January 19, 2010

ULNRC-05673

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
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
REGARDING PROPOSED REVISION TO TECHNICAL SPECIFICATION
5.5.16, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM"
<u>(LICENSE AMENDMENT REQUEST LDCN 09-0008)</u>**

References: 1. Ameren UE Letter ULNRC-05598, "Proposed Revision to
Technical Specification 5.5.16, 'Containment Leakage Rate Testing
Program,' (License Amendment Request LDCN 09-008)," dated
March 20, 2009
2. NRC E-mail Request for Additional Information on the License
Amendment Request to Modify Technical Specification 5.5.16,
"Containment Leakage Rate Testing Program," for One-time
Extension of Integrate Leak Rate Test Interval (TAC ME0986),
dated December 15, 2009

        By letter dated March 20, 2009 (Reference 1) and pursuant to 10 CFR 50.90,
AmerenUE (Union Electric) submitted a license amendment request (LAR) to
incorporate proposed changes to Technical Specification (TS) 5.5.16, "Containment
Leakage Rate Testing Program," which establishes the program for leakage rate
testing of the containment, as required by 10 CFR 50.54, "Conditions of licenses,"
Section (o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage
Testing for Water-Cooled Power Reactors," Option B, "Performance Based
Requirements," as modified by approved exemptions. Specifically, AmerenUE
proposed a one-time five-year deferral of the containment Type A integrated leak rate
test from once in ten years to once in 15 years.

From its ongoing review of this proposed revision to TS 5.5.16, the NRC staff has transmitted a request for additional information (RAI), per Reference 2, containing several questions/requests for which responses from AmerenUE are needed in order to support completion of the NRC's review. Accordingly, this letter provides AmerenUE's response to the NRC's RAI in Attachment 1. Within the attachment, each of the individual questions/requests contained in the RAI is stated and immediately followed with AmerenUE's response. Text from the NRC's RAI is shown in italics.

Responding to the NRC's RAI does not require changes to be made to the proposed changes for TS 5.5.16. Further, the response to the NRC's RAI does not change the evaluations provided in the license amendment request, including the determination of no significant hazards consideration. This letter does not contain commitments.

AmerenUE appreciates the NRC staff's continued review of the proposed revision to TS 5.5.16. If there are any questions, please contact Tom Elwood, Supervising Engineer, Regulatory Affairs and Licensing at (314) 225-1905.

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

Sincerely,

Executed on: __1/19/10__

Scott Sandbothe
Manager, Plant Support

KRA

Attachment: 1) Responses to NRC RAI Questions Regarding License Amendment
                Request LDCN 09-0008

cc: U.S. Nuclear Regulatory Commission (Original and 1 copy)
Attn: Document Control Desk
Washington, DC 20555-0001

Mr. Elmo E. Collins, Jr.
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
612 E . Lamar Blvd., Suite 400
Arlington, TX 76011-4125

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Mohan C. Thadani (2 copies)
Senior Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8G14
Washington, DC 20555-2738

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RESPONSES TO NRC RAI QUESTIONS REGARDING
LICENSE AMENDMENT REQUEST LDCN 09-0008

In its letter dated March 20, 2009, AmerenUE (the licensee) submitted a request to incorporate proposed changes to Technical Specification (TS) 5.5.16, "Containment Leakage Rate Testing Program," which establishes the program for leakage rate testing of the containment, as required by 10 CFR 50.54, "Conditions of licenses," Section (o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, "Performance Based Requirements," as modified by approved exemptions. Specifically, AmerenUE proposed a one-time five-year deferral of the containment Type A integrated leak rate test from once in ten years to once in 15 years. AmerenUE's request is currently under review by the NRC staff.

To continue the review of AmerenUE's LAR LDCN 09-0008, the NRC staff requests the following additional information. *The following RAIs are related to Table 2.1.3 of Attachment 5 of the subject application.*

1. *Item 3 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

   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, failure of essential service water (ESW) train 'B' to CCW heat exchanger 'B' hand valve (EFHV52) 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 then calculated to be 0.330.

   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 of CCW system prior to swapover to RHR recirculation, FTR-CCW-RHR-REC, is calculated to be 0.221.

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 be 1 and re-quantifying the same small fault trees in 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-1	1
EA-PSF-FC-SWESW8	OPERATORS FAIL TO RECOVER SW IN 8HRS DUE TO EQPT FAILURE	9.30E-2	1
EG-PSF-FC-CCWSYS	OPERATOR FAILS TO RECOVER CCW AFTER LOSS OF THE SYSTEM	3.30E-1	1
EG-REC-CCWSYSTEM	OPERATOR RECOVERS CCW SYSTEM AFTER SYSTEM LOSS	6.70E-1	0
FTR-CCW-RHR-REC	FAILURE TO RECOVER CCW PRIOR TO SWAP-OVER TO RHR RECIRC.	2.21E-1	1
SW-RECOVERD-2HRS	PROBABILITY OF SW RECOVERY 2 HRS AFTER LOSS	4.96E-1	4.08E-01
SW-RECOVERD-8HRS	PROBABILITY OF SW RECOVERY 8 HRS AFTER LOSS	8.80E-1	7.57E-01

With the new recovery/non-recovery probabilities, core damage frequency (CDF) is increased by 6.50E-7 per year, or about 1.5%, which is insignificant and should not impact the results of the original integrated leak rate test (ILRT) risk evaluation provided in Attachment 4 to Reference 1.

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. EG-27, Addendum 1, Revision 0, determined that the system time window for this recovery was about 46 hours. The likelihood of recovery, given this amount of time, is very high, but the PRA uses a failure probability of this recovery of 0.3, which is very conservative.

2. *Items 6, 7, 8, 11, 12, 20, and 21 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 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 CDF. The cutsets contained SW, ESW, or instrument air that would not have been 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-7/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 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 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-7/yr which was estimated to "increase CDF by about 1%," Callaway 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 CDF. These equations represent cooldown and depressurization. Callaway 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 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 nonconservatism. 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 of core uncovery probabilities estimated in the sensitivity analysis was just a round
number that was chosen to estimate its impact on CDF (which arrives at 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. In addition, page 44 of the WCAP discusses the use of a 30-minute time to initiate
cooldown and depressurization (CD&D). HFE OP-XHE-FO-DEPRES, which is used for
CD&D for all Callaway initiating events, used a system time window of 30 minutes
(calculation ZZ-278, Rev. 0, Add. 1). So, again, this indicates that the generic analyses are
acceptable for Callaway.

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 CDF will 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. In the Response to RAI #8, a sensitivity analysis was performed with a
25% increase of the baseline CDF and it yielded results such that none of the acceptance
criteria as defined in the ILRT risk assessment were exceeded. Therefore, this F&O on the

RCP seal LOCA model should not impact the results of the original ILRT risk evaluation as provided in Attachment 4 to Reference 1.

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 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.*

<u>Response</u>

PRA Notebook ZZ-177, Revision 0, documents the validation and verification of MAAP Version 3.0B PWR for Callaway. 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 10CFR50 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 10CFR 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 by using 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 broken SG. Case SGTR 1E assumed that five SG tubes would fail and Case SGTR 1F assumed that three SG tubes would fail. The results from the MAAP program are documented in Calculation ZZ-272, Revision 0, as follows. The core uncovery times for Cases 1B, 1E, and 1F are 43.6, 10.6, and 15.2 hours, respectively. 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 broken 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 broken 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 SGTR sequences, will be re-performed with the new MAAP4 model.

5. *Item 10 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 is not impacted by containment isolation capability. This F&O should not impact the results of the original ILRT risk evaluation provided in Attachment 4 to Reference 1. This was confirmed by the following sensitivity analysis. The SGTR event tree, T(SG), was reviewed and the SGTR sequences with failure to isolate the ruptured SG include T(SG)S09, S18, S20, S29, and S36. The total frequency of these sequences is 3.43E-7 per year. Removal of these sequences from the results will decrease CDF and large early release frequency (LERF). Table 5-1 presents the original release frequencies (with containment liner corrosion assumed) as a function of accident class for the base case (as in Table 5.2 in Attachment 4 of Reference 1) and the new release frequencies after removing the associated SGTR sequences from the results. The total CDF, and thus the frequencies for accident class 3a and class 3b, is decreased by about 1%. The frequency for accident class 8, which represents those containment bypass events such as interfacing system LOCA (ISLOCA) and SGTR, is decreased by about 60%.

Table 5-1 Changes in Release Frequencies for SGTR Sensitivity Analysis

Accident Classes (Containment Release Type)	Description	Original Frequency (per Rx-yr)	New Frequency (per Rx-yr)
		(With Corrosion)	(With Corrosion)
1	No Containment Failure	1.90E-05	1.92E-05
2	Large Isolation Failures (Failure to Close)	7.54E-09	7.54E-09
3a	Small Isolation Failures (liner breach)	3.93E-07	3.89E-07
3b	Large Isolation Failures (liner breach)	9.86E-08	9.74E-08
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.26E-05	2.24E-05
8	Bypass (Interfacing System LOCA)	5.94E-07	2.51E-07
CDF	All CET End States	4.27E-05	4.23E-05

Table 5-2 displays the original risk metrics results (with corrosion) as provided in Table 5.7 in Attachment 4 of Reference 1, and their corresponding new values with the above revised release frequencies. From Table 5-2, it can be seen that the new values of total dose rate and conditional containment failure probability (CCFP) are slightly smaller than those results in Reference 1 due to the small reduction of CDF, while the new LERF is significantly smaller than the results in Reference 1 (from about 50% smaller for the base case to about 30% smaller for the proposed interval, i.e., 1 in 15 years case). As can be seen in Table 5-2, there are no changes in the delta total dose rate and ΔCCFP, and very small changes in ΔLERF, which confirms the statement that this F&O should not impact the results of the original ILRT risk evaluation in Attachment 4 of Reference 1.

Table 5-2 Changes in Risk Metrics Results for SGTR Sensitivity Analysis

		Original Results (from submittal Table 5.7)			New Results		
		Base Case	Current Interval	Proposed Interval	Base Case	Current Interval	Proposed Interval
		(3 in 10 years)	(1 in 10 years)	(1 in 15 years)	(3 in 10 years)	(1 in 10 years)	(1 in 15 years)
Total Dose Rate (person-rem/yr)		32.57	32.61	32.63	32.00	32.04	32.06
Delta Total Dose Rate (person-rem/yr)	From 3 yr	N/A	0.04	0.06	N/A	0.04	0.06
	From 10 yr	N/A	N/A	0.02	N/A	N/A	0.02
LERF (per year)		7.00E-07	9.31E-07	1.10E-06	3.56E-07	5.85E-07	7.50E-07
ΔLERF (per year)	From 3 yr	N/A	2.31E-07	3.98E-07	N/A	2.29E-07	3.94E-07
	From 10 yr	N/A	N/A	1.66E-07	N/A	N/A	1.65E-07
CCFP		54.55%	55.09%	55.48%	53.81%	54.35%	54.74%
ΔCCFP	From 3 yr	N/A	0.54%	0.93%	N/A	0.54%	0.93%
	From 10 yr	N/A	N/A	0.39%	N/A	N/A	0.39%

6. *Item 14 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 Response to RAI #10). However, this data update had not yet been incorporated into the model that was used for the ILRT submittal (Reference 1).

It may 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. *Items 19 and 1 (from Table 2.2.1 of Attachment 5 to ULNRC-05598) 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.

<u>Response</u>

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. Capability Category II of Supporting Requirements QU-A3 and QU-E3 requires estimating the mean CDF and uncertainty intervals by taking into account the state-of-knowledge correlation. Yet, per the NRC Final Safety Evaluation for NEI TR 94-01, Revision 2, and EPRI 1009325, Revision 2, dated June 25, 2008, "Capability Category I of ASME RA-Sa-2003 shall be applied as the standard, since approximate values of CDF and LERF and their distribution among release categories are sufficient for use in the EPRI methodology." Since the ILRT submittal was considered to be a Capability Category I application which only requires a point estimate for CDF (and LERF), the "state-of-knowledge correlation" does not need to be addressed. Although it is a major contributor to LERF, an ISLOCA event is a containment bypass event and is not impacted by containment isolation capability. The related F&Os should not impact the results of the original ILRT risk evaluation provided in Attachment 4 to Reference 1, and as such no revision is provided.

8. *Item 23 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.*

<u>Response</u>

The Callaway PRA group performed and documented an uncertainty/sensitivity analysis in Calculation ZZ-267 and its addenda during each PRA update. This specific F&O (QU-10) in Item 23 identifies 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, was the referred to 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 so 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/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/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.

In the third PRA Update, 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 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 has been addressed in Addendum 2 of ZZ-267 for the second PRA Update. The point estimate of CDF for the second PRA Update is 2.45E-5 per year, while the mean value from the parametric uncertainty analysis is 2.99E-5 per year.

To address the above and other assumptions and uncertainties contained in the current Callaway PRA model, a sensitivity analysis was performed with a 25% increase of the baseline CDF from the fourth PRA Update. Table 8-1 presents the original release frequencies (with corrosion) as a function of accident class for the base case, as provided in Table 5.2 of Attachment 4 to Reference 1, and the new release frequencies after the 25% CDF increase. The frequencies for accident class 3a and class 3b which represent small and large containment isolation failures increased by 25%, along with the 25 % increase of total CDF.

Table 8-1 Changes in Release Frequencies for SGTR Sensitivity Analysis

Accident Classes (Containment Release Type)	Description	Original Frequency (per Rx-yr) (With Corrosion)	New Frequency (per Rx-yr) (With Corrosion)
1	No Containment Failure	1.90E-05	2.37E-05
2	Large Isolation Failures (Failure to Close)	7.54E-09	9.43E-09
3a	Small Isolation Failures (liner breach)	3.93E-07	4.91E-07
3b	Large Isolation Failures (liner breach)	9.86E-08	1.23E-7
4	Small Isolation Failures (failure to seal - Type B)	N/A	N/A
5	Small Isolation Failures (failure to seal - Type C)	N/A	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A	N/A
7	Failures Induced by Phenomena (Early and Late)	2.26E-05	2.83E-05
8	Bypass (Interfacing System LOCA)	5.94E-07	7.43E-07
CDF	All CET End States	4.27E-05	5.33E-05

Table 8-2 displays the original risk metrics results (with corrosion) as provided in Table 5.7 of Attachment 4 to Reference 1, and their corresponding new values after using the revised release frequencies from above. From Table 8-2, it can be seen that the new values of total dose rate are increased from about 32 person-rem/yr to about 41 person-rem/yr, while the delta total dose rate is changed slightly. CCFP values are slightly higher than those in the original submittal due to the increase of CDF; however, ΔCCFP stays the same. The LERF for the proposed ILRT interval, i.e., 1 in 15 years, increases from 1.10E-6/yr to 1.37E-6/yr. The $\Delta LERF_{15-10}$ due to changing from the current ILRT interval is 2.08E-7/yr in this case, and increased from the original 1.66E-7/yr provided in Attachment 4 to Reference 1.

None of the acceptance criteria are exceeded. Those acceptance criteria were defined in the ILRT risk assessment (Attachment 4 to Reference 1) as ΔLERF less than 1E-6/yr, increase in CCFP no greater than 1.5 percent, and an increase in population dose of no greater than 1.0 person-rem per year. The conclusions in the original ILRT risk assessment remain valid.

Table 8-2 Changes in Risk Metrics Results for SGTR Sensitivity Analysis

		Original Results (from submittal Table 5.7)			New Results		
		Base Case	Current Interval	Proposed Interval	Base Case	Current Interval	Proposed Interval
		(3 in 10 years)	(1 in 10 years)	(1 in 15 years)	(3 in 10 years)	(1 in 10 years)	(1 in 15 years)
Total Dose Rate (person-rem/yr)		32.57	32.61	32.63	40.73	40.78	40.81
Delta Total Dose Rate (person-rem/yr)	From 3 yr	N/A	0.04	0.06	N/A	0.05	0.08
	From 10 yr	N/A	N/A	0.02	N/A	N/A	0.03
LERF (per year)		7.00E-07	9.31E-07	1.10E-06	8.75E-07	1.16E-06	1.37E-06
ΔLERF (per year)	From 3 yr	N/A	2.31E-07	3.98E-07	N/A	2.89E-07	4.97E-07
	From 10 yr	N/A	N/A	1.66E-07	N/A	N/A	2.08E-07
CCFP		54.55%	55.09%	55.48%	54.61%	55.15%	55.54%
ΔCCFP	From 3 yr	N/A	0.54%	0.93%	N/A	0.54%	0.93%
	From 10 yr	N/A	N/A	0.39%	N/A	N/A	0.39%

9. *Item 24 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 listed results in Callaway's sequence quantification document, 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 this 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." These fifteen core damage sequences account for about 88% of the total non-flooding core damage frequency at Callaway. Seven cutsets with frequencies greater than 1E-6 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 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 Reference 1, when the Fourth Callaway PRA Update was completed in early 2006. As described in Table 2.2.1 of Attachment 5 to Reference 1, none of the five remaining F&Os would impact the ILRT risk assessment in Attachment 4 to Reference 1.

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 F&Os and twenty-six Level B F&Os were generated from the analysis. With regard to potential impact of these F&Os, none of the open/remaining F&Os from the gap assessment would impact the ILRT risk assessment provided in Attachment 4 to Reference 1 because, as previously discussed, the ILRT submittal is considered to be a Capability Category I application.

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 generated during the WOG peer review and gap assessment. The total workload of the PRA RG 1.200 Upgrade Project was estimated as about 12,000 man-hours. To support the transition of the Callaway fire protection program to NFPA 805, the PRA Upgrade Project was divided into three phases: A, B, and C. Phase 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. Phase A and B were completed early 2009 by application of about 2,700 man-hours. Callaway is currently working on Phase C which has a workload of 9,300 man-hours. It was started in April 2009. The target completion date for the project is March 2011. All the

remaining open F&Os, including any open C and D F&Os, are anticipated to be resolved upon the completion of the project.