AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

May 14, 2010

ULNRC-05704

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)

References: 1. ULNRC-05665 dated November 25, 2009
2. ULNRC-05694 dated April 22, 2010

In Reference 1 above, 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.

Reference 2 provided additional information requested by the NRC's PSA Branch during the amendment acceptance review.

During the NRC staff's review another 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 Reference 1 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: 5/14/2010

Scott A. M

Scott A. Maglio
Regulatory Affairs Manager

Attachment 1: Responses to NRC Request for Additional Information (RAI) – Questions Regarding License Amendment Request LDCN 09-0039

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

Index and send hardcopy to QA File A160.0761

Hardcopy:

Certrec Corporation
4200 South Hulen, Suite 422
Fort Worth, TX 76109
(Certrec receives ALL attachments as long as they are non-safeguards and may be publicly disclosed.)

Electronic distribution for the following can be made via Tech Spec ULNRC Distribution:

A. C. Heflin
F. M. Diya
L. S. Sandbothe
C. O. Reasoner III
S. A. Maglio
S. L. Gallagher
T. L. Woodward (NSRB)
T. B. Elwood
G. G. Yates
Ms. Diane M. Hooper (WCNOC)
Mr. Dennis Buschbaum (Luminant Power)
Mr. Ron Barnes (APS)
Mr. Tom Baldwin (PG&E)
Mr. Wayne Harrison (STPNOC)
Mr. John O'Neill (Pillsbury, Winthrop, Shaw, Pittman LLP)
Missouri Public Service Commission
Mr. Dru Buntin (DNR)

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)
QUESTIONS REGARDING LICENSE AMENDMENT REQUEST LDCN 09-0039

By letter dated November 25, 2009 (i.e., letter ULNRC-05665), AmerenUE submitted a request to amend Technical Specification 3.3.2. That amendment request would add a new Required Action for the restoration of an inoperable Balance of Plant (BOP) Engineered Safety Feature Actuation System (ESFAS) train to operable status within 24 hours.

The NRC staff has completed its initial review of the Union Electric Company's application for extending the Callaway Plant, Unit 1, Completion Time for Technical Specification 3.3.2 [Condition Q] *to 24 hours. This change would add a new Required Action (Q.1) for the restoration of an inoperable BOP ESFAS train to operable status within 24 hours. In addition, the Completion Time for TS 3.3.2 Required Actions J.1 and O.1 would be extended to 24 hours.*

Based on our review of the application, the NRC staff has identified areas where additional information is needed to complete our review. The request for additional information is provided below. Please provide your responses by May 15, 2010, so that the NRC staff can complete the requested action in a timely manner.

Question 1 Internal Events CDF

The Callaway IPE reported a total CDF of 5.9 E-05 /yr. The internal flooding contribution was about 30% of this CDF value (~ 4.2 E-05 /yr) which is the baseline reported value for internal flooding CDF (section 4.1.1 Attachment 1 page 18). Please provide additional information regarding each of the revisions of the PRA since the IPE, including the plant and model changes, success criteria and supporting analysis changes, and associated baseline CDF and LERF.

Response:

It should be clarified that Chapter 7 of the Callaway Individual Plant Examination (IPE) report (ULNRC-02703 dated 9-29-92) documented an internal flooding contribution to the core damage frequency (CDF) of 1.78E-05/yr, which was 30.48% of the total CDF (5.846E-05/yr), as opposed to the 4.2E-05/yr value cited in this question. Nevertheless, it is true that the total CDF reported in 1992 has decreased over the four probabilistic risk assessment (PRA) updates that have occurred since then, as discussed hereinafter.

The values for CDF, large early release frequency (LERF), and flooding percentage, as well as information describing the model changes, are documented in the IPE revisions and in the appropriate addenda to calculations ZZ-267 for system quantification and ZZ-470 for the LERF model.

The Callaway IPE was completed and the results were submitted to the NRC in September 1992. The current Level 1 PRA model is documented in the 4th PRA update, and the 5th PRA update project is currently underway. A summary of the previous PRA CDF updates is provided in the following table.

	IPE CDF (yr^{-1})	First PRA Update CDF (yr^{-1})	Second PRA Update CDF (yr^{-1})	Third PRA Update CDF (yr^{-1})	Forth PRA Update CDF (yr^{-1})
Document Date	1992	1999	2000	2004	2006
Non-Floods	4.02E-05	3.32E-05	2.45E-05	3.47E-05	4.22E-05
Floods	1.78E-05	5.95E-06	5.95E-06	9.14E-06	9.14E-06
Other	4.73E-07	4.73E-07	4.73E-07	4.73E-07	4.73E-07
Total CDF	5.85E-05	3.96E-05	3.09E-05	4.43E-05	5.18E-05
"Other" is comprised of reactor vessel rupture and interfacing systems LOCAs.					

A summary of the previous PRA LERF updates is provided in the following table.

	IPE LERF (yr^{-1})	First PRA Update LERF (yr^{-1})	Second PRA Update LERF (yr^{-1})
Document Date	1992	2000	2001
PDS 23 Frequency	7.11E-09	9.31E-10	2.28E-09
PDS 24 Frequency	7.56E-09	3.52E-09	4.33E-09
Combined PDS 30 and 51 Frequency	1.56E-08	3.57E-09	2.79E-10
PDS 81 (SGTR) Frequency	4.56E-07	2.41E-07	2.40E-07
PDS 80 (ISL) Frequency	1.73E-07	1.73E-07	1.73E-07
Total LERF	6.59E-07	4.22E-07	4.20E-07

1st PRA Update

The first PRA update was completed in February 1999. Callaway PRA calculation ZZ-267, Revision 0, Addendum 1, documented the sequence quantification for the first update. That calculation addendum was generated to support Task 2.6 of the Callaway PRA Update 1, which involved revising, executing, and debugging of the sequence quantification batch input files.

The reduction in the flooding-initiated CDF was primarily due to (1) a reduction in selected flood initiator frequencies, (2) the actual calculation of the conditional core damage probability (CCDP), given a flood, as opposed to assuming a CCDP of 1.0, and (3) credit taken for the normal charging pump (NCP). The reduction in the non-flooding-initiated CDF was primarily due to (a) lower initiating event frequencies, (b) lower test/maintenance probabilities (i.e., shorter times that trains of equipment were in test/maintenance) and (c) credit taken for the NCP.

The changes for the models of systems documented in the following Callaway PRA calculations were incorporated in the first PRA update.

(1) Calculation BG-33, Revision 0, Addendum 1, was generated to document changes made to the RCP seal cooling fault tree (FT), pursuant to the Update Division Action Plan that was issued on May 20, 1998. The RCP seal cooling FT was updated to incorporate two (2) plant modifications:

- Callaway Modification Package (CMP) 89-1028 in which valves BGHV8357A, B were changed from solenoid-operated valves (SOVs) to motor-operated valves (MOVs).
- CMP 92-1010 in which the positive displacement charging pump (PDP) was replaced with a centrifugal charging pump (PBG04), also referred to as the normal charging pump, or NCP.

(2) Calculation EA-03, Revision 0, Addendum 1, documented a change made to the PRA "All Service Water" fault trees. The change was made to the fault trees to incorporate a plant maintenance-related configuration which was either not practiced when the IPE was developed or was overlooked at that time. The plant configuration referred to is the draining of one (1) train of Essential Service Water (EF system) for the performance of maintenance.

(3) Calculation EF-15, Revision 0, Addendum 1, documented two (2) changes made to the Essential Service Water (ESW) system fault trees, pursuant to the first PRA update. Those changes were:

- Addition of a failure to recover valve EFHV0059 event to the ‘A’ train ESW fault trees, and
- Creation of two fault trees (ETNAX and ETWBX) in order to break logic loops at a lower level in the fault trees.

(4) For the Component Cooling Water (CCW) system, calculation EG-16, Revision 0, Addendum 1, was generated to correct a modeling oversight error which was documented in Corrective Action Program (CAP) document SOS (no longer used acronym for “Suggestion, Occurrence, Solution”) 92-2031. That oversight was the omission, from the CCW train ‘A’ fault tree model, of a failure of check valve EGV003 to open following a loss of offsite power (LOSP) event.

(5) Calculation EP-10, Revision 0, Addendum 1, was generated to support Task 2.4 of the Callaway PRA Update 1 as described in the Update Division Action Plan that was issued on May 20, 1998. Task 2.1 of the plan identified Probabilistic Risk Assessment Evaluation Request (PRAER) 94-007 as requiring an update in the accumulator fault tree model. PRAER 94-007 was performed to support OL Amendment No. 1150 in which (what is now) Technical Specification 3.5.l Condition A was added and Condition B was revised to allow a 24-hour Completion Time (also called allowed outage time or AOT at the time Callaway License Amendment 91 was approved on August 5, 1994). PRAER 94-007 evaluated the longer AOT by modeling test and maintenance in the accumulator fault tree.

(6) Calculation NB-03, Revision 0, Addendum 1, documented changes made to the Callaway emergency diesel generator (DGN) fault trees in order to incorporate a change to the fuel oil transfer pump start logic, implemented via CMP 88-1004. That CMP changed the start logic such that each fuel oil transfer pump, PJEOIA(B), runs when its associated DGN, NE01(2), runs.

(7) Calculation NK-06, Revision 0, Addendum 1, was generated to support Task 2.4 of the Callaway PRA Update 1 as described in the Update Division Action Plan that was issued on May 20, 1998. Task 2.1 of the plan identified PRAER 94-018 as requiring an update in the DC bus fault tree models. PRAER 94-018 was performed to support CMP 92-1014. That CMP installed two swing battery chargers in the NK system.

(8) Calculation ZZ-253, Revision 1, was generated to support Task 2.4 of the Callaway PRA Update 1 Plan that was issued on May 20, 1998. Task 2.1 of the plan identified PRAER 98-102 as requiring an update in the secondary plant depressurization fault tree models. PRAER 98-102 evaluated a change to what is now the LCO for Technical Specification 3.7.4. The LCO was changed from requiring three operable atmospheric steam dump (ASD) valves to requiring four operable ASD lines (a line includes the ASD valve and the associated block valve).

(9) Calculation ZZ-257, Revision 1, Addendum 1 (addendum to the original Initiating Event (IE) frequency calculation) documented the methods and results of the IE frequency update. The following IE frequencies were updated.

T_3 - Turbine Trip/Reactor Trip
T_2 - Loss of Main Feedwater
T_{SG} - Steam Generator Tube Rupture (SGTR)
T_1 - Loss of Offsite Power (LOSP)
T_{DC} - Loss of a Vital 120 VDC (NK01 or NK04) Buss
S_3 - Very Small LOCA

Also updated were the frequencies for a T_2/T_3 - initiated ATWS event and a T_1 - initiated ATWS event.

(10) Calculation ZZ-275, Revision 0, Addendum 1, was issued to document changes made to the Callaway IPE Level 1 PRA event trees pursuant to the first PRA update. Specific changes made to the event trees were described. The revised event trees were provided. Also attached to and described in this calculation addendum are the following items associated with revision of the event trees:

- Revised OCL files for the (revised) T_C and T_{SW} event trees.
- A fault tree for use in quantifying the "N_{CP}" event, which was added to the revised T_C and T_{SW} event trees.
- Marked-up sections of the batch input file showing the changes made to that file to quantify the "N_{CP}" event for solution of the revised T_C and T_{SW} event trees.

(11) Calculation ZZ-462, Revision 0, documented the model, input information, and methodology used to quantify the core damage frequencies (CDFs) due to postulated floods in selected areas of the plant. In each of these areas, the postulated source of the flood was an ESW line. Therefore, in addition to flooded equipment and cable, an ESW train was assumed to fail. The calculated CDFs, due to each of these postulated floods, were also documented in this calculation.

(12) Calculation ZZ-266, Revision 0, Addendum 1, documented the update and, in the case of the NK battery chargers, creation of test and maintenance (TM) basic event probabilities used in the updated Callaway PRA model. This data update task was performed pursuant to Task 2.5 of the "Plan for the First Update of the Callaway IPE PRA." Calculation ZZ-266, Revision 0, Addendum 2, added a printout of the revised UEALL.BED data file.

(13) Calculation ZZ-470, Revision 0, generated the Callaway Large Early Release Frequency (LERF) model. Containment failure frequencies and quantitative release calculations were performed for Callaway during the Level 2 evaluation of the Individual Plant Examination (IPE). The containment failure sequences ranged from no failure, to

early failure, to long term containment failure. The release calculations ranged from small to large releases. The release calculations ranged from small to large releases.

The 4.22E-07 yr^{-1} calculated LERF for Callaway using the first PRA update database was 36% less than the LERF calculated from the IPE data.

2nd PRA Update

Callaway PRA calculation ZZ-267, Revision 0, Addendum 2 was generated to support Self-Assessment Action Items of the Callaway PSA Self-Assessment Report that was issued September 15, 2000. That report required a requantification of the PRA that incorporated several corrections/updates. The major changes that were incorporated included: a correction to a fault tree, new LOCA (large, intermediate, and small) initiating event frequencies, a change to the Service Water fault tree, revision of the diesel generator mission times, and addition of several transfer sequences to the core damage equation.

The reduction in the non-flooding-initiated CDF was primarily due to the lower initiating event frequencies for three LOCA sizes (large, intermediate, and small) and a change in the Service Water fault tree.

The changes for the models of systems documented in the following Callaway PRA calculations were incorporated in the second PRA update.

(1) Calculation EA-06, Revision 0, Addendum 2, documented a revision made to the SVCWTRA and SVCWTRAX fault trees, during the "PRAWDT2" update.

(2) Calculation ZZ-257, Revision 0, Addendum 2, was generated to support Self-Assessment Action Item No. 2 of the Callaway PSA Self-Assessment Report that was issued September 15, 2000**.** A result of that report was a requantification of the PRA that incorporated several corrections/updates, including the one documented in this addendum. Action Item No. 2 of the report addressed the update of LOCA initiating event (IE) frequencies. The following IE frequencies were updated:

A - Large break LOCA
S1 - Medium break LOCA
S2 - Small break LOCA

(3) Calculation ZZ-258, Revision 0, Addendum 1, was generated to support Self-Assessment Action Item No. 1 of the Callaway PSA Self-Assessment Report that was issued September 15, 2000. A result of that report was a requantification of the PRA that incorporated several corrections/updates, including the one documented in this addendum. PRAER 00-114 was identified as finding an error in the **14HPIlS.LGC** fault tree. That error was corrected. PRAER 00-114 documented the Callaway input provided

to the Westinghouse Owners Group in support of the allowed outage time (AOT) extension program for the diesel generators (DGNs).

(4) Calculation ZZ-266, Revision 0, Addendum 3, documented changes to the revised UEALL.BED data file as result of the second PRA update.

(5) Calculation ZZ-470, Revision 0, Addendum 1, was generated to update the Large Early Release Frequency (LERF) model based on the second requantification of the PRA.

The revised Callaway large early release frequency (LERF), determined via the second PRA update, was 4.20E-07 yr^{-1}. When compared to the first PRA update LERF of 4.22E-07 yr^{-1}, this represented a decrease of approximately 0.5%.

<u>3rd PRA Update</u>

Callaway PRA calculation ZZ-267, Revision 0, Addendum 3, was generated to requantify the Callaway PRA core damage sequences with the incorporation of the third PRA update information and changes.

The third PRA update increased the flooding CDF based on information included in the first PRA update (Calculation ZZ-466, Revision 0, Addendum 1). The flooding CDF increase was mainly tied to PRAER 02-170 which evaluated the impact of adequate floor drainage and determined the effects of drain blockage and inoperable sump pumps (where sump pumps are installed in the flood areas evaluated in the plant).

The increase in the non-flooding initiated CDF was influenced to a large degree by higher initiating event frequencies. New industry data dictated increased initiating event frequencies for three LOCA sizes (Large, Intermediate, and Small) and for the Steam Generator Tube Rupture event. A change in methodology using industry data (versus using the large LOCA frequency) for all Secondary Line Breaks resulted in increased initiating event frequencies for these events as well. The addition of the potential for Service Water system (EA system) strainer plugging resulted in the increased initiating event frequency for the Loss of All Service Water event.

New industry data, along with Callaway Plant-specific data, resulted in a reduction in the Loss of Offsite Power initiating event frequency in the third PRA update.

The Station Blackout event was also impacted by the reduction in the Loss of Offsite Power initiating event frequency as well as by the increased reliability of Callaway's EDGs (Emergency Diesel Generators) as reflected in lower EDG fail-to-run probabilities generated with updated Callaway-specific failure data. However, those benefits were off-set by two methodology changes. The first methodology change was related to the calculation of the probability of recovery of AC power, and resulted in a significant increase in fail-to-recover AC power probabilities. The second methodology change added a basic event to specifically account for EDG fail-to-run common cause failures (CCFs). The revised EDG CCF probabilities were significantly larger than the combined EDG CCF probability used in the second PRA update. The end result was an increase in the Station Blackout event CDF. These methodology changes were required, however, in order to make the SBO modeling more robust.

The major changes incorporated in the third PRA update included:

(1) The addition of steam generator blowdown isolation and AFW "smart valve" failures to the AFW fault tree (AL-04, Revision 1, Addendum 2, "Auxiliary Feedwater System Fault Tree Model").

(2) Adding additional CCW failures to the RCP seal cooling fault tree (BG-33, Revision 0, Addendum 3, "RCP Seal Cooling Fault Tree").

(3) The expansion of rotating component common cause failures into common cause failure-to-start and failure-to-run failure modes (ZZ-126, Revision 0, Addendum 1, "Expansion of Rotating Component Common Cause Failure Basic Events in the System Fault Trees").

(4) New initiating event frequencies (ZZ-257, Revision 0, Addendum 4, "PRA Initiating Event Frequencies for the Third PRA Update").

(5) A revision to the loss of all component cooling water event tree to question recovery of CCW prior to questioning the availability of RHR recirculation (EG-27, Revision 0, Addendum 1 "Calculation of CCW System Recovery" and ZZ-275, Revision 0, Addendum 2, "Revision to Loss of All CCW Event Tree").

(6) The recalculation of station blackout (SBO) failure-to-recover AC power probabilities (ZZ-276, Revision 1, "Determination of Offsite Power Failure-to-Recover Probabilities for Use in Station Blackout Quantification").

(7) The use of updated component failure rate data, component common cause data, and test and maintenance unavailability data (ZZ-266, Revision 0, Addendum 4, Addendum 6 and Addendum 7, "Basic Event Data (BED) File for the Third PRA Model Update").

<u>4th PRA Update</u>

Callaway PRA calculation ZZ-267, Revision 0, Addendum 4, was generated to requantify the Callaway PRA core damage sequences with the incorporation of fourth PRA update information and changes.

The increase in the non-flooding initiated CDF was influenced to a large degree by higher human error probabilities that were calculated for the risk-significant human failure events during the human reliability analysis (HRA) Update documented in ZZ-278, Revision 0, Addendum 1, “Callaway IPE / PRA Human Error Calculation.” The HRA Update reflected an updated human reliability analysis using a currently-accepted methodology.

The major changes that this update incorporated include:

(1) Added actuation failures and DC power dependency to the main steam isolation fault tree (AB-11, Revision 0, Addendum 1, “Callaway PRA - Failure of Main Steam Isolation Fault Tree”).

(2) Added actuation and main feedwater regulating valve (MFRV) failures to the main feedwater isolation fault trees. Changed the top level success criteria to isolation of 4-of-4 steam generators for the **MFWISOL1.LGC** fault tree (AE-29, Revision 0, Addendum 2, “Failure of Main Feedwater Isolation Fault Tree”).

(3) Deleted the top level event of failure to terminate SI from the secondary line break event trees [T(MSI), T(MSO), and T(FLD)]. Replaced this top level event with an event of failure to reclose the PZR PORVs or safety valves following water relief through the valves. This necessitated the creation of a new fault tree, **PWR.LGC**. (BB-92, Revision 1, Addendum 2, “Failure of PZR PORV or Safety Valve to Reclose After Reactor Trip or Secondary Break Fault Trees” and ZZ-275, Revision 0, Addendum 3, “Callaway IPE - Level 1 Event Trees”).

(4) Added logic to the CCW train B fault tree to incorporate a new basic event for the probability that EFHV0052 is closed (EG-16, Revision 0, Addendum 2, “CCW Trains A & B Fault Trees”).

(5) Changed the actuation logic in the RHR cold leg recirculation fault tree back to its original form. This necessitated a modification to BED file **SBOPRFB.BED** (EJ-19, Revision 0, Addendum 1, “RHR System Cold Leg Recirculation Mode Fault Tree Model”).

(6) Modified the Loss of All Service Water [T(SW)] event tree to remove a branch that questioned service water restoration at 2 hours for events with successful decay heat removal via AFW. Added a branch that questioned core uncovery for events with AFW, with RHR injection, and with successful service water recovery at 8 hours. Added a branch that questioned core uncovery for events without AFW, but with successful service water recovery at 2 hours (ZZ-258, Revision 0, Addendum 2, “Quantification Fault Tree Models” and ZZ-275, Revision 0, Addendum 3, “Callaway IPE - Level 1

Event Trees").

(7) Modified the **DAM-1.LGC** fault tree to remove small LOCA [S(2)] cutsets that include basic event OP-XHE-FO-CCWRHX. The failure to initiate CCW flow to the RHR heat exchanger, for the S(2) event, was captured in basic event OP-XHE-FO-ECRLS2 (ZZ-264, Revision 0, Addendum 4, "Callaway PRA - Disallowed Maintenance Fault Tree" and ZZ-278, Revision 0, Addendum 1, "Callaway IPE / PRA Human Error Calculation").

(8) Modified the Intermediate LOCA [S(1)] event tree to add a branch that questioned accumulator injection for events with successful high head injection (ZZ-275, Revision 0, Addendum 3, "Callaway IPE - Level 1 Event Trees").

(9) Modified the Main Steamline Break Outside Containment [T(MSO)] and Main Steamline Break Inside Containment [T(MSI)] event trees to modify branching to address F&O AS-4 concerns with success criteria (ZZ-275, Revision 0, Addendum 3, "Callaway IPE - Level 1 Event Trees").

(10) Returned to a 1-hour AC power recovery probability (from 2 hours) for SBO sequences S21 to S26 (ZZ-276, Revision 1, Addendum 1, "Determination of Offsite Power Failure-to-Recover Probabilities for Use in Station Blackout Quantification").

(11) Updated the Human Reliability Analysis (HRA). Several sets of individual, independent human failure event (HFE) basic events were determined to not be independent (within the set). These sets of basic events were replaced by new, single HFE basic events. Numerous fault trees were modified to incorporate these new HFE basic events. Additionally, the updated HRA recalculated human error probabilities (HEPs) for the risk-significant HFEs. These new HEPs were incorporated into the basic event database (BED) file (AL-04, Revision 1, Addendum 4; BB-95, Revision 1, Addendum 1; BB-98, Revision 0, Addendum 1; BG-32, Revision 0, Addendum 1; BG-33, Revision 0, Addendum 4; EA-06, Revision 0, Addendum 4; EA-07, Revision 0, Addendum 2; EG-16, Revision 0, Addendum 3; EJ-19, Revision 0, Addendum 1; EJ-20, Revision 0, Addendum 1; EM-02, Revision 1, Addendum 1; EM-03, Revision 1, Addendum 1; EM-04, Revision 1, Addendum 1; NB-03, Revision 0, Addendum 3; NK-06, Revision 0, Addendum 2; ZZ-258, Revision 0, Addendum 2; ZZ-263, Revision 0, Addendum 1; ZZ-266, Revision 0, Addendum 8 and ZZ-278, Revision 0, Addendum 1).

Question 2 Bayesian Analysis

On page 17 Section 4.1.1 of Attachment 1, the BOP ESFAS train failure rate was estimated using Bayesian analysis, but no details or guidelines were provided. Please provide additional information regarding how the Bayesian analyses were performed and the data used in the analyses.

Response:

See the response to question 8 in ULNRC-05694 dated April 22, 2010. That information is repeated here with an additional summary paragraph added at the conclusion of this response.

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 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 value 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 by 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-2. 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 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 of λ is assumed as the Jeffrey's non-informative prior distribution 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). Using a Bayesian update, the failure rate of BOP ESFAS $\lambda = (14+0.5)/(122.48+0) = 1.18E\text{-}01\ yr^{-1}$ 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 $\lambda = (14+1.0)/(122.48+0) = 1.225E\text{-}01\ yr^{-1}$ and MTTF = 8.17 years. The 5th percentile of λ is 7.12E-02 and the 95^{th} percentile of λ is 1.91E-01.

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 conservative failure rate of 1.225E-01 yr^{-1} was used in the PRA evaluation reported in ULNRC-05665.

Summary

The failure rate was used to evaluate the yearly average impact on the plant risk due to a failure of a SA036D (or SA036E) BOP ESFAS cabinet. On average over a year the potential unavailability of the 24-hour Completion Time extension from the failure is 1.225E-01*24/8760 =3.355E-04, which is added to the following basic events: AL-ICC-AF-AFAS4, AL-ICC-AF-LOSP4, and AMSACFAILS. The resulting plant risk in CDF is 4.213E-05. This value is the same as the normal operation baseline risk. In addition, basic events for the containment purge system, VT-PND-FT-VTHZ04, VT-PND-FT-VTHZ11 and MNPURGVLVSOPEN, were adjusted by adding the unavailability 3.355E-04. The resulting failure probability of containment isolation is the same 3.551E-03 and the LERF change shows increase 0.01%; therefore, the resulting LERF is almost unchanged. From the above analysis, although the BOP ESFAS CT extension can potentially affect the plant risk, the potential impact of the CT extension unavailability on the plant baseline risk would be negligible on the yearly average basis.

Question 3 Operator Performance

It was stated in Attachment 1 of the submittal that continued operator training of the effect of completion time extension is planned. No details were provided about the nature of training or supporting procedures. Please describe how the proposed amendment impacts the human error probabilities and how that was reflected in the model.

Response:

The discussion of operator training on page 8 of Attachment 1 to ULNRC-05665 (original license amendment request) refers to the training performed for every license amendment as a part of License Operator Continuing Training. This training is not unique to this specific amendment request. That is why no details were provided. With respect to supporting procedures, OTS-SA-00001, “Operation of Engineered Safety Feature Actuation System,” provides direction for de-energizing and re-energizing Engineered Safety Feature Actuation System (ESFAS) cabinets and identifies applicable

Technical Specification Conditions to be entered following power supply failure or removal from service.

With one train of the auxiliary feedwater actuation signal (AFAS) circuitry of the BOP ESFAS out of service (OOS) during the requested 24-hour Completion Time, the operators would attempt to manually initiate AFW flow if the other BOP ESFAS train failed coincident with the occurrence of a reactor trip or safety injection signal.
In addition, the plant would technically be in LCO 3.0.3 in this situation. The motor-driven AFW pumps and the turbine-driven AFW pump are checked to be running at Step 8 of E-0, "Reactor Trip or Safety Injection," and total AFW flow to the steam generators is checked at Step 10 of E-0. Therefore, the operators will initiate, and verify, AFW flow quickly after transient initiation. This manual backup to the automatic AFW actuation signal is modeled in the Callaway PRA. The change to a Completion Time of 24 hours in TS 3.3.2 Condition Q (plus 6 hours to be in MODE 3) from no allowed outage or restoration time (plus 6 hours to be in MODE 3) does not impact the operator's ability to manually start an AFW pump. The Risk Achievement Worth (RAW) for the basic event of failure to manually start the turbine-driven AFW pump is unaffected by the proposed amendment and the RAW for the basic event of failure to manually start the motor-driven AFW pumps decreases for the proposed amendment.

Question 4 Common Cause Failure

The Sorensen Power Supplies and cards used in the BOP ESFAS appear to exist in other systems at the Callaway station. Please describe how inter-system CCF is addressed in the PRA for this potential condition and, if appropriate, please perform sensitivity analyses that address this potential inter-system CCF.

Response:

This risk-informed submittal requires a PRA model that meets Capability Category II of the ASME Standard. The ASME Standard does not require modeling of inter-system common cause failures for Capability Category II.

Page 32 of RG 1.200, Revision 2, says that PRA Capability Category II (CC-II) is adequate for the majority of risk-informed applications. This is the expectation for the subject amendment that affects only TS 3.3.2. In addition, Table B-4 of RG 1.200, Revision 2, says that Supporting Requirement SY-B2 is not required for CC-I or CC-II.

NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," was submitted to the NRC on 11/13/06 and was approved generically on 5/17/07. That NEI topical report was also approved during the plant-specific lead plant RI-4b demonstration pilot for STP. Item 2 of Section 2.3.4 in NEI 06-09 requires that a plant must meet CC-II to pursue RMTS. Logic dictates that if CC-II is sufficient for an entire RMTS conversion using RI-4b for every LCO with

PRA-modeled structures, systems, and components, then CC-II should be sufficient for the proposed amendment which affects TS 3.3.2 Functions 6.c, 6.g, and 6.h in a single LCO.

Question 5 Internal Fire

In the submittal's section 4.1.2 of Attachment 1 it was stated that "fire in the control room is dominated by human action including manual actuation." It was concluded in the evaluations that out of service BOP ESFAS does not impact the ability of the operator to manually actuate AFWS, and no change in risk was estimated. The Callaway IPEEE study actually identified the main control room as the most significant contributor to fire risk. Please discuss the credit given to the control room operator manual actions relevant to the baseline risk assessment.

Response:

Section 4.3.6 of the Callaway IPEEE describes the control room fire evaluation. The dominate control room fire sequence consists of:

- control room fire AND
- control room evacuation required AND
- human error to safely shutdown from the auxiliary shutdown panel.

A failure probability of 0.06 was used for the human error to safely shutdown from the auxiliary shutdown panel. The IPEEE indicates that this failure probability was obtained from EPRI TR-104031, "Fire Risk Analysis Implementation Guide." Section 4.1.2 of Attachment 1 to the submittal shows that the baseline AFW unavailability is 3.616E-04, while the AFW unavailability with one train of AFAS OOS is 4.862E-04. The increase in unavailability of the AFW system, due to an AFAS train OOS, is 1.246E-04. The human error failure probability of 0.06 is more than two orders of magnitude larger than the AFW unavailability probabilities.