AmerenUE
Callaway Plant

PO Box 620
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April 22, 2010

ULNRC-05694

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
UNION ELECTRIC CO.
APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE NPF-30
COMPLETION TIME EXTENSIONS FOR TS 3.3.2
ENGINEERED SAFETY FEATURE ACTUATION SYSTEM (ESFAS)
INSTRUMENTATION FUNCTIONS
TAC NO. ME2822 (LDCN 09-0039)

Reference: ULNRC-05665 dated November 25, 2009

In the above reference, AmerenUE submitted an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant.

That amendment application proposed changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Action System (ESFAS) Instrumentation," that would add a new Required Action Q.1 to require restoration of an inoperable Balance of Plant ESFAS (BOP ESFAS) train to OPERABLE status within 24 hours. Currently, Condition Q of TS 3.3.2 for Function 6.c of TS Table 3.3.2-1 requires the plant to enter a shutdown track to MODE 3 within 6 hours and to MODE 4 within 12 hours with no allowed outage time provided for restoration. In addition, the Completion Times for TS 3.3.2 Required Actions J.1 and O.1 to trip inoperable channels that provide inputs to BOP ESFAS would also be extended to 24 hours. Shutdown track Completion Times to be in MODES 3 and 4 would be increased to reflect these longer restoration times.

During the NRC staff's acceptance review a request for additional information (RAI) was identified. Attachment 1 provides the requested information. The information provided in Attachment 1 does not affect the licensing evaluations submitted in the referenced application or alter their conclusions.

AmerenUE continues to request approval of this proposed license amendment prior to November 20, 2010. AmerenUE further requests that the license amendment be made effective upon NRC issuance to be implemented within 90 days. As was the case with the referenced application, no commitments are contained in this correspondence. If you have any questions on this amendment application or the attached information, please contact me at (573) 676-8719 or Mr. Thomas Elwood at (314) 225-1905.

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

Very truly yours,

Executed on: 4/22/2010

Scott A. M

Scott A. Maglio
Regulatory Affairs Manager

Attachment 1: RAI Responses

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
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RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
QUESTIONS REGARDING LICENSE AMENDMENT REQUEST LDCN 09-0039

In its letter dated November 25, 2009 (i.e., letter ULNRC-05665), AmerenUE (the licensee) submitted a request to incorporate proposed changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Specifically, AmerenUE proposed Completion Time extensions that would allow 24 hours for equipment restoration in Conditions J, O, and Q of TS 3.3.2 for functions 6.c, 6.g, and 6.h of TS Table 3.3.2-1. AmerenUE's request is currently under review by the NRC staff.

In a separate letter dated March 20, 2009 (i.e., letter ULNRC-05598), AmerenUE submitted a license amendment request (LDCN 09-0008, TAC No. ME0986) for a one-time extension of the containment Type A integrated leak rate test (ILRT) interval for Callaway Plant. That request was approved via License Amendment 195 as issued by NRC letter dated March 17, 2010. During the NRC review of that license amendment request an RAI was identified in which information was requested pertaining to the plant probabilistic risk assessment (PRA) model developed for Callaway Plant. The requested information was tied to the disposition of peer review and gap analysis findings.

In order to continue the review of AmerenUE's license amendment request LDCN 09-0039 (BOP ESFAS amendment request), the NRC staff requested that the following questions addressed during the review of LDCN 09-0008 (15-year ILRT extension request) also be addressed with respect to LDCN 09-0039 where applicable. The following Requests for Additional Information are tied to gap analysis finding item numbers included in Table 2.1.3 of Attachment 5 to ULNRC-05598 dated March 20, 2009 and peer review finding item numbers in Table 2.2.1 of Attachment 5 to ULNRC-05598 (15-year ILRT extension request) with cross-references, where applicable (in parentheses), to peer review and gap analysis finding item numbers in Tables 1 and 2, respectively, of Attachment 6 to ULNRC-05665 (BOP ESFAS amendment request).

1. Item 3 (no corresponding gap analysis item number in Table 2 of Attachment 6 to ULNRC-05665 – see response below) identifies credit for repair of hardware faults for certain initiator models without sufficient analysis and data, including repairs of (common cause failures) CCFs. To support credit for repairs in the PRA model, the licensee must have identified repair rates and times for the specific components and failures for which the repair is credited. Please provide the non-recovery / non-repair probabilities applied in the PRA model and their bases. Also, please provide a sensitivity calculation for this application that takes no credit for the repair of hardware faults, including associated CCFs.

Response:

Based on the schedule for PRA Update 5 at the time of the BOP ESFAS amendment request submittal, and the planned closure of all Initiating Event (IE) peer review findings in Update 5, no WOG Peer Review IE Facts and Observations (F&Os) were included in Table 1 of Attachment 6 to ULNRC-05665 nor were any Scientech Gap Analysis IE Findings/Observations (F/Os) included in Table 2 of Attachment 6 to ULNRC-05665. However, since the supporting risk analysis for ULNRC-05665 was based on PRA Update 4, ILRT RAI questions #1 and #7 related to IE peer review and gap analysis findings will be addressed in this submittal for BOP ESFAS.

Callaway PRA Calculation EG-27, Revision 0, documents the calculations used to determine the probabilities of recovery and non-recovery for the component cooling water (CCW) system. The calculation determines that CCW must be recovered in 1 to 2 hours to prevent core damage according to MAAP code analysis. Based on engineering judgment, the postulated failure of the essential service water (ESW) train 'B' to CCW heat exchanger 'B' supply valve (EFHV0052) to open due to either valve failure or loss of power on the power supply (MCC NG04C) and failure of the operator to align ESW to CCW heat exchanger 'B' could be recovered in 1 hour, as could dependent failures of the CCW pumps. The non-recovery probability of CCW system, EG-PSF-FC-CCWSYS, is calculated to be 0.330.

Callaway PRA Calculation EG-27, Addendum 1, Revision 0, determines the probability for failure to recover CCW prior to transfer to residual heat removal (RHR) recirculation in the T(C), Loss of All Component Cooling Water, event tree. Those cutsets containing CCW heat exchanger 'B' in test/maintenance are considered to be recoverable during the period before swapover to recirculation mode. The non-recovery probability for the CCW system prior to swapover to RHR recirculation, FTR-CCW-RHR-REC, is calculated to be 0.221.

Callaway PRA Calculation EA-08, Revision 0, documents the calculations used to determine the probabilities of recovery and non-recovery for the service water (SW) system at 2 and 8 hours after a complete loss of service water. The Loss of All Service Water initiating event quantification was examined and it was determined that the dominant contributor to the initiating event frequency was failure to correctly place ESW in service following a loss of service water. The remaining portion of the initiating event frequency was due to random and common cause equipment failures. Based on engineering judgment, it was estimated that ESW valve failures to transfer, ESW pump failures to start, and ESW-related dependent failures were all recoverable in 2 hours, while service water pump failures to start and run, service water pump discharge valve failures, ESW valve failures, ESW pump failures, and ESW system-related dependent failures were all recoverable in 8 hours. The probability of SW non-recovery due to equipment failures in 2 hours, EA-PSF-FC-SWESW, is calculated to be 0.352. The probability of SW non-recovery due to equipment failures in 8 hours, EA-PSF-FC-SWESW8, is calculated to be 0.093. The total probability of service water recovery in 2 hours, SW-RECOVERED-2HRS, is determined to be 0.496. The total probability of service water recovery in 8 hours, SW-RECOVERED-8HRS, is determined to be 0.880.

Table 1-1 lists the names, descriptions, old probabilities of the related basic events discussed above, as well as new probabilities as a result of a sensitivity analysis which has been performed to take no credit for the above repair of hardware faults, i.e., the values for the non-recovery probabilities were set as 1 and the value for the recovery probability as 0. (The total probabilities of service water recovery in 2 hours and in 8 hours are determined by setting the non-recovery probability due to equipment failure to a value of 1 and re-quantifying the same small fault trees in Calculation EA-08, Revision 0.)

Table 1-1, Changes in Non-recovery/Non-repair Probabilities for Sensitivity Analysis

Basic Event	Description	Old Probability	New Probability
EA-PSF-FC-SWESW	OPERATORS FAIL TO RECOVER SW IN 2HRS DUE TO EQPT FAILURE	3.52E-01	1
EA-PSF-FC-SWESW8	OPERATORS FAIL TO RECOVER SW IN 8HRS DUE TO EQPT FAILURE	9.30E-02	1
EG-PSF-FC-CCWSYS	OPERATOR FAILS TO RECOVER CCW AFTER LOSS OF THE SYSTEM	3.30E-01	1
EG-REC-CCWSYSTEM	OPERATOR RECOVERS CCW SYSTEM AFTER SYSTEM LOSS	6.70E-01	0
FTR-CCW-RHR-REC	FAILURE TO RECOVER CCW PRIOR TO SWAP- OVER TO RHR RECIRC.	2.21E-01	1
SW-RECOVERD-2HRS	PROBABILITY OF SW RECOVERY 2 HRS AFTER LOSS	4.96E-01	4.08E-01
SW-RECOVERD-8HRS	PROBABILITY OF SW RECOVERY 8 HRS AFTER LOSS	8.80E-01	7.57E-01

With the new recovery/non-recovery probabilities, the core damage frequency (CDF) is increased by 6.50E-07 per year, or about 1.5%, which is insignificant and should not impact the conclusions of the original BOP ESFAS risk evaluation provided in Section 4.1 of Attachment 1 to ULNRC-05665. Also note that most of the CDF increase of 1.5% determined for this sensitivity case was due to the basic event FTR-CCW-RHR-REC which has a RAW value of 1.01. Calculation EG-27, Addendum 1, Revision 0, determined the system time window for this recovery to be about 46 hours. The likelihood of recovery, given this amount of time, is very high, but the PRA uses a failure probability for this recovery of 0.3, which is very conservative.

2. Items 6, 7, 8, 11, 12, 20, and 21 (F/Os AS-1, AS-2, AS-3, AS-7, SY-1, QU-3, and QU-4 in Table 2 of Attachment 6 to ULNRC-05665) all identify apparent fundamental logic errors in the fault tree / event tree structure of the PRA model, including failure to properly treat dependencies, invalid placement of human error events in the logic, credit for systems which would not be available given the sequence (i.e., station blackout crediting main feedwater, loss of service water

crediting instrument air). The dispositions state that correction of these items has been determined to increase CDF by about 1%. Please provide the basis for this conclusion, including exactly how the 1% increase was determined. Describe how the extent of condition of these logic errors was investigated to ensure other instances do not exist in other places within the PRA. In addition, please revise the PRA to address these F/Os and provide revised results.

Response:

All of the findings/observations (F/Os) that this RAI addresses speak to several equations that were found to contain erroneous cutsets. Callaway staff reviewed all of the PRAUPDT4 equations to determine the extent of condition. Only equations L2SW-M (TDAFP for Loss of All SW), L2T1S (TDAFP for SBO), O1SW-M (Cooldown and Depress for Loss of All SW), and O1T1S (Cooldown and Depress for SBO) had erroneous cutsets that, if removed, could increase the CDF. The cutsets credited SW, ESW, or instrument air availability, but these systems would not be available given either a Loss of All SW or a station blackout (SBO).

To ascertain the potential impact on CDF due to the erroneous cutsets in equation L2SW-M, sequence equation T(SW)S23 was reviewed. This sequence equation would contain any erroneous cutsets stemming from equation L2SW-M. It is possible that sequence equations T(SW)S19 to S22 could also contain these erroneous cutsets. However, service water was recovered at 2 hours for these sequences. Thus, failure of SW, ESW, or instrument air could be in these sequence equations and this would be acceptable due to the recovery. Sequence equation T(SW)S23 was opened in the WinNUPRA sensitivity module and the probabilities for basic events related to the pressure transmitters (AL-PST), ESW pumps, and human errors AL-XHE-FO-AFWESW and AL-XHE-MC-CST were set to 1.0 to reflect the fact that ESW would not be available as a suction source for the turbine-driven auxiliary feedwater pump (TDAFP). The sequence frequency went up by about 2E-07/yr.

To ascertain the potential impact on CDF due to the erroneous cutsets in equation L2T1S, sequence equation T(1S)S26 was reviewed. This sequence equation would contain any erroneous cutsets stemming from equation L2T1S. It is possible that sequence equations T(1S)S22 to S25 could also contain these erroneous cutsets. However, power was recovered at 1 hour for these sequences, making equipment potentially available again. Sequence equation T(1S)S26 was opened in the WinNUPRA sensitivity module to confirm that the AL-PST and AL-XHE issues discussed in the previous paragraph do not exist in this equation. The ESW failures that do exist in this equation all stem from the SBO initiating event cutset equation T(1)S08. Thus, there is no impact to the model or CDF due to this issue.

To ascertain the potential impact on CDF due to the erroneous cutsets in equation O1SW-M, sequence equation T(SW)S17 was reviewed. This sequence equation would contain any erroneous cutsets stemming from equation O1SW-M. It is possible that sequence equations T(SW)S13 to S16 could also contain these erroneous cutsets. However, service water was recovered at 8 hours for these sequences. Thus, failure of SW, ESW, or instrument air could be in these sequence equations and this would be acceptable due to the recovery. Sequence equation T(SW)S17 was opened in the WinNUPRA sensitivity module to confirm that there are no basic events associated with SW, ESW, or instrument air (other than one SW basic event that represents the failure to recover SW at 8 hours). Thus, there is no impact to the model or CDF due to this issue.

To ascertain the potential impact on CDF due to the erroneous cutsets in equation O1T1S, sequence equation T(1S)S20 was reviewed. This sequence equation would contain any erroneous cutsets stemming from equation O1T1S. It is possible that sequence equations T(1S)S12 to S19 could also contain these erroneous cutsets.

However, power was recovered at 8 or 10 hours for these sequences, making equipment potentially available again. Sequence equation T(1S)S20 was opened in the WinNUPRA sensitivity module to confirm that basic events associated with ESW or standby generation (diesel generators) all stem from the SBO initiating event cutset equation T(1)S08. Thus, there is no impact to the model or CDF due to this issue.

Since the only impact noted above was a sequence frequency increase of about 2E-07/yr which was estimated to "increase CDF by about 1%," Callaway staff did not need to revise the PRA model for this submittal to address these F/Os, and there are no new results to provide.

Note that equations O1SW-M, O1C-M, and O1CT1-M contain an erroneous cutset that, if removed, would decrease the CDF. These equations represent cooldown and depressurization. The Callaway PRA models a human error, OP-XHE-FO-DEPRESS, to fail this function. However, due to a modeling problem, human error FB-XHE-FO-FANDB also shows up as failing cooldown and depressurization for these equations. Deleting this erroneous basic event from the above equations would result in a decrease in CDF.

3. Item 9 (F/O AS-4 in Table 2 of Attachment 6 to ULNRC-05665) addresses the use of an inaccurate reactor coolant pump seal LOCA model. The disposition states that core uncovery probabilities were increased by 25% resulting in a 1.5% increase in CDF. Typically, the seal LOCA model is used to determine the time to core uncovery, which is then used to estimate the offsite power recovery probability, and higher leak rate scenarios. Although low probability, these seal LOCAs tend to dominate the risk. Please provide the basis for selection of a 25% increase used in the sensitivity study, and how this is known to bound the seal LOCA non-conservatism. In addition, please revise the PRA to reflect the WOG 2000 model if Callaway has high temperature seals installed in all its pumps such that the WOG 2000 model is applicable; otherwise, use the conservative seal

LOCA model accepted by the NRC (i.e., the Rhodes model) and provide revised results.

Response:

The current Callaway PRA uses a reactor coolant pump (RCP) seal loss of coolant accident (LOCA) model based on WCAP-10541. In the Station Blackout, Loss of All CCW, and Loss of All Service Water event trees, core uncovery is questioned before the ability to provide reactor coolant system (RCS) makeup is recovered such that RCS makeup is only addressed if the core has remained covered. Callaway PRA Calculation BB-97, Revision 0, documents the determination of various probabilities of core uncovery due to an RCP seal LOCA based on the WCAP-10541 model. A more recent RCP seal LOCA model can be found in the WOG 2000 model which is documented in WCAP-15603 and WCAP-16141. During the original disposition of F/Os for the ILRT submittal, a sensitivity analysis was performed to address this source of uncertainty related to different seal LOCA modeling. The 25% increase in core uncovery probabilities estimated in the sensitivity analysis was just a round number that was chosen to estimate its impact on CDF (which corresponds to approximately a 1.5% increase in CDF).

To further investigate the uncertainty due to the seal LOCA modeling, a close review of the WOG 2000 model was performed. WCAP-16141, Appendix A, Table 2 provides key plant characteristics. A review of this table indicates that the generic analyses are acceptable for Callaway Plant. In addition, page 44 of the WCAP discusses the use of a 30-minute time period to initiate cooldown and depressurization (CD&D). Human failure event (HFE) OP-XHE-FO-DEPRES, which is used for CD&D for all Callaway initiating events, used a system time window of 30 minutes (Callaway PRA Calculation ZZ-278, Rev. 0, Addendum 1). This also indicates that the generic analyses are acceptable for Callaway Plant.

WCAP-16141, Appendix A, Table 5 provides time to core uncovery for various RCP seal leakage rates and other conditions. Time to core uncovery can be assumed to represent the required time for AC power recovery. The Callaway SBO event tree credits 8-hour and 12-hour AC power recovery times for sequences with successful CD&D. It credits 8-hour and 10-hour AC power recovery times for sequences with failure of CD&D. It credits a 1-hour offsite AC power recovery time plus a 1-hour onsite restoration time (total of 2 hours to restore power) for sequences with the failure of auxiliary feedwater (AFW).

A sensitivity analysis was performed with the core uncovery probabilities set to one for those scenarios in which the times to core uncovery in the WCAP table are less than the corresponding Callaway AC power recovery times (2, 8, 10, 12 hours in the SBO event tree). The sensitivity analysis indicates that the CDF would increase 5-6% compared to the baseline CDF. Note that this is conservative because one could actually define AC recovery times (less than 8, 10, or 12 hours) that would be successful for the cases being identified as not meeting the recovery times in the sensitivity analysis. Therefore, this F/O on the RCP seal LOCA model should not impact the conclusions of the original BOP ESFAS risk evaluation provided in Section 4.1 of Attachment 1 to ULNRC-05665.

In the ongoing Callaway PRA RG 1.200 Upgrade Project, the current WCAP-10541 RCP seal LOCA model will be updated and replaced with the WOG 2000 RCP seal LOCA model.

4. Item 10 F&O (F/O AS-6 in Table 2 of Attachment 6 to ULNRC-05665) questions the validity of MAAP 3 for addressing the SGTR sequence with failure to isolate. Please provide the basis for the validity of MAAP 3 for addressing this sequence.

Response:

Callaway PRA Calculation ZZ-177, Revision 0, documents the validation and verification of MAAP Version 3.0B PWR for Callaway Plant. The MAAP Version 3.0B is a computer code which simulates light water reactor system response to accident initiation events. The MAAP 3.0B PWR code was prepared as a part of the Industry Degraded Core Rulemaking (IDCOR) program to investigate the physical phenomena that might occur in the event of a serious light water reactor accident leading to core damage, possible reactor pressure vessel failure, and possible containment failure and depressurization. MAAP includes models for all the important phenomena that might occur in a serious light water reactor accident. The MAAP 3.0B PWR code was maintained by Fauske & Associates Incorporated (FAI) for the Electric Power Research Institute (EPRI).

As a Quality Assured code, MAAP 3.0B was maintained in conformance with 10 CFR 50 Appendix B. A Design Review and an independent verification and validation were undertaken by EPRI for the code revision released in June 1990. All code changes since that date were made under the FAI Quality Assurance Plan in conformance with 10 CFR 50 Appendix B.

The NRC has reviewed and evaluated MAAP 3.0B (PWR & BWR) with the results documented by Brookhaven National Laboratory. The review concludes that MAAP 3.0B PWR has adequate models to address important behavior during severe accidents and is adequate for predicting thermal-hydraulic behavior prior to core damage. The review does include recommendations for utilities not to use MAAP for determining success criteria after clad damage, or to provide justification if using MAAP for certain thermal-hydraulic conditions such as for an anticipated transient without scram (ATWS); however, there are no specific recommendations regarding using MAAP for determining success criteria in steam generator tube rupture (SGTR) scenarios.

Callaway PRA Calculation ZZ-272, Revision 0, documents the Callaway Individual Plant Examination (IPE) Level 2 MAAP analyses using the MAAP 3B code. Eight cases have been run for different SGTR scenarios. Case SGTR 1B is a single steam generator tube rupture in the cold leg side of the SG with a stuck-open secondary atmospheric relief valve and the assumption that the operator fails to isolate the ruptured SG. Case SGTR 1E assumes that five SG tubes would fail and Case SGTR 1F assumes that three SG tubes would fail. Based on the results from the MAAP program as documented in Calculation ZZ-272, Revision 0, the core uncovery times for Cases 1B, 1E, and 1F are 43.6, 10.6, and 15.2 hours, respectively and the core melt times are 59.5, 12.9, and 19.2 hours, respectively. Case SGTR 2 is a single SG tube rupture with the assumption that the operator isolates the ruptured SG successfully. No core damage occurs in this case. The SGTR results from MAAP 3B are reasonable. Thus, retaining the SGTR sequences with failure to isolate the ruptured SG is reasonable and acceptable.

In the ongoing Callaway PRA RG 1.200 Upgrade Project, the MAAP 3.0B model will be upgraded and replaced with the latest MAAP4 model, i.e., MAAP 4.0.7. The previous success criteria analyses, including those in the SGTR sequences, will be re-performed with the new MAAP4 model.

5. Item 10 (F/O AS-6 in Table 2 of Attachment 6 to ULNRC-05665) states it is conservative to assume the SGTR sequence automatically goes to LERF. However, for this application, conservatively assuming events result in LERF is non-conservative, since it masks the intact containment frequency, and reduces the delta LERF. Address this item for this application accounting for the non-conservative impact.

Response:

The SGTR event is a containment bypass event, and the effects on the reported large early release metrics in ULNRC-05665 would be increased by no more than a factor of 4.1 by the proposed Completion Time extension due to the impact on equipment important for SGTR mitigation that is actuated by BOP ESFAS such as AFW, SG blowdown and sample line isolation, and containment mini-purge isolation. This F/O would not invalidate the risk-related conclusions of the original BOP ESFAS risk evaluation. The risk analyses performed for ULNRC-05665 changed all of the pertinent basic events that could be impacted by the BOP ESFAS submittal, and the analyses account for the impacts on the SGTR sequences, so the impact of retaining the STGR sequences on the conclusions is negligible.

6. Item 14 (F/O DA-2 in Table 2 of Attachment 6 to ULNRC-05665) identifies an improper treatment of data. The response indicates a recent update using the correct method per the standard was performed. It is not clear why this item is not therefore resolved if the data has been updated. Clarify this apparent inconsistency.

Response:

The data update has been performed as part of the ongoing Callaway PRA RG 1.200 Upgrade Project, specifically in Phase A of the project (refer to the response to RAI #10 below). However, this data update had not yet been incorporated into the model that was used for the BOP ESFAS risk evaluation provided in Attachment 1 to ULNRC-05665. It should be noted that from the data update it was determined that the original data group estimations were correct, essentially making the F/O invalid.

7. Item 19 (F/O QU-1 in Table 2 of Attachment 6 to ULNRC-05665) and Item 1 from Table 2.2.1 of Attachment 5 to ULNRC-05598 (no corresponding gap analysis finding item number in Table 1 of Attachment 6 to ULNRC-05665 – see the response to question #1 above) identify the failure to consider the "state of knowledge correlation." The disposition states that this only impacts the uncertainty analysis. This is fundamentally not true. The standard requires that quantification of CDF and LERF consider correlated data. This is especially significant for evaluation of ISLOCA, where the primary failure mode leading to overpressurization of low pressure piping involves coincident failure of two or more redundant identical isolation valves. Neglecting the data correlation has the potential to significantly underestimate the overall frequency of the event. Identifying that these events are not significant (when quantified with the non-conservative error) does not justify that they would not become significant once the error is corrected. Please provide the basis for why this error is known to be insignificant, especially with regards to the interfacing LOCA contribution. In addition, please revise the PRA to specifically address these F&Os by including the "state of knowledge correlation" and provide revised results.

Response:

Note that the state-of-knowledge correlation is defined in ASME/ANS RA-Sa-2009 as the correlation that arises between sample values when performing uncertainty analysis for cutsets consisting of basic events using a sampling approach (such as the Monte Carlo method). When taken into account for each sample, this results in the same value being used for all basic event probabilities to which the same data applies.

State-of-knowledge about the BOP ESFAS train failure rate was considered and its uncertainty was evaluated in the risk evaluation. To assess the state-of-knowledge concerning the failure rate that is expressed in term of prior distributions, two methods, Maximum Likelihood Estimation and Bayesian Estimation, and two prior distributions, Jeffrey's non-informative prior and positive uniform distributions, were used to estimate the failure rate. The difference between the resulting failure rates does not exceed 10%. A failure rate of 1.225E-01 per year was used in the risk evaluation (see also the response to question #8 below). To assess the maximum impact of the state-of-knowledge correlation on the BOP ESFAS trains, basic events AL-ICC-AF-AFAS1 and AL-ICC-

AF-AFAS4 were both set to the same value of 1.0 (which is the maximum probability value a sampling approach can achieve and represents an extreme case in which there are no AFAS signals to BOP ESFAS trains A and B via separation groups 1 and 4). The resulting conditional CDF would increase 16 times, therefore it could lead to the most limiting risk metric ICCDP increasing to 3.07E-07, which is still less than the RG 1.177 acceptance criterion (5E-07).

Although ISLOCA is a major contributor to LERF, it is a small contributor to CDF and no mitigation is credited. An ISLOCA event is a containment bypass event and is not impacted by the functions actuated by BOP ESFAS. Therefore, the related F/Os are very unlikely to impact the conclusions of the original BOP ESFAS risk evaluation in ULNRC-05665.

8. Item 23 (F/O QU-10 in Table 2 of Attachment 6 to ULNRC-05665) indicates that key assumptions and key sources of uncertainty that influence the current quantification is not addressed in a coherent manner. The disposition indicates that this is solely a documentation issue without any basis for how the licensee determined that there were no assumptions or uncertainties that could impact this application. Please provide a discussion of the key assumptions and key sources of uncertainty that could impact this application and how the licensee has addressed these key assumptions and key sources of uncertainty (e.g., by conducting additional sensitivity studies) and, as necessary, please provide any additional sensitivity study results.

Response:

The Callaway PRA group performed and documented an uncertainty/sensitivity analysis in Callaway PRA Calculation ZZ-267 and its addenda during each PRA update. This specific F/O (QU-10) indicates that "Key assumptions and key sources of uncertainty which influence the current quantification are not addressed in a coherent manner in the documentation." ZZ-267, Addendum 4, Revision 0, is the referenced documentation of "the current quantification" which updated the sequence quantification for the Fourth Callaway PRA Update. Section 3.7 of the calculation performs the uncertainty/sensitivity analysis, but unlike the similar analyses in previous PRA updates, the parametric uncertainty was not addressed. Due to resource constraints, the model sensitivity to human error probabilities and initiating event frequencies was not evaluated.

As documented in various addenda of ZZ-267, the following key assumptions and key sources of uncertainty could impact the Callaway PRA model estimation of CDF and, therefore, impact this application.

The test/maintenance (T/M) probabilities used in the Callaway PRA are based on historical plant data. Using the historical plant T/M data for current CDF calculations introduces uncertainty to the results due to possible changes in T/M practices and probabilities. To evaluate CDF sensitivity to T/M data, two sensitivity cases were run for the Fourth PRA Update. The probabilities of all T/M basic events were set to zero or doubled in each case. The resulting CDF values were 25% below or above the baseline CDF value, which indicates that CDF can vary significantly due to changes in T/M practices and probabilities.

The initiating event (IE) frequencies used in the Callaway PRA are based upon credible methodologies and data sources. However, IE frequencies do represent a potential source of uncertainty. Since each cutset in the Callaway core damage equation contains one IE, if all IE frequencies were doubled, the Callaway CDF would double. Conversely, if all IE frequencies were reduced by one half, the CDF would decrease by 50%. With regard to individual IEs, IE-T1, Loss of Offsite Power, is by far the largest contributor to Callaway CDF. The methodology used to determine the frequency of IE-T1 for Callaway was to Bayesian update the generic loss of offsite power (LOOP) frequency with Callaway-specific experience (i.e., no losses of offsite power). Should Callaway actually experience a LOOP event in the future, the frequency of IE-T1 would increase appreciably, resulting in an appreciable increase in baseline CDF due to the high Fussell-Vesely value of IE-T1 (54.7% in the Fourth PRA Update) for Callaway Plant.

In the Third PRA Update, a sensitivity analysis was performed for human error probabilities (HEPs). The probabilities of all human error (i.e., "XHE") basic events were set to zero or doubled in the sensitivity case. The resulting CDFs were 30% below or 40% above the corresponding baseline CDF value, which indicates that the Callaway CDF is sensitive to the HEPs used. Callaway Plant has updated its human reliability analysis (HRA) for the Fourth PRA Update and documented it in ZZ-278, Addendum 1, Revision 0. The HRA update was in accordance with the requirements of Capability Category II of the ASME PRA Standard Ra-S-2003.

Another key source of CDF uncertainty is the uncertainty due to the component failure data used in the PRA. It is assumed that the generic industry component failure data used in the PRA model are reasonable representations of the component failure probabilities for Callaway Plant components when 1) generic component failure data is applied directly, or 2) generic component failure data is applied as a prior distribution for Bayesian updating with plant-specific data. This source of uncertainty can be assessed using the WinNUPRA uncertainty module. However, this would require an up-to-date parameter (PRM) file, which was not prepared due to resource constraints during the Fourth PRA Update. Parametric uncertainty was addressed in Addendum 2 of ZZ-267 for the Second PRA Update. The point estimate CDF for the Second PRA Update was 2.45E-05 per year, while the mean value from the parametric uncertainty analysis was 2.99E-05 per year.

The following uncertainty analysis was performed in support of the BOP ESFAS amendment but was not included in the ULNRC-05665 submittal.

Uncertainty

The BOP ESFAS cabinet failure is a rare event with the failure rate λ. To assess the impact of uncertainty from the methods and prior distributions on the failure rate, two methods were used to estimate the failure rate, Maximum Likelihood Estimation and Bayesian Estimation, and two prior distributions were used, Jeffrey's non-informative prior and positive uniform distributions. The failure rate calculated by the Bayesian method was used in the risk calculations reported in ULNRC-05665.

The Bayesian method is commonly used in PRA, and in this case the Bayesian method would yield a more conservative number and allows the incorporation of operating experience and engineering judgments.

Maximum Likelihood Estimation (MLE)

Using MLE, the failure rate is the total number of events (14) discussed in ULNRC-05665 (Attachment 1 page 17) divided the total service time 122.48 years, thus, $\lambda = 14/122.48 = 0.1143\ yr^{-1}$. If the 14^{th} failure is discounted as an early infant mortality failure, and it is excluded in the standard deviation (STD) calculation, the STD = 8.90E-02. The Mean Time to Failure, MTTF = $1/\lambda$ = 8.75 years.

Bayesian Estimation

If the BOP ESFAS cabinet failure is assumed as a random failure with a Poisson distribution, its failure rate λ follows a Gamma distribution of shape Gamma(a, b). The failure rate is developed based on a Bayesian probability calculation for a rare event. Two non-informative prior distributions of failure rate were evaluated as follows.

(1) The prior distribution is assumed as the Jeffrey's non-informative prior distribution of λ and the likelihood of the observation is the Poisson distribution. Jeffrey's prior distribution of λ for the Poisson distribution has the density shape of Gamma(1/2,0) density. Using a Bayesian update, the failure rate of BOP ESFAS $\lambda = (14+0.5)/(122.48+0) = 1.18E-01\ yr^{-1}$ and the MTTF = 8.45 years. The 5th percentile of λ is 6.69E-02 and the 95th percentile of λ is 1.94E-01.

(2) The prior distribution is assumed as a positive uniform distribution with the shape of Gamma(1,0) density. Using a Bayesian update, the failure rate $\lambda = (14+1.0)/(122.48+0) = 1.225E-01\ yr^{-1}$ and MTTF = 8.17 years. The 5th percentile of λ is 7.12E-02 and the 95th percentile of λ is 1.91E-0.

From the above evaluation, using different methods and prior distributions, the difference between the resulting failure rates would not exceed 10%, therefore, it does not affect the conclusions of the PRA evaluation for the 24-hour BOP ESFAS Completion Time. The failure rate of 1.225E-01 yr^{-1} was used in the PRA evaluation reported in ULNRC-05665.

9. Item 24 (F/O QU-11 in Table 2 of Attachment 6 to ULNRC-05665) indicates that the licensee did not use the ASME definition of significant and the licensee dispositions this item as being solely a documentation issue. The staff disagrees that not including upwards of 7% of the results is a documentation issue. Please provide revised results that meet the ASME definition of significant.

Response:

This is a documentation issue. All results above the truncation limit are retained in the CDF and LERF equations. The ASME definition of a significant sequence / cutset specifies that the aggregate contribution is 95% of the total CDF and each individual sequence / cutset contributes greater than 1% to the CDF. The results listed in Callaway's sequence quantification document, Calculation ZZ-267, Addendum 4, Revision 0, do not list all of the cutsets or sequences that would meet the "95% of the total CDF" portion of the ASME Standard definition of significant. Section 3.2 of that calculation lists fifteen core damage sequences, each contributing greater than one percent to the non-flooding internal events CDF, as the "dominant core damage sequences." Those fifteen core damage sequences account for about 88% of the total non-flooding core damage frequency at Callaway Plant. Seven cutsets with frequencies greater than 1E-06 per year were listed in this section as the significant cutsets. To meet the ASME definition of significant cutset and accident sequence, this section should have listed more cutsets and core damage sequences than those seven cutsets and fifteen sequences currently present, such that the listing reflects no less than 95% of the total CDF. However, the additional cutsets and sequences would not have individually contributed greater than one percent to the Callaway CDF. Again, this is only a documentation issue as all results are retained in the core damage equations.

10. There are numerous B F&Os (significant and should be resolved by next update of PRA) and one A F&O (highly significant and should be resolved immediately) that remain open many years after the peer review and gap analysis. This is not consistent with the expectations of the peer review process and the staff. Please provide a schedule and commitment for the resolution of all remaining open F&Os, including any open C and D F&Os.

Response:

There have been two external reviews of the Callaway PRA model. The Westinghouse Owners Group (WOG) performed a peer review of the Callaway PRA in accordance with NEI 00-02 during the week of November 5 to 10, 2000. The final review report was issued in January 2002. There were four Level A Facts and Observations (F&Os) and twenty-eight Level B F&Os identified during the process. Resolution of all F&Os from the peer review was completed with five exceptions, listed in Table 2.2.1 of Attachment 5 to ULNRC-05598 (ILRT amendment request), when the Fourth Callaway PRA Update was completed in early 2006. As described in Table 1 of Attachment 6 to

ULNRC-05665, none of the remaining F&Os would impact the BOP ESFAS risk evaluation provided in Section 4.1 of Attachment 1 to ULNRC-05665.

Scientech was contracted to perform a gap assessment of the Callaway PRA in May 2006. The purpose of this assessment was to identify gaps between the Callaway PRA and Capability Category II of the ASME Standard (draft at the time). No importance Level A Findings/Observations (F/Os) and twenty-six Level B F/Os were generated from the analysis. With regard to the potential impact of these F/Os, none of the remaining open F/Os from the gap assessment would impact the BOP ESFAS risk evaluation provided in Section 4.1 of Attachment 1 to ULNRC-05665.

A Callaway Plant PRA Model Upgrade Plan was prepared in early 2007. The plan will address the internal events PRA quality gaps including all the remaining open F&Os and F/Os generated during the WOG peer review and gap assessment, respectively. The total workload of the PRA RG 1.200 Upgrade Project was estimated to be about 12,000 man-hours. To support the transition of the Callaway Plant fire protection program to NFPA 805, the PRA Upgrade Project was divided into three phases: A, B, and C. Phases A and B address those quality gaps of the plan that have significant impact on the fire PRA. Phase C addresses the other gaps in the plan. Phases A and B were completed in early 2009 by application of about 2,700 man-hours. Callaway Plant is currently working on Phase C which has an estimated workload of 9,300 man-hours. It was started in April 2009. The target completion date for the project is March 2011. All of the remaining open F&Os and F/Os, including any open C and D items, are anticipated to be resolved upon the completion of the project.