified through an adequate analysis or examination of data."

The table below provides the applicable events and their current probabilities.

Basic Event	Description	Probability
FTR-CCW-RHR-REC	Failure to recover Component Cooling Water (CCW) prior to swap-over to RHR recirculation	0.221
EG-PSF-FC-CCWSYS	Operator fails to recover CCW after loss of the system	0.330
EA-PSF-FC-SWESW	Operators fail to recover Service Water (SW) in 2 hours due to equipment failure	0.352
ES-PSF-FC-SWESW8	Operators fail to recover SW in 8 hours due to equipment failure	9.30E-2

These basic events are included in fault trees used to represent event tree headings for CCW and SW recovery in the loss of all CCW and loss of all SW event trees, respectively. Development of the event probabilities included a review of the loss of CCW and loss of SW initiator cutsets by an individual in the PRA group with previous System Engineering experience, who made a judgment as to which components/failure modes could be repaired, and the timeframe required for the repairs. Thus, contrary to the finding, the credited repair of hardware faults in the recovery of CCW and SW did involve analysis by a qualified individual. Regarding the "lack of sufficient data" aspect of the finding, pertinent IPE-era reference documents providing system recovery probabilities were used; however, it is not clear that these documents were ever published such that they would constitute a valid reference today.

To respond to this RAI question, a sensitivity case was run in which the above basic event probabilities were set to 1.0, i.e., no credit taken for the recovery. The baseline Callaway Plant internal events CDF increased by only 1.56 percent. Therefore, this finding would not preclude implementation of a surveillance frequency control program.

5. **In Attachment 2 Table 1, findings associated with requirement AS-B1 identify that initiating event impacts on mitigating functions may not be properly captured. This has been dispositioned based on a sensitivity analysis done on a prior application for Emergency Service Water. These findings point to a potential fundamental flaw in the PRA model if initiator impacts are not properly addressed in the logic. Since the scope of the findings is not identified, the staff requests additional specific details for these findings and a demonstration that the scope and impact of these findings is not significant for this application.**

Response:

The table below provides the text of the findings in question, as well as the bases for concluding that the findings do not preclude implementation of a surveillance frequency control program at Callaway Plant.

Finding No.	Finding	Basis that the finding does not preclude implementation of a surveillance frequency control program.
AS-1	"Event Tree T(SW) function L2SW-M should evaluate the TDAFW pump with no functioning SW/ESW equipment. The cutsets for this function include failures of the ESW pumps and human action failures for alignment of SW/ESW. Since the initiator fails all SW/ESW, the logic should not include these events. A similar situation exists for function L2T1s. Event Tree T(SW) function O1SW-M includes a FANDB operator error which does not belong in the function. A similar situation exists for functions O1C-M, O1CT1-M, and O1SW-M."	This finding relates to a relatively small number of erroneous cutsets that were found by the reviewer in a small number of cutset equations representing event tree headings. For the updated/upgraded PRA model, the cited errors were thoroughly investigated, and minor changes were made to the applicable logic models, where necessary, to correct the errors. As stated, the errors were noted in cutset equations representing event tree headings. They had not previously been noted during reviews of actual core damage cutsets generated from the PRA model. As a reasonably thorough review of cutsets was performed following quantification of the current model, and the cited errors were not noted, the errors do not result in any significant impact on the core damage results. Also, as noted in the original license amendment request, a sensitivity evaluation performed for the previously approved one-time Essential Service Water (ESW) Completion Time (CT) extension application determined that correction of findings AS-1, -3 and -7 would result in only a 1% increase in the PRA Model Update 4 baseline

		CDF. Based on the above discussion, this finding would not preclude implementation of a surveillance frequency control program.
AS-3	"The method of event tree analysis for support system initiators does not appear to correctly capture the failed dependencies in the mitigating systems for some support system IE's. A single basic event is used for the initiating event. House events are included in the fault trees to turn off the affected trains when a support system is not available. It is not clear there are sufficient support systems modeled in the main feedwater and non-safety service water to fail these systems when their support systems are unavailable. This may occur in Tsw, Tnk01, and Tnk04. The cutsets for Tsw, Tnk01, Tnk04, and Tccw should be checked to search for systems that would be failed by the loss of the initiator, and then modify the fault trees to include the appropriate house events to disable these systems."	The finding is related to SR AS-B1 of the ASME/ANS PRA Standard which, in effect, requires that dependencies of the mitigating systems on the initiating events (in particular, support system initiators) be included in the event tree and/or fault tree models, such that the dependencies are adequately addressed. The finding cites no specific examples, e.g., cutsets, where a mitigating system, which would be failed by the occurrence of an initiator, has been credited to mitigate the initiator. In response to the finding statements "....It is not clear there are sufficient support systems modeled in the main feedwater and non-safety service water to fail these systems when their support systems are unavailable. This may occur in Tsw, Tnk01, and Tnk04.....", the following information is provided: • The main feedwater and non-safety service water fault trees were reviewed for external transfers. Both trees have the appropriate support system dependencies modeled in terms of external transfers, especially in light of the item below. • Contrary to the excerpted finding statements cited above,

		main feedwater is not credited (i.e., is not used in the event trees) for mitigation of T_{SW}, T_{NK1} or T_{NK4}. In addition, non-safety service water is only credited following a T_{SW} event for sequences in which service water has been recovered and non-safety service water has no dependency on NK01 or NK04. Irrespective of the bulleted information above, and in response to the last sentence of the finding, in order to further investigate this finding, core damage cutset files for the cited initiators, from PRA Update 4, were reviewed. The review focused on the validity of cutsets, and looked for any cutsets reflecting credit for mitigation by a system that would not be available due to the occurrence of the initiator. No such cutsets were found. All cutsets reviewed were correct. Based on the above discussion, no further action relative to the finding is required, and the finding does not preclude implementation of a surveillance frequency control program.
AS-5	"Room cooling requirements for the switchgear rooms for SBO should be re-evaluated to consider the actual heat loads in the rooms during SBO."	This finding suggests re-evaluation of the switchgear room cooling requirements for SBO conditions. It has since been determined that switchgear room cooling is not required for SBO conditions. Use of the current PRA model, which requires switchgear room cooling, would be conservative. Therefore, this finding would not preclude implementation of a surveillance

		frequency control program.
AS-7	"Specific errors are as noted below: • Function O1T1S in the SBO event tree contains basic events for MFW and SW as a backup source for water to SGs if the TDP fails. The problem occurs in the SECDEP fault tree which asks for GMFX100 but does not have any logic to cancel the gate in SBO. There are no events in the MFX fault tree which will cancel it in the event of an SBO, either. Also, in MFW.lgc, gate GMFW413 – the SVC system – will be failed by LOSP, but comes through the link in the SBO function. Back-up sources of water to the SG are modeled at a high level, often only represented by an HEP. There needs to be either, a) support systems developed which will be failed by LOSP or AC power, or b) house event logic to fail these for SBO. • The AFW function on the TSW event tree – (L2SW-M) – has recovery factors for ESW as a suction source to the turbine driven AFW pump. (AL-XHE-FO-AFWESW). ESW is failed by the initiator, but the IE is a basic event, not cutsets. Need to represent the initiator as a support system fault tree, OR need to include house events in the AFW function to fail the cross-tie to the ESW system after a Loss of ESW. • In TSW event tree, function O1SW-M has an event (AE-XHE-FO-MFWFLO) for failure of MFW as back up to AFW.	The errors identified were investigated, and corrections were made to the updated/upgraded PRA logic model. As stated for finding AS-1, the errors were noted in cutset equations representing event tree headings, but were not noted during the post-quantification review of core damage cutsets from the current PRA model. Thus, the errors do not have a significant impact on the PRA results. Also as noted for finding AS-1, a sensitivity evaluation performed for the previously approved one-time ESW Completion Time (CT) extension application determined that correction of findings AS-1, -3 and -7 would result in only a 1% increase in the Update 4 baseline CDF. Based on the above discussion, this finding would not preclude implementation of a surveillance frequency control program.

	MFW is unavailable after loss of SW. Need to include support systems for MFW or insert house events in fault tree to turn off MFW for loss of TSW."	.

6. **In Attachment 2 Table 1, finding AS-4 states that the reactor coolant pump (RCP) seal LOCA model is out-of-date. The disposition of this item identifies the use of an "older-vintage" model.**

    a) **The staff requests that the specific basis for the current seal LOCA model be identified.**

    Response:

    The RCP seal LOCA model used in the current Callaway internal events PRA model of record is based on WCAP-10541, Rev. 2, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power." See the response to b) below, regarding the basis for using the WCAP-10541, Revision 2, RCP seal model.

    b) **If the updated model referred to in the submittal uses the correct seal LOCA model, identify this model and provide an assessment of its impact on the overall results compared to the existing older model.**

    Response:

    The updated/upgraded Callaway internal events PRA model uses the "WOG2000" RCP seal LOCA model, based on the implementation guidance of WCAP-16141, "RCP Seal Leakage PRA Model Implementation Guidelines for Westinghouse PWRs." Since the updated PRA model has not yet been quantified, an actual assessment of the impact of use of the WOG2000 seal LOCA model cannot be made. However, the sensitivity described in the response to RAI question 2 for finding AS-4 provides an indication of the sensitivity of the Callaway Plant internal events PRA to the RCP seal LOCA model used. The results of this sensitivity indicate that use of the older seal LOCA model would not preclude implementation of a surveillance frequency control program.

7. **In Attachment 2 Table 1, finding QU-1 identifies that the correlated data probabilities are not accounted for during quantification, and the disposition identifies that this will be addressed by sensitivity studies. The staff requests that a discussion of the significance of this finding be provided in order to conclude that sensitivity analyses are an adequate means to address the issue.**

Response:

The actual text of this finding is: "The current quantification does not include an uncertainty calculation to account for the "state-of-knowledge" correlation between event probabilities. The structure exists to perform this correlation within WinNUPRA but at the current time it has not been done."

When performing a parametric uncertainty analysis using a Monte Carlo sampling approach, such as that used by WinNUPRA, Callaway Plant's PRA software, the state-of-knowledge correlation is accounted for by using the same sample value for each basic event whose probability is estimated using the same data. In the current Callaway Plant internal events PRA model, the state-of-knowledge correlation is not accounted for in parametric uncertainty analyses. This limitation, however, does not impact use of the STI evaluation guidance of NEI 04-10, Rev. 1. That is, global parametric uncertainty analysis is not part of, or used in, the NEI 04-10, Rev. 1 guidance. Therefore, this finding would not preclude implementation of a surveillance frequency control program at the Callaway Plant.

Note that Attachment 2 of the original license amendment request indicated that this finding/gap can be addressed, if necessary, with sensitivity studies. In fact, it is unlikely that such studies would actually be required when evaluating a STI change.

8. **In Attachment 2 Table 1, finding LE-1 identifies that some LERF contributors are not addressed by the PRA model. The staff requests identification of the missing scope items and justification be provided that their contribution to LERF would not be significant in order to conclude that sensitivity analyses are an adequate means to address the issue.**

Response:

For clarity, the following is the actual text of finding LE-1.

"Probability of containment isolation failure leading to LERF does not contain a term to represent undetected, residual failures in containment structural integrity. This has been estimated at 5E-3 in NUREG/CR-4550. Failure of containment isolation is derived by fault tree analysis of the containment isolation combinations on the penetration paths. There are three LERF split fractions with probabilities of 7.7E-4. If the 5E-3 was added to this, the split fraction would change, although LERF would not move significantly. Split fractions for induced SGTR and HPME were not explicitly stated in the documentation available for review."

Regarding the lack of a term for undetected, residual failures in containment structural integrity item, the finding itself concludes that this item would not have a significant impact on the calculated baseline LERF. It is also noted that inclusion of a term for undetected, residual failures in containment structural integrity does not appear to be required by the ASME/ANS PRA Standard.

Additionally, induced steam generator tube rupture and high pressure melt ejection are considered in the current Callaway LERF model.

Based on the above discussion, this finding would not preclude implementation a surveillance frequency control program.

9. **In Attachment 2 Table 1, several findings against internal flooding are addressed stating that NEI 04-10 allows the use of qualitative or bounding analyses to address internal flood contributors. NEI 04-10, Revision 1, specifies Regulatory Guide 1.200, Revision 1, as the governing document for PRA technical adequacy. Internal flooding initiators are specifically addressed in the internal events standard endorsed by this regulatory guide. Therefore, based on this discussion, the use of a quantitative PRA evaluation is required, and from staff review of the application, qualitative or bounding evaluations are not addressed. The staff requests licensee disposition of internal flooding findings from the peer review be re-evaluated.**

Response:

Based on additional review and investigation of the NEI 04-10, Rev. 1, guidance, Callaway Plant concurs that STI evaluations must address the contribution from internal flooding risk quantitatively. The general process that Callaway Plant will use to quantitatively determine the internal flooding risk contribution associated with an STI increase is described below. In addition, in response to the staff's specific RAI request that Callaway Plant re-evaluate the disposition of the internal flooding findings for this application, the following information is provided.

Finding IF-1: This finding states, "This requirement [i.e., IF-D5 and IF-D5a] is met to Category I. The flood initiating event frequencies are based on generic pipe break frequencies. No plant specific experience is considered in the determination of the flooding initiator frequencies. Plant experience at the time the flooding analysis was performed was 0 events. Documentation of the plant specific considerations used in the development of the scenarios needs to be added as discussed in SR IF-D5a."

The current Callaway Plant internal flooding analysis uses generic pipe break frequencies from EPRI TR-102266, "Pipe Failure Study Update," April 1993. No plant-specific experience was factored into these pipe break frequencies for the current Callaway Plant internal flooding analysis. The updated internal flooding analysis uses generic flood initiator frequencies from EPRI 1013141, "Pipe Rupture Frequencies for Internal Flooding PRAs, Revision 1," March 2006, which are Bayesian-updated with Callaway Plant specific experience. When determining the internal flooding risk contribution due to a proposed STI extension, the internal flood initiator frequencies from the updated flooding analysis will be used. This will serve to address this finding.

Finding IF-2: This finding states, "This requirement [i.e., IF-E3a] is not met at any

Category. The Category I/II screening quantitative criteria in the standard is 1E-09/year. ZZ-466 screening criteria was 1E-06/yr."

This finding is somewhat misleading in that the current Callaway Plant internal flooding analysis uses a criterion of 1E-6 per year, not as a screening threshold, but instead as a criterion for developing a more detailed analysis for the flood area in question. The calculated core damage frequency for each flood area analyzed in the current flooding analysis was included in the total internal flooding CDF value. No flood areas were screened from, i.e., not included in, the current internal flooding analysis based on the 1E-6 per year criterion. Therefore, this finding does not preclude implementation of a surveillance frequency control program. However, since the intent of the updated flooding analysis is to meet Capability Category II of the Standard, it would be prudent to compare the flood areas identified and quantified in the updated flooding analysis to those in the current flooding analysis, and add to or subtract from the set of flood areas in the current flooding analysis, as appropriate, prior to quantifying the internal flooding contribution to core damage risk associated with a proposed STI increase. This step is included in the process described later in this RAI response. (See page 16.)

<u>Finding IF-3</u>: This finding states, "This requirement [i.e., IF-C6 and IF-C8] is met to Category I only. ZZ-466 allows the operator intervention and mitigation for floods that take 30 minutes or longer. Isolation and available manpower are not specifically addressed. Isolation and available manpower should be considered and documented with the revised screening discussed in F&O IF-2."

The current Callaway Plant internal flooding analysis credits operator intervention and mitigation of some floods. The human error probabilities (HEPs) used were 1.0 (if the critical flood height was reached in less than 30 minutes), 0.5 (if the critical flood height was reached in between 30 minutes and one hour) or 0.1 (if the critical flood height was reached in greater than one hour). These HEPs were applied without regard to certain performance shaping factors that are required to be considered by the Standard to meet Capability Category II requirements. This is the basis for the finding. However, since the HEPs in question were applied to the flood initiator frequencies in the current flooding analysis, and Callaway Plant will use the flood initiator frequencies from the updated internal flooding analysis for STI evaluations, which do not credit operator mitigation of floods, this finding is not relevant to this application.

<u>Finding IF-4</u>: This finding states, "If additional human failure events are required to support quantification of flood scenarios, PERFORM any human reliability analysis in accordance with the applicable requirements described in Tables 4.5.5-2(e) through Table 4.5.5-2(h). This requirement is not met. The HEP values used in ZZ-466 are not developed from a human reliability analysis."

The HEP values used in ZZ-466, noted in the finding, are those described for Finding IF-3, above. As the finding notes, these HEPs were not developed from a human reliability analysis, per se. As noted for finding IF-3, the HEPs in question were applied to the flood initiator frequencies in the current flooding analysis. However, Callaway Plant will use the flood initiator frequencies from the updated internal flooding analysis for STI evaluations, which do not credit operator mitigation of floods. Therefore, this finding is not relevant to this license amendment request.

Finding IF-5: This finding states, "For each defined flood area and each flood source, IDENTIFY those automatic or operator responses that have the ability to terminate or contain the flood propagation. This requirement is not met. ZZ-466 treats operator response in a generic sense."

As the finding notes, the current Callaway Plant internal flooding analysis treats operator response (to terminate a flood) in a generic sense. The associated Supporting Requirement (SR) of the Standard requires, in part, that these operator actions be treated in a flood area- and flood source-specific fashion. Thus, this SR is not met in the current Callaway Plant internal flooding analysis. Again, however, since the current internal flooding analysis applies credit for the operator actions in question to selected flood initiating event frequencies, and since for STI evaluations the internal flooding contribution to risk will be determined using flood initiator frequencies which do not credit these operator actions from the updated internal flooding analysis, this finding is not relevant to this application.

Finding IF-6: This finding states, "For each flood scenario, REVIEW the LERF analysis to confirm applicability of the LERF sequences. If appropriate LERF sequences do not exist, MODIFY the LERF analysis as necessary to account for any unique flood-induced scenarios or phenomena in accordance with the applicable requirements described in paragraph 4.5.9. This requirement is not met. The internal flooding sequences are not considered in the LERF analysis."

Note that this finding is similar to finding L2-1 of the WOG PRA Peer Review.

As indicated previously in these RAI responses, it is anticipated that LERF will rarely be the limiting risk metric relative to STI extension evaluations for Callaway Plant. However, the internal flooding-initiated contribution to LERF risk, associated with a proposed STI increase, will be estimated by multiplying the flooding-initiated CDF increase determined for the STI increase by an appropriate conditional probability of LERF. Thus, the impact of the proposed STI increase on flood-initiated LERF will be included in the STI extension evaluations. This will serve to address this finding.

General Process that Callaway Plant Will Use to Quantitatively Determine the Internal Flooding Risk Contribution Associated with an STI Increase:

It is anticipated that the upgraded Callaway Plant internal flooding PRA model will be available to support STI extension evaluations by the time that this amendment request is approved and implemented. The upgraded internal flooding model is intended to meet Capability Category II of the Standard. However, if there is an interim period during which the current internal flooding model would need to be used for STI extension evaluations, the following general process would be used for determining the core damage and large early release risk metrics associated with an STI increase.

a. Internal flooding initiating event frequencies from the updated Callaway Plant internal flooding PRA model will be used. These flood initiator frequencies include plant-specific experience and do not credit operator mitigation actions. Use of these initiator frequencies serves to address findings IF-1, IF-3, IF-4 and IF-5.

b. The screening quantification results from the upgraded internal flooding analysis will be evaluated to identify any flood areas that should be added to or subtracted from the current flooding analysis prior to quantification of the internal flooding contribution to core damage risk for the STI change under consideration. This step is related to finding IF-2.

c. Parameters in the current internal flooding PRA model, reflecting the STI increase under evaluation, will be adjusted using the NEI 04-10, Rev. 1, guidance. The internal flooding PRA will then be quantified to determine the core damage risk increase associated with the STI increase under evaluation.

d. The core damage frequency adjustment determined via step c, above, will be multiplied by an appropriate conditional probability of large early release to obtain an estimate of the increase in large early release frequency. An appropriate conditional probability of large early release will be selected based on consideration of the impact of the STI increase under consideration. For example, if the candidate STI increase could increase the likelihood of containment bypass, an applicable conditional probability of large early release will be selected and used. This step serves to address finding IF-6.