June 3, 2005

LICENSEE: Union Electric Company

FACILITY: Callaway Plant

SUBJECT: SUMMARY OF MEETING WITH UNION ELECTRIC COMPANY ON MAY 18, 2005, FOR THE LICENSEE TO DISCUSS ITS DRAFT RESPONSES TO THE REQUEST FOR ADDITIONAL INFORMATION ON ITS APPLICATION DATED SEPTEMBER 17, 2004, TO REPLACE THE STEAM GENERATORS IN THE FALL OF 2005 (TAC NO. MC4437)

A meeting was held on Wednesday, May 18, 2005, between the Nuclear Regulatory Commission (NRC) staff and representatives of the Union Electric Company, the licensee for the Callaway Plant. The meeting was requested by the licensee for it to discuss any questions from the NRC staff on some of its draft responses to part of the request for additional information (RAI) issued by the NRC on May 4, 2005 (Agencywide Documents Access Management System (ADAMS) Accession No. ML051150102). The RAI is based on the licensee's application dated September 17, 2004, which was submitted to support the planned replacement of the steam generators (RSGs) in Refueling Outage (Refuel) 14 in the fall of 2005. This RAI includes questions from the Reactor Systems Branch (RSB) review of the licensee's application and the meeting was requested, as stated in the meeting notice, to (1) discuss the latest revision of TSTF-449 and (2) explain the licensee's responses to the RSB questions. Questions from NRC branches, other than the RSB, were not addressed in the meeting. A notice of this meeting was issued on May 4, 2005.

Enclosure 1 is the list of attendees. There was no handout from the licensee or from the NRC staff. The agenda (which was given in the meeting notice) was the following:

1. Introduction
2. Licensee's Application Dated September 17, 2004 (ADAMS Accession Nos. ML042870364, ML042860352, and ML042870372)
    a. Responses to Questions from the Reactor Systems Branch
    b. Technical Specification (TS) Changes in Current Revision of Technical Specification Task Force (TSTF) - 449, "Steam Generator Tube Integrity"
3. Closing Comments
4. Public Comments (no one attended from the public)
5. Adjourn Meeting

For TSTF-449, the licensee had been informed that the Notice of Availability in the *Federal Register* for TSTF-449, Revision 4, was May 6, 2005, Volume 70, Number 87, Pages 24126 to 24127. Also, the TSTF can be accessed on the NRC Website under "CLIIP [NRC

Consolidated Line Item Improvement Process] STS [Standard Technical Specification] Changes Available for Adoption." TSTF-449, Revision 4, that includes the up-to-date TS pages related to steam generator tube integrity, is part of the licensee's application dated September 17, 2004. The licensee agreed that the TSTF did not have to be discussed further in the meeting.

In Enclosure 2 are the emails sent to the licensee with questions that were finally issued in the RAI dated May 4, 2005. The first email sent only RSB questions. The second email also included questions from (1) the instrumentation and controls section of the electrical branch and (2) the containment branch. The RSB questions in the second email were a revised set of questions with respect to the questions in the first email. The official questions from these three branches for the licensee's application are in the RAI letter, which are essentially the questions in the second email, only with editorial corrections. The questions were provided earlier to the licensee than the RAI letter because the license amendment is needed for the upcoming fall 2005 refueling outage, which is expected to begin in September 2005.

In Enclosure 3 are the licensee's draft responses to the questions in the RAI letter. These responses were discussed in the meeting only if the NRC staff had questions on the responses. If additional information was needed to be added to the responses, the NRC staff identified this in the discussion on the response. Enclosure 3 contains proprietary information in the responses for questions 15 and 16, and this information is not in this meeting summary. The licensee stated before the meeting that the proprietary information in the responses to these questions is contained in WCAP-16259-P, "Callaway Replacement Steam Generator Program NSSS Licensing Report," which is covered by the NRC staff's letter dated November 10, 2004, to Westinghouse Electric Company, on withholding information in this WCAP from public disclosure. The licensee submitted proprietary and non-proprietary versions of WCAP-16259-P in its application dated September 17, 2004.

In Enclosure 4 are four figures that are the plots referenced in the draft response to Question 46.

The licensee stated that because, as discussed in the response to Question 38, the time critical validations of the new steam generator tube rupture scenarios are to be timed in early June 2005, it expected to submit the formal responses to the RSB questions by June 17, 2005. The NRC staff stated that this date was acceptable.

Following the NRC staff discussion on the licensee's draft responses, the meeting was adjourned. There were no decisions made by the NRC staff related to the licensee's proposed TS changes submitted in the September 17, 2004, application.

During the meeting discussions, the licensee may have provided additional information to the NRC staff that is not in Enclosures 3 and 4. Because of this, the licensee's responses the NRC staff's questions that will be used to complete the staff's evaluation of the application will be that given in the licensee's formal RAI response letter, to be submitted by June 17, 2005, and not

the information provided to the NRC staff in Enclosures 3 and 4 of this meeting summary, or in the discussions conducted during this meeting.

/RA/
Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. List of Meeting Attendees
2. Questions Sent to Licensee in Two Emails
3. Draft Responses to Reactor Systems Branch Questions
4. Plots Referenced in License's Draft Response [in Enclosure 3] to Question 46

cc w/encls: See next page

the information provided to the NRC staff in Enclosures 3 and 4 of this meeting summary, or in the discussions conducted during this meeting.

/RA/
Jack Donohew, Senior Project Manager, Section 2
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cc w/encls: See next page

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LIST OF MEETING ATTENDEES

STEAM GENERATOR REPLACEMENT IN REFUELING OUTAGE 14 IN FALL 2005

MEETING OF MAY 18, 2005, WITH CALLAWAY PLANT

NAME	AFFILIATION
J. Donohew	NRC/NRR/PDIV-2
J. Nakoski	NRC/NRR/SRXB
F. Orr	NRC/NRR/SRXB
K. Desai	NRC/NRR/SRXB
S. Miranda	NRC/NRR/SRXB
L. Ward	NRC/NRR/SRXB
D. Shafer	UE
B. Yates	UE
K. Mills	UE
D. Martin	UE
J. Andrachek	Westinghouse
R. Schoff	Westinghouse
M. Watson	Westinghouse
J. Fontes	Westinghouse
C. Boyd	Westinghouse

Where:

NRC	=	Nuclear Regulatory Commission
NRR	=	Office of Nuclear Reactor Regulation
PDIV-2	=	Project Directorate IV-Section 2
SRXB	=	Reactor Systems Branch
UE	=	Union Electric Company
Westinghouse	=	Westinghouse Electric Company

QUESTIONS SENT TO LICENSEE IN TWO EMAILS

DATED APRIL 19 AND 22, 2005

The first email dated April 19, 2005, only sent the questions from the reactor systems branch review; however, the second email also had the questions from the instrumentation and controls review and the containment integrity review of the licensee's application. The request for additional information (RAI) letter issued May 4, 2005, to the licensee had the questions from all three reviews; however, only the responses to the reactor systems branch were discussed in the meeting.

ENCLOSURE 2

QUESTIONS SENT BY EMAIL DATED APRIL 19, 2005

CALLAWAY PLANT
STEAM GENERATOR REPLACEMENT- WCAP 16265
REQUEST FOR ADDITIONAL INFORMATION
TAC NO. MC4437
([REACTOR SYSTEMS BRANCH QUESTIONS ONLY])

1. Section 1.5.1, discusses the Analysis Methodologies and Computer Codes used for non-LOCA transient analyses for the Callaway Plant replacement steam generators (RSG) project. RETRAN-02, which has been generically approved by the NRC staff for non-LOCA transient analyses, was used for the first-time for non-LOCA transient analyses at the Callaway Plant. Please explain the quality assurance process used to verify that RETRAN-02 was adequately used at the Callaway Plant and show that the Callaway nodalization modeling is consistent with the Westinghouse 4-loop plant nodalization model of WCAP-14882-P-A. If the modeling of the Callaway deviated from the plant model in the WCAP-14882-P-A, justify their technical validity.

2. Table 6.3-4 lists a summary of initial conditions and computer codes used in revised RSG analyses. These computer codes have been approved by the staff for accidents and non-LOCA transient analyses. For each computer code, the staff's safety evaluation lists the staff positions and limitations for its application. List the staff positions or limitations for ANC and VIPRE computer code and address how you satisfy these requirements for RSG conditions at the Callaway Plant.

3. Table 2-1 lists the thermal design parameters for RSG conditions. Please provide a tabulation of the expected operating and safety analyses values assumed in the safety analyses to demonstrate that the safety analyses assumptions are adequate or conservative.

4. Provide a quantitative evaluation of the time needed for plant cooldown to cold shutdown conditions, per RSB BTP 5-1 (natural circulation cooldown using only safety grade equipment), and for plant cooldown per the requirements of 10 CFR 50, Appendix R (regarding fire protection) for the Callaway plant at the RSG revised RCS operating conditions.

5. Section 4.2.2 discusses steam generator blowdown system. The minimum full-load steam pressure could be as low as 867 psia or 41 psi lower than the current minimum full-load pressure of 908 psia. This decrease in blowdown system inlet pressure will impact the required maximum lift of the blowdown flow control valves. Demonstrate and justify that the blowdown flow control valves are adequate for the range of NSSS design parameters approved for the RSGs. Please provide MP 00-1013 report which discusses blowdown control valves.

6. Section 4.3.2 discusses pressure control component sizing. It does not provide any information regarding pressurizer safety valves sizing. Discuss the design basis of the pressurizer safety valves sizing at the Callaway Plant. Are they sized according to the

method described in SRP Section 5.2.2, Section II, A, which is based upon the assumptions of a reactor trip on the second reactor trip signal? Verify the adequacy of the pressurizer safety valves for the RSG conditions using methods that are consistent with the current licensing basis for the Callaway Plant.

7. Section 4.3.5 discusses low-temperature overpressure protection. It is stated that the design-basis Mass Input (MI) transient remains applicable for the RSG program and Heat Input (HI) transient re-analysis was performed. The revised LTOP setpoints were developed using the results of HI transient and the current MI results. Provide the results and assumptions (temperature difference between SG and RCS prior to start a RCP) used in the reanalysis of HI transient

   Describe the basis for the wide-range pressure uncertainty for LTOP which has increased from 85 to 93 psig for the RSG program and its effects on LTOP design setpoints.

   The MI transient is limiting for RCS temperature less than about 180 degree F. The current LTOP arming temperature of 275 degree F remains applicable for the RSG program. Describe which transient is more limiting between180 degree F and 275 degree F and its effects on LTOP design setpoints

8. Please confirm that only safety grade systems and components are credited in the re-analyses of all accidents and transients for the RSG conditions at the Callaway plant.

9. Provide a quantitative evaluation of the impacts of the RSG conditions on the ability to cope with a Station Blackout (SBO) event. The evaluation should address the capabilities of the condensate storage tank, turbine driven auxiliary feedwater pump, station batteries, and backup air supplies for air operated valves for decay heat removal and reactor coolant system (RCS) cooldown during the time period of an SBO.

10. The Callaway SG replacement package does not discuss the effect of the SG replacement on Callaway LOCA analyses. Page 22 of 51 of Attachment 1 indicates that LOCA analyses are not affected by the new SGs. However, the discussion on pages 19 and 20 of 51 of Attachment 1 indicates that the calculated peak containment pressure for a LOCA event is greater than 1 psi lower with the new SGs than previously analyzed. This would seem to increase the calculated LBLOCA peak cladding temperature. Please discuss the effect of the SG replacement on LBLOCA events.

11. In looking for other possible discussions, the staff consulted and reviewed the Callaway Plant "Amended 10 CFR 50.46 Thirty Day Report...," dated March 9, 2005. This 30-day report is unacceptable and does not provide a basis for the SG discussion above. The 30-day report must be corrected. Reference to WCAP-13451 (unapproved) explains but cannot justify the 30-day report. The information in Attachment 2 indicates that the summations (for LBLOCA and SBLOCA) of errors and changes were not done in conformance with 10 CFR 50.46 (a)(3)(I). It appears to the staff that Callaway must submit a proposed schedule for reanalyses of LBLOCA and SBLOCA to establish new Analyses of Record (AORs). This item will affect the staff's evaluation of item 10 above.

12. The Callaway submittal provides a summary of LOCA analysis parameters. This submittal also refers to information specific to the LBLOCA analyses performed to define the licensing basis for Callaway LBLOCA. The staff requests further information to address the programmatic requirements of 10 CFR 50.46(c). To show that the referenced generically approved LOCA analysis methodologies (continue to) apply specifically to the Callaway plant, provide a statement that Ameren UE and its vendor have ongoing processes which assure that the ranges and values of the input parameters for the Callaway LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters. Furthermore, if the Callaway plant-specific analyses are based on the model and/or analyses of any other plant, then justify that the model or analyses apply to Callaway. (e.g., if the other design has a different vessel internals design the model wouldn't apply to Callaway.)

13. The LOCA submittals did not address slot breaks at the top and side of the pipe. Please justify why these breaks are not considered for the Callaway LBLOCA submittal.

14. The Callaway submittal states that the W LOCBART extension method was used to address downcomer boiling for Callaway. This method has not been approved at this time for generic application. Show that this method is applicable to Callaway by:

    a. Provide comparative results (PCT vs Time Graph) for the most limiting Callaway LOCA scenario, using the LOCBART extension method and WCOBRA/TRAC:

    b. Provide the LBLOCA analysis results tables and graphs to at least 1600 second show that stable and sustained quench is established.

    c. Indicate on the graph the period of downcomer boiling for each calculation.

15. It is not clear from LBLOCA and SBLOCA figures what specific upper core plate is used for Callaway. Identify the specific upper core plate design used in Callaway.

16. Tables 6.2.1-5 and Table 6.2.2-5 provide LBLOCA and SBLOCA analyses results for the Callaway RSG. Please provide all results (PCT, maximum local oxidation, and total hydrogen generation) for both LBLOCA and SBLOCA. For maximum local oxidation include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also include the results for fuel resident from previous cycles.

17. Please provide a table listing the key assumptions and input parameter values for all accident analyses in the licensing bases of Callaway, before and after the proposed license amendment. Include the core kinetics assumptions used for the steam line break analyses.

18. Use the table of Section 3.1 in [Reference 1], which compares the key design parameters of the Framatome Model 73/19T RSGs to the design parameters of the OSGs, to discuss how differences in these design parameters would affect analyses and evaluations of the Callaway accidents and transients.

19. Based upon operating history at Callaway and other PWRs, discuss the benefits of removing the Trip Time Delay (TTD) system. Would there be any expected increase in feedwater system-related reactor trips at low power levels?

20. Based upon operating history at Callaway and other PWRs, discuss the benefits of retaining the Environmental Allowance Modifier (EAM) system.

<u>Section 6.3.3</u> <u>Steam System Piping Failure (analysis)</u>

21. Section 6.3.3.1 states, "The effective throat diameter of the flow restriction nozzles of 6.0315 inches is considerably smaller than the diameter of the main steam pipe. These restrictions are located in the outlet nozzles of the steam generators and serve to limit the maximum steam flow for any break at any location." This is equivalent to a throat area of about 0.20 ft^2, much less than the 1.4 ft^2throat area of current Westinghouse steam generators with integral flow restrictions. Section 6.3.3.2 states, "Since the steam generators are provided with integral flow restrictions with a 1.39 ft^2 throat area, any rupture with a break greater than 1.39 ft^2, regardless of the location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.39 ft^2 break." Please reconcile these statements.

    Note that flow restriction nozzles with an effective throat diameter of 16.0315 inches (if the 6.0315 inch value is a typo) would have an equivalent throat area of 1.40 ft^2. Please identify the maximum steam line break size that is analyzed for Callaway, and relate it to the steam pipe and flow nozzle dimensions.

22. Section 6.3.3.2: Explain how the coefficient assumption was revised for the RSG analysis to improve the core physics prediction of the point kinetics core model.

23. Section 6.3.3.2: Explain how all reactivity physics parameters are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

24. Section 6.3.3.3 states that the minimum DNBR does not go below the limit value at any time during the transient. What are the calculated minimum DNBR values? Please provide transient plots of DNBR for both Callaway HZP steam line break cases.

25. Table 6.3.3-1 indicates that (for both HZP cases) the low steam line pressure setpoint is reached 2 seconds after the steam line rupture. Adding 2 seconds for signal processing and 15 seconds for MSIV stroke time results in an MSIV closure time of 19 seconds; but the Table lists MSIV closure at 18 seconds. Please explain how the 18 second value was determined.

26. Please provide steam generator mass transient plots, for the faulted and intact steam generators, for both HZP steam line break cases.

27. Please include auxiliary feedwater flow plots, for the faulted and intact steam generators, in the feedwater flow transient plots for both HZP steam line break cases.

Section 6.3.5 Loss of Non-Emergency AC Power to the Station Auxiliaries/Loss of Normal Feedwater Flow (analysis)

28. Section 6.3.5.2 states, "The dual-analysis approach has been previously used by Westinghouse in one other LONF analysis of a Westinghouse-designed plant. That previous analysis has been accepted by the NRC and the NRC will again review the results of this dual-analysis approach as part of the RSG license amendment." Please reference this LONF analysis and NRC acceptance.

Section 6.3.10 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (analysis)

29. Discuss, or cite discussions of the effect the SG replacement would have upon the core limits, protection lines, and Overtemperature *T trip setpoint calculations used in the Uncontrolled RCCA Bank Withdrawal at Power event analyses.

30. What is the minimum possible reactivity insertion rate for Callaway? Provide analysis results for this minimum possible reactivity insertion rate assuming various initial power levels and minimum reactivity feedback, to show that the pressurizer would not fill before the reactor is tripped.

31. Section 6.3.10.4 states, "The RCS pressure safety analysis limit of 2,748.5 psia is confirmed to be met via a generic evaluation." Please cite the generic evaluation and verify that the evaluation (1) does not credit operation of pressurizer spray or PORVs, and (2) applies to Callaway.

Section 6.3.14 Inadvertent Actuation of ECCS at Power (analysis)

32. How was the current licensing basis for the Inadvertent Actuation of ECCS at Power event established (e.g., by 10CFR50.59 evaluation or by staff review and approval)? If by 10CFR50.59 evaluation, then please provide a copy of the 10CFR50.59 evaluation. If by staff review and approval, then please cite the effective license amendment.

33. The pressurizer power-operated relief valves are predicted to open when the pressurizer is water-solid. Are they expected to reseat properly?

    a. If yes, then (1) state how they, and their associated discharge piping, have been qualified for water relief during an Inadvertent Actuation of ECCS at Power event, (2) verify that the automatic control circuitry of these valves meets Class 1E requirements, and (3) indicate the PORV opening setpoint, setpoint tolerance, and surveillance requirements for operation under water relief conditions.

    b. If no, then explain how the analysis results demonstrate that the event will not develop into a more serious event, e.g., a small break LOCA.

34. Please supply the ECCS flow delivery rate, as a function of RCS pressure, that was assumed for the Inadvertent Actuation of ECCS at Power event analysis. Compare this flow delivery rate to that assumed in the current licensing basis analyses.

Section 6.3.15 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (evaluation)

35. Section 15.5.2 of the Callaway FSAR [Reference 2] concludes that the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory will not fill the pressurizer before the operator can terminate the transient, at about 15 minutes. Provide information to demonstrate that the operator can terminate the transient, as the result of following established procedures, before the calculated pressurizer fill time.

Section 6.3.17 Anticipated Transients Without SCRAM (analysis)

36. The Anticipated Transients Without Scram (ATWS) analyses of [Reference 3] and [Reference 4] are based upon the design parameters of Westinghouse steam generator series 44 and 51, and models D and F, not the Framatome Model 73/19T design. Explain how the referenced Westinghouse ATWS analyses and methods apply to Callaway, as equipped with the RSGs.

37. Section 6.3.0.6 indicates that, "LOFTRAN remains the system transient code for the analyses of ..... the anticipated transients without scram (ATWS, FSAR 15.8)." LOFTRAN has a single-node steam generator shell side model. Please describe how LOFTRAN models the primary-to-secondary heat transfer rate as the shell-side water level drops and the steam generator tubes are exposed, for the Framatome Model 73/19T steam generator design.

38. Provide transient plots and sequence of events tables denoting the time and value of peak RCS pressure for analyses of the Callaway Loss of Feedwater and Loss of Load ATWS events, assuming the design characteristics of the Framatome Model 73/19T RSGs.

39. List and explain the analysis assumptions (e.g., moderator temperature coefficient and initial steam generator mass) used in the ATWS analysis.

40. Verify that the maximum differential pressure, across the tubesheet and tubes of the Framatome Model 73/19T RSGs, matches or exceeds the value listed in Appendix C of WCAP-8330 [Reference 3].

Section 6.4 Steam Generator Tube Rupture (analysis)

41. Show that all assumed operator action times are verified in Callaway simulator exercises.

42. What is the tube rupture size (in sq ft or sq in) that is analyzed? Are all tubes in the RSGs of the same diameter?

43. Can the steam line support the weight of water in the steam line up to the MSIV?

44. Confirm that there is not an intermediate value of initial T_{avg} that will produce more severe results for the SG tube rupture event.

45. Why are the pressurizer heaters not assumed to be operating prior to reactor trip?

46. Section 6.4.1.2 states that, "Feedwater isolation is completed 4.3 seconds after reactor trip/SI." What is the basis for the 4.3 second value, why is this time not 17 seconds, like all the other applicable events, listed in Table 6.3-6? Why is the SG Tube Rupture not listed in Table 6.3-6?

47. One of the assumptions in Section 6.4.2.2 is, "Additional active failure: The ruptured steam generator's safety valve fails partially open (5-percent effective area) after water relief." If the steam generator's safety valve is not qualified for water relief, then the valve would be assumed to stick open, as a consequential failure, not as an additional active failure. What are the analysis results of this case, assuming a steam generator safety valve that sticks open following water relief?

REFERENCES

1. Letter No. ULNRC-05056, K.D. Young, AmerenUE, to USNRC, "Docket No. 50-483, Union Electric Company, Callaway Plant, Technical Specification Revisions Associated with the Steam Generator Replacement Project," September 17, 2004.

2. "Callaway Nuclear Plant Final Safety Analysis Report," Rev. OL-13, May 2003.

3. WCAP-8330, Westinghouse Anticipated Transients Without Trip Analysis, August 1974.

4. Westinghouse Letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

QUESTIONS SENT BY EMAIL DATED APRIL 22, 2005

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO THE CALLAWAY STEAM GENERATOR REPLACEMENT

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

By letter dated September 17, 2004, Union Electric Company (the licensee) requested NRC approval for changes to the Technical Specifications (TSs) for the Callaway Plant, Unit 1 (Callaway) to support the installation of the replacement steam generators (RSGs) in the Fall of 2005 in Refueling Outage 14. Based on its review of the licensee's application dated September 17, 2004, in the areas of instrumentation and controls, and reactor systems, the NRC staff requests the following additional information.

[This email included the questions from the instrumentation and controls review and the containment integrity review of the licensee's application. These questions were not included in the first email.]

Instrumentation and Controls Review:

1. Provide the setpoint calculation documentation for the following protection functions which have allowable values (AV) being revised in this license amendment request:

   (1) Steam Generator Water Level Low-Low (TS Table 3.3.1-1 Functions 14a. and 14b., and TS Table 3.3.2-1 Function 5e.(1), 5e.(2), 6d.(1) and 6d.(2))

   (2) Steam Line Pressure Low (TS Table 3.3.2-1 Functions 1e. And 4e.(1))

   (3) Steam Generator Water Level High-High (TS Table 3.3.2-1 Function 5c.)

2. The TSs define Limiting Safety System Settings (LSSS) as an allowable value (AV). During reviews of proposed license amendments that contain changes to LSSS setpoints, the NRC staff identified concerns regarding the method used by some licensees to determine the AVs. AVs are identified in the TSs as LSSS to provide acceptance criteria for determination of instrument channel operability during periodic surveillance testing. The NRC staff's concern relates to one of the three methods for determining the AV as described in the Instrument Society of America (ISA) recommended practice ISA-RP67.04-1994, Part II, "Methodologies for Determination of Setpoints for Nuclear Safety-Related Instrumentation."

The NRC staff has determined that to ensure a plant will operate in accordance with the assumptions upon which the plant safety analyses have been based, additional information is required regardless of the methodology used to establish LSSS values in technical specifications. Details about the NRC staff's concerns are available on the NRC's public website under ADAMS Accession Numbers ML041690604, ML041810346, and ML050670025.

In Order for the NRC staff to assess the acceptability of the license amendment request related to this issue, the NRC staff requests the following additional information:

a. Describe the setpoint methodology used to establish AVs associated with LSSS setpoints.

b. In discussing the methodology used, address the following questions regarding the use of the methodology:

    (1) Discuss how the methodology and controls you have in place ensure that the analytical limit (AL) associated with an LSSS will not be exceeded (the AL is a surrogate that ensures the safety limits will not be exceeded). Include in your discussion information on the controls you employ to ensure the trip setpoint established after completing periodic surveillances satisfies your methodology. If the controls are located in a document other than the TS, discuss how those controls satisfy the requirements of 10 CFR 50.36.

    (2) Discuss how the TS surveillances ensure the operability of the instrument channel. This should include a discussion on how the surveillance test results relate to the technical specification AV and describe how these are used to determine the operability of the instrument channel. If the requirements for determining operability of the LSSS instrument being tested are in a document other than the TS (e.g., plant test procedure), discuss how this meets the requirements of 10 CFR 50.36.

c. In discussing the methodology, the following explicit regulatory commitments and proposed TS changes are needed for the NRC staff to complete its review of the methodology:

    (1) Commitment is provided to adopt the final Technical Specification Task Force (TSTF) Technical Specification change adopted by NRC for plant TSs to come into conformance with the existing understanding of the requirements of 10 CFR 50.36.

    (2) Commitment to assess the operability of tested instrumentation based on the previous as-left instrument setting and accounting for the uncertainties associated with the test or calibration.

    (3) A revision to the TSs for the LSSS being changed by the license amendment request to incorporate a footnote that states: "The as-left

instrument setting shall be returned to a setting within the tolerance band of the trip setpoint established to protect the safety limit."

3. In Attachment 1 to the application, Page 7 of 51, Section 3.2, "TTD Elimination," states that upon NRC approval of the amendment request, the 7300 Process Protection System will be modified to eliminate the trip time delay (TTD) circuitry. Provide detailed justification and related protection system logic changes including any markup drawings for the proposed changes. Is there any precedent for these changes?

4. Discuss and identify any safety-related instrumentation change for this steam generator replacement.

5. Describe the locations of the new steam generator level instrument tapes and their impact to the steam generator level setpoint calculation described in item 1 above.

6. Identify any environmental data change (include reference leg data) after the steam generator replacement and discuss the impact on the steam generator level setpoint calculation described in item 1 above.

<u>Reactor Systems Branch (RSB) Review</u>:

General:

7. State (1) is the RETRAN-2 nodalization used in the Callaway RSG analysis in agreement with the corresponding nodalization in WCAP-14882PA and (b) does the analysis for the RSG using the ANC and the VIPRE codes satisfy the staff limitations on these codes. Explain if there are deviations.

8. Section 4.2.2 discusses the SG blowdown system. The minimum full-load steam pressure could be as low as 867 psia i.e. 41 psi lower than the current full load pressure of 908 psia. This decrease in blowdown system inlet pressure will impact the required maximum lift of the blowdown flow control valves. Justify that the valves are adequate for the range of the NSSS design parameters approved for the RSG. Provide the document MP 00-1013 which discusses the blowdown control valves.

9. Provide a quantitative evaluation of the impacts of the RSG conditions on the ability to cope with a Station Blackout (SBO) event. The evaluation should address the capabilities of the condensate storage tank, turbine driven auxiliary feedwater pump, station batteries, and backup air supplies for air operated valves for decay heat removal and reactor coolant system (RCS) cooldown during the time period of an SBO.

Low Temperature Overpressure (LTOP) Protection:

10. For Section 4.3.5 on low temperature overpressure (LTOP), does the introduction of the RSG affect the LTOP design basis? If it does please explain.

11. The SG replacement package does not discuss the effect of the SG replacement on overpressure protection analyses (Standard Review Plan (SRP) 5.2.2, Section II, A). It

is apparent that RSGs would have some effect on overpressure scenarios. Identify the present Callaway overpressure protection basis (is this WCAP-7769) and discuss the impact, if any, of the SG replacement on this basis.

Loss of Coolant Accident (LOCA) Analyses:

12. The SG replacement package does not discuss the effect of the SG replacement on the Callaway LOCA analyses. Page 22 of 51 of Attachment 1 indicates that LOCA analyses are not affected by the new SGs. However, the discussion on pages 19 and 20 of 51 of Attachment 1 indicates that the calculated peak containment pressure for a LOCA event is greater than 1 psi lower with the new SGs than previously analyzed. This would seem to increase the calculated LBLOCA peak cladding temperature. Discuss the effect of the SG replacement on LBLOCA events.

13. The Callaway submittal (WCAP-16265, Section 6.2) provides a summary of LOCA analysis parameters. This submittal also refers to information specific to the large break LOCA (LBLOCA) analyses performed to define the licensing basis for the Callaway LBLOCA. The staff requests the following information to address the programmatic requirements of 10 CFR 50.46©). To show that the referenced generically approved LOCA analysis methodologies continue to apply specifically to the Callaway plant, provide a statement that Ameren UE and its vendor have ongoing processes which assure that the ranges and values of the input parameters for the Callaway LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters. Furthermore, if the Callaway plant-specific analyses are based on the model and or analyses of any other plant, then justify that the model or analyses apply to Callaway (e.g. if the other design has a different vessel internals design the model wouldn't apply to Callaway).

14. The Callaway LOCA submittals did not address slot breaks at the top and side of the cold leg discharge pipe. Justify why these breaks are not considered for the Callaway LOCA submittal.

15. The Callaway submittal states that the Westinghouse LOCBART extension method was used to address downcomer boiling for Callaway. This method has not been approved at this time for generic application. Show that this method is applicable to Callaway by:

    a. Provide comparative results (peak clad temperature (PCT) vs time graph) for the most limiting Callaway LOCA scenario using the LOCBART extension method and WCOBRA/TRAC.

    b. Provide the LBLOCA analysis results tables and graphs to at least 1600 seconds to show that stable and sustained quench is established.

    c. Indicate on the graph the period of downcomer boiling for each calculation.

16. Tables 6.2.1-5 and 6.2.2-5 of WCAP-16265, Section 6.2 (Attachment 1), provide LBLOCA and small break LOCA (SBLOCA) analyses results (PCT, maximum local oxidation, and total hydrogen generation) for the Callaway RSG. For maximum local

oxidation include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also include the limiting local oxidation results for fuel resident from previous cycles.

17. Provide a table listing the key assumptions and input parameter values for all accident analyses in this license amendment request, before and after the proposed license amendment. Include the core kinetics assumptions used for the steam line break analyses.

18. Use the table of Section 3.1 in Attachment I in the application, which compares the key design parameters of the Framatome Model 73/19T RSGs to the design parameters of the old SGs (OSGs) to discuss how differences in these design parameters would affect analyses and evaluations of the Callaway accidents and transients.

19. Based upon operating history at Callaway and other pressurized water reactors (PWRs), explain the benefits of removing the TTD system, and retaining the environmental allowance monitor (EAM) system, in terms of the predicted effect, if any, upon the frequency of unnecessary reactor trips at Callaway.

<u>Section 6.3.3</u> <u>Steam System Piping Failure (analysis)</u>:

20. Section 6.3.3.1 states, "The effective throat diameter of the flow restrictor nozzles of 6.0315 inches is considerably smaller than the diameter of the main steam pipe. These restrictors are located in the outlet nozzles of the steam generators and serve to limit the maximum steam flow for any break at any location." This is equivalent to a throat area of about 0.2 ft^2, or a much less than half the typical 1.4 ft^2 throat area of current Westinghouse steam generators with integral flow restrictors. Section 6.3.3.2 states, "Since the steam generators are provided with integral flow restrictors with a 1.39 ft^2 throat area, any rupture with a break greater than 1.39 ft^2, regardless of the location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.39 ft^2 break." Please reconcile these statements.

    Note that flow restrictor nozzles with an effective throat diameter of 16.0315 inches (if the 6.0315 inch value is a typo) would have an equivalent throat area of 1.40 ft^2. Please identify the maximum steam line break size that is analyzed for Callaway, and relate it to the steam pipe and flow nozzle dimensions.

21. For Section 6.3.3.2:

    (a) Explain how the coefficient assumption was revised for the RSG analysis to improve the core physics prediction of the point kinetics core model.

    (b) Explain how all reactivity physics parameters are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

22. Table 6.3.3-1 indicates that, for both hot zero power (HZP) cases, the low steam line pressure setpoint is reached 2 seconds after the steam line rupture. Adding 2 seconds for signal processing and 15 seconds for main steam isolation valve (MSIV) stroke time

results in an MSIV closure time of 19 seconds; but the Table lists MSIV closure at 18 seconds. Explain how the 18 second value was determined.

23. Provide transient plots depicting steam generator mass and auxiliary feedwater flow, for the faulted and intact steam generators, for both HZP steam line break cases.

Section 6.3.5 Loss of Non-Emergency AC Power to the Station Auxiliaries/Loss of Normal Feedwater Flow (analysis):

24. Section 6.3.5.2 states, "The dual-analysis approach has been previously used by Westinghouse in one other LONF [loss of normal feedwater] analysis of a Westinghouse-designed plant. That previous analysis has been accepted by the NRC and the NRC will again review the results of this dual-analysis approach as part of the RSG license amendment." Reference this LONF analysis and the NRC acceptance documentation.

Section 6.3.10 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power:

25. Discuss, or cite discussions of the effect the SG replacement would have upon the core limits, protection lines, and Overtemperature *T trip setpoint calculations used in the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power event analyses.

26. What is the minimum possible reactivity insertion rate for Callaway? Provide analysis results for this minimum possible reactivity insertion rate, assuming various initial power levels and minimum reactivity feedback, to show that the pressurizer would not fill before the reactor is tripped.

27. Section 6.3.10.4 states, "The [reactor coolant system] RCS pressure safety analysis limit of 2,748.5 psia is confirmed to be met via a generic evaluation." Cite the generic evaluation and verify that the evaluation (1) does not credit operation of spray or power operated control valves (PORVs), and (2) applies to Callaway with the RSGs.

Section 6.3.14 Inadvertent Actuation of ECCS at Power (analysis):

28. How was the current licensing basis for the Inadvertent Actuation of Emergency Core Cooling System (ECCS) at Power event established (i.e., by 10 CFR 50.59 evaluation or by staff review and approval). If by 10 CFR 50.59 evaluation, provide a copy of the evaluation. If by staff review and approval, then please cite the license amendment.

29. The pressurizer PORVs are predicted to open when the pressurizer is water-solid. Are they expected to reseat properly?

    a. If yes, then (1) state how they, and their associated discharge piping, have been qualified for water relief during an Inadvertent Actuation of ECCS at Power event, (2) verify that the automatic control circuitry of these valves meets Class 1E requirements, and (3) indicate the PORV opening setpoint, setpoint tolerance, and surveillance requirements for operation under water relief conditions.

b. If no, then explain how the analysis results demonstrate that the event will not develop into a more serious event, e.g., a SBLOCA.

30. Provide the ECCS flow delivery rate, as a function of RCS pressure, that was assumed for the Inadvertent Actuation of ECCS at Power event analysis. Compare this flow delivery rate to that assumed in the current licensing basis analyses.

Section 6.3.15 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (evaluation):

31. Section 15.5.2 of the Callaway Final Safety Analysis Report (FSAR) concludes that the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory will not fill the pressurizer before the operator can terminate the transient, at about 15 minutes. Provide information to demonstrate that the operator can terminate the transient, as the result of following established procedures, before the calculated pressurizer fill time.

Section 6.3.17 Anticipated Transients Without SCRAM (analysis):

32. The Anticipated Transients Without SCRAM (ATWS) analyses of (1) WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," dated August 1974, and (2) Westinghouse letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," dated December 1979, are based upon the design parameters of Westinghouse steam generator series 44 and 51, and models D and F, not the Framatome Model 73/19T design. Explain how the referenced Westinghouse ATWS analyses and methods apply to Callaway, as equipped with the RSGs.

33. Section 6.3.0.6 indicates that, "LOFTRAN remains the system transient code for the analyses of ..... the anticipated transients without scram (ATWS, FSAR 15.8)." LOFTRAN has a single-node steam generator shell side model. Describe how LOFTRAN models the primary-to-secondary heat transfer rate as the shell-side water level drops and the steam generator tubes are exposed, for the Framatome Model 73/19T steam generator design.

34. Provide transient plots and sequence of events tables denoting the time and value of peak RCS pressure for analyses of the Callaway Loss of Feedwater and Loss of Load ATWS events, assuming the design characteristics of the Framatome Model 73/19T RSGs.

35. List and explain the analysis assumptions (e.g., moderator temperature coefficient and initial steam generator mass) used in the ATWS analysis.

36. Verify that the maximum differential pressure, across the tubesheet and tubes of the Framatome Model 73/19T RSGs, matches or exceeds the value listed in Appendix C of WCAP-8330.

Section 6.4 Steam Generator Tube Rupture (analysis):

37. Show that all assumed operator action times are verified in Callaway simulator exercises.

38. What is the tube rupture size (in sq ft or sq in) that is analyzed? Are all tubes in the RSGs of the same diameter?

39. Confirm that there is not an intermediate value of initial average temperature (T_{avg}) that will produce more severe results for the SG tube rupture event.

40. Why are pressurizer heaters not assumed to be operating prior to reactor trip?

41. Section 6.4.1.2 states that, "Feedwater isolation is completed 4.3 seconds after reactor trip/SI." What is the basis for the 4.3 second value and why is this time not 17 seconds, like all the other applicable events listed in Table 6.3-6? Why is the SG Tube Rupture event not listed in Table 6.3-6?

42. One of the assumptions in Section 6.4.2.2 is, "Additional active failure: The ruptured steam generator’s safety valve fails partially open (5-percent effective area) after water relief." If the steam generator’s safety valve is not qualified for water relief, then the valve would be assumed to stick open, as a consequential failure, not as an additional active failure. What are the analysis results of this case, assuming a steam generator safety valve that sticks open following water relief?

43. Does the steam generator replacement affect the system dynamics and the timing for boric acid precipitation following large break LOCAs. Explain.

44. In the SBLOCA analyses, while the PCT for the worst break is low, review of the analysis results suggests that a break size between 3 and 4 inches would be more limiting. Since accumulator injection terminates the clad temperature rise for the 4-inch break, a slightly smaller break wherein the RCS pressure just remains of the accumulator actuation pressure could produce a higher PCT. Even though the two-phase level would be higher than that for the 4 inch break, uncovery would persist for a potentially much longer period of time causing the PCT and clad oxidation percentage to increase beyond that for the 4 inch break. Based on these considerations, explain the rationale for assuring the 4-inch break is the limiting break when accumulator injection terminates the clad heat-up.

45. Table 6.2.2-4 identifies loop seal clearing times for each break. Identify the loop seals that cleared and was there any residual water predicted to remain in any of the loop seal piping horizontal sections. Explain.

Containment Integrity Review:

46. In the response to question 2.I of the licensee's submittal dated February11, 2005, it is stated that "the values for containment volume, heat sink areas, and ... included conservatisms." Provide examples of these conservatisms.

DRAFT RESPONSES TO REACTOR SYSTEMS BRANCH QUESTIONS

PROVIDED BY LICENSEE FOR MAY 18. 2005, MEETING

ENCLOSURE 3

DRAFT RESPONSES TO REACTOR SYSTEMS BRANCH QUESTIONS

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO THE CALLAWAY STEAM GENERATOR REPLACEMENT

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

By letter dated September 17, 2004, Union Electric Company (the licensee) requested NRC approval for changes to the Technical Specifications (TSs) for the Callaway Plant, Unit 1 (Callaway) to support the installation of the replacement steam generators (RSGs) in the Fall of 2005 in Refueling Outage 14. Based on its review of the licensee's application dated September 17, 2004, in the areas of instrumentation and controls, and reactor systems, the NRC staff requests the following additional information.

[The numbering of questions follows the questions listed in the email dated April 22, 2005. Some of the information provided by the licensee is proprietary, as was stated in the notice for meeting. This information is not provided here. Only the responses to the Reactor Systems Branch questions are provided.]

Reactor Systems Branch (RSB) Review

General

7. State (a) is the RETRAN-2 nodalization used in the Callaway RSG analysis in agreement with the corresponding nodalization in WCAP-14882-P-A and (b) does the analysis for the RSG using the ANC and the VIPRE codes satisfy the staff limitations on these codes. Explain if there are deviations.

Response to question 7:

(a) The nodalization modeling used for the Callaway non-LOCA analyses is consistent with that of WCAP-14882-P-A; no changes to the generic nodalization model were made.

(b) ANC Code: ANC (WCAP-11596-P-A and WCAP-10965-P-A) is a PWR neutronics code. Analysis for the Callaway RSG does not impact the application of ANC and is within SER compliance.

VIPRE: In the SER on WCAP-14565 (Sung, Y. et al., “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” WCAP-14565-P-A, October 1999), the NRC concludes that the Westinghouse VIPRE model is acceptable for licensing applications provided that a plant specific application meets four conditions. The compliance with those conditions is addressed below.

VIPRE SER Condition 1: Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

Compliance:

The CHF correlations, the plant specific hot channel factors for enthalpy rise and other fuel-dependent parameters that have been used in DNBR analyses for Callaway RSG have all been previously used and approved.

VIPRE SER Condition 2: Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

Compliance:

The core boundary conditions for the VIPRE calculations are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE as discussed in the safety evaluation. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in WCAP-9272/9273.

VIPRE SER Condition 3: The NRC staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using the WRB-1, WRB-2 and WRB-2M correlations. The DNBR limits for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.

Compliance:

The WRB-2 and W-3 correlations used for Callaway RSG have been previously approved by the NRC. No other CHF correlations are used.

VIPRE SER Condition 4: Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC staff's generic review of VIPRE did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.

Compliance:

For the Callaway RSG, the use of VIPRE in the post-CHF region is limited to the peak cladding temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embattlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient;
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation;
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further ensured with the following input:

- Fuel rod input based on the maximum fuel temperature at the given power;
- The hot spot power factor was equal to or greater than the design linear heat rate;
- Uncertainties were applied to the initial operating conditions in the limiting direction.

8. Section 4.2.2 discusses the SG blowdown system. The minimum full-load steam pressure could be as low as 867 psia i.e. 41 psi lower than the current full load pressure of 908 psia. This decrease in blowdown system inlet pressure will impact the required maximum lift of the blowdown flow control valves. Justify that the valves are adequate for the range of the NSSS design parameters approved for the RSG. Provide the document MP 00-1013 which discusses the blowdown control valves.

Response to question 8:

As discussed during the April 14, 2005 telecon with NRC staff, Callaway Modification Package (MP) 00-1013 is a large engineering document that covers the component modifications for the RSG project. MP 03-1028, which implements the design, fabrication, and field work required to disconnect the blowdown piping from the original steam generators and reconnect the blowdown piping to the replacement steam generators, is referenced in MP 00-1013. The following information from MP 03-1028 is provided regarding the impact on the Steam Generator blowdown system and flow control valves:

> *The scope of this modification does not affect any operating parameters for the SGBS. The flow controlling throttle valves will not need to have their operating positions change. The estimated difference in pressure drop between the existing and the proposed rerouted section of piping is only about 3.7 psid out of the overall pressure drop of 950 psid (assuming 150 psia flash tank pressure, 1100 psia steam pressure). This resistance change of 0.4% will not be apparent to the operators nor will it affect existing procedures. Accordingly, the operation of the SG blowdown system will be identical in form, fit and function with the Replacement Steam Generator as existed with the Original Steam Generator.*

In addition, the operating experience for the Steam Generator Blowdown System was reviewed. The SG blowdown flow control valves are currently positioned at 2-3 turns from closed at full power. The full open position is approximately 5 turns. The current differential pressure across the throttle valve is approximately 800 psid. For the RSG, the minimum differential pressure across the throttle valve will be approximately 700 psid. Based on this and review of the differences between the original steam generators and replacement steam generators, it is expected that the blowdown flow control valves have plenty of available lift to allow operation at the minimum full-load steam pressure.

9. Table 2-1 lists the thermal design parameters for RSG conditions. Please provide a tabulation of the expected operating and safety analyses values assumed in the safety analyses to demonstrate that the safety analyses assumptions are adequate or conservative.

Response to question 9:

The design parameters listed in Table 2-1 are the bounding set of conditions set for all the NSSS analyses that were performed. They establish a range of possible operating conditions (i.e., T-avg between 570.7°F and 588.4°F and steam generator tube plugging between 0 and 5%) that provide adequate conservatism for the analyses while at the same time providing operating flexibility. The safety analyses then use the most conservative set of design parameters for their analysis, based on which conditions are known to be conservative for that analysis. If it was not clear what case would be most limiting, the analysis was performed for more than one case and the most limiting case was documented in LAR Appendix A. Where appropriate, uncertainties were also added onto the nominal values assumed (for instance, T-avg uncertainty applied to the nominal T-avg used from Table 2-1) to make sure the analysis was conservative.

The specific design parameters used in each of the safety analyses can be found in the sections of LAR Appendix A (WCAP-16265) as listed below:

- Large-Break LOCA: Section 6.2.1 and Table 6.2.1-1.
- Small-Break LOCA: Section 6.2.2 and Table 6.2.2-1.
- Non-LOCA Transients: Section 6.3 and Tables 6.3-3 and 6.3-4. More details on the design parameters used in each event-specific analysis are also provided within each sub-section of 6.3.
- Steam Generator Tube Rupture: Sections 6.4.1.2 and 6.4.2.2.

The plant is limited to operation within the bounding design conditions specified in Table 2-1 such that the safety analyses will adequately cover the actual plant operation with the RSGs.

10. Discuss the effects of the RSG conditions on the station blackout coping analysis for Callaway.

Response to question 10:

The Callaway condensate storage tank (CST) is the preferred non-safety related source of water to the steam generators. The steam generators must remove decay and the sensible heat. The essential service water (ESW) system provides the safety-related source of water for the steam generators. Technical Specification (TS) 3.7.6 states that the CST must store a minimum of 281,000 gallons to be OPERABLE in MODES 1, 2 and 3. The TS Bases for LCO 3.7.6 indicates that the useable volume must be sufficient to remove decay heat and cooldown the reactor coolant system (RCS) during a four-hour SBO coping period. The required water volume for this event is 158,000 gallons over the four-hour coping period based on an RCS cooldown to a MODE 3 temperature less than 406‹F.

The summary of the Callaway SBO Coping Assessment is included in FSAR Appendix 8.3A. The following subsections are specifically addressed:

- 8.3 A.5.1 Condensate Inventory for Decay Heat Removal. The CST inventory capacity requirement for the RSG metal mass and current licensed power (decay heat) is bounded by the Technical Specification requirement.
- 8.3A5.2 Class 1E Battery (ies) Capacity is not impacted by RSG implementation. The previous determination regarding emergency batteries remains valid.
- 8.3A5.3 Compressed Air is not impacted by RSG implementation. The previous determination regarding compressed air remains valid.
- 8.3A5.4 Effects of Loss of Ventilation is not impacted by RSG implementation. The previous determination regarding ventilation remains valid.
- 8.3A5.5 Containment Isolation is not impacted by RSG implementation. The previous determination regarding containment isolation remains valid.
- 8.3A5.6 Reactor Coolant Inventory is not impacted by RSG implementation. The previous determination regarding reactor coolant inventory remains valid.

Summary

The only impact from RSG implementation to the SBO Coping Assessment is the increase in sensible heat from the increased RSG metal mass. The overall impact from increased sensible heat due to RSG implementation is minimal, since decay heat dominates the cooling requirements during an SBO event. The effects of the RSG project are bounded by the current licensing bases (FSAR Appendix 8.3A.5 and NUMARC 87-00).

<u>Low Temperature Overpressure (LTOP) Protection</u>

11. Section 4.3.5 discusses low-temperature overpressure protection. It is stated that the design-basis Mass Input (MI) transient remains applicable for the RSG program and Heat Input (HI) transient reanalysis was performed. The revised LTOP setpoints were developed using the results of HI transient and the current MI results. Provide the results and assumptions (temperature difference between SG and RCS prior to start a RCP) used in the reanalysis of HI transient.

    Describe the basis for the wide-range pressure uncertainty for LTOP which has increased from 85 to 93 psig for the RSG program and its effects on LTOP design setpoints.

    The MI transient is limiting for RCS temperature less than about 180 degree F. The current LTOP arming temperature of 275 degree F remains applicable for the RSG program. Describe which transient is more limiting between 180 degree F and 275 degree F and its effects on LTOP design setpoints.

Response to question 11:

At Callaway, in lieu of "LTOP" we use the term Cold Overpressure Mitigation System (COMS) as specified in TS LCO 3.4.12.

The temperature difference between the SG and the RCS prior to an RCP start assumed in the Heat Input (HI) transient re-analysis was 50‹F, the same as the current design basis heat input for COMS analysis. This temperature difference is the same as that specified in LCO Notes for TS 3.4.6 and TS 3.4.7 for RCP startup.

The results of the HI transient reanalysis showed that the current COMS maximum allowable setpoints remain bounding with the RSG. The current COMS arming temperature also remains valid for RSG.

Callaway-specific instrumentation, procedures, and practices were reviewed and uncertainty terms were determined for the wide range pressure transmitters and process racks. These terms were combined using the Westinghouse Square Root Sum of the Squares (SRSS) methodology. The Callaway system uses a diverse instrumentation system which has both a Barton transmitter channel and a Veritrak transmitter channel. Both channels were evaluated and the most limiting/conservative channel uncertainty was used as input to the COMS setpoint. The resultant uncertainty for the Veritrak channel is 3.1 % of span and for the Barton channel is 2.5 % span. For a 3000 psig span this corresponds to 93 psig and 75 psig, respectively. Therefore, 93 psig was used as the limiting input to the COMS setpoint.

The Heat Input (HI) transient was slightly more limiting between the RCS temperatures of 180‹F and 275‹F for the COMS setpoints analysis. As an example, the maximum allowable PORV setpoint based on the Mass Input (MI) transient was calculated as 759 psig while it was 758 psig based on the HI transient at an RCS temperature of 200‹F. Similarly, the maximum allowable PORV setpoint was 759 psig for MI but it was 750 psig for HI at an RCS temperature of 250‹F. As stated above, the calculated values bound the maximum allowable PORV setpoints specified in the current Pressure and Temperature Limits Report (PTLR), which is not being revised for the RSG project.

12. The SG replacement package does not discuss the effect of the SG replacement on overpressure protection analyses (Standard Review Plan (SRP) 5.2.2, Section II, A). It is apparent that RSGs would have some effect on overpressure scenarios. Identify the present Callaway overpressure protection basis (is this WCAP-7769) and discuss the impact, if any, of the SG replacement on this basis.

Response to question 12:

SRP 5.2.2 specifically requires that the second safety grade reactor trip signal be credited for safety valve sizing calculations. This is consistent with the safety valve sizing procedure discussed in Section 2 of WCAP-7769. WCAP-7769 states, "For the sizing, main feedwater flow is maintained and no credit for reactor trip is taken." This analysis is typically performed prior to construction of the plant to provide a basis for the capacity requirements for the safety valves and the requirement of SRP 5.2.2 provides a conservative basis for the number and design of the valves.

However, WCAP-7769 goes on to say, "After determining the required safety valve relief capacities, as described above, the loss of load transient is again analyzed for the case where main feedwater flow is lost when steam flow to the turbine is lost... For this case, the bases for analysis are the same as described above except that credit is taken for Doppler feedback and appropriate reactor trip, other than direct reactor trip on turbine trip." This describes the analysis performed in Chapter 15 of the FSAR which verifies that the overpressure limits are satisfied with the current/latest design.

The analyses performed in support of the Callaway RSG Program are not safety valve sizing calculations - no changes are being made to the safety valves as a result of this program. The Loss of External Electrical Load / Turbine Trip analysis performed for the RSG Program, presented in Section 6.3.4, demonstrates that the safety valves are adequately sized to maintain peak primary pressure below 110% of design which satisfies the requirements of GDC-15. GDC-15 applies to "any condition of normal operation, including anticipated operational occurrences" which does not include a common mode failure of the first safety grade reactor trip signal.

WCAP-7769 remains a valid FSAR reference in that it discusses the methodology used for the FSAR Loss of External Electrical Load / Turbine Trip analysis for Callaway. The Loss of External Load / Turbine Trip RCS overpressure analysis is performed to demonstrate that in the event of a sudden loss of the secondary heat sink, the associated increase in reactor coolant system temperature does not result in overpressurization of the RCS system.

<u>Loss of Coolant Accident (LOCA) Analyses</u>

13. The SG replacement package does not discuss the effect of the SG replacement on the Callaway LOCA analyses. Page 22 of 51 of Attachment 1 indicates that LOCA analyses are not affected by the new SGs. However, the discussion on pages 19 and 20 of 51 of Attachment 1 indicates that the calculated peak containment pressure for a LOCA event is greater than 1 psi lower with the new SGs than previously analyzed. This would seem to increase the calculated LBLOCA peak cladding temperature. Discuss the effect of the SG replacement on LBLOCA events.

Response to question 13:

As discussed during the 4/14/05 telecon between NRC staff and AmerenUE, the RSG License Amendment Request (LAR) Attachment 1 page 22 of 51 covers only the elimination of the Trip Time Delay (TTD), as denoted by the title of Section 4.2 of Attachment 1. The noted page, therefore, states that the LOCA analyses are not affected by TTD elimination. For a discussion of the effect of the SG replacement on LBLOCA events, refer to Section 6.2 of LAR Appendix A and the RAIs hereafter numbered as questions 14 - 17 and 44 - 46.

14. The Callaway submittal (WCAP-16265, Section 6.2) provides a summary of LOCA analysis parameters. This submittal also refers to information specific to the large break LOCA (LBLOCA) analyses performed to define the licensing basis for the Callaway LBLOCA. The staff requests the following information to address the programmatic requirements of 10 CFR 50.46(c). To show that the referenced generically approved LOCA analysis methodologies continue to apply specifically to the Callaway plant,

provide a statement that Ameren UE and its vendor have ongoing processes which assure that the ranges and values of the input parameters for the Callaway LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters. Furthermore, if the Callaway plant-specific analyses are based on the model and or analyses of any other plant, then justify that the model or analyses apply to Callaway (e.g. if the other design has a different vessel internals design the model wouldn't apply to Callaway).

Response to question 14:

AmerenUE and Westinghouse follow a process described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985, through which the operating parameters are assured of being within the parameters and conditions assumed in the accident analyses. WCAP-9272-P-A is cited in Technical Specification 5.6.5.b. This process is called the Reload Safety Evaluation (RSE) process which is completed prior to each operating cycle at Callaway. AmerenUE identifies the planned operating conditions, current plant configuration, and current Operating License status including any approved amendments for the upcoming cycle in a Reload Safety and Licensing Checklist (RSLC). Based on that input and the cycle energy requirements, Westinghouse completes a Reload Safety Analysis Checklist (RSAC) and a Reload Safety Evaluation (RSE) that demonstrates the operating conditions are bounded by the assumptions in the safety analyses of record for Callaway. If necessary, additional safety analyses are performed to ensure that they are conservative relative to the plant operating conditions. During development of the RSE, which includes an evaluation of the new core design against the requirements of 10 CFR 50.59, specific attention is given to methodology changes that require prior NRC review and approval. AmerenUE controls on the reload design process are contained in procedure EDP-ZZ-00014, "Reload Design Control and Coordination."

The Callaway plant-specific LOCA analyses are not based on the model or analyses of any other plant. The analyses are Callaway-specific.

15. The Callaway LOCA submittals did not address slot breaks at the top and side of the cold leg discharge pipe. Justify why these breaks are not considered for the Callaway LOCA submittal.

Response to question 15:

References 15-1 through 15-4 cited in the response to question 15 are identified at the end of the response.

Break location, type and size were considered during the development of the Westinghouse LBLOCA Evaluation Models (EM). It was concluded that cold leg guillotine breaks are limiting for Westinghouse plant designs which utilize the Appendix K EMs. As such, only cold leg guillotine type breaks are specifically analyzed in the calculation of the LBLOCA PCT with the BASH EM.

For Small Break LOCA (SBLOCA) events, the effects of break location have been generically evaluated as part of the application of the NOTRUMP Evaluation Model (Reference 15-1). This document concluded that a break in the Reactor Coolant System (RCS) cold leg was limiting.

Additionally, the effects of break orientation were considered during the evaluation of Safety Injection in the Broken Loop and application of the COSI Condensation Model (Reference 15-2). This work concluded that a break oriented at the bottom of the RCS cold leg piping was limiting with respect to Peak Cladding Temperature (PCT).

While these references specifically address the short-term response to the LOCA break spectrum, the long-term effects associated with potential Reactor Coolant Pump (RCP) suction crossover leg (loop seal) re-plugging core uncovery is addressed in the following.

A review of the analysis conditions associated with potential core uncovery due to loop seal re-plugging has previously been performed in Reference 15-3. Reference 15-3 documents the Westinghouse position with regards to the potential for Inadequate Core Cooling (ICC) scenarios following Large and Intermediate Break LOCAs as a result of loop seal re-plugging. Reference 15-3 concludes the following:

- *The reactor coolant system response following a LOCA is a dynamic process and the expected response in the long term is similar to the response that occurs in the short term. This short term response has been analyzed extensively through computer analysis and tests and is well documented.*

- *Consideration of the physical mechanisms for liquid plugging of the pump suction leg U-bend piping following large and intermediate break LOCAs at realistic decay heat levels precludes quasi steady-state inadequate core cooling conditions.*

- *It is important to emphasize that the operator guidance provided in the Emergency Response Guidelines includes actions to be take in the event of an indication of a challenge to adequate core cooling following a LOCA.*

A review of the key contributors associated with long-term loop seal plugging core uncovery scenarios, under LOCA conditions, was performed as part of Reference 15-4 including a review of pertinent experimental data.

[Proprietary Information]

From References 15-3 and 15-4 it can be concluded that post-LOCA core uncovery scenarios as a result of loop seal re-plugging do not constitute a significant concern to Callaway plant safety.

<u>References</u>:

15-1. WCAP-11145-P-A, “Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With the NOTRUMP Code”, S. D. Rupprecht, et al., 1986.

15-2. WCAP-10054-P Addendum 2, Revision 1, “Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model”, C. M. Thompson, et al., July 1997.

15-3. OG-87-37, "Westinghouse Owners Group (WOG) Post LOCA Long Term Cooling, Letter from Roger Newton (WOG) to Thomas Murley (NRC)", August 26, 1987.

15-4. NSD-NRC-97-5092, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery," Letter from N. J. Liparulo (W) to NRC, March, 1997.

16. The Callaway submittal states that the Westinghouse LOCBART extension method was used to address downcomer boiling for Callaway. This method has not been approved at this time for generic application. Show that this method is applicable to Callaway by:

a. Provide comparative results (peak clad temperature (PCT) vs time graph) for the most limiting Callaway LOCA scenario using the LOCBART extension method and WCOBRA/TRAC.

b. Provide the LBLOCA analysis results tables and graphs to at least 1600 seconds to show that stable and sustained quench is established.

c. Indicate on the graph the period of downcomer boiling for each calculation.

Response to question 16:

References 16-1 through 16-4 cited in the response to question 16 are identified at the end of the response.

a. A WCOBRA/TRAC model for Callaway is not available; please see the response to question 16.c for related information.

b. Please see the response to question 16.c.

c. The BASH Evaluation Model (BASH-EM) as described in Reference 16-1 was designed to analyze [Proprietary Information].

The LOCBART Transient Extension Method (Reference 16-2) was utilized in the Callaway replacement steam generator (RSG) BASH-EM analysis to support the demonstration of transient termination. In order to address request for additional information (RAI) #16, the cases that produced the maximum peak cladding temperature and maximum local oxidation have been further extended to hot assembly quench. The latest version of LOCBART was used to incorporate the effects of 10 CFR 50.46 changes made since the original analysis was completed, resulting in a very small increase in peak cladding temperature and maximum local oxidation to 1939°F and 6.5%. These results supersede the results reported on page 6-6 of LAR Appendix A (1938°F and 6.4%) and these revised values will be used to establish the analysis-of-record results for the large break LOCA analysis.

The enclosed Figures 16-1 through 16-3 show the hot rod peak cladding temperature, the hot rod cladding temperature at the peak cladding temperature elevation, and the hot rod local oxidation at the maximum oxidation elevation for the maximum peak cladding temperature case. Figures 16-4 through 16-6 show the same parameters for the maximum local oxidation case. Downcomer boiling begins at [proprietary information] for the maximum peak cladding temperature case and [proprietary information] for the maximum local oxidation case, as denoted by the open squares on each figure. Hot assembly quench occurs at [proprietary information] for the maximum peak cladding temperature case and [proprietary information] for the maximum local oxidation case.

Reference 16-3 describes BASH-EM and Best Estimate LOCA (BELOCA) calculations for a 3-loop plant with a dry atmospheric containment and a 4-loop plant with an ice condenser containment that demonstrate significant conservatism in BASH-EM relative to a best estimate plus uncertainties approach. These comparisons were based on the original BELOCA Code Qualification Document (CQD) methodology (Reference 16-4), and additional margin would be anticipated from applying the recently approved ASTRUM methodology (Reference 16-5). While the exact magnitude of the differences between BASH-EM and BELOCA may vary somewhat for Callaway, the overall conclusion that BASH-EM is significantly conservative would remain applicable.

Based on the preceding information, extending the Callaway BASH-EM analysis beyond the onset of downcomer boiling [proprietary information]. Even with the substantial conservatism that exists in BASH-EM relative to a best estimate plus uncertainties approach, significant margin has been retained to the applicable acceptance limits of 10 CFR 50.46. Based on these results, it is concluded that the LOCBART Transient Extension Method is applicable to Callaway and can be used to perform an engineering assessment of transient termination for the RSG LBLOCA analysis without requiring prior NRC approval of the proposed generic revision to the BASH-EM. As such, the licensing basis LBLOCA methodology continues to be WCAP-10266-P-A, Revision 2, March 1987 as cited in TS 5.6.5.b.3.

References:

16-1. WCAP-10266-P-A, Revision 2, “The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code,” March 1987.

16-2. WCAP-10266-P-A, Revision 2, Addendum 3, “Incorporation of the LOCBART Transient Extension Method into the 1981 Westinghouse Large Break LOCA Evaluation Model with BASH (BASH-EM),” December 2002.

16-3. LTR-NRC-05-2, “Presentation Material for January 25, 2005 Meeting Regarding WCAP-10266-P-A, Revision 2, Addendum 3 (Proprietary/Non-Proprietary),” January 24, 2005.

16-4. WCAP-12945-P-A Volume I (Revision 2) and Volumes II-V (Revision 1), “Code Qualification Document for Best Estimate LOCA Analysis,” March 1998.

16-5. WCAP-16009-P-A, “Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” January 2005.

Figures 16-1 to 16-6: [These figures contain proprietary information and are not provided.]

17. Tables 6.2.1-5 and 6.2.2-5 of WCAP-16265, Section 6.2 (Attachment 1), provide LBLOCA and small break LOCA (SBLOCA) analyses results (PCT, maximum local oxidation, and total hydrogen generation) for the Callaway RSG. For maximum local oxidation include consideration of both pre-existing and post-LOCA oxidation, and cladding outside and post-rupture inside oxidation. Also include the limiting local oxidation results for fuel resident from previous cycles.

Response to question 17:

The results (peak cladding temperature, maximum local oxidation and total hydrogen generation) for the Callaway LBLOCA and SBLOCA RSG analyses are provided in Table 17-1 below. (The SBLOCA results are obtained from the RSG submittal, and the LBLOCA results are obtained from the response to RAI #16.) Additional information regarding the bases for the maximum local oxidation, including consideration of both pre-existing and post-LOCA oxidation, cladding outside and post-rupture inside oxidation is discussed below.

Table 17-1 Callaway RSG LOCA Analysis Results

	LBLOCA	SBLOCA
Peak Cladding Temperature	1939°F	1043°F
Maximum Local Oxidation	Pre-transient = 0% Transient = 6.5%	Pre-transient = 0% Transient = 0.02%
Total Hydrogen Generation	<1%	<1%

<u>LBLOCA</u>

As shown in Table 17-1, the transient maximum local oxidation calculated for the Callaway RSG large break LOCA analysis is 6.5%. This transient maximum local oxidation was predicted to occur at the burst elevation, such that the metal-water reaction occurred on both the inner and outer cladding surfaces.

The maximum local oxidation was calculated for fresh fuel, at the beginning of the cycle. This represents the maximum amount of transient oxidation that could occur at any time in life. As burnup increases, the transient oxidation decreases for the following reasons:

1) The cladding creeps down towards the fuel pellets, due to the system pressure exceeding the rod internal pressure. This will reduce the initial stored energy at the hot spot by several hundred degrees relatively early in the first cycle of operation, and will tend to reduce the transient oxidation.

2) Later in life, the clad creep-down benefit still remains effective. In addition, with increasing irradiation, the power production from the fuel will naturally decrease as a result of depletion of the fissionable isotopes. Reductions in achievable peaking factors in the burned fuel relative to the fresh fuel are realized before the middle of the second cycle of operation. The achievable linear heat rates decrease steadily from this point until the fuel is discharged, at which point the transient oxidation will be negligible.

The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for each of the fuel designs that will be included in the RSG cores is < 16%. The actual upper bound values predicted for the RSG fuel design are expected to be well below this value.

Based on the above discussion, the transient oxidation decreases from a maximum of 6.5% at BOL to a negligible value at EOL, while the pre-transient oxidation increases from zero at BOL to a conservative maximum of <16% at EOL. Additional BASH-EM calculations were performed at intermediate burnups, accounting for burnup effects on fuel performance data (primarily initial stored energy and rod internal pressure). These calculations support the conclusion that the sum of the transient and pre-transient oxidation remains below 16% at all times in life. This confirms Callaway conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

<u>SBLOCA</u>

As part of the Callaway RSG, a new SBLOCA analysis was performed resulting in a peak cladding temperature of 1043°F. Because of low clad temperatures, fuel rod burst was not predicted to occur, and the maximum transient oxidation was only 0.02%.

The pre-transient oxidation increases with burnup, from zero at beginning of life (BOL) to a maximum value at the discharge of the fuel (end of life, or EOL). The design limit 95% upper bound value for each of the fuel designs that will be included in the RSG cores is < 16%. The actual upper bound values predicted for each of the fuel designs are expected to be well below this value. Because the transient oxidation is so low, the sum of the transient and pre-transient oxidation remains below 16% at all times in life. This confirms Callaway conformance with the 10 CFR 50.46 acceptance criterion for local oxidation.

18. Provide a table listing the key assumptions and input parameter values for all accident analyses in this license amendment request, before and after the proposed license amendment. Include the core kinetics assumptions used for the steam line break analyses.

Response to question 18:

Key assumptions for the safety analyses before and after the proposed amendment can be found in the attached markup to FSAR Table 15.0-2. In addition, Appendix A of the LAR provides the key input assumptions used in the various safety analyses. Specifically, the following sections of LAR Appendix A (WCAP-16265) include the key inputs for those analyses:

- Large-Break LOCA: Section 6.2.1 and Table 6.2.1-1
- Small-Break LOCA: Section 6.2.2 and Table 6.2.2-1
- Non-LOCA Transients: Section 6.3 and Tables 6.3-3 and 6.3-4. More details on the input assumptions used in each event-specific analysis are also provided within each sub-section of 6.3.
- Steam Generator Tube Rupture: Sections 6.4.1.2 and 6.4.2.2

Additional information regarding the non-LOCA analyses is provided hereafter.

The non-LOCA safety analysis methodology used in the Callaway RSG analyses models the various control systems only when their operation yields worse (more limiting) transient results. For those events where operation of control systems may be of importance to the transient, a discussion of assumptions made regarding the control systems is presented in the Method of Analysis section specific to each event.

LAR Appendix A Table 6.3-3 presents the plant-specific initial condition values assumed in the Callaway RSG non-LOCA analyses. Nominal values are presented for those events analyzed using the RTDP DNB methodology since the initial condition uncertainties are statistically included in the DNBR safety analysis limit. Limiting (low and high end) values, including initial condition uncertainties, are also presented for events where the RTDP methodology is not employed. Table 6.3-4 summarizes the initial condition assumptions made on an event-specific basis, also identifying whether RTDP methodology was used or not.

Specific to the Inadvertent ECCS Actuation at Power event, the case analyzed to investigate the event's DNBR transient assumes a nominal high T-avg value (588.4ºF) because RTDP was used, and since a high T-avg value is always conservative with respect to DNBR. Linked to this initial T-avg value is an initial pressurizer water level of 65% span (60% nominal plus 5% level uncertainty).

For the pressurizer filling case, since the limiting condition could not be identified up front, two separate cases were considered and only the more limiting of the two was presented in the licensing report and FSAR. The cases considered were one where the initial T-avg was set at the nominal value of 588.4ºF minus uncertainties (lower initial T-avg maximizes RCS mass and minimizes time to pressurizer filling) and the corresponding initial pressurizer water level of 65% span (60% span nominal value plus 5% uncertainty); the second case assumed an initial T-avg set at the low nominal value of 570.7ºF minus uncertainties (the lower initial T-avg further maximizes initial RCS mass) with a corresponding initial pressurizer water level of 43% span (38% span nominal value plus 5% uncertainty; the lower initial pressurizer level offsets the higher initial mass). Both cases were analyzed and their results showed that the second case was slightly more limiting. Note that the initial condition variation combined with the transient conditions and control system assumptions affect the final results. Therefore, separate cases are modeled to ensure the most limiting conditions are analyzed. This limiting case is presented in LAR Appendix A.

For the steam line break core kinetics assumptions, see the attached mark-ups to FSAR page 15.1-17 and new FSAR Figures 15.1-11 and 15.1-14 to see how they changed with the RSG.

The core kinetics assumptions are also discussed in section 6.3.3.2 of LAR Appendix A (WCAP-16265).

19. Use the table of Section 3.1 in Attachment 1 in the application, which compares the key design parameters of the Framatome Model 73/19T RSGs to the design parameters of the old SGs (OSGs) to discuss how differences in these design parameters would affect analyses and evaluations of the Callaway accidents and transients.

Response to question 19:

The table in Section 3.1 of Attachment 1 of the application provides a comparison of the key characteristics of the two SG models. The accident analyses completed by Westinghouse did not compare the differences in SG design parameters. The RSG design parameters and conditions were modeled in the Westinghouse codes with the intent of demonstrating the applicable acceptance criteria were met. As described in LAR Appendix A (WCAP-16265), all acceptance criteria were met.

In general, it is difficult to say how specific RSG design parameters affected the analyses since the analyses also incorporated other input assumption changes (for example, revised DNBR correlation, updated auxiliary feedwater flows, etc. for the non-LOCA events) that make a direct comparison of the updated analysis results and the current analysis results (LOFTRAN-based for non-LOCA) difficult.

Overall, it can be said that the greater cooling capability of the replacement steam generators has resulted in margin gained for long-term heatup events such as the feedline break and loss of normal feedwater (with offsite power available) events. The increased primary side SG tube volume has had a slightly more limiting effect on the loss of normal feedwater event under natural circulation conditions (with loss of offsite power), as it increased the overall RCS water volume that subsequently expands into the pressurizer, reducing the margin to pressurizer filling. In any case, these effects can not be solely attributed to differences in SG design as they are also combined with other benefits and penalties resulting from other non-SG related changes.

The remaining non-LOCA events that were not explicitly reanalyzed for this RSG program were determined to either not be sensitive to SG changes or remain limiting as currently analyzed. This last scenario is that of the boron dilution event. For the boron dilution event, it was determined that the higher primary side SG tube volume would result in an increase in the total active mixing volume for the event. Since a lower active mixing volume is conservative for this event, it was determined that the current analysis of record would remain applicable to Callaway following implementation of the RSG Program.

For the steam generator tube rupture (SGTR) event, there were also other input assumptions that were changed at the same time as the steam generator design changes. So it is difficult to say how the RSG itself affected the analysis. However, it can be said that the SGTR analysis is very dependent on the flow area of the tube. Since the RSG has a larger tube inside diameter and longer tubes than the OSG, the analysis would yield worse results due to the greater amount of flow that would be forced out the break.

20. Based upon operating history at Callaway and other pressurized water reactors (PWRs), explain the benefits of removing the TTD system, and retaining the environmental allowance monitor (EAM) system, in terms of the predicted effect, if any, upon the frequency of unnecessary reactor trips at Callaway.

Response to question 20:

The Trip Time Delay (TTD) feature was originally installed at Callaway as a result of a program developed by the Westinghouse Owners Group to provide the plant operators a short time delay to manually correct and stabilize low steam generator level before an automatic reactor trip at low power levels. The replacement steam generators for Callaway are, by design, less susceptible to low power water level fluctuations and lessons learned at Callaway have shown that better and tighter control of feedwater heating will also provide better control of steam generator level. These two factors, when combined, reduce the likelihood of automatic reactor trips due to low steam generator level.

Justification for the TTD circuitry elimination is presented in Sections 3.2 (pages 7 and 8 of 51) and 4.2 (pages 20 and 21 of 51) of Attachment 1 to ULNRC-05056. Deleting the TTD circuitry will result in less design complexity and less required surveillance testing. Parts obsolescence concerns with these 7300 Process Protection System cards will be reduced by eliminating this circuitry from the design. Reduced surveillance testing will result in substantial man-hour savings since we will no longer have to verify 32 PROM logic card time delays (16 channels x 2 power level-dependent time delays) at least every 6 months during channel COTs.

Retaining the EAM circuitry will allow plant operation with SG water levels that are further distanced from the low-low level trip setpoints under normal containment environmental conditions (< 1.5 psig). This amendment application is not asking for changes to that portion of the RTS or ESFAS circuitry. The drawing changes included in the response to question #3 above clearly show that the EAM circuitry is separate from the TTD circuitry.

We do not expect any impact on reactor trip or ESFAS actuation frequency after the TTD circuitry is eliminated.

Section 6.3.3 Steam System Piping Failure Analysis

21. Section 6.3.3.1 states, “The effective throat diameter of the flow restrictor nozzles of 6.0315 inches is considerably smaller than the diameter of the main steam pipe. These restrictors are located in the outlet nozzles of the steam generators and serve to limit the maximum steam flow for any break at any location.” This is equivalent to a throat area of about 0.2 ft^2, or a much less than half the typical 1.4 ft^2 throat area of current Westinghouse steam generators with integral flow restrictors. Section 6.3.3.2 states, “Since the steam generators are provided with integral flow restrictors with a 1.39 ft^2 throat area, any rupture with a break greater than 1.39 ft^2, regardless of the location, would have the same effect on the nuclear steam supply system (NSSS) as the 1.39 ft^2 break.” Please reconcile these statements.

Note that flow restrictor nozzles with an effective throat diameter of 16.0315 inches (if the 6.0315 inch value is a typo) would have an equivalent throat area of 1.40 ft^2. Please identify the maximum steam line break size that is analyzed for Callaway, and relate it to the steam pipe and flow nozzle dimensions.

Response to question 21:

The comment is correct. The value of 6.0315 inches cited on page 6-81 of LAR Appendix A (WCAP-16265) is a typographical error. The correct value is 16.0315 inches.

The analysis presented in Section 6.3.3 does indeed model a limiting break area of 1.39 ft^2, consistent with an effective throat diameter of 16.0315 inches (the 6.0315 inch value is a typo). The remaining analysis description continues to apply.

The effective throat diameter of each flow restrictor nozzle is 6.0315 inches which is equivalent to a throat area of about 0.2 ft^2. There are seven restrictors arranged in a cluster.

22. For Section 6.3.3.2:

    (a) Explain how the coefficient assumption was revised for the RSG analysis to improve the core physics prediction of the point kinetics core model.

    (b) Explain how all reactivity physics parameters are weighted toward the core sector exposed to the greatest cooldown from the faulted loop.

Response to question 22:

(a) The analysis of the HZP SLB involves an iterative process utilizing the RETRAN and ANC codes. The process is one where the RETRAN model is first run using conservative, worst stuck rod and limiting core reactivity feedback coefficients provided by Core Design. The resultant transient core heat flux, RCS temperatures and pressure and boron concentration are then utilized in a more detailed core design physics model in ANC to determine if the RETRAN-calculated return to power is in agreement with the ANC prediction. If needed, the process is repeated until the power mismatch between the two codes falls within the acceptable limits of the detailed core design models. The final transient statepoints (core heat flux, RCS temperatures and pressure and boron concentration) are then used in the VIPRE code to calculate the DNBR transient for this event.

(b) The core fluid density used to calculate the reactivity feedback is weighted towards the core sector associated with the faulted loop. Instead of using the average of the four core sectors to calculate the overall core fluid density, RETRAN (via user inputs) assumes a higher contribution from the faulted loop sector. This results in a reduced core density, more reactivity feedback, and a higher return to power.

23. Table 6.3.3-1 indicates that, for both hot zero power (HZP) cases, the low steam line pressure setpoint is reached 2 seconds after the steam line rupture. Adding 2 seconds for signal processing and 15 seconds for main steam isolation valve (MSIV) stroke time

results in an MSIV closure time of 19 seconds; but the Table lists MSIV closure at 18 seconds. Explain how the 18 second value was determined.

Response to question 23:

Table 6.3.3-1 provides the time when the low steam line pressure setpoint is reached in both the faulted *and* unfaulted loops. The setpoint is reached at ~0.5 seconds in the faulted loop and ~2.0 seconds in the unfaulted loop. The longest time, ~2.0 seconds, is reported in Table 6.3.3-1. Main steam line isolation for all loops then occurs 17 seconds after the setpoint is reached in the faulted loop (2 second signal delay plus 15 second valve stroke time). This total time is 17.5 seconds which was rounded up to 18 seconds for the report.

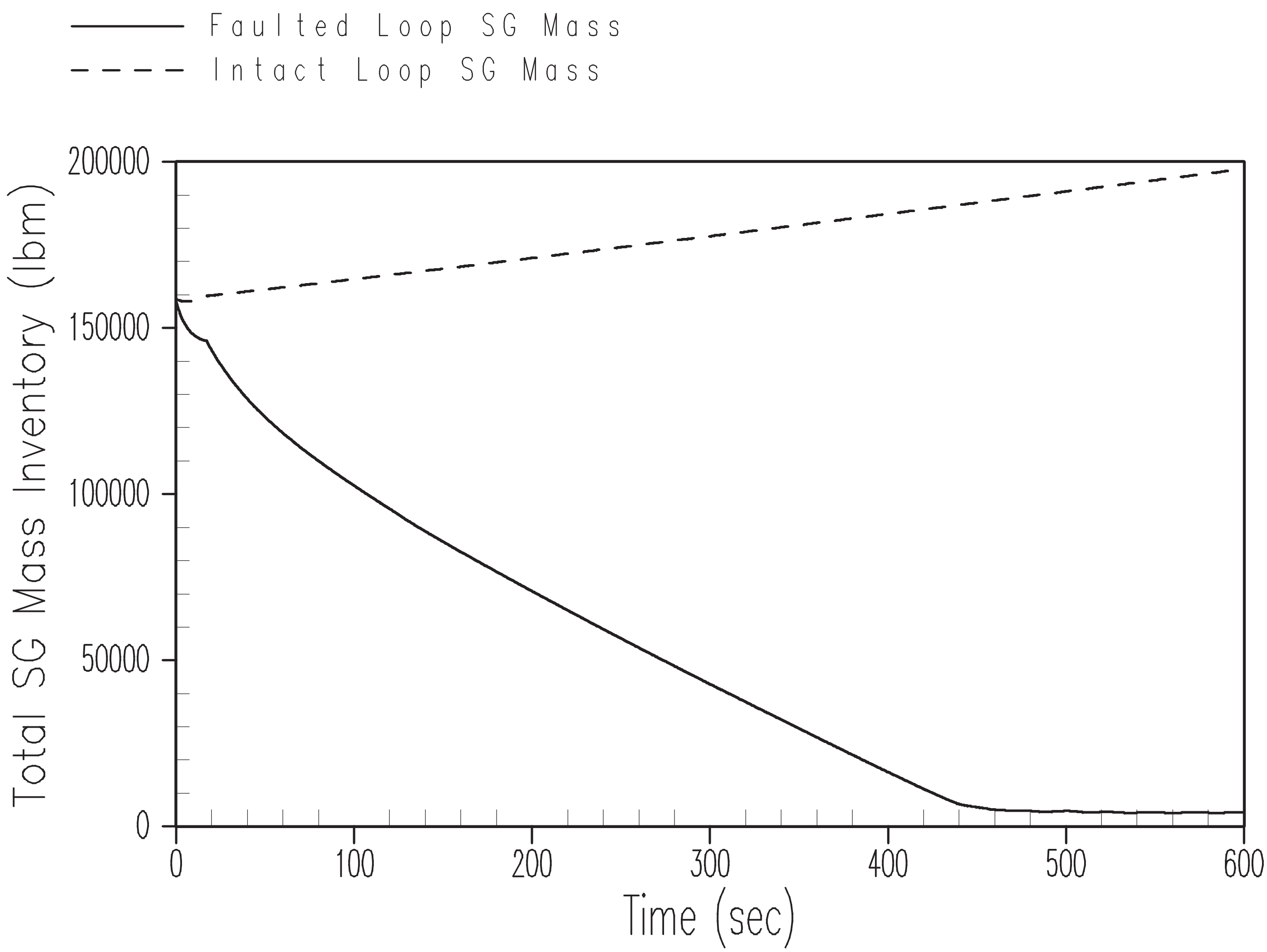
24. Provide transient plots depicting steam generator mass and auxiliary feedwater flow, for the faulted and intact steam generators, for both HZP steam line break cases.

Response to question 24:

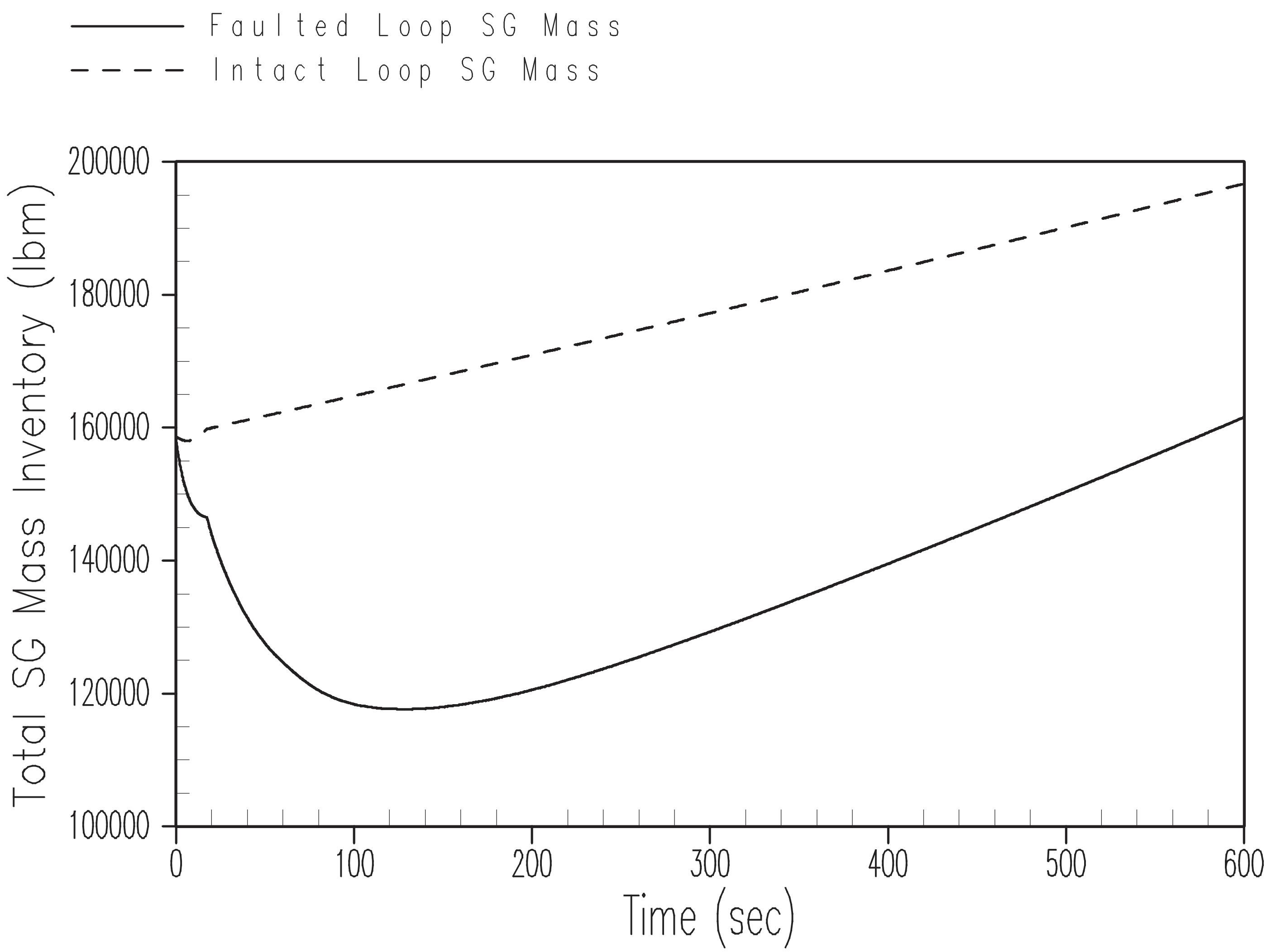
The requested transient plots for SG mass and AFW flow are provided below [on the next three pages].

[The response to Question 25 begins on page 23.]

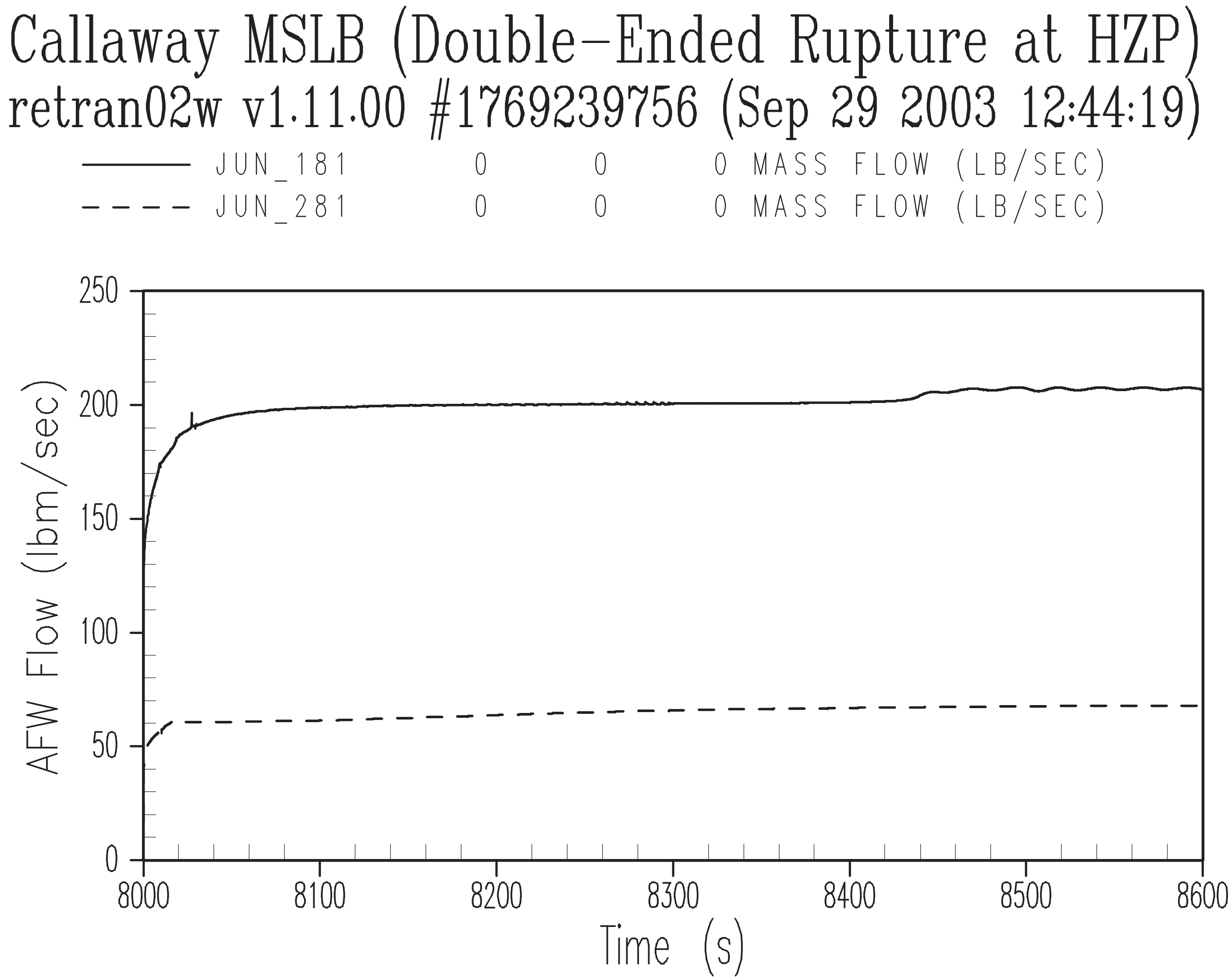
HZP Steamline Break Transient with Offsite Power Available, Double-Ended Rupture
Total SG Mass versus Time



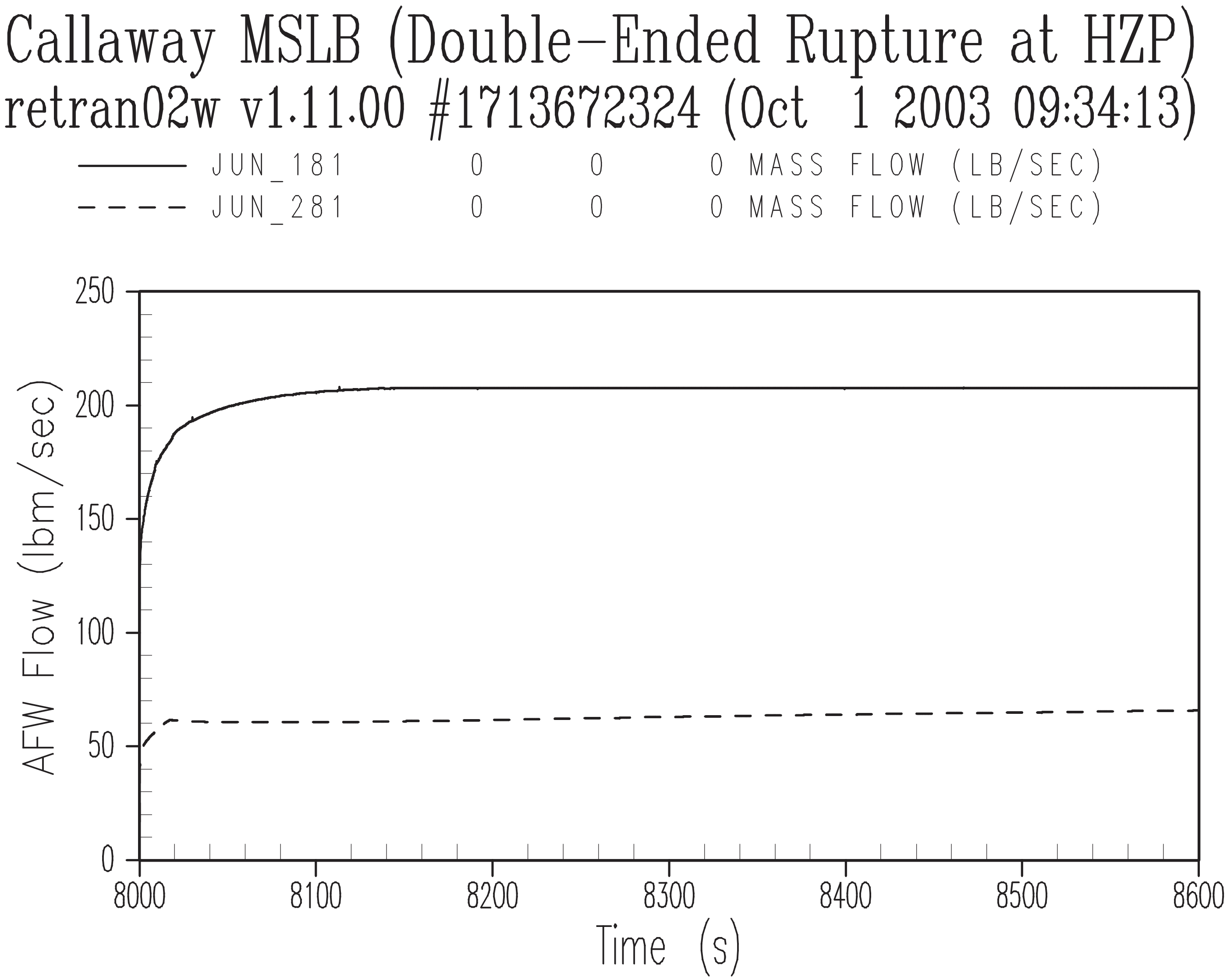
HZP Steamline Break Transient without Offsite Power Available, Double-Ended Rupture
Total SG Mass versus Time



HZP Steamline Break Transient with Offsite Power Available, Double-Ended Rupture
AFW Flow versus Time



HZP Steamline Break Transient without Offsite Power Available, Double-Ended Rupture
AFW Flow versus Time



JUN_181 = Faulted Loop
JUN_281 = Unfaulted Loop

Section 6.3.5 Loss of Non-Emergency AC Power to the Station Auxiliaries/Loss of Normal Feedwater Flow Analysis

25. Section 6.3.5.2 states, “The dual-analysis approach has been previously used by Westinghouse in one other LONF (loss of normal feedwater) analysis of a Westinghouse-designed plant. That previous analysis has been accepted by the NRC and the NRC will again review the results of this dual-analysis approach as part of the RSG license amendment. Reference this LONF analysis and the NRC acceptance documentation.

Response to question 25:

The dual-analysis approach has been previously used by Westinghouse for the Millstone Unit 3 loss of normal feedwater analysis. Although it is not clear if NRC approval was received for the approach and explicitly documented in an NRC Safety Evaluation, a review of this plant's FSAR clearly indicates that two separate analyses were performed. The current Millstone Unit 3 FSAR licensing basis for the Auxiliary Feedwater System (AFWS), Section 10.4.9.1, Item 17b (Page 10.4-47 Revision 16.2) discusses the system reliability. This reliability analysis is a best-estimate type analysis of the Loss of Normal Feedwater (LONF) event that shows that the AFWS operation with one auxiliary feedwater (AFW) pump operating meets the reliability acceptance criteria. Millstone FSAR Section 15.2.7.2, Items 5 and 6 (Page 15.2-14 Revision 16), for LONF describes the AFW system assumptions. In particular, the Millstone FSAR analysis assumes the turbine driven AFW pump as the single failure and that both motor driven pumps are providing flow.

The NRC recently completed the review of a Loss of Normal Feedwater (LONF) analysis where a similar dual-analysis approach had been used. The analysis was performed in support of the Indian Point 3 (IP3) power uprating. The NRC Safety Evaluation (SE) was attached to IP3 Amendment Number 225, TAC Number MC3552, Patrick D. Milano (NRC) to Michael Kansler (Entergy), ADAMS Accession Number ML050600380, dated March 24, 2005. SE Section 3.2.2.11.2.6 discusses the LONF analysis.

Section 6.3.10 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

26. Discuss, or cite discussions of the effect the SG replacement would have upon the core limits, protection lines, and Overtemperature ΔT trip setpoint calculations used in the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power event analyses.

Response to question 26:

The effect of a typical SG replacement on the core thermal limits is minimal. In Callaway's case, the core thermal limits were recalculated to incorporate other non-RSG effects that had previously been addressed via stand alone evaluations and the upgrade in DNB methodology from ITDP to RTDP. Also, as part of the Callaway RSG Program, the calculation of the Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) protection lines was revisited and, as a result, it was confirmed that the existing OTΔT and OPΔT setpoints remained valid. The adequacy of the OTΔT setpoints to ensure that the DNB design basis is met is confirmed via analyses, including that of the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power (RWAP) event.

27. What is the minimum possible reactivity insertion rate for Callaway? Provide analysis results for this minimum possible reactivity insertion rate, assuming various initial power levels and minimum reactivity feedback, to show that the pressurizer would not fill before the reactor is tripped.

Response to question 27:

The RWAP cases most likely to result in pressurizer filling, as currently analyzed for Callaway, are those that assume a low level of reactivity insertion. Westinghouse typically uses the criterion of preventing pressurizer filling to demonstrate that various Condition II incidents would not generate a more serious incident without other incidents occurring independently. The RWAP analyses performed do not credit the high pressurizer water level trip. Only High Neutron Flux and Overtemperature DT are explicitly credited. The high pressurizer water level trip would preclude pressurizer filling; thus, the generic Westinghouse analysis methodology for the RWAP event does not consider pressurizer filling as an acceptance criterion for this event. Furthermore, the approach to a water-solid condition for these low reactivity insertion RWAP cases would be slow, making the response time for said reactor trip function not as critical (continued demonstration of the operability of the trip function would suffice for the purposes of this analysis). Based on this, the RWAP analysis is performed for the primary purpose of demonstrating that the adequacy of the High Neutron Flux and OTΔT reactor trip functions in preventing the violation of the DNBR safety analysis limit, without explicit consideration of the high pressurizer water level reactor trip function, which would yield slightly less limiting DNBR results.

28. Section 6.3.10.4 states, “The (reactor coolant system) RCS pressure safety analysis limit of 2,748.5 psia is confirmed to be met via a generic evaluation.” Cite the generic evaluation and verify that the evaluation (1) does not credit operation of spray or power operated control valves (PORVs), and (2) applies to Callaway with the RSGs.

Response to question 28:

As explained in the item above, the RWAP analysis is performed for the primary purpose of demonstrating that the adequacy of the High Neutron Flux and OTΔT reactor trip functions in preventing the violation of the DNBR safety analysis limit. For that reason, pressure control mechanisms, such as pressurizer sprays and power-operated relief valves (PORVs), are assumed to operate as designed to minimize the calculated DNBR for the event. To address RCS overpressurization concerns for this event, Westinghouse performed a generic analysis of this event without assuming operation of these control systems, utilizing bounding values for several key input parameters. The results of this generic analysis, which considers 2-loop, 3-loop and 4-loop Westinghouse-designed plants, demonstrate that adequate protection would be provided through the use of the high neutron flux and high pressurizer pressure reactor trip functions in conjunction with the positive flux rate trip (PFRT) reactor trip function. This last function is typically not explicitly modeled in safety analysis since most utilities do not perform response time testing on it. However, the generic work performed to address RWAP overpressurization takes credit for the function’s availability (confirmed by surveillance requirements that ensures its operability) without placing any requirement on its response time. The generic work concluded that the presence of these three protection functions ensure that overpressurization following a RWAP, assuming all automatic pressure control features are unavailable, is bounded by other Condition II transients. As such, the analysis methodology for this event focuses on the limiting acceptance criterion for this event (i.e., DBNR) and assumes both pressurizer sprays and PORVs are available.

A review of the key input assumptions made in the generic overpressurization analysis was performed as part of the Callaway RSG Program and it was confirmed that the generic analysis continued to apply to Callaway with the RSGs.

It should be noted that the generic method described above has been reviewed by the US NRC in a recent submittal for the Diablo Canyon Nuclear Plant.

Section 6.3.14 Inadvertent Actuation of ECCS at Power Analysis

29. How was the current licensing basis for the Inadvertent Actuation of Emergency Core Cooling System (ECCS) at Power event established (i.e., by 10 CFR 50.59 evaluation or by staff review and approval). If by 10 CFR 50.59 evaluation, provide a copy of the evaluation. If by staff review and approval, then please cite the license amendment.

Response to question 29:

This was approved by the NRC staff in License Amendment 137 dated 9/25/00, based on ULNRC-04258 dated 5/25/00. See especially NRC Safety Evaluation pages 3-5.

30. The pressurizer PORVs are predicted to open when the pressurizer is water-solid. Are they expected to reseat properly?

    a. If yes, then (1) state how they, and their associated discharge piping, have been qualified for water relief during an Inadvertent Actuation of ECCS at Power event, (2) verify that the automatic control circuitry of these valves meets Class 1E requirements, and (3) indicate the PORV opening setpoint, setpoint tolerance, and surveillance requirements for operation under water relief conditions.

    b. If no, then explain how the analysis results demonstrate that the event will not develop into a more serious event, e.g., a SBLOCA.

Response to question 30:

a.(1)

Yes, the pressurizer power-operated relief valves (PORVs) are expected to reseat properly following water relief during an Inadvertent Actuation of ECCS at Power event. Based on the PORV design for water or steam flow, the PORVs are expected to reseat under the full range of design conditions. Operating License Amendment 137 (which approved ULNRC-04258 dated May 25, 2000) provides details of a modification implemented at Callaway to upgrade the automatic PORV actuation circuitry to fully Class 1E. The PORVs will automatically close after resetting of the pressurizer pressure high signal. As discussed in Technical Specification Bases 3.4.11, block valves are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV.

No changes were required to the previous pressurizer safety or relief piping system analysis inputs as a result of the modification to replace steam generators. Westinghouse previously performed analysis for the qualification of the pressurizer safety and relief piping. Hydraulic forcing functions were generated assuming the simultaneous opening of either the safety valves or the relief valves, associated loop seal water slug discharge, and included water discharge transients when the relief valves were utilized for cold overpressure mitigation. These thermal hydraulic transients represented the worst applicable loading cases for the piping and supports. The analysis summary indicates the operability and structural integrity of the piping system have been ensured for all applicable loadings and load combinations including all pertinent safety and relief valve discharge cases. These conclusions were documented by the NRC in their Safety Evaluation for NUREG-0737 Item II.D.1, Thomas W. Alexion to Donald F. Schnell, dated September 10, 1987.

a.(2)

The automatic pressure control circuitry for the pressurizer PORVs was upgraded during Refuel 11 (spring 2001) to fully Class 1E during the implementation of License Amendment 137 dated 9/25/00, based on the design described in ULNRC-04258 dated 5/25/00.

a.(3)

With the implementation of Callaway License Amendment 137 during Refuel 11 (spring 2001), the PORVs now open at 2350 psia with a reset of 20 psi (2.5% of span). The setting tolerance is 0.035 VDC which corresponds to 2.8 psi. ESFAS Function 9 was added to TS LCO 3.3.2 Table 3.3.2-1 to address operability and surveillance requirements associated with this ESFAS trip function.

The PORV setpoint is not a critical parameter for the safety analysis performed in support of the Callaway RSG Program provided it ensures that the PORV opens at a lower pressure than the pressurizer safety valves minus any applicable uncertainties. In that manner, water relief through the safety valves is always precluded.

The Reactor Coolant System Power Operated Relief Valves (BBPCV0455A and BBPCV0456A) are surveillance tested in accordance with Callaway procedure OSP-BB-V002A. This procedure demonstrates the operability of the RCS PORVs per TS 5.5.8 (Inservice Testing Program), T/S SR 3.4.11.2, and TS SR 3.3.2.14 for Table 3.3.2-1 Function 9.a. The actuation instrumentation is tested per the surveillance requirements noted for Function 9 in TS Table 3.3.2-1, as approved by NRC in Callaway License Amendment 137.

There are no surveillance requirements for RCS PORVs to specifically demonstrate operability under water relief conditions nor was any such requirement imposed by the NRC staff in approving Amendment 137.

31. Provide the ECCS flow delivery rate, as a function of RCS pressure, that was assumed for the Inadvertent Actuation of ECCS at Power event analysis. Compare this flow delivery rate to that assumed in the current licensing basis analyses.

Response to question 31:

The actual maximum ECCS flows modeled in the updated Callaway analysis of the Inadvertent ECCS Actuation at Power event are presented below. The flows are the same as those assumed in the current analysis of record for this event. As explained in Section 6.3.14, operator action to terminate flow from the Normal Charging Pump is assumed 6 minutes into the transient. ECCS flows are provided for conditions with and without NCP flow. The data in the first column represents the flow provided by both charging pumps (CCPs) and the normal charging pump (NCP). The data in the second column represents the maximum flow after the normal charging pump is secured by operator action.

RCS Pressure (psia)	Total SI flow with NCP (gpm)	Total SI flow without NCP (gpm)
2015	441.5	389
2115	408	358
2215	372	323
2250	357	310
2315	331	285
2415	283	239
2515	231	189

These flow delivery rates remain unchanged from the current licensing basis.

Section 6.3.15 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory Evaluation

32. Section 15.5.2 of the Callaway Final Safety Analysis Report (FSAR) concludes that the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory will not fill the pressurizer before the operator can terminate the transient, at about 15 minutes. Provide information to demonstrate that the operator can terminate the transient, as the result of following established procedures, before the calculated pressurizer fill time.

Response to question 32:

Operator response times were tested per procedure APA-ZZ-00395. APA-ZZ-00395 maintains a program to ensure operator response times credited in the analysis are valid. During the validation of the recently upgraded EOPs, this event was simulated for a 3-man control crew on August 18, 2004. The time for the crew to terminate charging was 4 minutes and 12 seconds from the start of the event. The operators responded before the pressurizer high level alarm occurred.

Section 6.3.17 Anticipated Transients Without SCRAM Aanalysis

33. The Anticipated Transients Without SCRAM (ATWS) analyses of (1) WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis," dated August 1974, and (2) Westinghouse letter NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," dated December 1979, are based upon the design parameters of Westinghouse steam generator series 44 and 51, and models D and F, not the Framatome Model 73/19T design. Explain how the referenced Westinghouse ATWS analyses and methods apply to Callaway, as equipped with the RSGs.

Response to question 33:

The referenced Westinghouse ATWS analyses were revisited to support the Callaway RSG Program. The Loss of Load and Loss of Normal Feedwater ATWS cases were reanalyzed, explicitly incorporating the Framatome Model 73/19T steam generators into the model. Given the similarities between the Westinghouse SG design and the replacement SG, the methods used remain applicable. The results presented in Section 6.3.17.3 are specific to the updated ATWS analyses with the Framatome steam generators.

34. Section 6.3.0.6 indicates that, "LOFTRAN remains the system transient code for the analyses of ..... the anticipated transients without scram (ATWS, FSAR 15.8)." LOFTRAN has a single-node steam generator shell side model. Describe how LOFTRAN models the primary-to-secondary heat transfer rate as the shell-side water level drops and the steam generator tubes are exposed, for the Framatome Model 73/19T steam generator design.

Response to question 34:

In the ATWS analyses of NS-TMA-2182, the transient primary-to-secondary heat transfer rate of LOFTRAN is calculated in conjunction with a multi-node steam generator model using the TRANFLO code (see Section 4.5 of NS-TMA-2182). A similar approach was used for the Callaway steam generator replacement except the TRANFLO code was replaced with the NOTRUMP code.

Calculation of the ATWS transient primary-to-secondary heat transfer was performed by developing a multi-node NOTRUMP model of the Framatome Model 73/19T steam generator. The model included nine axial fluid nodes for the secondary side of the tube bundle to calculate the primary-to-secondary heat transfer rate as the shell-side water level drops and the steam generator tubes are exposed. The amount of heat transfer during the ATWS event was translated into a curve of overall tube bundle UA versus steam generator fluid mass. This curve can be input to LOFTRAN and can then be used in place of the LOFTRAN steam generator heat transfer calculation.

An initial LOFTRAN run which assumes a representative estimate of the tube bundle overall UA versus liquid mass is performed. Steam generator primary side tube conditions (flow, inlet enthalpy and pressure), secondary side feedwater flow and steam flow from LOFTRAN are input to the NOTRUMP model as boundary conditions. The NOTRUMP model then calculates a new curve of UA versus secondary side fluid mass. This new curve is then used as input to LOFTRAN and the process is repeated until convergence of steam generator state parameters between the LOFTRAN and NOTRUMP models occurs.

35. Provide transient plots and sequence of events tables denoting the time and value of peak RCS pressure for analyses of the Callaway Loss of Feedwater and Loss of Load ATWS events, assuming the design characteristics of the Framatome Model 73/19T RSGs.

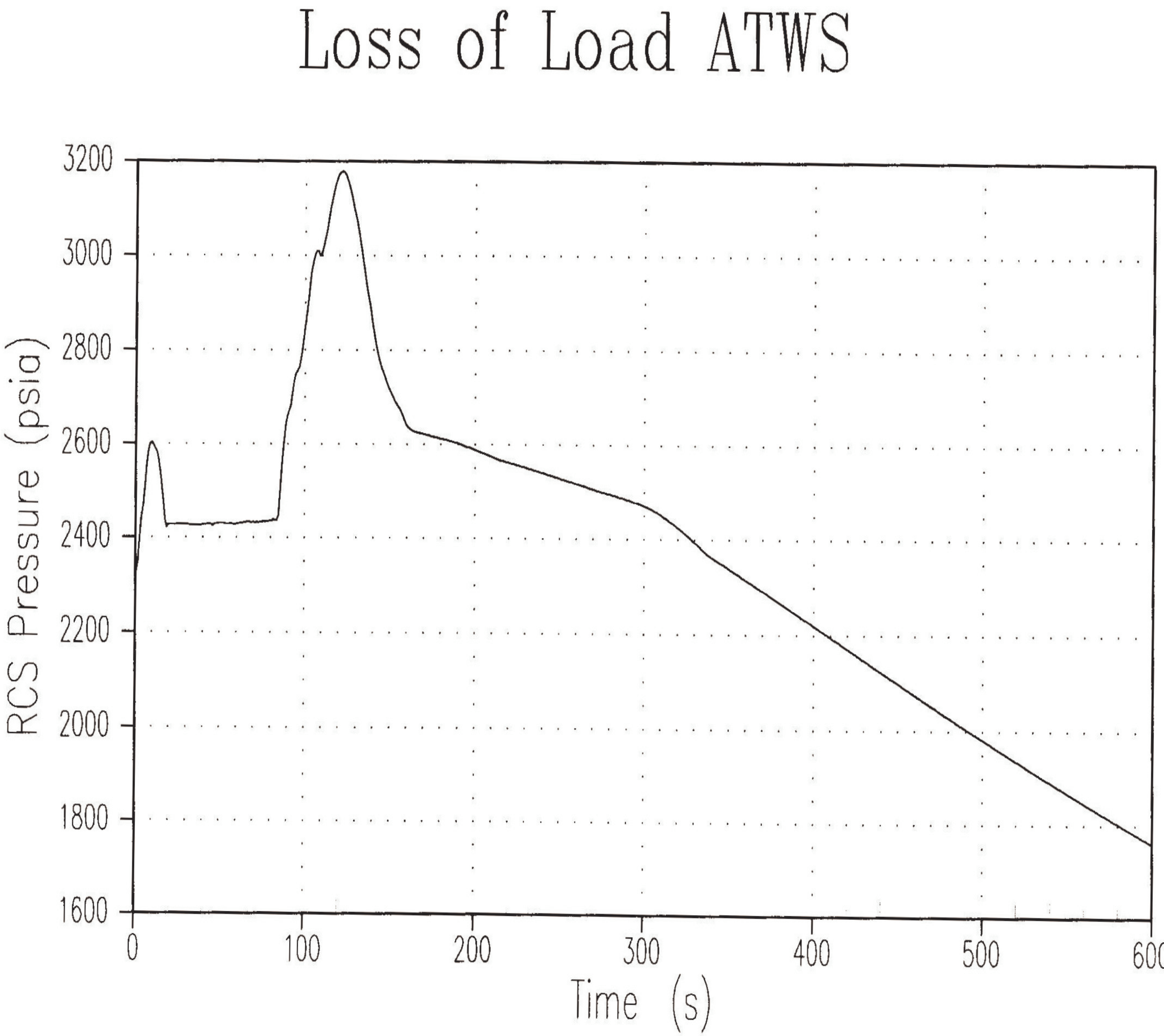
Response to question 35:

Time Sequence of Events
Loss of Load ATWS

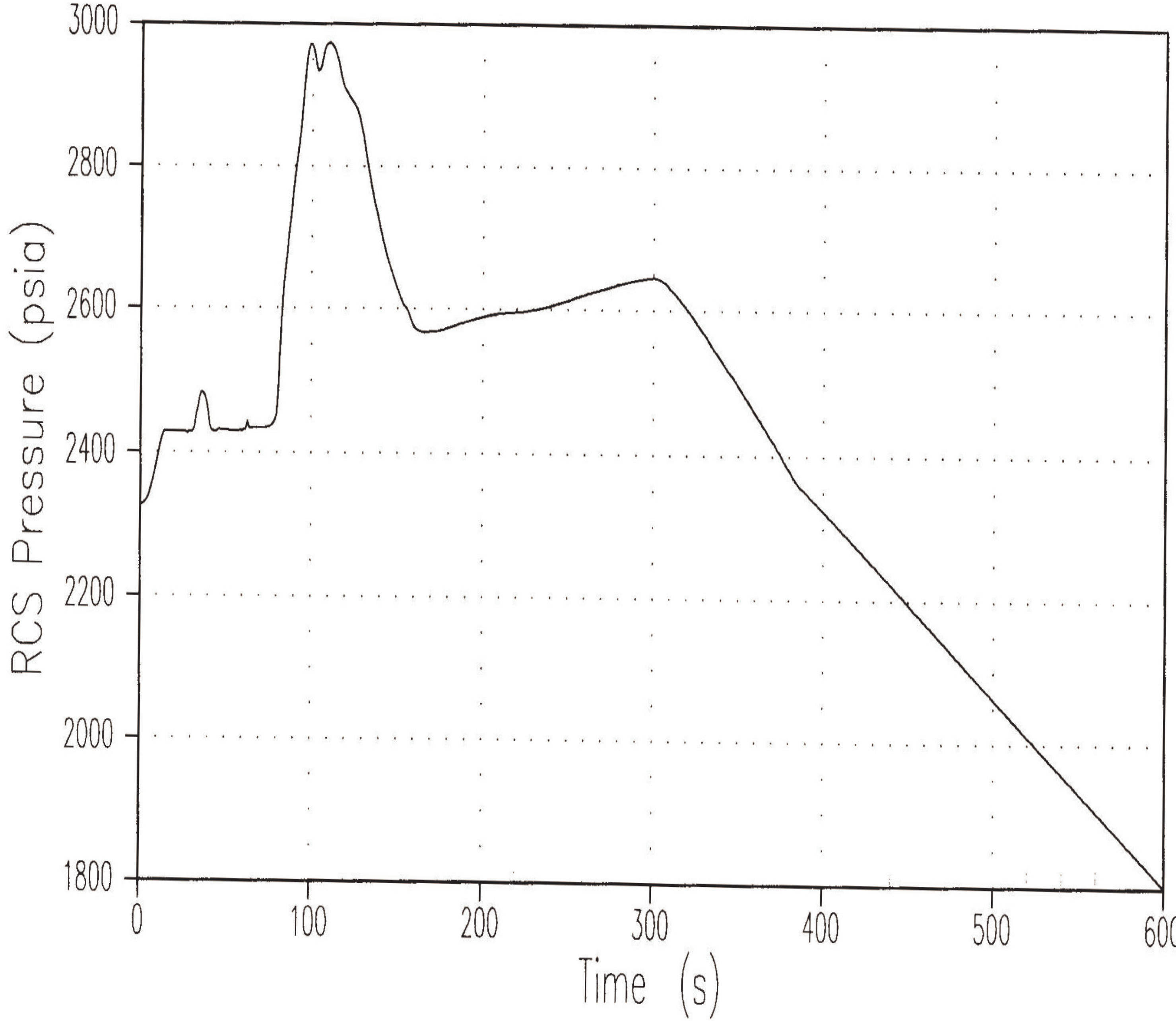
Event	Time (sec)
Turbine trip occurs	1.0
FW flow terminated	4.0
AFW flow initiated	60.0
Peak RCS pressure reached (3177 psia) (versus RCS pressure limit of 3215 psia)	122.0

Time Sequence of Events
Loss of Feedwater ATWS

Event	Time (sec)
FW flow terminated	4.0
Turbine trip	30.0
AFW flow initiated	60.0
Peak RCS pressure reached (2973 psia) (versus RCS pressure limit of 3215 psia)	110.0


Loss of Load ATWS
RCS Pressure (psia)
3200
3000
2800
2600
2400
2200
2000
1800
1600
0
100
200
300
400
500
Time (s)

LONF ATWS


3000
2800
2600
2400
2200
2000
1800
RCS Pressure (psia)
0
100
200
300
400
500
600
Time (s)

36. List and explain the analysis assumptions (e.g., moderator temperature coefficient and initial steam generator mass) used in the ATWS analysis.

Response to question 36:

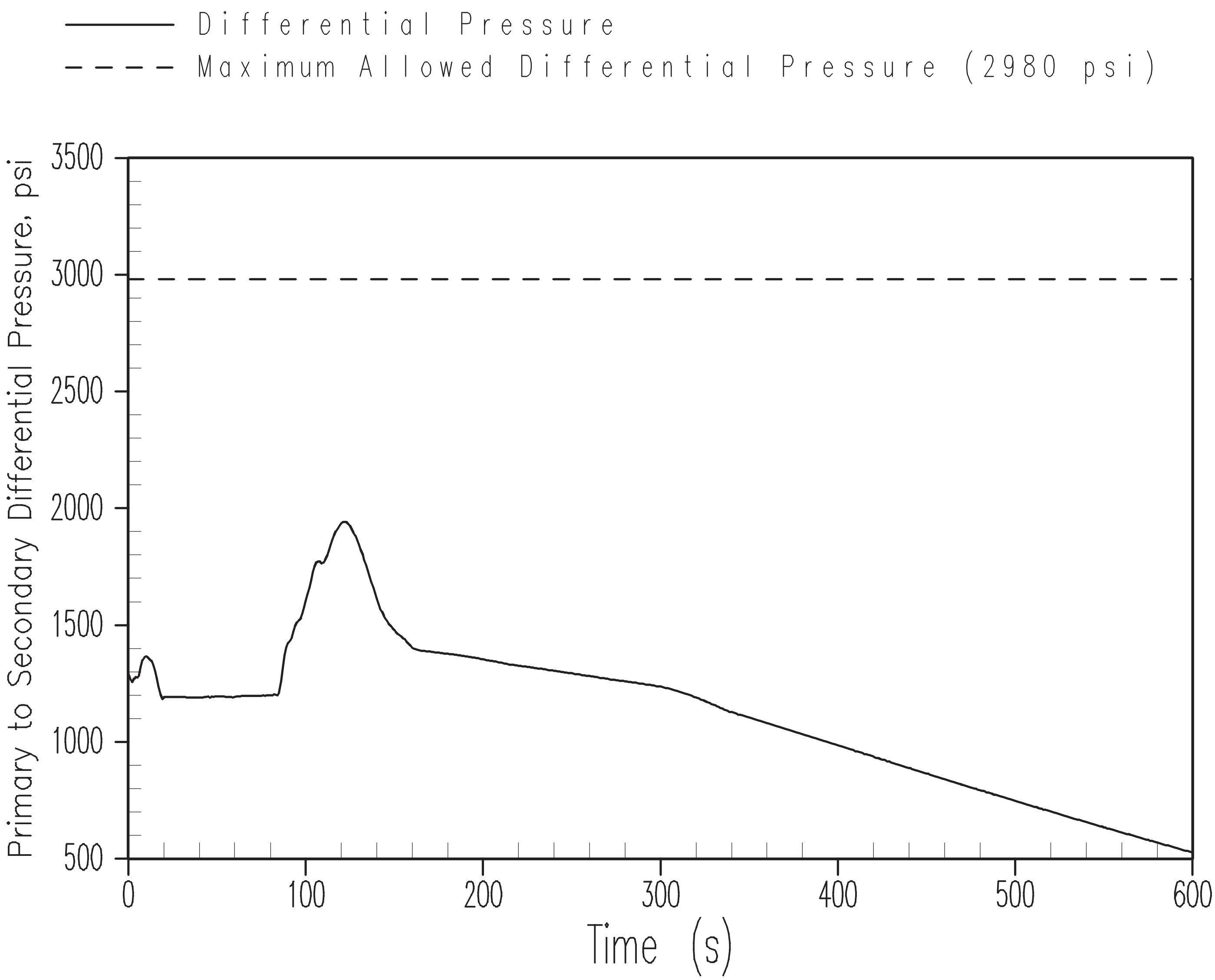
The ATWS analysis assumes a moderator temperature coefficient of -8 pcm/°F which is bounding for 95% of a representative cycle. The initial steam generator mass assumed is consistent with the full-power nominal SG level of 51.3% NRS.

37. Verify that the maximum differential pressure, across the tubesheet and tubes of the Framatome Model 73/19T RSGs, matches or exceeds the value listed in Appendix C of WCAP-8330.

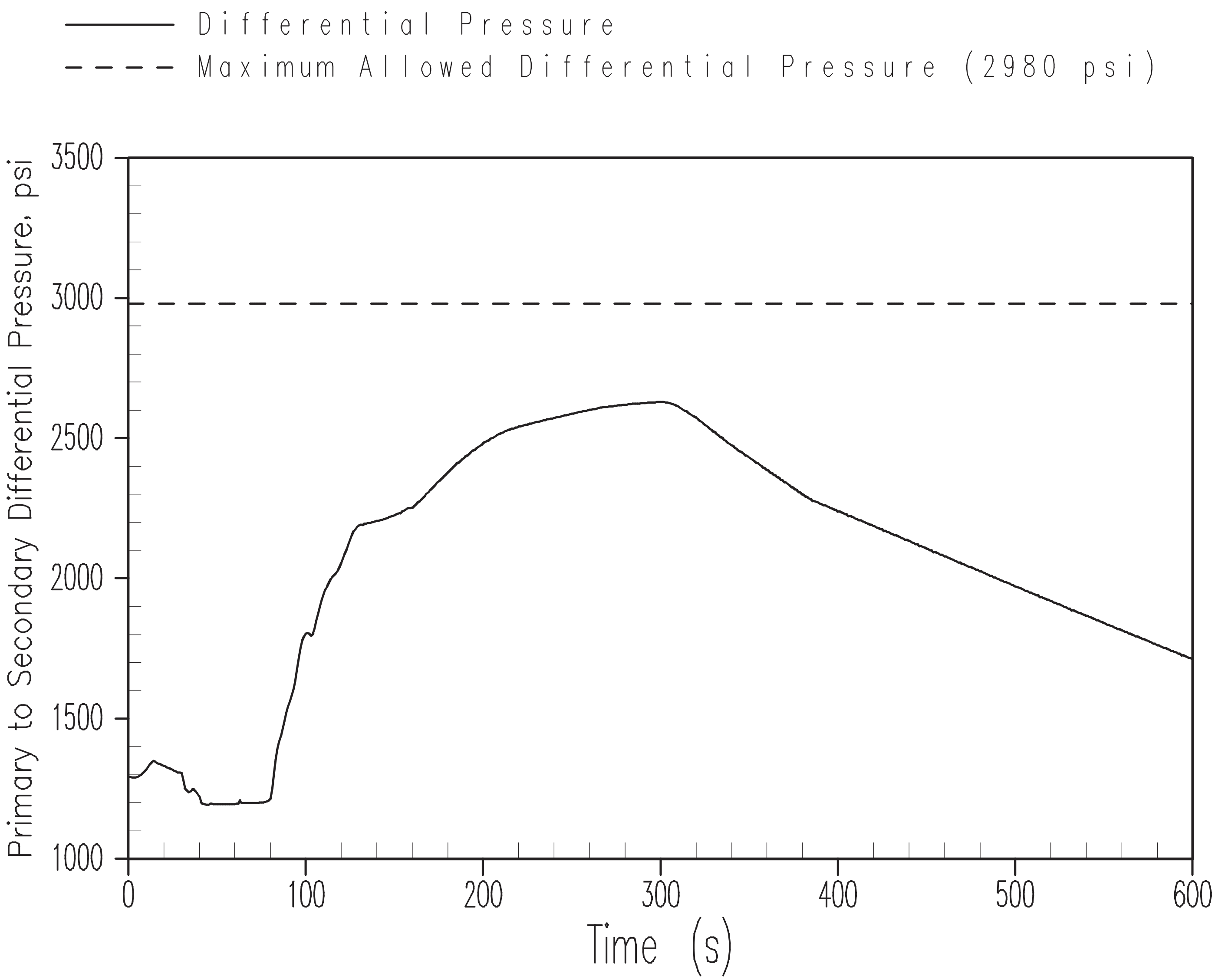
Response to question 37:

Appendix C of WCAP-8330 and Section 6.0 of Westinghouse letter NS-TMA-2182 indicate that the maximum allowed differential pressure across the tubes or the tube sheet is 2980 psi. The limiting ATWS events are the loss of load and loss of normal feedwater transients. Analysis input from NS-TMA-2182 for these events was modified to reflect the Framatome Model 73/19T RSGs with the current Callaway full power conditions. The figures below show the primary to secondary side differential pressure for the loss of load and loss of normal feedwater ATWS transients. The maximum primary to secondary differential pressure for the loss of load transient is 1943 psi; the maximum primary to secondary differential pressure for the loss of normal feedwater transient is 2630 psi. For both cases, the differential pressure was maintained below the 2980 psi limit.

Loss of Load ATWS
Maximum SG Differential Pressure



Loss of Normal Feedwater ATWS
Maximum SG Differential Pressure



Section 6.4 Steam Generator Tube Rupture (analysis):

38. Show that all assumed operator action times are verified in Callaway simulator exercises.

Response to question 38:

The Emergency Operating Procedure (EOP) Steering Committee is a standing committee with a charter to manage the operator response time validation of the RSG EOPs. The RSG EOP setpoints have been communicated to the EOP group and the RSG upgrade is in progress. Time critical validations of the new SGTR scenarios will be timed in June 2005 on one licensed crew following completion of the RSG SGTR simulator model upgrade. After successful completion of time critical validations and prior to the RSG EOP revisions being issued, all licensed operators will receive training on the RSG EOP revisions.

39. What is the tube rupture size (in sq ft or sq in) that is analyzed? Are all tubes in the RSGs of the same diameter?

Response to question 39:

The tube area modeled is 0.002405 ft^2. All of the tubes in the RSGs are of the same diameter.

40. Confirm that there is not an intermediate value of initial average temperature ($T_{avg)}$ that will produce more severe results for the SG tube rupture event.

Response to question 40:

For the SGTR case with a stuck-open atmospheric steam dump valve, a high initial T-avg is modeled to maximize flashing of break flow and steam releases to the atmosphere. A sensitivity case modeling a low initial T-avg was included, since a lower initial T-avg maximizes the break flow. An intermediate T-avg would not result in higher releases than those calculated for the High T-avg case or a higher break flow rate than that calculated for the Low T-avg case. The dose analysis was performed with data selected to conservatively bound the flashing fraction and steam releases from the High T-avg case and the break flow rate from the low T-avg case. This is discussed in Section 6.4.1.4 of LAR Appendix A. The SGTR case with failure of the ruptured steam generator AFW control valve leading to overfill and water relief maximizes the inventory in the ruptured steam generator and the water releases by using modeling assumptions that result in low steam releases and high break flow. Both of these are obtained by modeling a low initial T-avg.

41. Why are pressurizer heaters not assumed to be operating prior to reactor trip?

Response to question 41:

The NRC-approved SGTR methodology for Callaway, which was followed for the RSG analysis, does not model the pressurizer heaters. The SGTR methodology was approved in Operating License Amendment 159 issued on March 11, 2004.

42. Section 6.4.1.2 states that, "Feedwater isolation is completed 4.3 seconds after reactor trip/SI." What is the basis for the 4.3 second value and why is this time not 17 seconds, like all the other applicable events listed in Table 6.3-6? Why is the SG Tube Rupture event not listed in Table 6.3-6?

Response to question 42:

For the SGTR case with a stuck-open atmospheric steam dump valve, earlier feedwater isolation is conservative. The value is taken from the original Callaway SGTR analysis that was used as the basis for the RSG analysis. It includes a 2.3 second delay after reactor trip/rod motion (and assumed loss of offsite power) and then 2 seconds for the valve to completely close. The SGTR with failure of the ruptured steam generator AFW control valve leading to overfill and water relief conservatively modeled the 17 second delay in feedwater isolation to maximize the ruptured steam generator inventory. This assumption is included in Section 6.4.2.2 of LAR Appendix A.

Table 6.3-6 contains information related to the various non-LOCA analyses, which are summarized in section 6.3. The SGTR analysis is addressed as a separate event from the non-LOCA events. The information related to the SGTR event is summarized in Sections 6.4.1.2 and 6.4.2.2.

For the SGTR case that assumes the atmospheric steam dump valve associated with the faulted SG fails open, main feedwater isolation valve (MFIV) closure in 4.3 seconds is assumed based on the following from FSAR page 15.6-11:

> "g. Prior to reactor trip, the normal feedwater matches the steam flow in the intact steam generators. For the ruptured steam generator, the total feed flow (including the break flow) matches the steam flow. The feedwater isolation signal occurs 2.3 seconds after reactor trip and the feedwater isolation valves stroke closed within 2.0 seconds. These are the minimum expected delay and stroke time, respectively, which tend to decrease heat removal from the RCS resulting in higher RCS temperatures and pressures. This results in maximum flashed fraction and break flow."

43. One of the assumptions in Section 6.4.2.2 is, "Additional active failure: The ruptured steam generator's safety valve fails partially open (5-percent effective area) after water relief." If the steam generator's safety valve is not qualified for water relief, then the valve would be assumed to stick open, as a consequential failure, not as an additional active failure. What are the analysis results of this case, assuming a steam generator safety valve that sticks open following water relief?

Response to question 43:

The overfill analysis presented in the present submittal does represent the results of a main steam safety valve sticking open 5-percent following water relief. This flow area value was mandated as part of Question 1 from NRC staff requests for additional information (RAI) 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. NRC approved the findings of this response letter in their Safety Evaluation dated August 6, 1990.

The basis of the 5-percent flow area was more recently covered during the RAI process for Operating License Amendment 159 which was issued on March 11, 2004. AmerenUE addressed the 5-percent flow area value as a part of ULNRC-04928.

Classifying the valve's sticking 5-percent open following water relief as a consequential failure, instead of an active failure, will not affect the results of the analyses which were previously submitted with single failures postulated.

44. Does the steam generator replacement affect the system dynamics and the timing for boric acid precipitation following large break LOCAs. Explain.

Response to question 44:

After a large break LOCA the RCS depressurizes rapidly to near containment pressure. Reflood and core quench would occur early relative to the time period required for the core boric acid concentration to approach the solubility limit. After reflood and core quench the system is in a quasi-equilibrium state and it is during this period when the boric acid concentration only begins to increase to significant levels. The RSGs would have no significant effect on the boric acid concentration during this quasi-equilibrium period since the rate of boric acid accumulation is dependent primarily on the boron concentration of the injected SI and the steaming rate in the core. The increase in primary side volume for the RSGs would have a small beneficial impact. Since the pre-LOCA RCS inventory is a dilution source for the sump boric acid solution, the increased RSG primary side volume would slightly decrease the sump boric acid concentration and therefore would decrease the rate at which boric acid accumulates in the core region.

45. In the SBLOCA analyses, while the PCT for the worst break is low, review of the analysis results suggests that a break size between 3 and 4 inches would be more limiting. Since accumulator injection terminates the clad temperature rise for the 4-inch break, a slightly smaller break wherein the RCS pressure just remains of the accumulator actuation pressure could produce a higher PCT. Even though the two-phase level would be higher than that for the 4 inch break, uncovery would persist for a potentially much longer period of time causing the PCT and clad oxidation percentage to increase beyond that for the 4 inch break. Based on these considerations, explain the rationale for assuring the 4-inch break is the limiting break when accumulator injection terminates the clad heat-up.

Response to question 45:

References 45-1 through 45-4 cited in the response to question 45 are identified at the end of the response.

The basis methodology of the NOTRUMP evaluation model is documented in References 45-1 and 45-2. Section 5-3-2 of Reference 45-2 states "A break spectrum, with the Appendix K conservatism in the loop seal steam venting response assured, is presented in this section of the report. This evaluation model break spectrum consists of cold leg breaks of equivalent diameters of 2, 3, 4, 5, and 6 inches." Additionally, References 45-1 and 45-3, investigated break spectrums of 2, 3, 4 and in some cases 6 inches. It should be noted that Reference 45-3 was published in response to post TMI II action items to demonstrate continued compliance to 10 CFR 50.46 on a generic basis. In this environment, the review of the NOTRUMP EM was carried out under significant scrutiny. In all these generic licensing submittals, the NRC staff issued Safety Evaluation Reports which did not question the resolution of the break spectrum. In addition, with the introduction of the original ECCS evaluation models in 1974, Westinghouse performed sensitivity studies (Reference 45-4) which included break size variations of 2, 3, 4, 6 inch and larger equivalent diameters. Since then, Westinghouse has always analyzed SBLOCA break spectrums consisting of these increments. Thus, the practice of the application of the NOTRUMP EM is to stay within the resolution boundaries of this break spectrum.

The break spectrum analyzed for Callaway was based on 2, 3, 4, and 6 inch cases. The 5 inch case is not reported since break sizes above 4 inches typically demonstrate good depressurization characteristics that allow a rapid amount of both accumulator injection and pumped ECCS inventory. This depressurization behavior is considered captured in the 6 inch case and to an even greater extent as break size is increased. This is illustrated in Figure 5-3-1 of Reference 45-2 and has been demonstrated many times before and since. As such, the break spectrum analyzed is considered adequate for an evaluation model developed to Appendix K standards and conservatism.

<u>References</u>:

45-1 WCAP-10054-P-A, Addendum 2, Rev 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997.

45-2 WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

45-3 WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," October 1986.

45-4 WCAP-8356, "Westinghouse Emergency Core Cooling System – Plant Sensitivity Studies," July, 1974.

46. Table 6.2.2-4 identifies loop seal clearing times for each break. Identify the loop seals that cleared and was there any residual water predicted to remain in any of the loop seal piping horizontal sections. Explain.

Response to question 46:

The number of loop seals cleared for the 2, 3 and 4 inch break sizes is only the faulted loop due to application of the loop seal restriction. Thus the number of loop seals that cleared is 1. For the 6 inch case, both the faulted and lumped loop seals clear, however, the lumped loop seal starts to re-plug at about 500 seconds.

The horizontal section of the reactor coolant pump (RCP) suction cross-over leg in the NOTRUMP EM is modeled as part of two control volumes. One captures the uphill portion of the cross-over leg and ½ the horizontal section and the other the outlet plenum of the steam generator and ½ of the horizontal section.

The Table below summarizes the amount of combined liquid mass present (after loop seal clearing) in the two nodes described above. This information was obtained from the ASCII output. Note the edit frequency for the 2 inch case does not have this information in this time frame.

Time (sec)	2 inch	3 inch	4 inch	6 inch*
500	N/A	-	414 lbm	74.5 / 838 lbm
1000	N/A	905 lbm	132 lbm	676 / 6799 lbm
1500	N/A	721 lbm	116 lbm	
2000	N/A	138 lbm	112 lbm	
2500	N/A	137		

* Faulted loop / Intact loop(s)

In addition, the attached plots summarize the mixture level with respect to time as requested. [The plots are in Enclosure 4.]

Containment Integrity Review

47. In the response to question 2.i of the licensee's submittal dated February 11, 2005, it is stated that "the values for containment volume, heat sink areas, and ... included conservatisms." Provide examples of these conservatisms.

Response to question 47:

The containment volume, heat sink areas, and thickness input values were taken from the previous licensing basis containment pressure/temperature (P/T) analysis and are typically biased low to maximize the pressure for the containment DBA [design basis accident] evaluation model. The minimum free volume in the containment DBA evaluation model is calculated by subtracting the uncertainty in the measured free volume data. Likewise, the heat sink input for the containment DBA evaluation model is calculated by subtracting the uncertainty in the measured heat sink area and/or thickness.

The containment volume and heat sink area values used in the Callaway containment DBA evaluation model are compared in the table below with the containment volume and heat sink values used in the Callaway LOCA PCT evaluation model to give an example of the conservative input bias. The containment volume and heat sink input values used in the LOCA PCT evaluation model are biased high to minimize the containment backpressure.

	LOCA PCT	Containment DBA
Containment Volume (ft^3)	2,700,000	2,500,000
Heat Sink Areas (ft^2)		
Containment Cylinder	64919	58807
Containment Dome	34129	30806

Based on this comparison, the uncertainty in the measured containment volume is about 100,000 ft^3 and the uncertainty in the measured containment building heat sink area is about 5%. The uncertainty was subtracted from the nominal measured values to yield the conservative containment DBA evaluation model input values.

PLOTS REFERENCED IN LICENSE'S DRAFT RESPONSE TO QUESTION 46

(ADAMS Accession No. ML051430154)

ENCLOSURE 4

Callaway Plant, Unit 1

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