AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

December 12, 2003

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Ladies and Gentlemen: ULNRC-04928

DOCKET NUMBER 50-483
CALLAWAY PLANT
UNION ELECTRIC COMPANY
RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION
PROPOSED REVISION TO TECHNICAL SPECIFICATION 1.1, "DEFINITIONS"; TECHNICAL SPECIFICATION 3.7.3 "MAIN FEEDWATER ISOLATION VALVES (MFIVs)"; AND STEAM GENERATOR TUBE RUPTURE WITH OVERFILL RE-ANALYSIS

Reference: ULNRC-04592, dated June 27, 2003

The referenced letter transmitted to the NRC AmerenUE's subject proposed license amendment request. During its review, the NRC staff made several requests for additional information via e-mail, telephone conference, and during a meeting held with the NRC on November 12, 2003. AmerenUE has responded via e-mail, telephone conference, and during the meeting with the NRC. However, this letter provides formal transmittal of AmerenUE responses to the requests for additional information. The Attachment to this letter provides a composite of all NRC staff requests for additional information and AmerenUE's responses to those requests.

In addition, AmerenUE is revising information provided in the original submittal which was transmitted to the NRC in the referenced letter. First, Page 4 of 33 of Attachment 2 to the referenced letter describes the closure of the main feedwater isolation valve with the replacement actuator. It contains the statement: "After a 30 second time delay, solenoid valves MV5 and MV6 will go to an energized state (closed or pressurized position), preventing any leakage from the LPC".

Following assembly of each MFIV system medium actuator, a hot functional test was performed using Callaway's spare MFIV body. Through this testing it was found that the MFIV would close within 60 seconds with as little as 0 psig of system

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pressure. Based upon this testing, the time delay to energize MV5 and MV6 was changed to 60 seconds to ensure MFIV closure during a feed water line break with no system pressure available.

In order to ensure the MFIVs will close during a feed water line break with no system pressure available, MV5 and MV6 will go to an energized state (closed or pressurized position) after a 60 second time delay. This change in the time delay to energize MV5 and MV6 does not affect the MFIV 15 second stroke time with sufficient system pressure available. The change in solenoid closure time is reflected in the AmerenUE response to request for additional information concerning the feed water line break event. This change has no impact on evaluation results for the affected accident scenario.

Second, the minimum DNBR for the Steam Line Break of 2.072 for Cycle 13, which was previously reported in ULNRC-04592, was incorrect. The reported value for Cycle 13 should have been a minimum DNBR of 2.90. Regardless of this reporting error, the analyses for all Steam System Piping Failures continue to satisfy their acceptance criteria and the conclusions presented in the FSAR for these Non-LOCA events remain valid. The minimum DNBR for SLB is calculated for each fuel cycle. Fuel Cycle 14 has a minimum DNBR of 2.35 as compared to the acceptance criteria of 1.50.

If you should have any questions on the above or attached, please contact Dave Shafer at (314) 554-3104 or Dwyla Walker at (314) 554-2126.

Very truly yours,

Keith D. Young

Keith D. Young
Manager, Regulatory Affairs

KDY/DJW/mlo
Attachment:

cc: U. S. Nuclear Regulatory Commission (Original and 1 copy)
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

Mr. Bruce S. Mallet
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

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

Mr. Jack N. Donohew (2 copies)
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop 7E1
Washington, DC 20555-2738

Manager, Electric Department
Missouri Public Service Commission
PO Box 360
Jefferson City, MO 65102

STATE OF MISSOURI)
) S S
COUNTY OF CALLWAY)

Keith D. Young, of lawful age, being first duly sworn upon oath says that he is Manager, Regulatory Affairs, for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Keith D. Young
Keith D. Young
Manager, Regulatory Affairs

SUBSCRIBED and sworn to before me this 12th day of December, 2003.

CATHY J. CRISP
MY COMM. EXP.
NOTARY SEAL
NOTARY PUBLIC
JAN. 29, 2006
STATE OF MISSOURI

Cathy J. Crisp
Notary Public
State of Missouri
Expiration 1-29-06

I. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED JULY 30, 2003

In reviewing your application dated June 27, 2003 (ULNRC-04592), which is revising the analysis of the steam generator tube rupture (SGTR) with overfill event, the staff has determined it needs the following information concerning the times listed in the table on page 17 of 33 of Attachment 2, "Evaluation," to the letter:

NRC Question 1:

The paragraph just below the table states that "Simulated control room exercises were performed in 2003 for this accident. The exercises have demonstrated that the operator action times that serve as inputs to the thermal-hydraulic analysis have increased above the times originally analyzed in the SGTR with overfill analysis presented to the NRC in [letter] ULNRC-1518, dated May 27, 1987 ..." The sentences in the application may mean that the times in the previous analysis were increased to those listed in the table solely to meet the increased operator action times observed in simulated control room exercises in 2003. Discuss this and explain what the times in the table are based upon.

AmerenUE Response to Question 1:

During the review and screening of the Feedwater Isolation Valve (MFIV) Actuator modification, it was determined that changing the MFIV isolation time from 5 to 15 seconds had the potential to adversely impact the SGTR-Overfill analysis. In the process of determining the sensitivity of Overfill consequences to the MFIV isolation time, it was identified that the assumed Operator action times used as inputs in the SGTR-Overfill analysis had not been maintained as currently valid.

The issues related to the validity of the SGTR-Overfill inputs were entered into the Callaway Plant's corrective action program. Additionally, a Licensee Event Report (LER 2003-003-00) was submitted to report this issue.

A complete re-analysis of the SGTR-Overfill sequence was performed. Re-validation of all analysis inputs was performed as a part of this re-analysis effort. During the re-analysis effort it was necessary to establish a new set of operator action times for use in the re-analysis. A series of simulator exercises were performed during the Spring of 2003 to establish the new operator action times. Additionally, AmerenUE and Westinghouse personnel reviewed the operator action times used by other Westinghouse plants to benchmark the new Callaway times to ensure that the new times were reasonable.

The analysis effort was then completed based on the new set of operator action times. These are the values provided in our License Amendment Request.

As will be discussed below, all operating crews have demonstrated that they are capable of satisfying these times.

NRC Question 2:

The NRC staff's evaluation dated March 30, 1987, of the Westinghouse Owners group WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," stipulated plant-specific criteria for assessing operator action times in the event of an SGTR. Address the criteria as updated below:

AmerenUE Response to Question 2:

It should be noted that the Callaway SGTR analysis is not performed using the methodology described in WCAP-10698. The Callaway SGTR analysis is based on the SNUPPS methodology which was originally developed for Callaway and Wolf Creek. The treatment of operator response times used in the SGTR analysis is discussed as follows:

NRC Question 2a:

Provide simulator and emergency operating procedure training related to a potential SGTR.

AmerenUE Response to Question 2a:

Callaway's Licensed Operator Training program provides training on the SGTR accident sequence and the associated emergency operating procedures (EOPs) used to respond. All Callaway Licensed Operators have been trained on the updated Procedure E-3 "Steam Generator Tube Rupture", using both classroom and simulator training sessions.

NRC Question 2b:

Using typical control room staff as participants in demonstration runs, show that the operator action times assumed in the SGTR analysis are realistic and achievable.

As requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide a table of SGTR operator response times for all crews from the simulator exercises. This includes measured times for the equipment operator to isolate auxiliary feedwater (AFW) flow to the faulted steam generator. In addition, provide a statement on how it was verified that the training simulator accurately modeled the initial conditions and critical parameters from the FSAR Chapter 15 Steam Generator Tube Rupture analysis.

AmerenUE Response to Question 2b:

All Callaway Plant operating crews have demonstrated that they can achieve the new SGTR-Overfill operator action times. This includes both on-shift and staff crews.

The AmerenUE Safety Analyses Engineers provided the Training Department with all of the critical attributes from the FSAR Chapter 15 SGTR overfill analysis to be used in the development of the SGTR overfill simulator training exercises. These included, but were not limited to, initial plant conditions, single failures, available equipment, equipment performance characteristics, credited RTS and ESFAS instrumentation, and event duration. The Safety Analyses Engineers then observed all of the SGTR overfill simulator training exercises to verify that the simulator results and operator actions were consistent with the FSAR Chapter 15 results.

The following tables list the response times demonstrated by Callaway Operations crews during simulator exercises:

Description	**Acceptable values**	**Crew A**	**Crew B**	**Crew C**	**Crew D**	**Crew E**	**Crew F**
T1 - AFW flow to ruptured S/G isolated	20	8	8	7	8	6	6
T2 - Initiate RCS cooldown	30	19	19	27	25	23	25
T3 - Complete RCS depressurization	40	29	29	30	35	35	37
T4 - SI terminated	45	34	33	36	40	40	39
T5 - RCS - S/G pressure equalized	60	43	49	51	58	59	54

Description	**Acceptable values**	**Crew G**	**Crew H**	**Crew I**	**Crew J**	**Crew K**	**Crew L**
T1 - AFW flow to ruptured S/G isolated	20	7	7	8	9	8	8
T2 - Initiate RCS cooldown	30	19	24	17	21	22	20
T3 - Complete RCS depressurization	40	30	37	29	31	36	28
T4 - SI terminated	45	35	42	33	37	38	34
T5 - RCS - S/G pressure equalized	60	45	58	46	39	49	45

NRC Question 2c:

Complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment and its impact on operator response times. All control room crews should demonstrate a response time which is less than the operator response time assumed in the analysis for the accident.

AmerenUE Response to Question 2c:

Overfill prevention is not demonstrated for Callaway. The analysis and operator action times are commensurate with mitigation of the consequences of an overfill event. All Callaway Plant operating crews have demonstrated that they can achieve the new SGTR-Overfill operator action times. This includes both on-shift and staff crews.

NRC Question 2d:

Describe the means the emergency operating procedures specify for identifying the steam generator (SG) with the ruptured tube, provide the expected time period for determining that SG, and discuss the effects on the duration of the accident.

AmerenUE Response to Question 2d:

E-0 "Reactor Trip or Safety Injection" is the initial procedure used following initiation of the accident sequence. The first time-critical diagnostic steps are those related with the transition from E-0 to E-3 "Steam Generator Tube Rupture." Diagnostic methods used by E-0 include:

- Process Radiation Monitors
- Sampling and Laboratory Analysis
- Uncontrolled Increase in Narrow Range Level for any Steam Generator

Exclusive reliance on sampling and laboratory analysis would result in delaying operator response to a large SGTR such as the Licensing Bases case. During the simulator exercises discussed previously that demonstrated that all crews could achieve the new analysis times, the Licensed Operators based their E-0 to E-3 transition on process radiation monitors and behavior of steam generator narrow range level indication.

Identification of the steam generator with the ruptured tube occurs following the E-0 to E-3 transition. Step 2 of E-3 provides the procedural diagnostic guidance to identify the ruptured steam generator. Identification of the ruptured steam generator is based on:

Unexpected increase in any SG narrow range level

OR

High activity in any SG sample

OR

High radiation from any SG Steamline
(This step would require a local Health Physics Technician to perform radiation surveys)

OR

High activity in any SG blowdown line sample.

As discussed earlier, exclusive reliance on laboratory analysis or local surveys would result in delaying operator response to a large SGTR such as the Licensing Bases case. During the simulator exercises discussed previously, which demonstrated that all crews could achieve the new analysis times, the Licensed Operators identified the ruptured steam generator based on behavior of the narrow range level.

Identification of the ruptured steam generator is not a step specifically modeled in the analysis. Therefore, the timing of this identification is not firmly established. Identification of the ruptured steam generator would occur prior to the rapid cooldown step which is assumed to occur at 30 minutes.

All crews demonstrated their capability to achieve the new assumed time values using the diagnostic methods specified by Callaway EOPs E-0 and E-3.

II. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED AUGUST 29, 2003

NRC Question 1:

Was a failure modes and effects analysis, or similar analysis, performed for the MFIV actuator modification and the motor driven auxiliary feedwater (AFW) pump discharge check valve replacement with "ARC" valves? If so, discuss the results.

AmerenUE Response to Question 1:

A. FAILURE ANALYSIS FOR MFIV ACTUATOR REPLACEMENT

The safety related function of the MFIV actuator is to close an MFIV in less than or equal to 15 seconds. Each actuator has two actuation trains capable of performing this function. The licensing basis for an MFIV actuator is that a single failure of any active component cannot prevent the actuator from performing its safety function. The following failure analysis discusses the actuator operation considering all possible failures. Along with the discussion below, refer to Attachment 8 of the license amendment request submittal (ULNRC-04592) for the MFIV Actuator Diagram for the failure analysis of the system medium actuator.

- ***Failure of 'A' ('B') Train ESFAS or 'A' ('B') Train MSFIS to Actuate***

  If the 'A' ('B') train of ESFAS or MSFIS fails to actuate, the associated UPC solenoid valves MV1 and MV3 (MV2 and MV4) will remain in an energized state. The associated LPC solenoid valve MV5 (MV6) will remain in a de-energized state. The solenoid valves in the opposite train, MV2 and MV4 (MV1 and MV3), will still de-energize, directing feedwater to the UPC. Both LPC solenoids will remain in a de-energized state (vented position) until the MFIV is closed. After a 60 second time delay, the actuated train LPC solenoid valve MV6 (MV5) will go to an energized state (closed or pressurized position). Under these conditions, feedwater will be vented through MV1 (MV2), while the opposite train solenoids, MV2 and MV4 (MV1 and MV3) will route feedwater to the UPC. The exhaust port from MV1 (MV2), however, is equipped with a 2 mm orifice sized to limit exhaust flow to an acceptable level. In addition, feedwater will be vented through either MV5 (MV6), whichever solenoid valve remains de-energized after 60 seconds. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 78 lbm/min (~10 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids.

- ***Inadvertent Actuation of 'A' ('B') Train ESFAS or 'A' ('B') Train MSFIS***

  If the 'A' ('B') train of ESFAS or MSFIS inadvertently actuates, the associated UPC solenoid valves MV1 and MV3 (MV2 and MV4) will go to a de-energized state. The associated LPC solenoid valve MV5 (MV6) will remain in a de-energized state. The solenoid valves in the opposite train, MV2 and MV4 (MV1 and MV3), will remain energized, and the opposite train LPC solenoid valve, MV6 (MV5) will remain de-energized. The de-energized UPC solenoid valves, MV1 and MV3 (MV2 and MV4), will direct feedwater to the UPC. The de-energized LPC solenoid valves MV5 and MV6 will remain in a de-energized condition (vented position) until the MFIV is closed. After a 60 second time delay, the actuated train LPC solenoid valve MV5 (MV6) will go to an energized state (closed or pressurized position). Under these conditions, feedwater will be vented through MV2 (MV1), while the actuated train solenoids, MV1 and MV3 (MV2 and MV4) will route feedwater to the UPC. The exhaust port from MV2 (MV1), however, is equipped with an orifice sized to limit exhaust flow to an acceptable level. In addition, feedwater will be vented through either MV5 (MV6), whichever solenoid valve remains de-energized after 60 seconds. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 78 lbm/min (~10 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids.

- ***Solenoid MV1 (MV2) Fails in the Energized State (Vented Position)***

  MV1 (MV2) is a three-way solenoid valve, which vents the UPC when energized and directs feedwater to the UPC when de-energized. Should the MFIV receive a close signal and MV1 (MV2) fails to de-energize, MV1 (MV2) will remain in the vented position. Under these conditions, feedwater will be vented through MV1 (MV2), while the opposite train solenoids, MV2 and MV4 (MV1 and MV3) will route feedwater to the UPC. Since the exhaust ports are connected, a portion of the feedwater from the opposite train will vent through MV1 (MV2). The exhaust port from MV1 (MV2), however, is equipped with an orifice sized to limit exhaust flow to an acceptable level. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 45 lbm/min (~6 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- ***Solenoid MV1 (MV2) Fails in the De-energized State (Pressurized Position)***

  This is the safe position, and will not adversely impact the ability of the actuator to close the MFIV in the required 15 seconds. If the MFIV is open and MV1 (MV2) de-energizes, feedwater will still be isolated by MV3 (MV4). Therefore, this single failure will not prevent the MFIV from closing in the required time or cause the MFIV to close creating a Reactor Trip.

- ***Solenoid MV3 (MV4) Fails in the Energized State (Closed Position)***

  MV3 (MV4) is a two-way solenoid valve, which isolates feedwater from the inlet to MV1 (MV2) when energized and directs feedwater to MV1 (MV2) when de-energized. Should the MFIV receive a close signal and MV3 (MV4) fails to de-energize, feedwater will still be directed to the UPC through the opposite train, MV2 and MV4 (MV1 and MV3). Therefore, this single failure will not prevent the MFIV from closing in 15 seconds or cause the MFIV to close creating a Reactor Trip.

- ***Solenoid MV3 (MV4) Fails in the De-energized State (Open Position)***

  This is the safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV3 (MV4) de-energizes, feedwater will still be isolated from the UPC by MV1 (MV2). Therefore, this single failure will not prevent the MFIV from closing or cause the MFIV to close creating a Reactor Trip.

- ***Solenoid MV5 Fails in the Energized State (Closed Position)***

  MV5 is a two-way solenoid valve, which isolates the LPC vent path when energized and provides a vent path for the LPC when de-energized. Should the MFIV receive a close signal and MV5 fails in the energized state (closed position), the LPC will still be vented through the opposite train, MV6. The MFIV will still close in the required 15 seconds with MV5 in the energized state (closed position).

- ***Solenoid MV5 Fails in the De-energized State (Open Position)***

  This is the primary safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV5 fails in the de-energized state (open position), the LPC will be vented through both MV5 and MV6. Therefore, this single failure will not prevent the MFIV from closing in the required time frame. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that up to 33 lbm/min (~4 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- ***Solenoid MV6 Fails in the Energized State (Pressurized Position)***

  MV6 is a three-way solenoid valve, which directs feedwater to the LPC when energized and provides a vent path for the LPC when de-energized. Should the MFIV receive a close signal and MV6 fails in the energized state (pressurized position), feedwater will be directed to the LPC if the system is pressurized. The LPC will still be vented through the opposite train solenoid, MV5. The inlet port to MV6, however, is equipped with an orifice sized to limit inlet flow to an acceptable level. The MFIV will still close in the required 15 seconds with MV6 in the energized state (pressurized position).

- ***Solenoid MV6 Fails in the De-energized State (Vented Position)***

  This is the primary safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV6 fails in the de-energized state (vented position), the LPC will be vented through both MV5 and MV6. Therefore, this single failure will not prevent the MFIV from closing in the required time frame. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that up to 33 lbm/min (~4 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- ***Loss of Lower Piston Chamber Vent Path***

  In order to perform the safety function of closing the MFIV in the required 15 seconds, the lower piston chamber must be vented. This is accomplished by providing two redundant vent paths through two LPC solenoid valves, MV5 and MV6, which are then tied to a common header. Two parallel vent paths are then provided from each MFIV vent header. The normal and preferred vent path is back to the condenser. A redundant vent path to a rupture disk, which discharges to an equipment/floor drain in Area 5, is also provided. If the non-safety-related path fails and the MFIV receives a close signal, once the LPC becomes pressurized, the rupture disk will break providing a vent path, ensuring the MFIV will close in the required 15 seconds.

- ***Dual Electrical Train Failure***

  In this design the separation of trains is on the order of one half inch where the wiring comes together on the switches. However, this design complies with the regulatory requirements by providing an insulating barrier for separation by using switches that have been qualified, and by providing a failure modes and effects analysis.

The separation barrier medium is a high temperature ceramic based insulating material. It will not burn. The published operating temperature is 2200 F for extended periods. Further, the sleeving provides electrical insulating properties of high electrical resistance at elevated temperatures, low shrinkage and low moisture absorption characteristics for an excellent electrical insulator.

The power supplies that feed the remote contacts for MSFIS are an ungrounded 48 VDC supply.

- ***Failures Related to Fast Close Switches***

Short to ground of all conductors – This scenario will not cause a power supply failure, since the power supplies are floating with respect to ground. It is possible that a fast close would be initiated, but that is the safeguard position of the valve and conservative.

Short together of all conductors – Same result as explained above.

Fire inside of switch cubicle – This scenario is not credible, since there are no heat sources. Shorting the 48 volt power supply to ground will not draw any currents, because the power supply is floating. Further, the fast close inputs to MSFIS only draw about 20 ma, which is about 1 watt of power dissipated in the MSFIS cabinets. This power is not enough heat to postulate a fire hazard at the switches.

Fire outside of cubicle – This scenario is the same threat as prior to the modification.

Switch breaks off – If the switch breaks off, the only possible outcome would be to cause a fast close on one or both trains of MSFIS, which is the safeguard position.

- ***Failures Related to Open/Close Switches***

Fire in hand switch resistor deck – It is conceivable to have a fire relating to the indicating lights on these switches. In this case the insulation on the opposite train wiring is protected by the high temperature sleeving. Even if the insulation did melt, the sleeving would provide the required electrical separation.

In the case of hypothesizing a complete failure of components, see the following discussion on shorting to ground.

Short to ground – Since the 48 volt power supplies of both trains are floating, there would be no safety consequence of shorting all wires on the switch together and to ground. An open command or a normal close command could result. However, these switches do not have a safety function; the safety function is provided by automatic actuation signals and the manual Fast Close switches. The safety functions override the logic and inputs from the switch.

Switch breaks off – The only possible scenario would be to create an open or close signal to one or both trains of a single valve. This switch has no safety function. The safety functions always override any inputs from this switch.

- ***Common Mode Software Failure (CMSF)***

Actuation control for the MFIVs is accomplished by the Mainsteam and Feedwater Isolation System (MSFIS), which is a Programmable Logic Controller (PLC)-based digital control system. Although the existing software will remain largely unmodified, one new module will be added to perform the new MFIV actuation logic. A common mode software failure (CMSF) could exist if both trains of PLCs have a simultaneous software malfunction and /or fault. As stated in the Safety Evaluation for Callaway License Amendment 117 (dated October 1, 1996), the possibility of a CMSF is reduced to a very low probability due to the high quality established throughout the software design process. Based on the simplicity of the new actuator design and the extent of the V&V performed on the new software module by the Developer, common mode software failures are no more likely with the new design than they were with the existing design.

Failure Occurs During Normal Operation							
Logic Output Failure State			*Solenoid Output Failure State*				
A	**B**	**C**	**A**	**B**	**C**	**Single Train Outcome**	**Common Mode Outcome**
0	0	0	1	1	1	Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	0	1	1	1	0	Sol A&B continue to isolate fluid to the upper piston chamber (UPC). The lower piston chamber (LPC) is vented through Sol C.. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	1	0	1	0	1	Sol A continues to isolate fluid to the UPC. Sol C energizes to block the lower piston vent path. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	1	1	1	0	0	Sol A continues to isolate fluid to the UPC. Sol C deenergizes to vent the LPC. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
1	0	0	0	1	1	Sol B continues to isolate fluid to the UPC. Sol C energizes to block the lower piston vent path. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
1	0	1	0	1	0	Sol B continues to isolate fluid to the UPC. Sol C deenergizes to vent the LPC. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above

Notes: 1) For the purposes of this evaluation, it is assumed that the PLC's fail in the output state shown. A logic output of '1' energizes the actuation relay and '0' deenergizes the actuation relay. A & B solenoids are energized when their actuation relays are deenergized, and C solenoid is energized with an energized actuation relay.

2) Solenoid A corresponds to MV1 or MV2, Solenoid B corresponds to MV3 or MV4 and Solenoid C corresponds to MV5 or MV6.

Failure Occurs During Normal Operation							
Logic Output Failure State			***Solenoid Output Failure State***				
A	**B**	**C**	**A**	**B**	**C**	**Single Train Outcome**	**Common Mode Outcome**
1	1	0	0	0	1	Sol A&B deenergizes to direct process fluid to the (UPC). Sol C energizes to block the lower piston vent path. MFIV closes valve within 15 seconds.	See discussion above
1	1	1	0	0	0	The deenergized Sol A&B direct process fluid to the UPC. Sol C vents the LPC. MFIV closes within 15 seconds.	See discussion above

Notes: 1) For the purposes of this evaluation, it is assumed that the PLC's fail in the output state shown. A logic output of '1' energizes the actuation relay and '0' deenergizes the actuation relay. A & B solenoids are energized when their actuation relays are eenergized, and C solenoid is energized with an energized actuation relay.
2) Solenoid A corresponds to MV1 or MV2, Solenoid B corresponds to MV3 or MV4 and Solenoid C corresponds to MV5 or MV6.

B. FAILURE ANALYSIS FOR AFW PUMP DISCHARGE "ARC" VALVE REPLACEMENT

The failure modes and effects analysis (FMEA) was considered for the modification to replace the MDAFP Discharge Check Valve with the automatic recirculation control check (ARC) valve. In summary, the only change between the existing system with a swing style check valve plus the recirculation line orifice and the replacement ARC valve is the interaction between the two performed by the armature in the replacement ARC valve. In the unlikely event that the armature that connects recirculation flow control and the lifting check disc were to fail (break), the result would be that the recirculation line of the ARC valve would fail open. When the valve fails to the open position, the system returns to the current configuration installed at Callaway, with flow through the recirculation line being continuous, but restricted by the bypass pressure reducer orifice of the ARC valve.

NRC Question 2:

Discuss if a MFIV will be able to close in the event of a feed water line break (FWLB). Address if the new MFIV actuators operate on system pressure and the possibility for a large FWLB to cause a loss of system pressure rapidly enough so that the associated MFIV does not have enough system pressure to close. The closed MFIV is needed to act as a pressure boundary for AFW injection.

AmerenUE Response to Question 2:

Feedwater isolation valve closure delays are not explicitly modeled in the loss of normal feedwater (LONF) or loss of AC power (LOAC) analyses. The assumed time for AFW delivery to the steam generators, which accounts for system actuation and piping purge delays, implies feedwater isolation since MFIV closure provides the pressure boundary for AFW injection into the steam generators. With regard to the feedwater line break (FWLB) analysis, a MFIV closure time of 68.2 seconds is currently listed in FSAR Table 15.2-1. This 68.2 seconds is being increased by 10 seconds in this amendment. However, the FWLB analysis is performed in a fashion similar to the LONF/LOAC analyses. As long as MFIV closure occurs within the assumed 60-second AFW actuation delay time, the results of these primary side heatup analyses are not impacted.

A FWLB could potentially result in a rapid secondary side depressurization down to a containment pressure greater than 0 psig. Following assembly of each MFIV system medium actuator, a hot functional test was performed using Callaway's spare MFIV body. Through this testing it was found the MFIV would close within 60 seconds with as little as 0 psig of secondary side system pressure. Therefore, the MFIVs will always close within 60 seconds in response to any LONF/LOAC or FWLB event.

The increased MFIV stroke time (15 seconds) has been evaluated for the MSLB core response and containment P/T analyses with no resultant impact on the conclusions of those analyses. For the MSLB core response analyses, main feedwater isolation is credited to limit the cooldown of the RCS. For the MSLB containment analyses, main feedwater isolation is credited to limit the main feedwater mass and energy release to containment. In both cases, it is conservative to assume main feedwater flow is maintained until feedwater isolation occurs. All conditions where main feedwater flow is maintained to the steam generators result in secondary side system pressures well in excess of 90 psig. Following assembly of each MFIV system medium actuator, a hot functional test was performed using Callaway's spare MFIV body. Through this testing it was found the MFIV would close within 15 seconds with as little as 90 psig of secondary side system pressure, using cold water (< 250°F) in the system. This testing also found the MFIV actuators would stroke much quicker when hot water (> 250°F) was used. Therefore, the FWIVs will always close within 15 seconds in response to any MSLB event.

III. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED AUGUST 29, 2003

The request for additional information is for the license amendment request in the licensee's application dated June 27, 2003, on the re-analysis of the steam generator tube rupture (SGTR) with overfill event. The following information is needed for the staff to determine that the analyzed radiological consequences of design basis accidents (DBAs) at Callaway, as modified by the proposed changes, meets regulatory requirements.

As explained in Regulatory Information Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," the NRC staff bases its finding on the acceptability of an amendment on its assessment of the licensee's analysis. For the NRC staff to make an acceptable finding, the licensee must provide adequate information regarding analysis assumptions, inputs, and methods in the submittal. If any of the requested information was previously docketed for Callaway, the licensee is requested to provide the specific citation in its response.

<u>NRC Question 1</u>:

The licensee did not address the impact of the proposed changes on the ability of the control room habitability systems to maintain doses to operators within the criteria of 10 CFR Part 50, Appendix A, General Design Criteria 19. The staff notes that the analysis of record overfill case thyroid dose (pre-incident spike) at the EAB was 24 rem and that this had increased to 46 rem in the new overfill case. Given the significant increase in the EAB dose, the staff suspects that the control room dose would have similarly increased. Respond to Question a or b below as applicable.

a. Provide a description of the assumptions, inputs, methods and results of the evaluation that demonstrates that GDC-19 will continue to be met.

b. If the licensee has not evaluated the control room dose but is relying on the dose being bounded by that determined for another accident, provide a justification that addresses the considerations in Paragraphs 7a through 7d of RIS 2001-19, as applicable.

In either case, please provide the information requested in item 1.a of Generic Letter 2003-01, as it applies to the SGTR with overfill, in your response. If you have already docketed your response to GL 2003-01, provide a citation to that response.

<u>AmerenUE Response to Question 1</u>:

The submitted changes to the SGTR Overfill analysis do not adversely affect Callaway's Licensing Bases Control Room radiological consequences.

Callaway's Licensing Bases is that LOCA provides the limiting radiological consequence to Control Room personnel. The only Control Room radiological consequences reported in

Callaway's FSAR are for the LOCA sequence. As a result, the analysis efforts were directed towards identifying the case that produces maximum offsite consequences.

Several offsite dose cases were performed by Westinghouse to identify the case that produces maximum offsite dose. The maximum offsite dose case is based on SI at initiation of the accident sequence. If SI occurs at initiation of the accident sequence, then the assumptions used in the LOCA analysis regarding the time of Control Room isolation are valid for the SGTR Overfill with SI at accident initiation. For this case, SGTR Overfill Control Room consequences are bounded by the FSAR reported value for LOCA.

The delayed SI cases analyzed by Westinghouse determined that SI would occur at approximately 6 minutes into the accident sequence. This is prior to the start of relief from the ruptured steam generator. Relief begins at approximately 11 minutes. Therefore, for the purposes of Control Room radiological consequences analyses, it is valid to assume that the Control Room would be isolated prior to the post-accident release of radioactivity to the environment.

Appendix F of SLNRC 86-01 describes the radiological methods used to calculate radioactivity releases to the atmosphere and offsite doses for SGTR events. One of the assumptions specified is that 100% of the iodine contained in the fraction of the break flow to the faulted SG that flashes upon reaching the secondary side is conservatively included even when the RETRAN analysis shows that no steam is released from the secondary side atmospheric steam dumps or safety valves. This assumption was intended to conservatively maximize offsite doses. It was not intended to imply that radioactivity release via the flashing pathway would occur prior to Control Room isolation.

The SGTR-Overfill release rates are less than those found in the LOCA analysis. The Control Room would be isolated prior to the initiation of release. Therefore, it is our evaluation that the SGTR-Overfill would not adversely affect the Licensing Bases post-accident Control Room consequences analysis contained in Callaway's FSAR.

With regards to items 7a through 7d of RIS 2001-19:

a. The control room design is often optimized for the DBA LOCA, and the protection afforded for other accident sequences may not be as advantageous. For example, in most designs, control room isolation is actuated by engineered safety feature (ESF) signals such as containment high pressure or safety injection (SI), or radiation monitors, or both. For accidents that rely on radiation monitor actuation, there may be a time delay in isolation that would not occur for the immediate SI signal that would result from a LOCA. In such cases, contaminated air would enter the control room for a longer period preceding isolation than it would for a LOCA.

AmerenUE Response to RIS 2001-19 Item a:

This was discussed in the above paragraphs.

a. The configuration of radiation monitors has an impact on their sensitivity. Ideally, the radiation monitors would be located outside in air ventilation intake ductwork. However, there are system designs that place the radiation monitor in recirculation ductwork or downstream of filters. There are also designs that use area radiation monitors. In these latter designs, the contaminated air continues to build up in the control room volume until the concentration is large enough to actuate the radiation monitor.

AmerenUE Response to RIS 2001-19 Item b:

GK-RE-04/05 located in the intake ductwork. They are not downstream of filters or located in recirculation ductwork.

a. In some cases, control room radiation monitor setpoints may have been based on external exposure concerns, for example, 2.5 mrem/hour, rather than thyroid dose from inhalation. The airborne concentration of radioiodines will likely cause elevated thyroid doses before reaching the concentration of all radionuclides necessary to alarm the monitor. This condition is typically seen with accidents that involve a high iodine-to-noble-gas ratio, such as main steam line breaks in PWRs.

AmerenUE Response to RIS 2001-19 Item c:

These radiation monitors are not relied on. Control Room isolation is initiated prior to the initiation of the release of radioactivity.

a. The distance between the control room and the release point, and the associated wind sectors, may be different for each postulated accident. These differences are usually not significant with regard to offsite doses, but may be significant for control room assessments because of the shorter distances typically involved. The X/Q for the DBA LOCA may not be applicable to other DBAs. A ground-level release associated with a non-LOCA event may be more limiting than the elevated release associated with LOCAs at plants with secondary containments or enclosure buildings.

AmerenUE Response to RIS 2001-19 Item d:

The Callaway FSAR does not include Control Room X/Q values for the main steam safety valves. However, we believe that use of the Reactor Building X/Q is an appropriate alternative. The Reactor Building X/Q does not credit any elevation. It is based on ground-level release assumptions.

<u>NRC Question 2</u>:

Figures 15.6-3P, 15.6-3.2d, and 15.6-3.2h of the submittal provide data for the intact steam generators (SGs). Discuss if the data represents each intact SG or the total for all intact SGs.

<u>AmerenUE Response to Question 2</u>:

The intact SG figures represent a lumped SG that is representative of the 3 intact SGs.

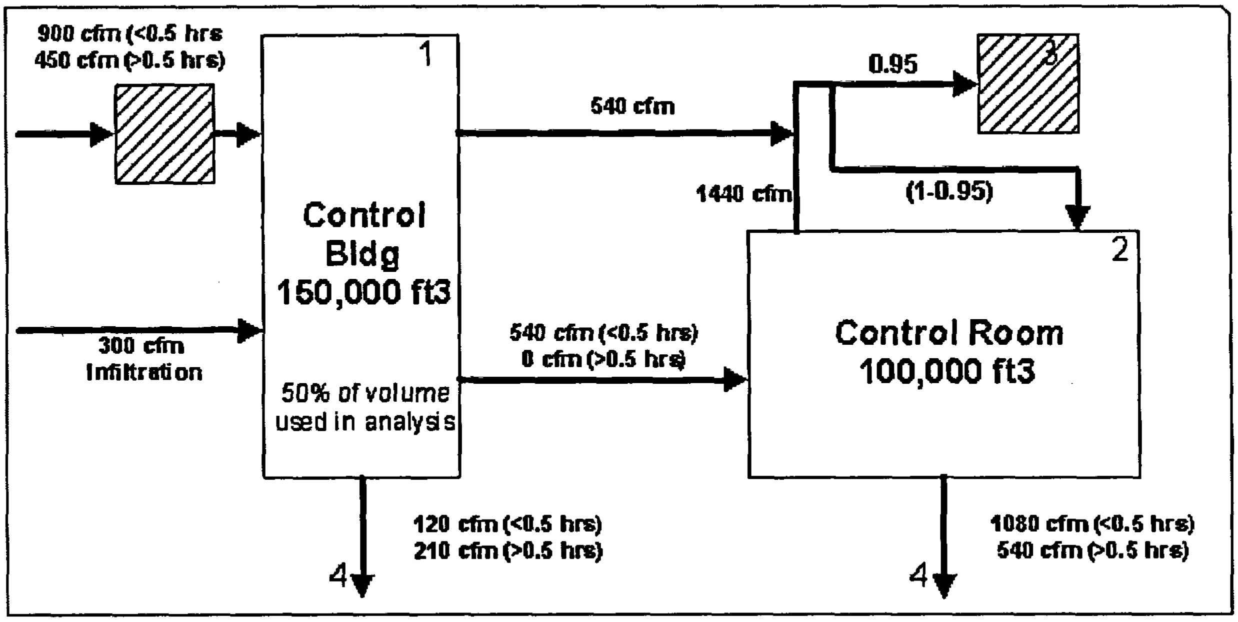
<u>NRC Question 3</u>:

The figure below represents the staff's interpretation of appropriate modeling of your control building and control room. Node 3 is the recirculation filter; node 4 is a "sink." Discuss the staff's interpretation and confirm if the staff's understanding is correct. In particular, discuss if the expected re-alignment will occur at 30 minutes for this event and provide the basis for this conclusion.

In addition, respond to the following questions transmitted to AmerenUE on November 18, 2003. AmerenUE provided an alternative figure from the FSAR for the control room/control building/environment flow interactions. Although the response identifies two flow paths not shown on the staff's figure, insufficient data is provided for the staff to determine the flow rate in these two paths.

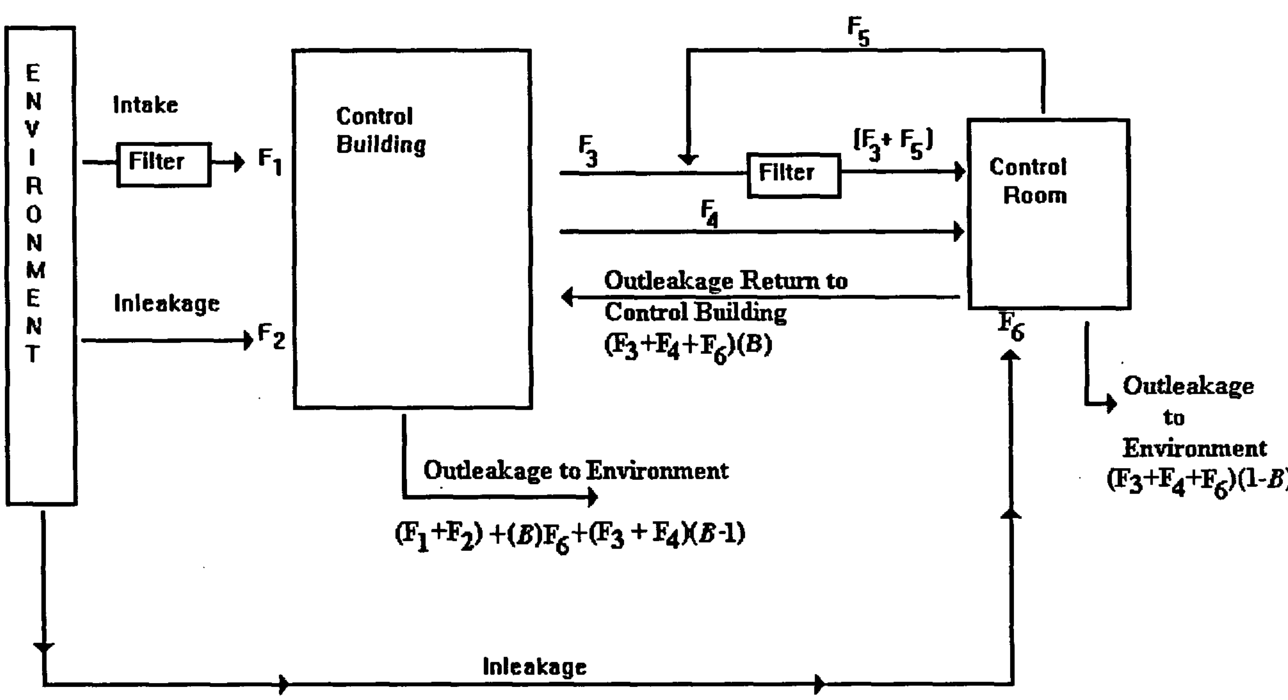
(1) What is the value of the direct inleakage to the control room (F6)? Is this equal to F2?

(2) What is the value of the constant "B" shown in the figure for determining outleakage from the control room returning to the control building? This constant is identified as the fraction of outleakage that returns to the control building. What is the constant based upon? Does the same value apply to the other two outleakage expressions as well?



AmerenUE Response to Question 3:

The staff's flow diagram does not match the one used by AmerenUE. The following diagram is taken from Section 15A of the Callaway FSAR:



The diagram has two flow paths not shown on the staff's diagram. The first is direct inleakage from the environment to the Control Room. The second represents Control Room outleakage returning to the Control Building.

Additionally, the flow values for the F_3, F_4, and F_5 flow paths shown in the staff's diagram are incorrect. These values are as follows:

F_3	440 cfm
F_4	440 cfm
F_5	1360 cfm

These values are based on the following:

The Control Room makeup flow from the Control Building is established at 400 cfm per train. The tolerance on this is +/- 10%. Higher flow values for the F_3 and F_4 flow paths produces more limiting results. Therefore, a value of 440 cfm is used in the analysis for these flow paths.

Total flow through the Control Room recirculation filter is limited to 2000 cfm. The tolerance on this is +/- 10%. Using the minimum value of 1800 cfm produces more limiting results. This 1800 cfm represents the sum of F_3 and F_5. Therefore, a value of 1360 is used for F_5. This minimizes the recirculation cleanup rate and provides conservative results.

The F6 value is assumed to be 10 cfm. This is different from the F2 value of 300 cfm. A 0.75 value is assumed for B. This is based on an engineering evaluation of Control Room sealing surfaces and the associated outleakage pathways. The B value is used to calculate Control Room outleakage to the environment, Control Room outleakage to the Control Building and Control Building outleakage to the environment.

<u>NRC Question 4</u>:

Discuss (1) the iodine appearance rates for I-131 to I-135, in Ci/hr, to which the multiplier of 335 will be applied and (2) the assumed duration of the accident induced spike.

<u>AmerenUE Response to Question 4</u>:

The iodine appearance rates are based on maximum allowable RCS DEI-131 levels and maximum letdown cleanup rates. The accident initiated spike is assumed to last for 8 hours.

<u>NRC Comment</u>:

In reviewing the licensee's application, the staff had the following comments on statements made by the licensee. First, with regard to the conclusion that 10 CFR 50.67 applied to this amendment, the staff believes that the requirements of 10 CFR 50.67 do not apply to this amendment request. This is based on the definition of "source term" in 10 CFR 50.2 and the statements of consideration for the final 10 CFR 50.67 rule (63 FR 71990 dated December 23, 1999). Second, the licensee has requested staff approval to use Regulatory Guide (RG) 1.195 for other licensing basis dose applications. The staff believes that 10 CFR 50.59 already provides the licensee with an adequate mechanism to implement the guidance of RG 1.195. The guide provides methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations; however, the guide also contains alternative methods that must be considered on a case-by-case basis. The staff considers blanket approval of the use of RG 1.195 would confer approval for each of these case-by-case situations, most of which are not considered relevant to the technical specification changes requested or to the re-analysis of the SGTR with overfill. If the licensee believes that the staff has misunderstood its rational in these two areas, the licensee should provide further explanation of its position.

AmerenUE Response to NRC Comment:

The AmerenUE discussion regarding the applicability of 10CFR50.59 to the Reg. Guide 1.195 methodologies was intended to reflect the pathway used in AmerenUE's evaluation regarding whether or not the 335 iodine spiking factor could be implemented without prior NRC approval. The 335 iodine spiking factor has been approved for use by another Licensee in their SGTR analysis. Part of our bases for concluding that implementation of the 335 spiking factor required prior NRC approval involved the position that source terms used in FSAR analyses are not exclusively regulated by 10CFR50.59.

AmerenUE is requesting approval for use of the 335 spiking factor in our SGTR analyses, and to use the ICRP-30 dose conversion factors, on a forward-fitting bases for all of our FSAR Chapter 15 radiological consequences analyses.

IV. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED SEPTEMBER 11, 2003

The following questions refer to the Main Steam Line Break Mass and Energy Release Analysis section of Attachment 2 to the June 27, 2003 letter concerning revision of Callaway TS Section 3.7.3.

NRC Question 1:

Given the increased MFIV closure time and the resulting longer steam generator dryout time for the MFW/AFW systems modifications, discuss the differences in analysis assumptions that result in the Bechtel containment analysis of containment pressure and temperature to remain bounding with respect to the CONTEMPT containment analysis.

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide a statement on which version of CONTEMPT was used to support the feedwater isolation valve modification.

AmerenUE Response to Question 1:

Two input assumptions have been revised since the original Bechtel containment analysis was performed. These are

- Condensate re-vaporization
- Time assumed for operator action to isolate Aux Feedwater to the affected steam generator

The original Bechtel analysis did not credit re-vaporization. Re-analysis first performed and incorporated into Callaway's FSAR as a part of Callaway's power uprating incorporated a credit of 8% re-vaporization, as allowed by NUREG 0588. The 8% revaporization is discussed in Section 6.2.1.4 of Callaway's FSAR.

The original Bechtel analysis assumed that operators did not isolate Aux Feedwater to the affected steam generator until 1800 seconds. However, the original FSAR Sections 10.4.9 (Auxiliary Feedwater System) and 15.0.13 (Operator Actions) stated that during a Main Steam Line Break, auxiliary feedwater to the faulted steam generator can be terminated within 10 minutes. FSAR Section 6.2.1.4.3.3 (Containment Pressure-Temperature Results) and Table 6.2.2-6a (Water Level Within the Reactor Building Following a MSLB) also state that termination of auxiliary feedwater can be accomplished in 10 minutes and that the 30-minute response time was used only to show conservatism in the Containment P/T Analysis. These statements remain in the Current FSAR for Callaway Plant.

The 10 minute isolation time was previously submitted for NRC review:

1. Bechtel letter BLSE-2422 provides discussion on "Secondary System Pipe Ruptures Inside Containment" as a proposed revision to PSAR Section 6.2.1.3.10. Section 6.2.1.3.10.4 (Auxiliary Feedwater) states that manual isolation of the auxiliary feedwater system is assumed at 600 seconds (10 minutes). This assumption was modeled into original containment analyses and incorporated into PSAR, Rev. 13.

2. The Auxiliary Feedwater System (FSAR Section 10.4.9) was submitted to the NRC for review against the Standard Review Plan, via letter SLNRC 81-39. The 10-minute operator action was included in Section 10.4.9.2.3.

The revised Containment analyses were performed using CONTEMPT LT-028.

NRC Question 2:

Describe the assumptions used in the CONTEMPT containment calculations. Discuss if the assumptions included the temperature flash assumption and if the Tagami and Uchida heat transfer correlations were used. Address what conservatisms are included in the calculation of structural heat sink areas and coatings, and in the value of the containment volume.

AmerenUE Response to Question 2:

The calculation of heat sink areas were reduced by either construction tolerances or a 10% factor for those items that did not have known tolerances. The containment volume listed in Callaway's FSAR of 2.5E6 ft^3 is the value used in the analysis. This value is approximately 4% lower than the calculated containment free volume. The Uchida heat transfer correlation is used in the analysis of MSLB cases. Callaway's limiting MSLB cases are insensitive to changes in the flashing modeling. This is because limiting pressure-temperature results are produced by split breaks which have no entrained moisture.

NRC Question 3:

Explain how the generic calculations of mass and energy release can be independent of feedwater (FW) assumptions and discuss if this is related to the discussion of Section 3.1.5 of WCAP 8822.

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide an explanation on steam generator dry-out and whether WCAP-8822 has been previously reviewed and approved by the NRC.

AmerenUE Response to Question 3:

The generic mass and energy release values provided by Westinghouse do not account for plant specific values. Plant specific values regarding feedwater assumptions were incorporated into the original Bechtel analysis as part of the steam generator dryout time calculation. This is the method discussed in Safety Analysis Standard 12.2, Section III.D. Safety Analysis Standard 12.2 is Appendix A of WCAP 8822. The revised main feedwater isolation time and auxiliary feedwater flowrate were used to calculate the new dryout time.

The re-analysis performed to support the longer main feedwater isolation times associated with the proposed modification did not involve any change to the currently FSAR-described methodologies regarding mass and energy release modeling.

FSAR Section 6.2.1.4.2, "Description of Blowdown Model" currently states:

A description of the blowdown model used is provided in Reference 6. This reference is the basis for the tables contained in Reference 7. WCAP 8822 and SNP 2035 are References 6 & 7, respectively of FSAR Section 6.2.1. WCAP 8822 describes a methodology for calculating post-accident mass and energy release. The Callaway mass and energy release values were obtained from a Westinghouse calculation note that was originally transmitted to Bechtel via SNP 2035. This calculation note was performed for Model F steam generators.

The Westinghouse mass and energy release values documented in SNP 2035, which were calculated using the methodology described in WCAP 8822 account for the initial mass of water contained within the affected steam generator. The Westinghouse methodology provides guidance that will generate the mass and energy release values for approximately the first 300 seconds of the accident sequence. Following this time, a dry-out calculation is performed by the containment analyst (originally Bechtel, now AmerenUE.) This is the portion of the transient where the mass of water associated with main feedwater and auxiliary feedwater are accounted for.

The proposed modification will not require any changes to the mass and energy release values listed in Tables 6.2.1-57A and 6.2.1-57B of Callaway's FSAR.

NRC Question 4:

It is stated that the peak containment pressure calculated by CONTEMPT remains below the Bechtel analysis for a steam line break with an increase in stroke time of the MFIVs and for an increase in the maximum auxiliary feedwater (AFW) flow due to the ARC valves when the cases are considered separately. This is stated in the last paragraph of the Main Steam Line Break Mass and Energy Release Analysis section (starting at page 21 of 33) of Attachment 2. Discuss and verify that when both effects are considered together the peak containment pressure remains below the Bechtel analysis for a steam line break.

<u>AmerenUE Response to Question 4</u>:

Both effects were considered and analyzed together. The original Bechtel Pressure-Temperature results remain bounding when the combined effects of the increased AFW flowrate and increased MFIV isolation time are considered together.

<u>NRC Question 5:</u>

This follow-on Question was transmitted to AmerenUE via e-mail on November 17, 2003: Topical Report WCAP 8822 states that the described methods are applicable to Model D steam generators, and, "with minor alterations," to Model 51 steam generators. However, Callaway uses Model F steam generators which the FSAR states are "similar in configuration" to the Model 51 steam generators. Describe the differences between the Callaway steam generators and the Model 51 steam generators and why the calculations of the topical report apply given those differences.

<u>AmerenUE Response to Question 5</u>:

The re-analysis performed to support the longer main feedwater isolation times associated with the proposed modification did not involve any change to the current FSAR-described methodologies regarding mass and energy release modeling.

FSAR Section 6.2.1.4.2, "Description of Blowdown Model" currently states:

A description of the blowdown model used is provided in Reference 6. This reference is the basis for the tables contained in Reference 7. WCAP 8822 and SNP 2035 are References 6 & 7, respectively of FSAR Section 6.2.1. WCAP 8822 describes a methodology for calculating post-accident mass and energy release. The Callaway mass and energy release values were obtained from a Westinghouse calculation note that was originally transmitted to Bechtel via SNP 2035. This calculation note was performed for Model F steam generators.

The Westinghouse mass and energy release values documented in SNP 2035, which were calculated using the methodology described in WCAP 8822 account for the initial mass of water contained within the affected steam generator. The Westinghouse methodology provides guidance that will generate the mass and energy release values for approximately the first 300 seconds of the accident sequence. Following this time, a dry-out calculation is performed by the containment analyst (originally Bechtel, now AmerenUE), where the mass of water associated with main feedwater and auxiliary feedwater are accounted for.

The proposed modification will not require any changes to the mass and energy release values listed in Tables 6.2.1-57A and 6.2.1-57B of Callaway's FSAR.

NRC Question 6:

This follow-on Question was transmitted to AmerenUE via e-mail on November 17, 2003: Verify that there were no changes to the containment calculation methods and input described in the FSAR and that used for the proposed license amendment in the June 27, 2003 letter. For example, Table 6.2.1-2 through Table 6.2.1-5.

AmerenUE Response to Question 6:

The mass and energy release values are calculated using the methodology in WCAP 8822. For the model F generators, calculations were performed using the WCAP methodology. The specific calculation used is CN-RPA-78-192, "Model F Steam Generator Mass and Energy Release Calculations." The results of this calculation were transmitted to Bechtel via letter SNP 2035. WCAP 8822 and SNP 2035 are References 6 & 7, respectively of FSAR Section 6.2.1.

FSAR Section 6.2.1.4.2, "Description of Blowdown Model" currently states:

A description of the blowdown model used is provided in Reference 6. This reference is the basis for the tables contained in Reference 7.

Therefore, it can be concluded that our current values are applicable to Model F steam generators, and that they were derived using the methodology currently described in the FSAR. The analysis performed for the proposed License Amendment was performed using FSAR described methodology, and the parameters in FSAR Tables 6.2.1-2 through 6.2.1-5 remain unaffected by the proposed change and our re-analysis.

V. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED OCTOBER 2, 2003

The following requests for information is based on the staff's review of the licensee's application submitted June 27, 2003 (ULNRC-04592) for the Callaway Plant:

<u>NRC Question 1</u>:

It is stated in the submittal that the SGTR re-analysis is performed with revised operator action times, inputs, and assumptions that are consistent with the current plant configuration and operation. Tabulate these changes and provide a justification for each change.

Based on the November 12, 2003 meeting between the NRC and AmerenUE, explain auxiliary feedwater (AFW) tables included in your response and provide a table of all the critical input parameters used in the steam generator tube rupture analysis. For each input parameter provided, state whether it either is the same as the original licensing bases assumption, more conservative than the original licensing bases assumption, or less conservative than the original licensing bases assumption. For those input parameters that are less conservative than the original licensing assumptions, provide bases for the changes.

<u>AmerenUE Response to Question 1</u>:

The following discussion provides a listing of inputs used in the new analysis. These inputs were validated to reflect current plant configuration.

The following new/revised assumptions are implemented in this new analysis.

- Initial conditions

- Vessel average temperature = 583.4°F - 5°F = 578.4°F. This reflects the Callaway Tavg band of 583.4°F - 588.4°F that is possible with a 5°F reduction in design Tavg.

- RCS flow = minimum measured flow = 382640 gpm. (prev. 374360 gpm)

- Steam generator pressure = 908 psia. (prev. 939 psia)

- Feedwater temperature = 390°F. (prev. 446°F)

- Decay heat model is changed to 80% of the 1979 ANS 2σ model used by transient analysis.

- The following AFW flow rates are modeled prior to partial/full isolation of AFW flow to the ruptured SG, to approximate the flows measured in the field:

- AFW flow to the ruptured SG before isolation of turbine driven AFW pump flow to the ruptured SG, at the intact SG pressure of 1235.7 psia, is used as a base. As the intact SG pressure drops the flow to the ruptured SG is reduced. This reduction is larger for higher ruptured SG pressures. A conservative model for this reduction is used based on the lowest ruptured SG pressure of 414.7 psia. This model is reflected in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)		Intact SG Pressure (psia)	Reduction in AFW to Ruptured SG (gpm)
414.7	1317.0		414.7	72.6
614.7	1214.0		614.7	55.4
814.7	1104.0		814.7	37.8
1014.7	982.0		1014.7	20.0
1139.7	895.0		1139.7	8.6
1235.7	823.0		1235.7	0.0

- AFW flow to intact SGs (total for the 3) before isolation of turbine driven AFW pump flow to the ruptured SG, at ruptured SG pressure of 1139.7 psia, is used. As the ruptured SG pressure drops the flow to the intact SGs is reduced. This reduction is small and is neglected:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1691.0
414.7	1576.0
614.7	1455.0
814.7	1326.0
1014.7	1186.0
1139.7	1091.0
1235.7	1013.0

- The following AFW flow rates are modeled to reflect the flow network analysis results after partial/full isolation of AFW flow to the ruptured SG:

- AFW flow to ruptured SG after isolation of turbine driven AFW pump flow to the ruptured SG is provided in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)
414.7	770.
614.7	712.
814.7	651.
1014.7	586.
1139.7	537.
1235.7	498.

- AFW flow to intact SGs (total for the 3) after isolation of turbine driven AFW pump flow to the ruptured steam generator, and after complete isolation of AFW to the ruptured SG is provided in the table below:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1760.
414.7	1656.
614.7	1546.
814.7	1425.
1014.7	1295.
1139.7	1205.
1235.7	1129.

- AFW temperature = 32°F. (prev. 70°F)
- AFW flow is initiated 5 seconds after reactor trip (to bound the field tested 12 second delay), with a 30-second ramp up to full flow.
- SI/charging flow temperature = 37°F. However, due to stability concerns the charging flow is modeled at 41°F. (prev. 50°F)
- SI action is initiated coincident with reactor trip. This was determined to provide limiting results relative to the scenario with SI actuation on low pressurizer pressure.
- MFIV isolation is initiated by the SI signal. This was determined to provide limiting results relative to the scenario with MFW isolation delayed until a high SG level signal was generated.
- MFIV closure is modeled as a step function with a 17 second delay.
- Operator actions modeled
    - Isolation of turbine driven AFW flow to the ruptured SG at 10 minutes from the start of the event.

- Isolation of all AFW flow to the ruptured SG at 20 minutes from the start of the event.

- Initiate cooldown by dumping steam from the lumped intact loop SG ARV after 30 minutes from reactor trip (which is at the start of the event).

- The cooldown is terminated when the core outlet temperature reaches the target temperature specified in the EOPs as a function of the ruptured SG pressure. The current EOP cooldown target temperature table (assuming normal containment and RCS temperatures per incore thermocouples) is:

SG Pressure (psig)	Target Temperature (°F)
> 1200	535
1100 - 1199	524
1000 - 999	512
900 – 999	499
800 – 899	485
700 – 799	470
600 – 699	450
500 – 599	430
430 - 499	410

- Initiate depressurization by dumping steam using pressurizer relief valve at a time such that the depressurization will be completed at 40 minutes from the start of the event.

- SI flow is terminated 5 minutes after the depressurization is completed. This is approximately 45 minutes from the start of the event.

- Depressurize by dumping steam using pressurizer relief valve following SI termination to terminate break flow at 60 minutes from the start of the event.

- Cooldown to RHR cut in is initiated after break flow is terminated.

- The break flow model is adjusted to model all flow at the enthalpy of the ruptured steam generator outlet header.

- The break flow flashing fraction is calculated assuming all break flow is at the ruptured loop hot leg temperature.

- The initial conditions are based on 10% steam generator tube plugging. However, the SG heat transfer model in RETRAN is based on 15% tube plugging and this is conservatively retained. A conservatively high initial secondary mass is assumed to bound 0% tube plugging.

- MSIV isolation on reactor trip and the assumed loss of offsite power has not been changed, although it could be significantly delayed based on the expected operator response. Early isolation of the MSIV allows the ruptured SG to depressurize due to the addition of the (maximum) AFW flow, while the intact SG pressure stays relatively high. This results in increased break flow to the ruptured SG, which is conservative. It also leads to higher AFW flow to the ruptured SG. If the MSIV would be left open, the SGs would tend to be at the same pressure, which would be closer to that of the intact SGs (which are lumped together in the RETRAN model). Also, with the MSIV open, overfilling the ruptured SG would not necessarily lead to water relief, since the water could go to the intact SG. The secondary pressure would not spike and the safety valve would not lift.

The time to close the MSIV is left at 1.5 seconds. As noted above early isolation is considered to be more limiting.

The above tables provide the auxiliary feedwater flowrate for three time periods of interest addressed in the analysis:

- The initial stage when the ruptured and intact steam generators are being fed by all three AFW pumps, and the MDAFP flow control valve for the ruptured steam generator is failed wide open.

- The second stage during which TDAFP flow to the ruptured steam generator has been isolated, but MDAFP flow continues.

- The third stage when the ruptured steam generator has been isolated, and feed to the intact steam generators is a function of pressure of the intact steam generators.

The following Tables provide all of the critical input parameters used in the steam generator tube rupture analysis along with comparisons to the parameters used in the original licensing bases assumptions.

CALLAWAY SGTR w/OVERFILL ANALYSIS Critical Input Parameters & Assumptions			
Parameter	Previous Value	New Value	Conservative? Less/More/Same
Most Limiting Single Failure	AFW Control Valve	AFW Control Valve	Same
Additional Active Failure	Ruptured S/G Safety Valve Sticks open 5%	Ruptured S/G Safety Valve Sticks open 5%	Same
Ruptured S/G ARV	Not operable	Not operable	Same
Tube rupture	Tube rupture, double-ended-guillotine at the cold leg tube sheet w/5% uncertainty.	Same	Same
Core Power	3565 MW	3565 MW	Same
PZR Level	65%	65%	Same
PZR Pressure	2280 psia	2280 psia	Same
RCS Average Temperature	583.4° F	578.4° F	More – Results in higher break flow
RCS Flow	374360 gpm	382640 gpm	More – Results in higher early stage break flow
S/G Steam Pressure	939 psia	908 psia	More – Results in higher break flow
Feedwater Temperature	446° F	390° F	Same – Little impact on S/G pressure
S/G Level	55 % NRS	55 % NRS	Same
S/G Tube Plugging	15%	15%	Same
Reactor trip time	T = 0	T = 0	Same
Loss of offsite power time	T = 0	T = 0	Same
MSIV isolation time	T = 0	T = 0	Same
MSIV stroke time	T = 1.5 s	T = 1.5 s	Same
MFIV isolation delay time	T = 5 s	T = 17 s	Plant modification, new value more limiting due to higher overfill mass
Decay heat model	0.8 × 1971 ANS model	0.8 × 1979 (2σ) ANS	More – higher decay heat model increase break flow

<table>
<tr><td>Ruptured S/G AFW flow (prior to securing TDAFP)</td><td><table><tr><th>S/G Press (psia)</th><th>AFW Flow (gpm)</th></tr><tr><td>700</td><td>995.5</td></tr><tr><td>800</td><td>931.1</td></tr><tr><td>900</td><td>862.2</td></tr><tr><td>1000</td><td>787.9</td></tr><tr><td>1125</td><td>684.8</td></tr></table></td><td><table><tr><th>S/G Press (psia)</th><th>AFW Flow (gpm)</th></tr><tr><td>414.7</td><td>1317.0</td></tr><tr><td>614.7</td><td>1214.0</td></tr><tr><td>814.7</td><td>1104.0</td></tr><tr><td>1014.7</td><td>982.0</td></tr><tr><td>1139.7</td><td>895.0</td></tr></table></td><td>More – higher AFW flow results in overfill sooner (new flows based on field testing data to develop max. performance pump curve & flow network modeling)</td></tr>
<tr><td>Ruptured S/G AFW flow (after securing TDAFP)</td><td>Not modeled, flow rates in above table assumed until isolation at T = 16 m</td><td><table><tr><th>S/G Press (psia)</th><th>AFW Flow (gpm)</th></tr><tr><td>414.7</td><td>769.3</td></tr><tr><td>614.7</td><td>711.6</td></tr><tr><td>814.7</td><td>651.0</td></tr><tr><td>1014.7</td><td>585.2</td></tr><tr><td>1139.7</td><td>536.6</td></tr></table></td><td>No significant change, lower flow rates are offset by longer time to isolate AFW, now T = 20 m</td></tr>
<tr><td>Intact S/G AFW flow</td><td>Constant 900 gpm total modeled</td><td><table><tr><th>S/G Press (psia)</th><th>AFW Flow (gpm)</th></tr><tr><td>1125</td><td>1205.2</td></tr><tr><td>1000</td><td>1295.4</td></tr><tr><td>800</td><td>1425.9</td></tr><tr><td>600</td><td>1546.0</td></tr><tr><td>400</td><td>1656.0</td></tr></table></td><td>More Accurate – little impact on overfill mass or break flow rate</td></tr>
<tr><td>AFW flow initiation</td><td>30-s delay to full flow</td><td>T = 5 s, 30-s total ramp up to full flow</td><td>More Accurate</td></tr>
<tr><td>Safety Injection initiation</td><td>15 seconds after SI signal at T = 0</td><td>15 seconds after SI signal at T = 0</td><td>Same</td></tr>
<tr><td>Intact S/G ARVs</td><td>Available for cooldown</td><td>Available for cooldown</td><td>Same</td></tr>
<tr><td>Operator actions modeled</td><td><table><tr><th>Description</th><th>Time</th></tr><tr><td>AFW flow to the ruptured S/G isolated</td><td>16</td></tr><tr><td>Initiate RCS cooldown</td><td>24</td></tr><tr><td>Complete RCS depressurization</td><td>35</td></tr><tr><td>SI terminated</td><td>38</td></tr><tr><td>RCS – S/G pressure equalized</td><td>43</td></tr></table></td><td><table><tr><th>Description</th><th>Time</th></tr><tr><td>AFW flow to the ruptured S/G isolated</td><td>20</td></tr><tr><td>Initiate RCS cooldown</td><td>30</td></tr><tr><td>Complete RCS depressurization</td><td>40</td></tr><tr><td>SI terminated</td><td>45</td></tr><tr><td>RCS – S/G pressure equalized</td><td>60</td></tr></table></td><td>More – longer times result in more overfill mass</td></tr>
<tr><td>Flashing Fraction</td><td>Hot Leg enthalpy assumed at the break</td><td>Hot Leg enthalpy assumed at the break</td><td>Same</td></tr>
<tr><td>S/G secondary side volume</td><td>5825 ft^3</td><td>5825 ft^3</td><td>Same</td></tr>
<tr><td>Steam header volume</td><td>100 ft^3</td><td>100 ft^3</td><td>Same - Intentionally reduced to cause overfill</td></tr>
<tr><td>PZR heaters & spray</td><td>Not modeled</td><td>Not modeled</td><td>Same</td></tr>
</table>

<u>NRC Question 2</u>:

Discuss the computer codes used in the thermal-hydraulic analysis of the SGTR event. If any of the computer codes are not approved by the NRC, provide a justification for their use in the analysis.

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide a statement that the RETRAN version used to support the steam generator tube rupture submittal is consistent with the version used to support the current licensing bases and that this is a plant-specific model.

<u>AmerenUE Response to Question 2</u>:

The thermal-hydraulic analysis was performed using RETRAN, which is an NRC approved code. The Westinghouse thermal-hydraulic re-analysis effort included validation that the version of RETRAN used by Westinghouse produced comparable results to the version used by Union Electric to support the SGTR analyses submitted for NRC review during1987 and approved by the NRC in 1988.

<u>NRC Question 3</u>:

Provide any revised emergency operating procedure (EOP) steps (in E-2 and E-3) that are related to the actions required for isolation of auxiliary feedwater (AFW) flow to the failed steam generator (SG).

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide the EOP step that throttles auxiliary feedwater flow early. It is unacceptable to base the analysis on an administrative procedure instead of an EOP.

<u>AmerenUE Response to Question 3</u>:

Emergency Operating Procedure E-3, Steam Generator Tube Rupture, was revised in support of the SGTR with overfill re-analysis. Procedure E-3 was revised so that the required operator action times provide a more aggressive response to the SGTR with overfill event. The revision included:

- Incorporation of Westinghouse recommended revisions which deleted a redundant step, reordered steps, and allow some parallel actions in order to improve the timely completion of operator actions

- Addition of several attachments to enhance the operating crew's ability to perform concurrent actions while progressing through the procedure

- Added and modified procedure notes to aid the crew in timely step completion

- Modified a step to isolate AFW flow to the ruptured SG in the case of the failed open AFW flow valve.

E-3 procedure changes were made with full validation on the simulator and were included in the training for all licensed operators.

The operator is directed to isolate AFW flow to a ruptured SG upon diagnosing the rupture by the foldout page of procedure E-0. See AmerenUE Response to NRC Question 4 for additional detail.

<u>NRC Question 4</u>:

For the case of an AFW control valve failing to its open position, discuss what operator actions are needed to isolate the failed SG from continued AFW flow injection (i.e., what backup valve needs to close and its location), including where these actions are performed.

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide a statement that the EOP steps include which isolation valves (by component ID number) must be closed to isolate auxiliary feedwater following a steam generator tube rupture event. Also state that there is redundant isolation valves located in series.

<u>AmerenUE Response to Question 4</u>:

Once the operator has identified the need to isolate AFW flow to the ruptured steam generator, he will attempt to close the flow control valves from the main control board. If this is unsuccessful, the operator has two options to complete the isolation. If adequate AFW flow can be maintained without the AFW pump associated with the flow to the ruptured steam generator, that pump may be stopped from the main control board. The alternative is to dispatch a local equipment operator to manually isolate AFW flow to the ruptured steam generator.

Callaway emergency procedure E-3, Step 4.b directs operators to stop feed flow to ruptured S/G(s). The foldout page to procedure E-0 directs operators to stop feed flow to ruptured SG(s). Step 4.b of procedure E-3 is reproduced below:

b. Stop feed flow to ruptured SG(s) .	b.
(1) CLOSE the MDAFW pump flow control valve to the ruptured SG. SG Flow CTRL VLV A AL-HK-7A B AL-HK-9A C AL-HK-11A D AL-HK-5A	(1) NOTE Once the failed flow control valve is isolated the AFP may be restarted • IF >300,000 lbm/hr flow can be maintained to the intact SG(s) without the associated MDAFP, place the associated MDAFP in Pull To Lock, and locally isolate its flow control valve. SG Flow CTRL VLV Manual ISO A AL-HV-7 ALV032 B AL-HV-9 ALV047 C AL-HV-11 ALV044 D AL-HV-5 ALV035 • If total AFW flow can not be maintained >300,000 lbm/hr without the associated MD AFW pump, then locally isolate the stuck open flow control valve. SG Flow CTRL VLV Manual ISO A AL-HV-7 ALV032 B AL-HV-9 ALV047 C AL-HV-11 ALV044 D AL-HV-5 ALV035
2) CLOSE the TDAFW pump flow control valve to the ruptured SG. SG Flow CTRL VLV A AL-HK-8A B AL-HK-10A C AL-HK-12A D AL-HK-6A	(2) NOTE Once the flow control valve is isolated restore the TDAFP to 3850 rpm. • If >300,000 lbm/hr can be maintained to the intact SG(s) without the TDAFP, reduce TDAFP speed <2000 rpm using FC HIK-313A and locally isolate the TDAFP flow control valve. SG Flow CTRL VLV Manual ISO A AL-HV-8 ALV056 B AL-HV-10 ALV066 C AL-HV-12 ALV071 D AL-HV-6 ALV061 • If total AFW flow can not be maintained >300,000 lbm/hr without the TDAFP, then locally isolate the flow control valve. SG Flow CTRL VLV Manual ISO A AL-HV-8 ALV056 B AL-HV-10 ALV066 C AL-HV-12 ALV071 D AL-HV-6 ALV061

NRC Question 5:

Discuss the integrity of the main steam line for SG overfill (i.e., with the entire main steam line up to the MSIV filled with water).

AmerenUE Response to Question 5:

A pipe stress analysis of the most highly stressed main steam line, which assumed the main steam line up to the MSIV was filled with water, was performed by Bechtel. This analysis found the piping and supports still met ASME Code Class 2 requirements and were acceptable for this condition.

NRC Question 6:

For the main steam line break (MSLB), discuss the effect of the proposed longer closure time (i.e., 15 seconds) of the main feedwater isolation valves (MFIVs).

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, provide details of the core response to various FSAR Chapter 15 events as it relates to DNBR for the feedwater isolation valve modification. Include specific numerical values where appropriate. Also, provide justification for all events where it is stated that the changes are insignificant or that there is no impact.

AmerenUE Response to Question 6:

As discussed in the submittal, the increase in MFIV stroke time potentially affects the main steam line break (MSLB) mass and energy (M&E) releases inside and outside containment.

The analysis of the MSLB M&E releases outside containment (FSAR Section 3B.4.2) assumes main feedwater isolation coincident with reactor trip, with no delays associated with instrumentation or valve stroke. This is a conservative assumption for this event. Quicker isolation of main feedwater flow produces more limiting Main Steam Tunnel pressure-temperature results due to minimized total mass addition to the SGs and resultant higher levels of superheat in the blowdown Mass and Energy (M&E) releases. Therefore, an increase in the MFIV closure time does not adversely impact this analysis.

The proposed increase in MFIV isolation time affects the key parameter of steam generator dry-out time in the MSLB inside containment analysis. This parameter is addressed in the original pressure-temperature calculations. Although slower valve closure time impacts mass and energy releases in general, the proposed stroke time increase of 10 seconds does not specifically impact the original calculated Mass and Energy (M&E) Releases. The original portion of the analysis is unaffected by the proposed change in MFIV isolation time.

The analysis of the MSLB M&E releases inside containment (FSAR Section 6.2.1.4) for Callaway is limiting at part-power conditions, resulting from a split rupture in a steam line. The analysis supporting the limiting MSLBs for containment response is a generic calculation performed by the NSSS supplier for the Model F steam generator design. The MSLB M&E releases for split ruptures are generic with no specific assumptions regarding time for main feedwater isolation (as well as other critical protection functions). It has been confirmed that the generic assumptions made in the original Mass and Energy Releases analysis bound the Callaway Plant proposed MFIV stroke time of 15 seconds.

Adjustments have been made to the generic M&E release values for specific Callaway Plant conditions. As previously stated, the key parameter affected by longer MFIV isolation time is steam generator dry-out time.

Post-accident steam generator dry-out is defined as the time when flow into the affected generator is equal to flow out of the generator, after the break has occurred. (In order to reach dry-out, the initial inventory must be depleted; break flow is then a function only of flow into the generator. Following dry-out, the magnitude of the break flow is not influenced by the secondary side water inventory). If dry-out occurs after the termination of AFW flow to the faulted steam generator, the mass release rate is set to zero following dry-out. If dry-out occurs prior to AFW termination, the mass release rate is set to the AFW flow rate. The mass release rate is then subsequently set to zero once AFW flow is terminated.

The proposed increase in MFIV stroke time would result in additional main feedwater mass being introduced into the affected steam generator. The additional mass would then be released to containment, which would delay dry-out of the affected generator. This would then provide the potential to lead to higher post-MSLB pressures or temperatures inside containment.

AmerenUE has performed a calculation to quantify the impact of the additional 10 seconds of main feedwater flow to the steam generators following initiation of the accident sequence. This calculation quantified the additional steam generator secondary side mass inventory, following a MSLB inside containment. Then CONTEMPT computer code runs were executed to determine the impact of the additional mass on post-MSLB containment pressures and temperatures. These CONTEMPT runs found that the proposed MFIV actuator replacement and associated increase in MFIV stroke time caused no adverse impact on post-MSLB containment pressures and temperatures. The calculated pressure and temperature profiles remain bounded by the analysis envelopes originally calculated by Bechtel.

The change in MFIV stroke time from 5 seconds to 15 seconds has a negligible impact on the Feedwater System Malfunctions that Result in an Increase in Feedwater Flow, the Steam System Piping Failure and the Small Break Loss-of-Coolant Accidents. The impact on these accidents has been evaluated by Westinghouse and is discussed below.

The acceptance criterion for the HFP Feedwater Flow Increase case is a minimum DNBR of 1.69. The current Analysis of Record for Callaway results in a minimum DNBR of 2.078, which is well above the minimum acceptable DNBR. The expected effect of delaying the MFIV stroke time would be to decrease the minimum DNBR to approximately 2.05. As can be seen, this small change is minimal when compared to the available margin to the safety analysis limit DNBR value of 1.69.

The minimum DNBR for the Steam Line Break of 2.072 for Cycle 13, which was previously reported in ULNRC-04592, was incorrect. The reported value for Cycle 13 should have been a minimum DNBR of 2.90. Regardless of this reporting error, the analyses for all Steam System Piping Failures continue to satisfy their acceptance criteria and the conclusions presented in the FSAR for these Non-LOCA events remain valid. The minimum DNBR for SLB is calculated for each fuel cycle. Fuel Cycle 14 has a minimum DNBR of 2.35 as compared to the acceptance criteria of 1.50. The effect of increasing the stroke time of the MFIVs to 15 seconds would be to see a slight decrease in the minimum calculated DNBR to approximately 2.30 for Cycle 14. The significant margin, combined with the determination that DNBR is insensitive to the change in MFIV stroke time or the change in increased AFW flow margin, demonstrates that the proposed modification to the MFIV actuators and the proposed installation of the ARC valves do not represent an unacceptable adverse impact on the SLB analyses.

The small break LOCA Analysis of Record modeled a MFIV stroke time of 5 seconds. Through further review and evaluation by Westinghouse, it was determined (stated) the effect of small changes on the secondary side of the steam generator will have no impact on the SBLOCA analysis results. As a result, it has been concluded that the proposed change will not affect the reported peak clad temperatures for the SBLOCA sequence. The current reported peak clad temperature of 1687 °F for the SBLOCA remains valid and continues to be well below the regulatory limit of 2200 °F.

<u>NRC Question 7</u>:

Following the SG overfill, it is assumed that the safety valve associated with the failed SG is stuck open with an effective flow area equal to 5% of the total safety valve flow area. Discuss the basis for the assumed effective flow area value.

<u>AmerenUE Response to Question 7</u>:

This flow area value was mandated as part of Question 1 from NRC staff requests for additional information transmitted to Union Electric in an NRC letter dated 11/12/86 from P. W. O'Connor to D. F. Schnell. Union Electric committed to use this value which was documented in our response to the NRC and transmitted in ULNRC 1518, dated May 27, 1987.

<u>NRC Question 8</u>:

In the sequence of events for a SGTR with overfill, it is indicated that the operator actions to terminate AFW flow from the turbine driven AFW (TDAFW) pump to the failed SG will be completed within 10 minutes following the event initiation. Discuss why these actions are expected to be completed within 10 minutes while the operator actions to isolate AFW flow from MDAFW pumps are expected to be completed within 20 minutes.

In addition, as requested at the meeting held on November 12, 2003 between AmerenUE and the NRC, (1) provide additional explanation as to why the motor-driven pump discharge valve failure is more limiting than the turbine-driven pump discharge valve failure; (2) identify recent NRC inspection report that reviewed the containment analysis assumption change

from 30 minutes to 10 minutes for operators to isolate auxiliary feedwater flow to the ruptured steam generator following a main steam line break. This issue was raised by the NRC inspection team during their review of the minimum containment cooler flow curve included in the Technical Specification Bases; and (3) verify that the referenced SLNRC 81-039 is the correct letter and include the date that it was transmitted to the NRC.

<u>AmerenUE Response to Question 8</u>:

The SGTR with overfill is a licensing bases event which includes a single failure. The single failure is the MDAFP flow control valve. The failed MDAFP flow control valve produces more limiting results than a failed TDAFP valve. So within 10 minutes, the operator diagnoses the need to isolate AFW to the ruptured generator. The TDAFP valve is assumed to respond and function properly when the RO closes it from the Main Control Board. The MDAFP flow control valve is the assumed failure, so the additional 10 minutes allows time to either dispatch a local operator, or to turn off the MDAFP.

The motor-driven auxiliary feedwater pump (MDAFP) discharge valve failure is more limiting than the turbine-driven auxiliary feedwater pump (TDAFP) discharge valve failure because of the higher feedwater flowrate entering the ruptured S/G. A higher feedwater flowrate has the effect of overfilling the ruptured S/G sooner and worsening offsite dose consequences. Although the TDAFP has a higher capacity than a MDAFP, the AFW piping configuration divides the TDAFP output among the four SGs. The AFW piping configuration for a single MDAFP directs the output to only two SGs. As an example taken from the input data, at 414 psia SG pressure, the ruptured SG can receive 547 gpm from the TDAFP and 770 gpm from a MDAFP. The MDAFP valve failure results in the highest flowrate.

Callaway Plant- NRC Inspection Report 50-483/01-04 And Exercise Of Enforcement Discretion, dated April 25, 2001 reviewed the containment analysis assumption change from 30 minutes to 10 minutes for operators to isolate AFW flow to the ruptured SG following a MSLB. The change was based on guidance given in Callaway administrative procedures. This guidance will be incorporated into Callaway E Procedures prior to implementation of the proposed license amendment.

The ten minute time to isolate the AFW flow to the ruptured SG was included in the original FSAR Section 10.4.9.2.3 which was transmitted to the NRC via letter SLNRC 81-039, dated June 3, 1981.

NRC Question 9:

Figure 15.6-3P, "Feedwater Flow Rate," of the proposed FSAR page changes, does not appear to be consistent with the time assumed for closure of the MFIVs (i.e., 15 seconds). Discuss how the FSAR figure is consistent with the assumed MFIV closure time.

AmerenUE Response to Question 9:

Figure 15.6-3P, "Feedwater Flow Rate", is only included in Attachment 6 of the submittal in order to show the placement of new Figures associated with the SGTR with overfill accident, which is being added into the FSAR. Figure 15.6-3P is not associated with the SGTR with overfill accident, but the SGTR with stuck-open ASD. The SGTR with stuck-open ASD is the current FSAR accident Analysis of Record as the most limiting in terms of radiological consequences. The SGTR with stuck-open ASD analysis is not impacted by the increase in MFIV isolation time and was not re-analyzed. The current analysis for SGTR with stuck-open ASD is more limiting in the assumption of MFIV isolation time. As discussed in the Callaway FSAR, Section 15.6.3.2.g, the analysis assumes the feedwater isolation signal occurs 2.3 seconds after reactor trip and the feedwater isolation valves stroke closed within 2.0 seconds. These are considered the minimum expected delay and stroke time, respectively, which decreases heat removal from the reactor coolant system resulting in higher reactor coolant system temperatures and pressures. This results in maximum flashed fraction and break flow in the analysis. In summary, quicker isolation of main feedwater flow causes an increase in break flow flashing and steam releases, resulting in more radioactivity released to the atmosphere and higher radiological consequences. As a result, the proposed increase in MFIV stroke time does not invalidate the results of the current FSAR SGTR with stuck-open ASD analysis.

NRC Question 10:

Discuss why the change to the MFIV closure time from 5 seconds to 15 seconds does not affect other events for which re-analyses of these event should be performed to support the proposed Technical Specification changes.

AmerenUE Response to Question 10:

As discussed in the license submittal, a complete review of FSAR Chapter 6.2 and Chapter 15 accident analyses was performed to evaluate the impact of the increase in MFIV closure time from 5 seconds to 15 seconds. In some of these accident analyses, the MFIV isolation time is not explicitly modeled or MFIV position is not an essential analysis consideration. These accident sequences or categories of sequences are listed in the submittal and are eliminated from further consideration for impact based on the increase in MFIV isolation time.

The remaining accident analyses were reviewed and evaluated for impact. These accidents are also listed in the submittal. Individual summaries are provided for each accident to discuss the evaluation and the results. In each case, the overall impact is determined to have little or low significance and re-analysis of the accident is not warranted. The accident scenario for SGTR with overfill is the exception. A complete re-analysis was performed for the SGTR with overfill accident.

AmerenUE is confident that the SGTR with overfill is the only accident scenario significantly impacted by the increase in MFIV closure time from 5 seconds to 15 seconds. The SGTR with overfill was re-analyzed in support of the proposed change.

VI. NRC REQUEST FOR ADDITIONAL INFORMATION TRANSMITTED VIA E-MAIL DATED OCTOBER 21, 2003

NRC Question 1:

This is an additional question for the SGTR re-analysis. In the email response from Dwyla Walker to me dated September 3, 2003, on the human factors questions, there are the following statements: 1. It should be noted that the Callaway SGTR analysis is not performed using the methodology described in WCAP-10698. The Callaway SGTR analysis is based on the SNUPPS methodology which was originally developed for Callaway and Wolf Creek." 2. "Overfill prevention is not demonstrated at Callaway. The analysis and operator action times are commensurate with mitigation of the consequences of an overfill event." Discuss why the SGTR analysis and operator actions are set for mitigation of the consequences of an overfill event instead of being set for prevention of the overfill event (i.e., WCAP-10698).

AmerenUE Response to Question 1:

WCAP-10698 has never been a licensing basis for Callaway Plant. The SGTR analysis forming the licensing basis for Callaway Plant is the SNUPPS methodology that was originally developed for Callaway and Wolf Creek. The WCAP was only mentioned in our response because it was cited as the basis for the NRC question. SGTR analyses for Callaway Plant demonstrate that overfill prevention is not possible, therefore, the analysis and operator action times are commensurate with mitigation of the consequences of an overfill event.