WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President, Engineering

December 14, 2007
ET 07-0060

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference: 1) Letter ET 07-0004, dated March 14, 2007, from T. J. Garrett, WCNOC, to USNRC

2) Letter ET 07-0050, dated October 16, 2007, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information Relating to Replacement of the Main Steam and Feedwater Isolation Valves and Controls

Gentlemen:

Reference 1 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)" based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls.

The Nuclear Regulatory Commission (NRC) provided by electronic mail on October 23, 2007, a request for additional information (RAI) associated with the non Loss-of-Coolant Accident (LOCA) analyses supporting the license amendment request submitted by Reference 1. Reference 2 provided responses to a RAI related to the non-LOCA analyses provided in Attachment I to Reference 1. This RAI is a follow-up to the RAI responses in Attachment I to Reference 2.

The additional information provided in the Attachment does not impact the conclusions of the No Significant Hazards Consideration provided in the Reference. In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Kansas State official.

A001
NRR

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment – Response to NRC Request for Additional Information

cc: E. E. Collins (NRC), w/a
T. A. Conley (KDHE), w/a
J. N. Donohew (NRC), w/a
V. G. Gaddy (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By_____
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 14th day of December, 2007.

RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010
NOTARY PUBLIC
OFFICIAL
SEAL
STATE OF KANSAS

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

The Nuclear Regulatory Commission (NRC) provided by electronic mail on October 23, 2007, a request for additional information (RAI) associated with the non-LOCA analysis supporting the license amendment request submitted by Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 07-0004 dated March 14, 2007. Letter ET 07-0004 provided a license amendment request that proposed revisions to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," and TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs)" based on a planned modification to replace the MSIVs and associated actuators, MFIVs and associated actuators, and replacement of the Main Steam and Feedwater Isolation System (MSFIS) controls. This RAI is a follow-up to the RAI responses in Attachment I to WCNOC letter ET 07-0050 dated October 16, 2007.

1. *The response to RAI-1 indicated that during the steam generator (SG) overfilling condition resulting from an increased feedwater event, the most likely mode of atmospheric relief valve (ARV) or safety valve failure is for the valve to fail to reseat completely. Under such a condition, the response indicated that the offsite dose releases are expected to be bounded by those for the steam generator tube rupture (SGTR) forced feedwater overfill event. Provide the bases to support that the offsite radiological consequences for the SGTR forced overfill event bound those for the increased SGTR feedwater overfill event.*

Response: It is important to note that during a steam generator tube rupture (SGTR) accident, a direct release path from the reactor coolant system (RCS) to the environment may be created such that primary coolant will bypass the steam generator (SG) liquid, and there will be neither dilution of the primary coolant iodine activity in the liquid, nor partitioning. Thus, the major contribution to the offsite radiological dose for the SGTR with Forced Overfill event is from the break flow from the primary to secondary that flashes immediately and escapes to the outside atmosphere.

A review of the dose consequence re-analysis results, as presented in Section 4.11 of letter ET 07-0004, dated March 14, 2007, reveals that the flashed break flow released through the direct release path contributes about 82% of the total thyroid dose (i.e., 51.8 rem) calculated for the SGTR forced overfill event with a pre-accident iodine spike scenario. This translates into an exposure of about 42.6 rem for the initial 2 hour period for persons located at the boundary of the exclusion area. The remaining 9.2 rem is contributed by the release of radioactivity from the unflashed break flow that is assumed to mix with the bulk water, the primary to secondary system leakage in the unaffected SGs and the initial radioactivity existing in the secondary system.

During the SG overfill situation, resulting from an increased feedwater event, there is no need to consider the contribution from a break flow that is assumed to immediately flash to vapor and be released to the environment with no mitigation. Thus, the offsite radiological consequences, for an increased feedwater event with the SG overfilling condition, is expected to be substantially lower than that resulting from a SGTR with Forced Overfill event.

Based on the preceding discussion, there is reasonable assurance that the calculated radiological consequences of a postulated overfill condition resulting from the increased feedwater event, assuming the primary coolant equilibrium iodine concentration permitted by the

Technical Specifications in combination with a pre-accident generated iodine spike, does not exceed a small fraction of the exposure guidelines as set forth in 10 CFR Part 100.

2. *The response to RAI-2 indicated that for the feedwater line break (FLB) and the SGTR re-analyses, the MSIV/MFIV closure models assuming the valves ramping closed linearly over a specific closure time period remain unchanged from that assumed in the analysis of record (AOR). For the steam line break (SLB) re-analysis, the AOR MSIV closure model of an instantaneous closure after a specific time delay is changed to ramp closed linearly over a specific closure time. For all three events, the closure times are changed from 10 seconds to 15 seconds.*

   Discuss the effects of the valve closure patterns (linear closure or step closure with a time delay) and closure times of the MSIV/MFIV on the results of analyses for the FLB, SGTR and SLB events. Also, provide valve performance test data and analysis to justify that: (1) the valve closure time of 15 seconds is the limiting value for pressure and temperature conditions predicted for the FLB, SGTR and SLB events; and (2) the assumption of the linear closure model is adequate for the replacement isolation valves in terms of predicating the peak reactor coolant system (RCS) for the FLB event, and the worst radiological releases for the SGTR event, and the minimum DNBR for the SLB event.

Response: The response to this question is separated into 3 items: valve closure patterns, valve performance test data – item (1), and valve performance test data – item (2).

<u>Valve Closure Patterns</u>

The MSIV/MFIV closure characteristics over a specific time period remain unchanged from the analysis of record (AOR) for the SGTR and feedline break (FLB) events. Subsequent to an examination of the replacement valve closure characteristics and considering the conservative closure time duration assumed in the other events analyzed, a similar closure model was adopted for the re-analysis of the steamline break (SLB) event. As discussed below (see Section entitled Valve Performance Test Data – Item (1)), a 15 second closure time duration was assumed.

Sensitivity studies have been performed to identify the effects of the valve closure patterns and MSIV/MFIV closure times on the re-analysis results for the FLB, SGTR and steamline break (SLB) events.

The transient responses of certain critical parameters with respect to the acceptance criteria for each event are determined and compared to the re-analysis results to establish the effects of the MSIV/MFIV closure patterns and closure times. The effects of the valve closure patterns and closure times on the specific transient are discussed as follows:

FLB:

Comparison of the plant parameters: core inlet pressure, hot leg temperature, pressurizer water volume, and SG pressure, following a FLB with a different MSIV closure pattern are shown in Figures 2-1 through 2-6. Figures 2-7 through 2-12 show the comparison of the system response following a postulated FLB with a MSIV closure time of 5 seconds or 15 seconds. As shown in these figures, the plant system response to the FLB event is similar for cases with different valve closure patterns and closure times. It is noted that the plant parameter response

starts to differ after the time at which the MSIVs are actuated on low steamline pressure. However, the critical parameters such as the maximum hot leg temperature, core inlet pressure and the pressurizer water volume occur prior to the time at which the MSIVs are actuated. Therefore, it is concluded that the model used for the valve closure patterns and closure times will have no impact on the margin to hot leg saturation, RCS pressurization, and pressurizer overfilling following a postulated FLB event.

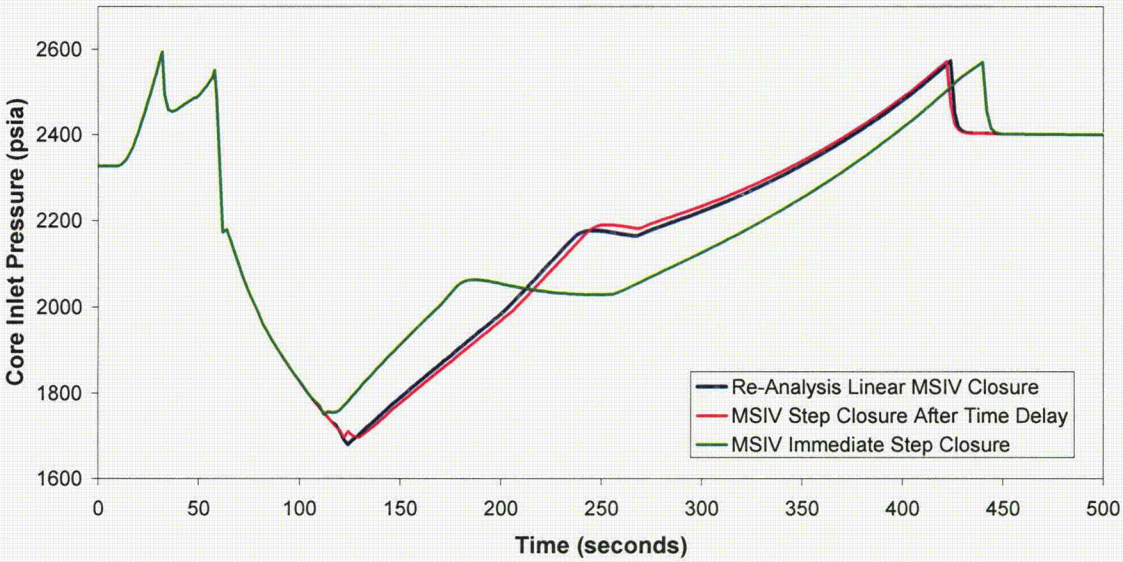

Figure 2-1: FLB - Core Inlet Pressure vs Time

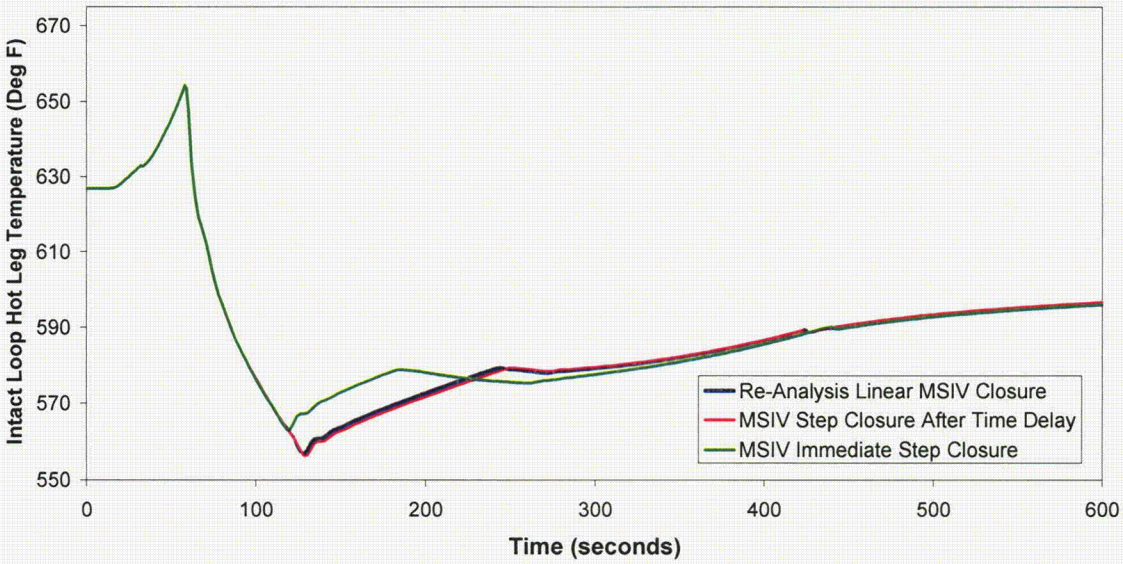

Figure 2-2: FLB - Intact Loop Hot Leg Temperature vs Time

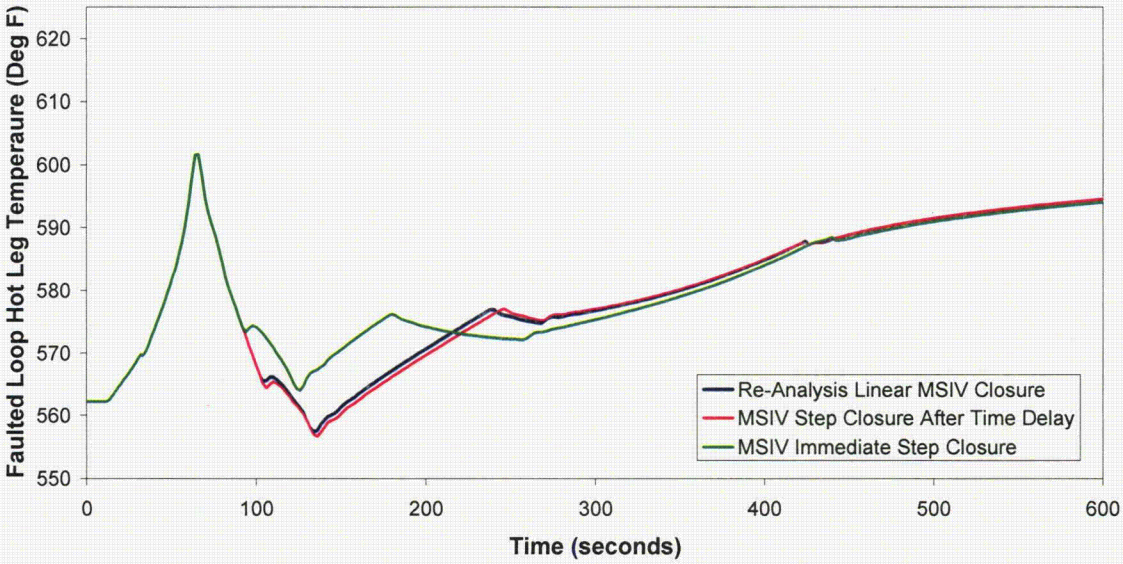

Figure 2-3: FLB - Faulted Loop Hot Leg Temperature vs Time

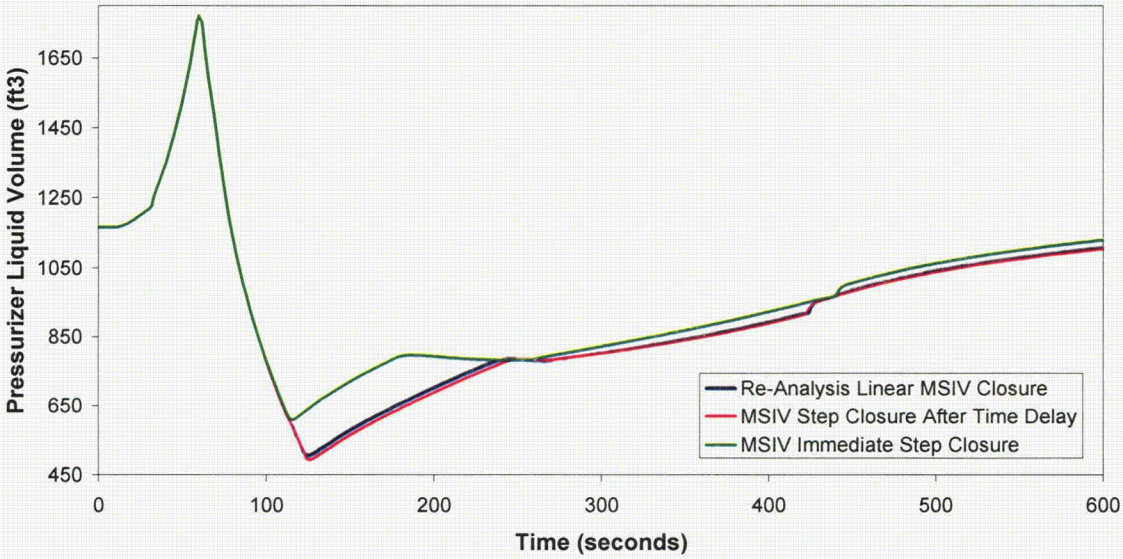

Figure 2-4: FLB - Pressurizer Liquid Volume vs Time

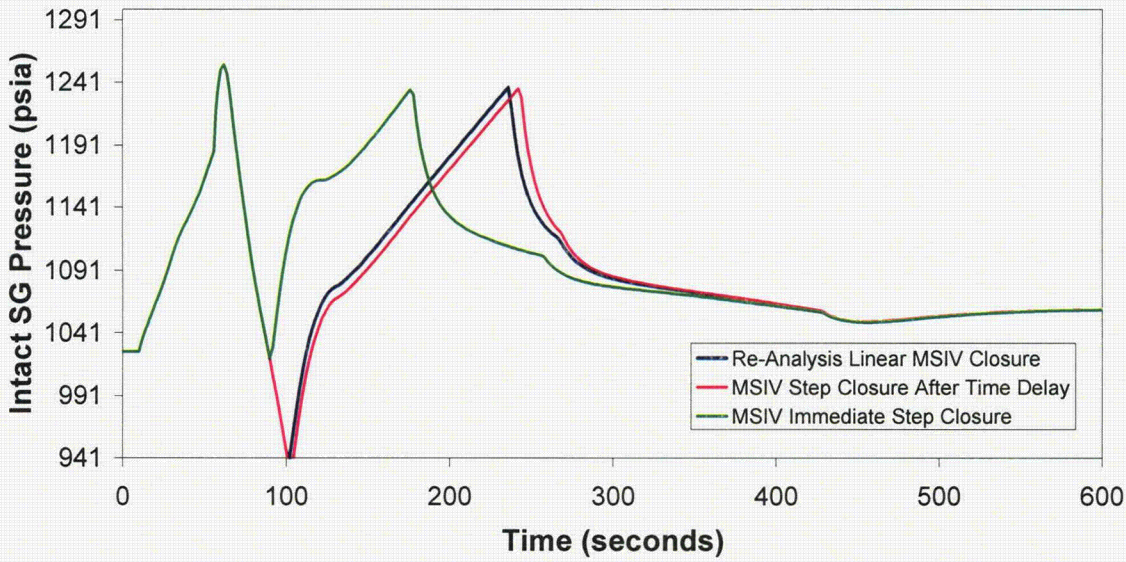

Figure 2-5: FLB - Intact SG Pressure vs Time

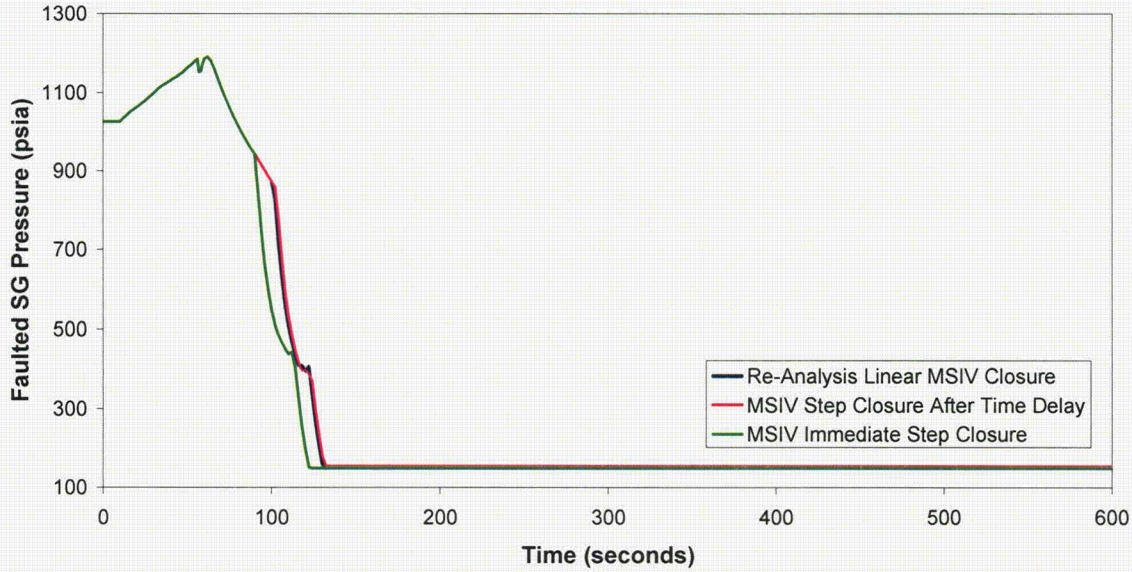

Figure 2-6: FLB - Faulted SG Pressure vs Time

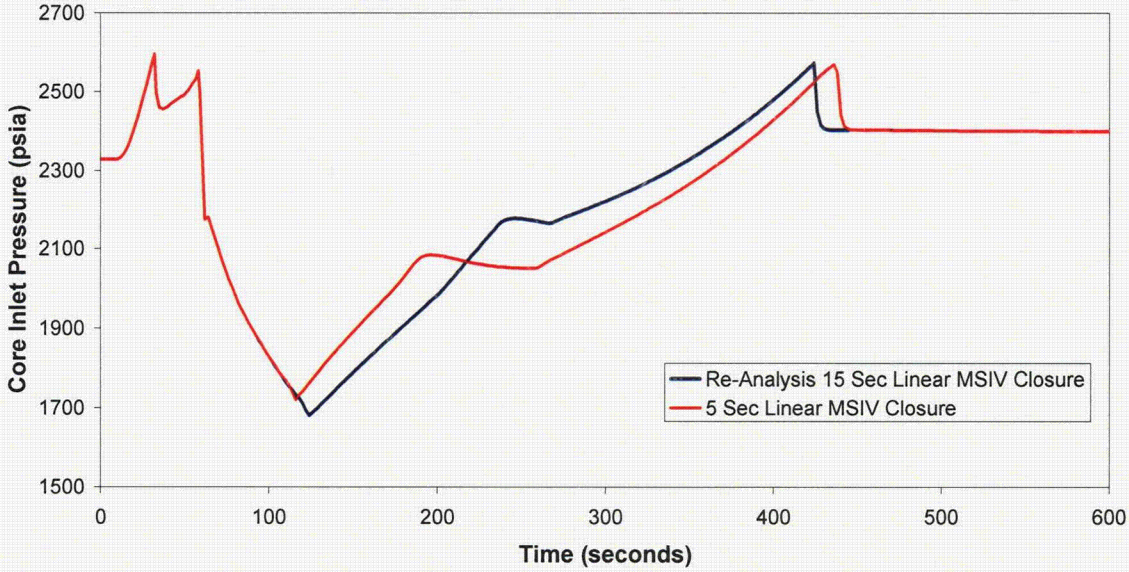

Figure 2-7: FLB - Core Inlet Pressure vs Time

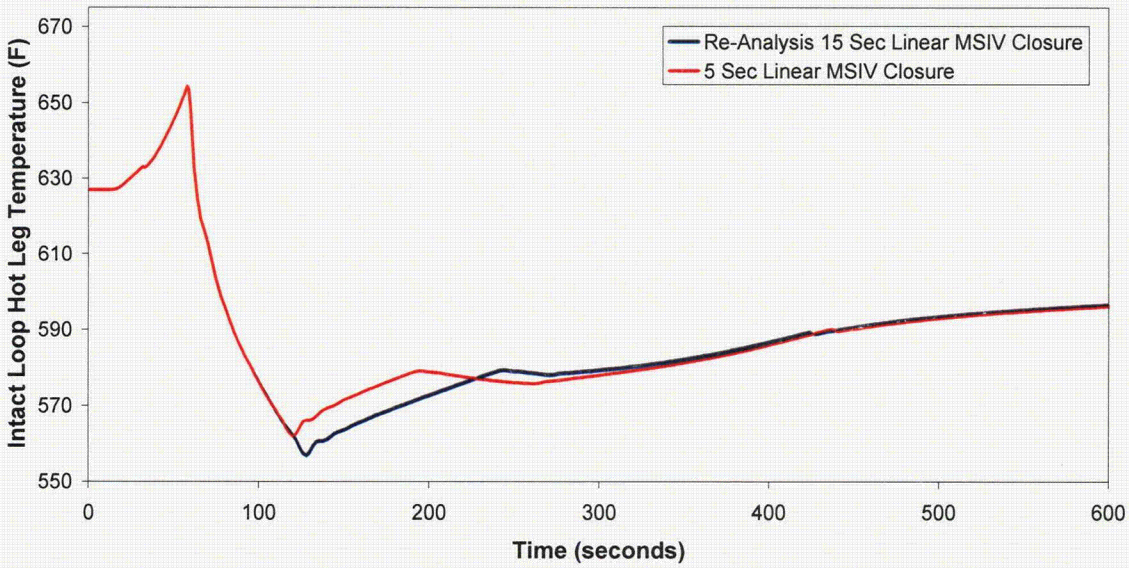

Figure 2-8: FLB - Intact Loop Hot Leg Temperature vs Time

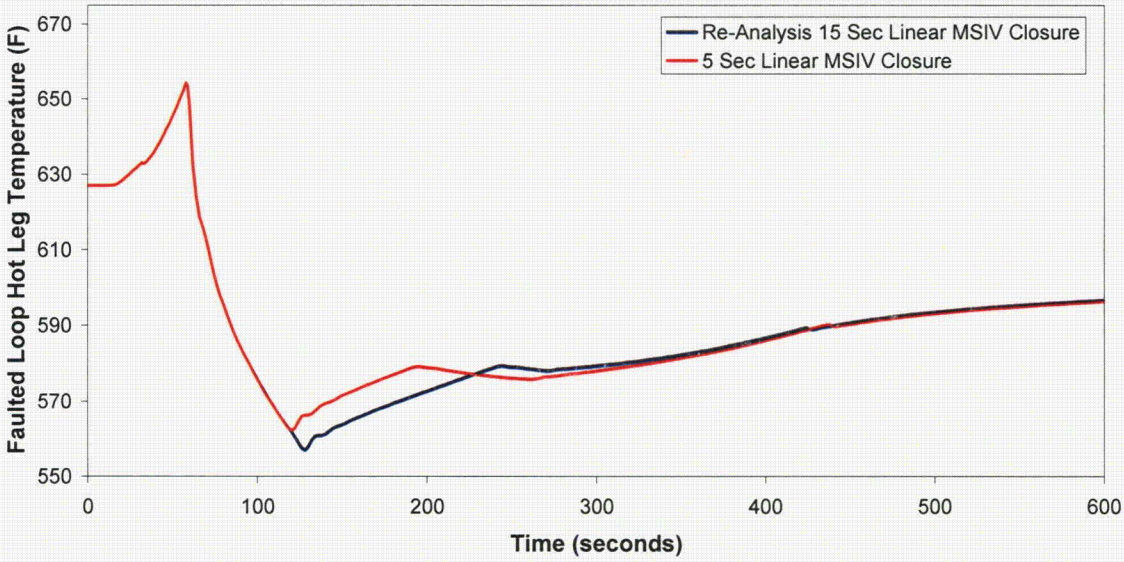

Figure 2-9: FLB - Faulted Loop Hot Leg Temperature vs Time

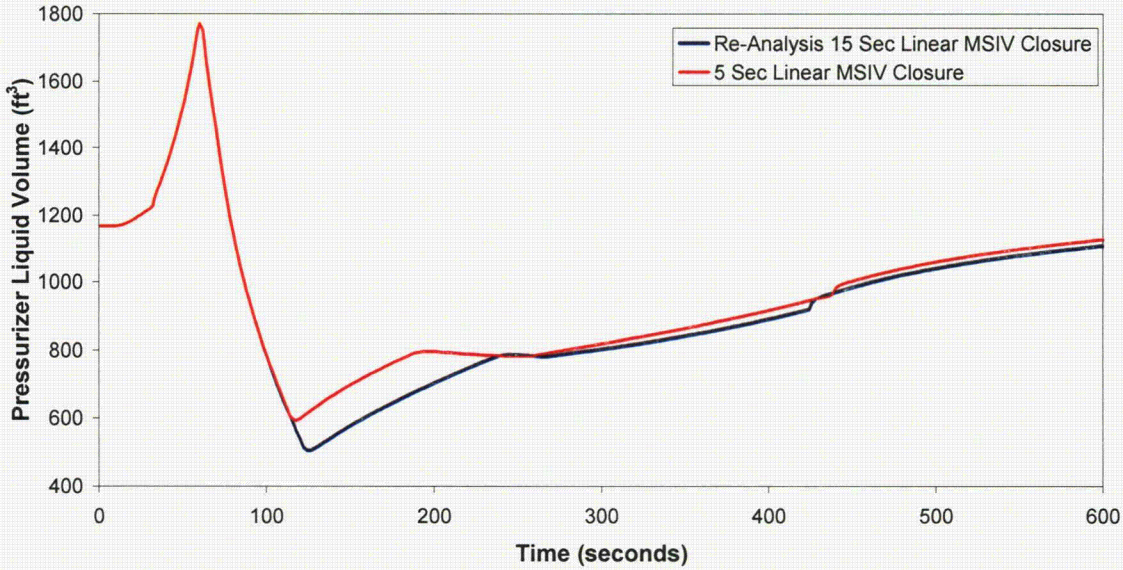

Figure 2-10: FLB - Pressurizer Liquid Volume vs Time

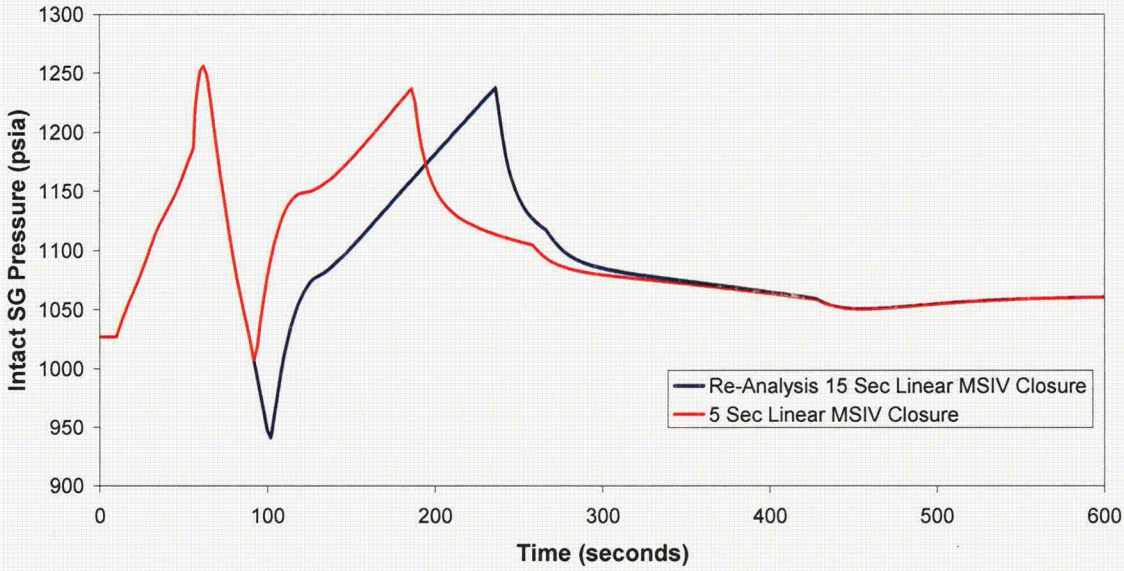

Figure 2-11: FLB - Intact SG Pressure vs Time

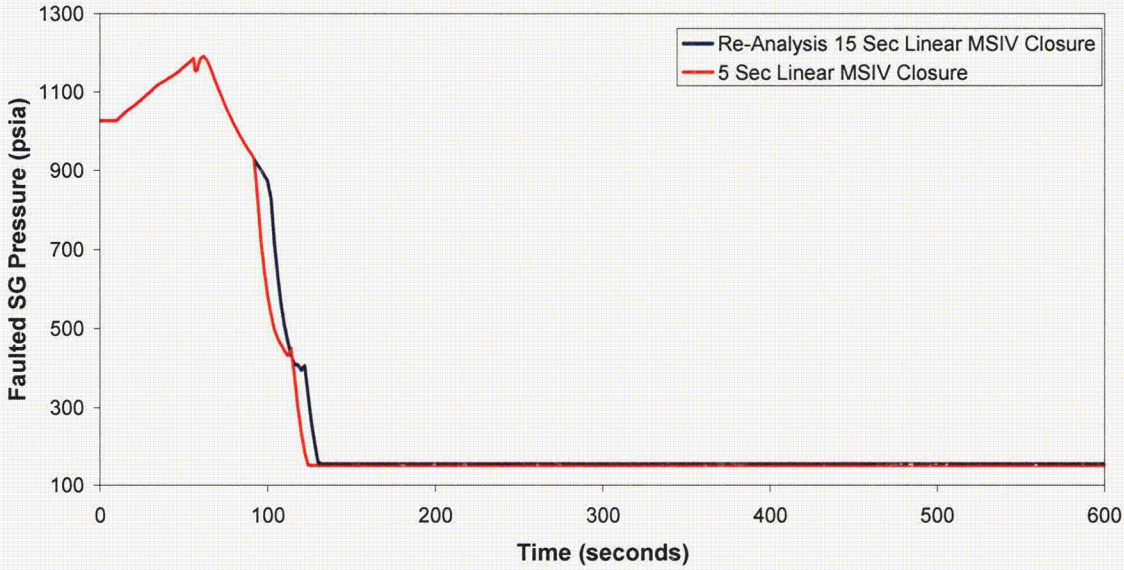

Figure 2-12: FLB- Faulted SG Pressure vs Time

SGTR:

Figures 2-13 through 2-16 presents the comparison between the differing valve closure patterns (linear closure or step closure with a time delay) as shown by the changes to the SGTR relevant plant parameters, such as: pressurizer pressure, differential break flow, integrated break flow, and faulted SG liquid volume.

Examination of the figures reveal that automatic function operation may result in slight changes during the transient progression, for instance, the faulted SG fills more rapidly (Figure 2-15) if the valve closure pattern of a step closure after time delay is assumed. Consequently, water relief through the safety valve begins earlier when the faulted SG and associated steam line up to the MSIV are filled with water. The sequence of events, with respect to the timing of the automatic and operator actions, for both cases are compared in Table 2-1. Note: The Table 4.5-2 Overfill Scenario values of ET 07-0004, Attachment I, should be the re-analysis with linear MFIV closure pattern values of Table 2.1.

However, the integrated break flow comparison, presented in Figure 2-16, shows that the comparative results are essentially identical and therefore, the effect of the differing valve closure pattern assumptions is negligible. Note: The response to question 4 below, indicated that the single-effect result of extending the valve closure time to a 15 second duration, for both SGTR events, was a slight reduction in the integrated mass released from the primary to secondary.

Table 2-1: Comparison of Sequence of Events for a SGTR with Forced Overfill Analysis

System Response/Operator Action	Time (seconds)	
	Re-Analysis Linear MFIV Closure	MFIV Step Closure After Time Delay
SGTR Occurs	0.0	0.0
Reactor Trip	153.9	153.9
AFW Injection	183.9	183.9
SI Signal	204.1	217.2
SI Enters RCS	219.1	232.2
Faulted SG Isolated	1284.1	1297.2
Initiate RCS Cooldown	2004.1	2017.2
Terminate RCS Cooldown	2388.2	2351.3
Initiate RCS Depressurization	2868.2	2831.3
Terminate RCS Depressurization	3032.8	2990.0
Terminate SI	3334.0	3290.0
Pressure Equalization	6214.9	6222.9

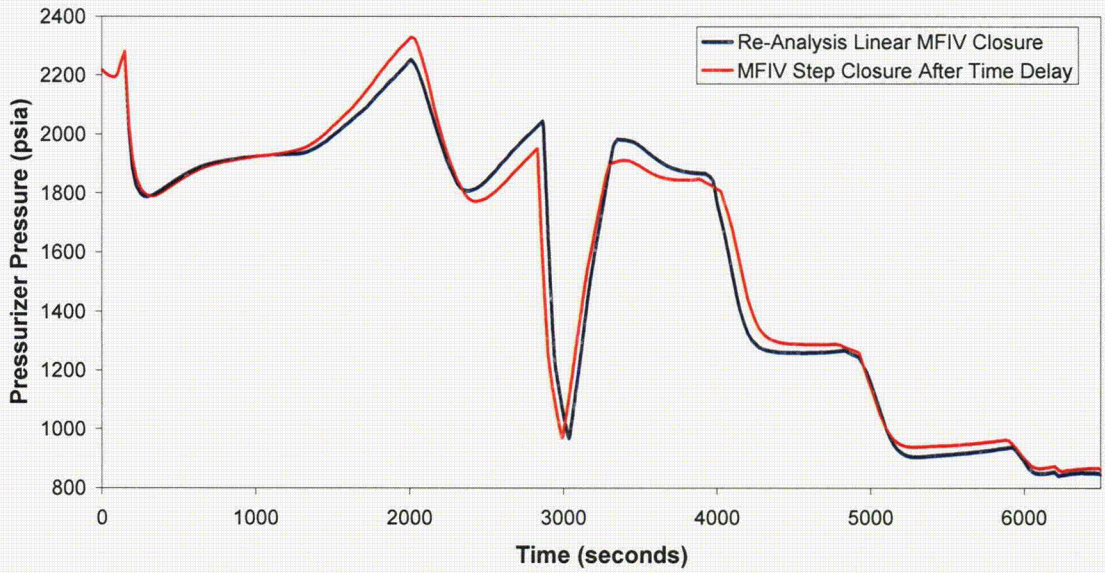

Figure 2-13: SGTR Overfill Analysis - Pressurizer Pressure vs Time

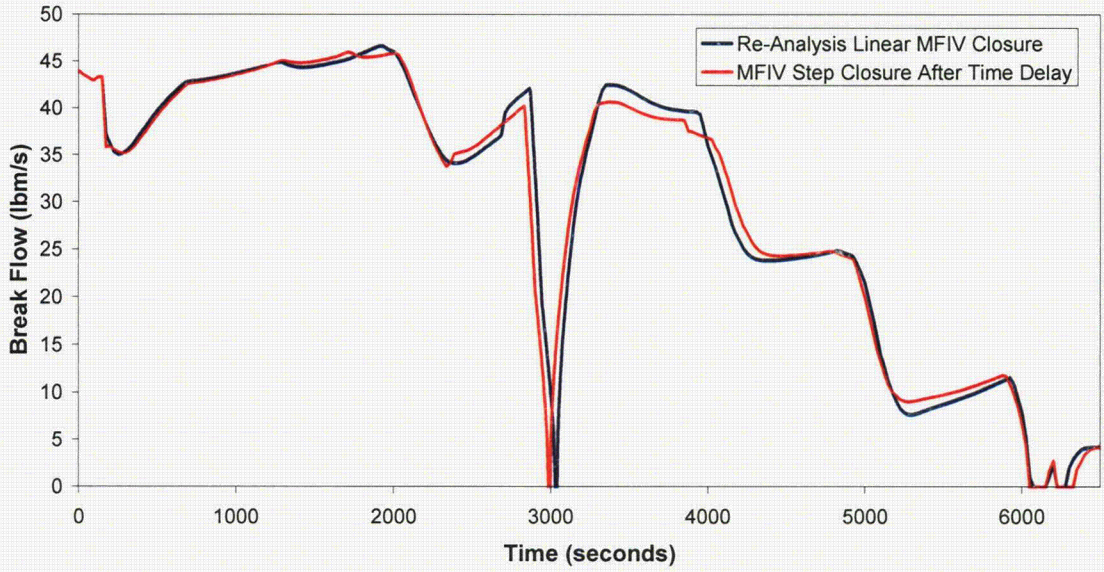

Figure 2-14: SGTR Overfill Analysis - Break Flow vs Time

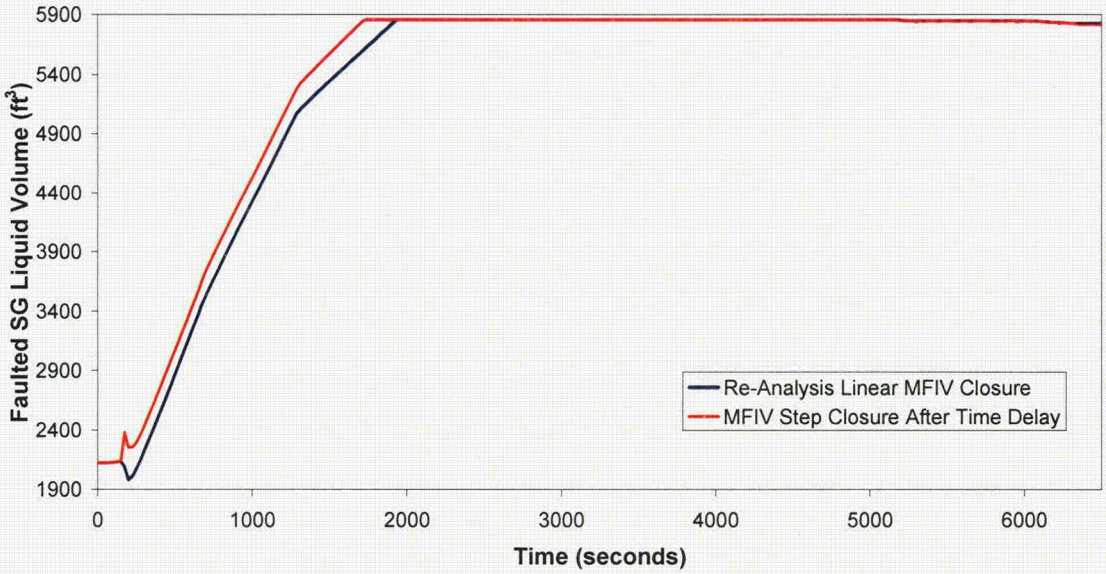

Figure 2-15: SGTR Overfill Analysis – Faulted SG Liquid Volume vs Time

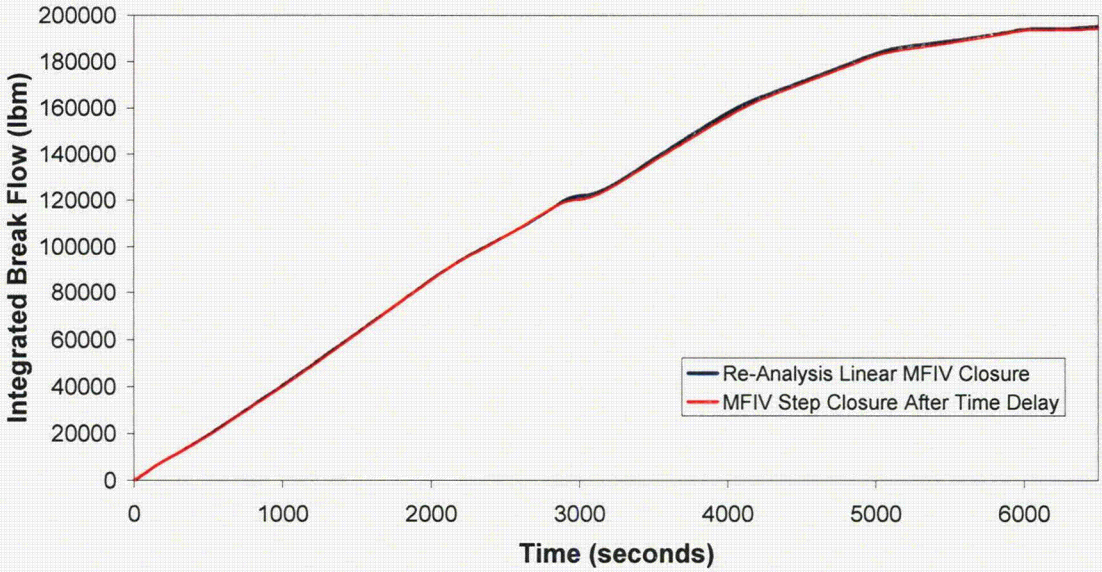

Figure 2-16: SGTR Overfill Analysis – Integrated Break Flow vs Time

SLB:

Figures 2-17 through 2-20 present the relevant plant parameters (normalized core power, core inlet pressure, core inlet enthalpy and core inlet flow) following a major steamline rupture with different MSIV and MFIV closure pattern (linear closure or step closure with a delay) combinations assumed.

Figures 2-21 through 2-23 present a comparison of the system response for the same parameters following a postulated SLB for the re-analysis closure pattern combination with an assumed closure period duration of either 5 or 15 seconds.

Table 2-2 presents the minimum departure from nucleate boiling ratio (DNBR) comparative results between a combination of MSIV and MFIV closure patterns.

Examination of the figures and table leads to the conclusion that the system response is as anticipated, with the assumed valve closure pattern combination corresponding to the greater cooldown of the primary causing the most rapid ascent to power and the most limiting minimum DNBR.

However, an inspection of the vendor supplied flow characteristics of the replacement MSIVs and MFIVs reveal that these characteristics are less limiting than the valve closure pattern assumed in the analysis for the SLB. In addition, Figures 2-21 through 2-23, and also Figures 2-24 and 2-25, provide certainty that the valve closure duration of 15 seconds assumed in the analyses is conservative.

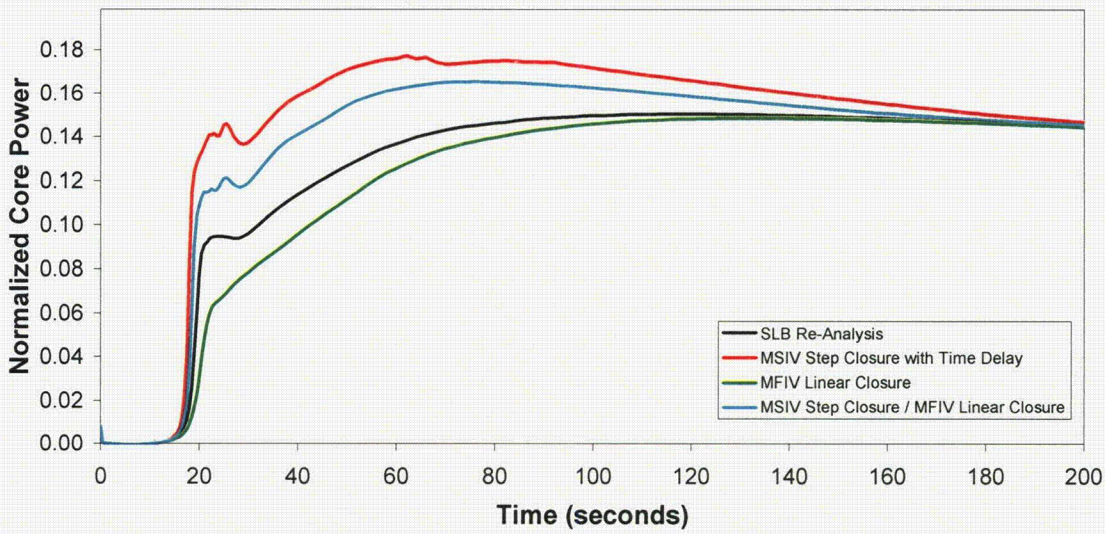

Figure 2-17: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Power vs Time

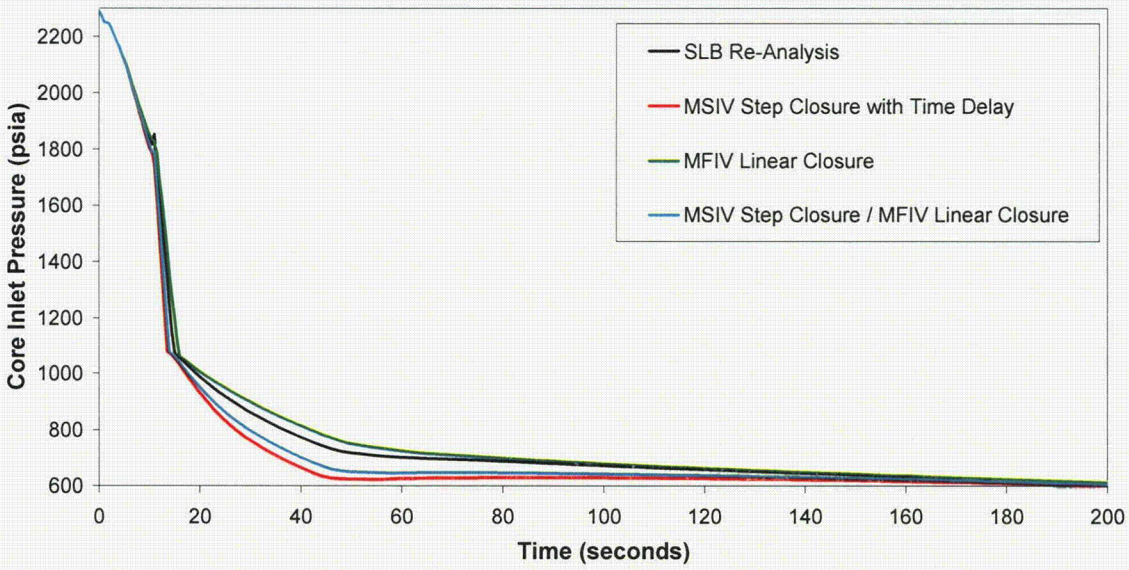

Figure 2-18: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Pressure vs Time

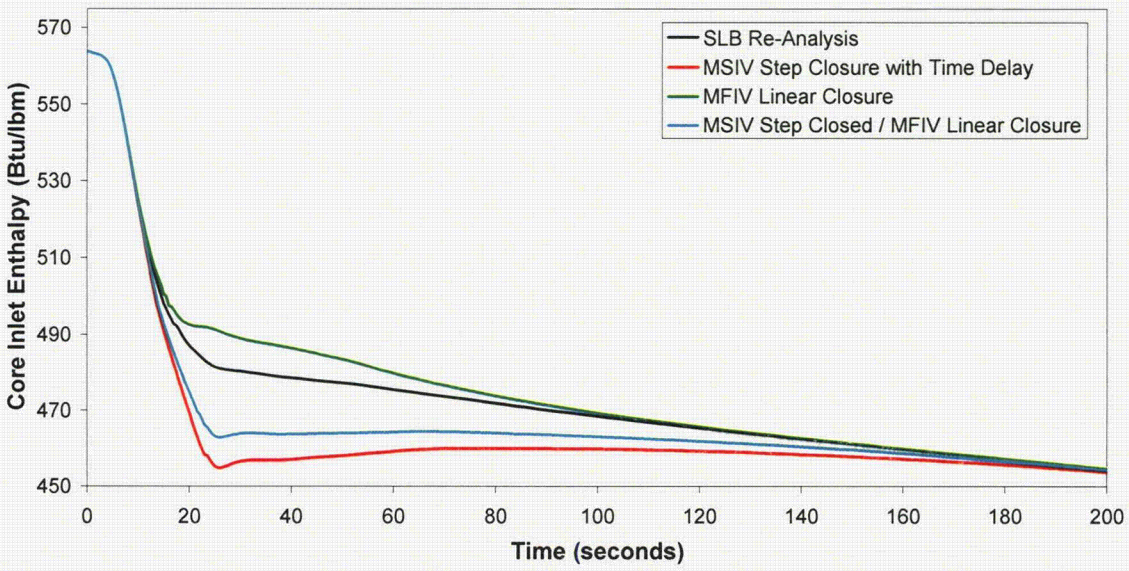

Figure 2-19: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Enthalpy vs Time

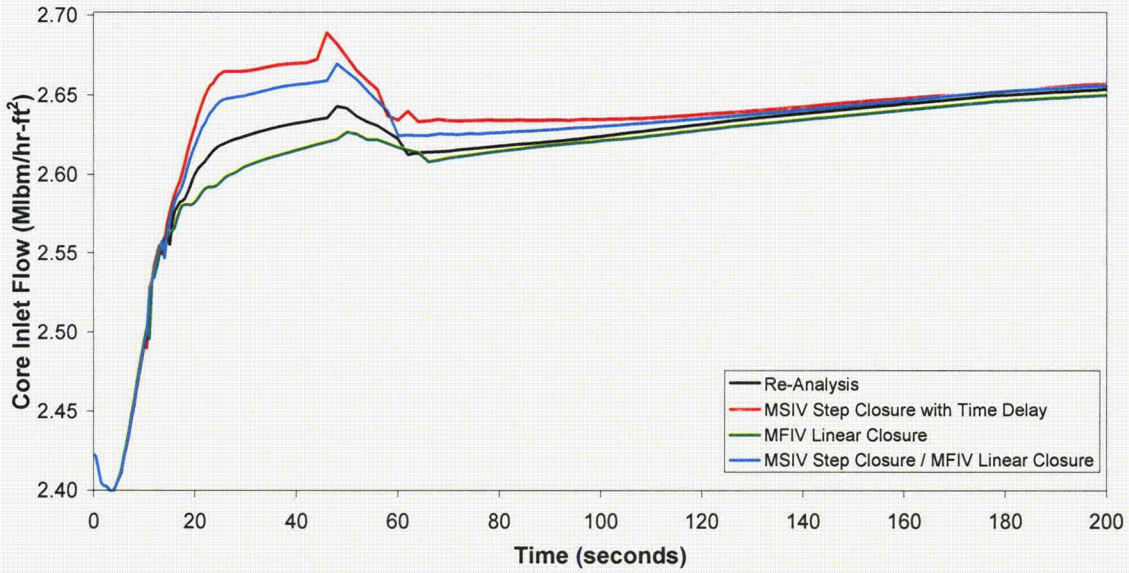

Figure 2-20: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Flow vs Time

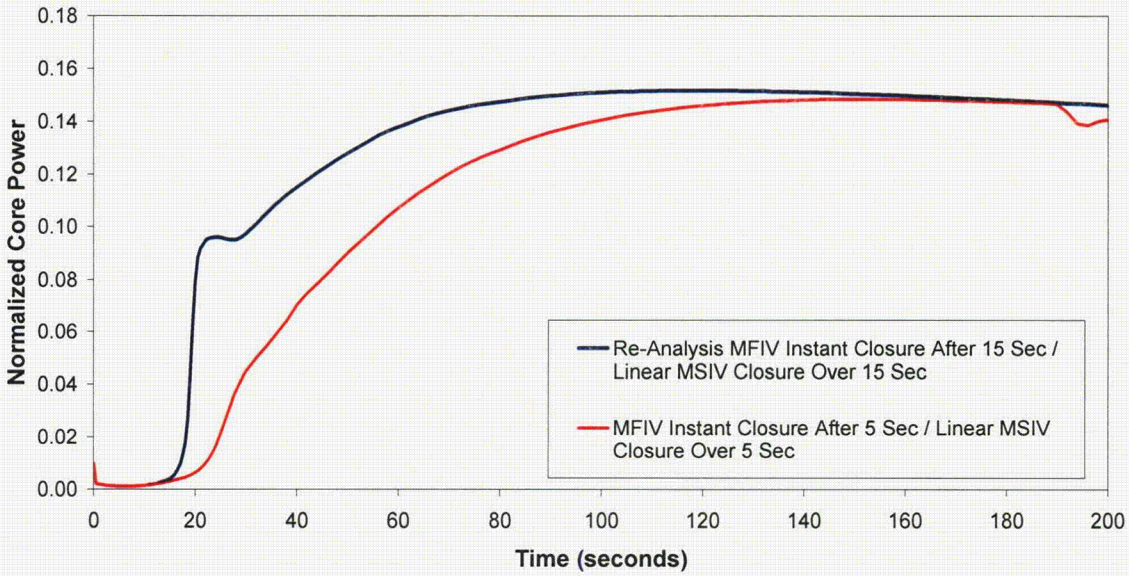

Figure 2-21: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Power vs Time

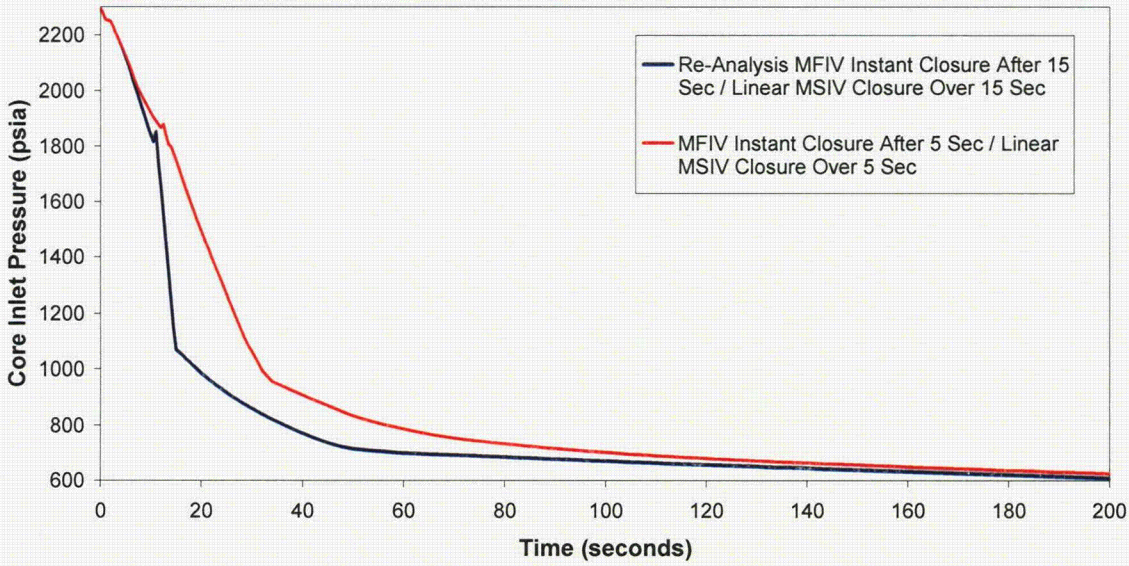

Figure 2-22: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Pressure vs Time

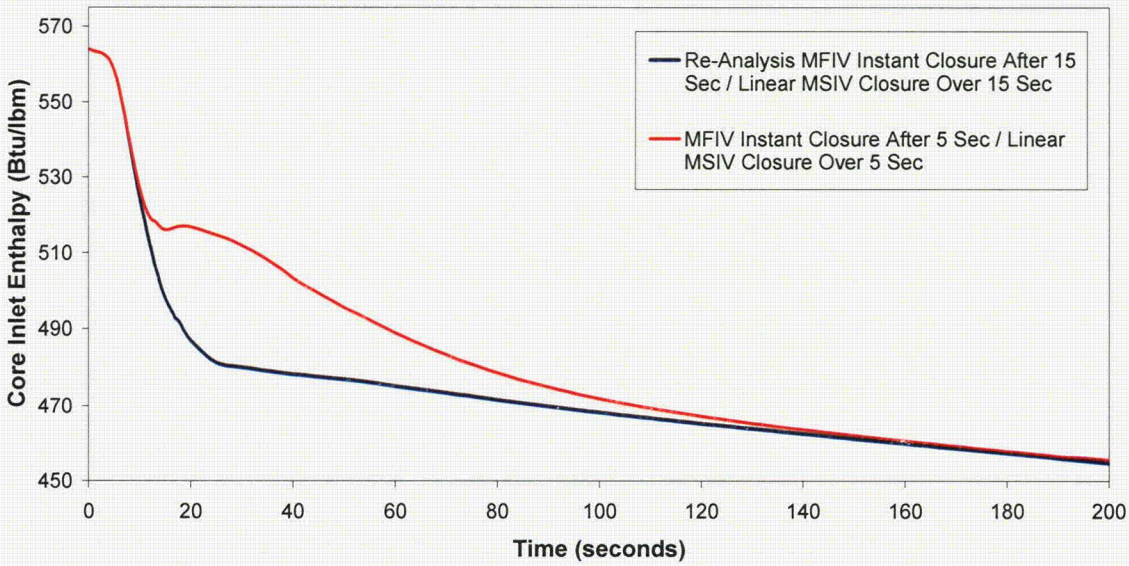

Figure 2-23: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Core Inlet Enthalpy vs Time

Table 2-2: Comparison of the Calculated Minimum DNBR Following a SLB Event with Different MSIV/MFIV Closure Patterns

SLB: Valve Closure Model	Mininum DNBR (Design Limit is 1.50)
Re-Analysis: Linear Closure of MSIVs and Step Closure of MFIVs with Time Delay	1.652
Linear Closure of MSIVs/MFIVs	2.211
Step Closure of MSIVs with Time Delay and Linear Closure of MFIVs	1.416
Step Closure of MSIVs/MFIVs with Time Delay	1.343
Current AOR for SLB	1.730

Valve Performance Test Data – Item (1)

As described in Section 3.0 of letter ET 07-0004, dated March 14, 2007, the MSIV/MFIV replacement actuators are system-medium actuators, with an inherent design feature being that the stroke time is a function of the system conditions.

Figures 2-24 and 2-25 present the single train and dual train valve performance test information, the acceptance limit for the surveillance test, and the safety analysis limit used for the FLB, SGTR, and SLB re-analyses.

From the information presented in the limiting FLB, SGTR, and SLB event pressure transient Figures 2-26, 2-27, 2-28 and 2-29, it is shown that the system pressures are significantly higher than 400 psia at the times the MSIVs and MFIVs are required to close. Thus, the stroke times of these valves are anticipated to be substantially lower than the 15 seconds assumed in the re-analyses for these events. Therefore, the closure time of 15 seconds assumed in the re-analyses is appropriate and conservative for the system conditions predicted for the FLB, SGTR and SLB events.

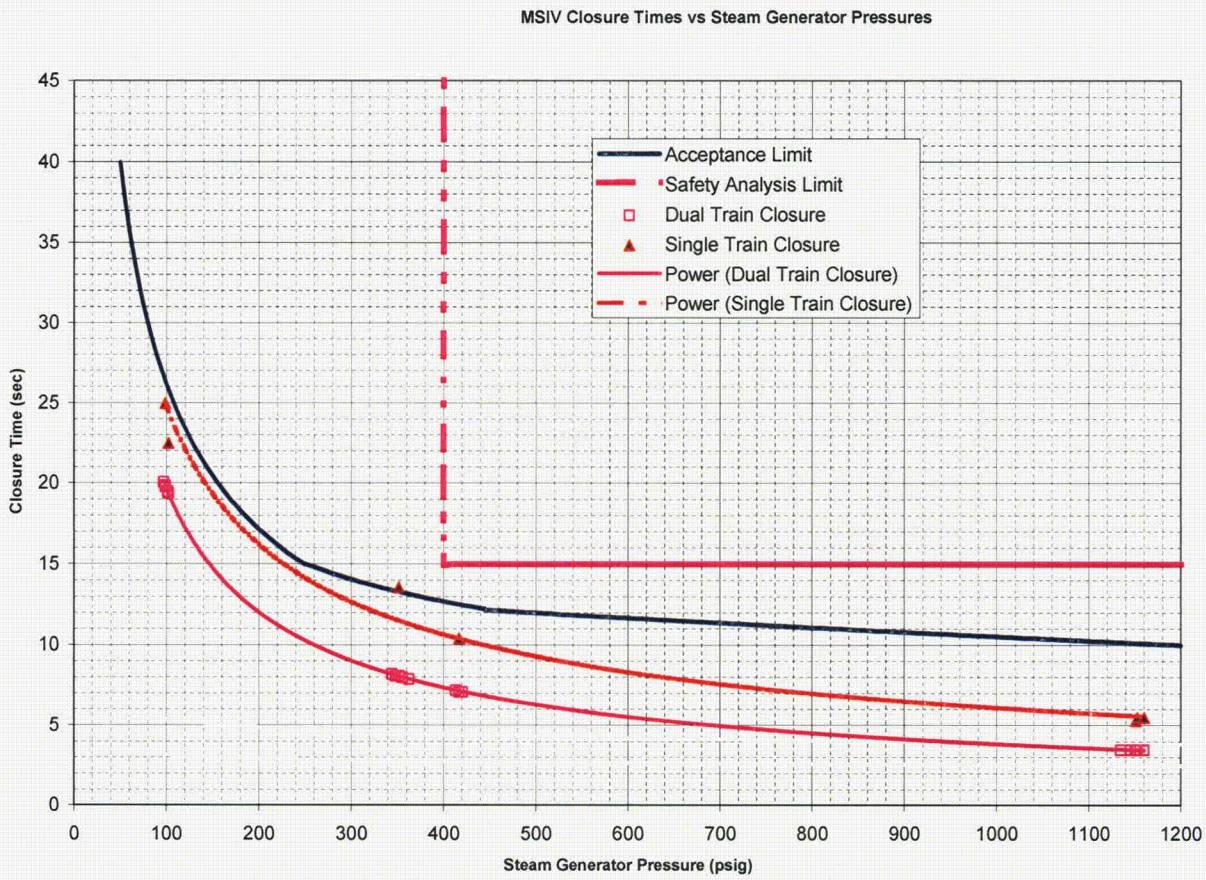

Figure 2-24: MSIV Performance Curves

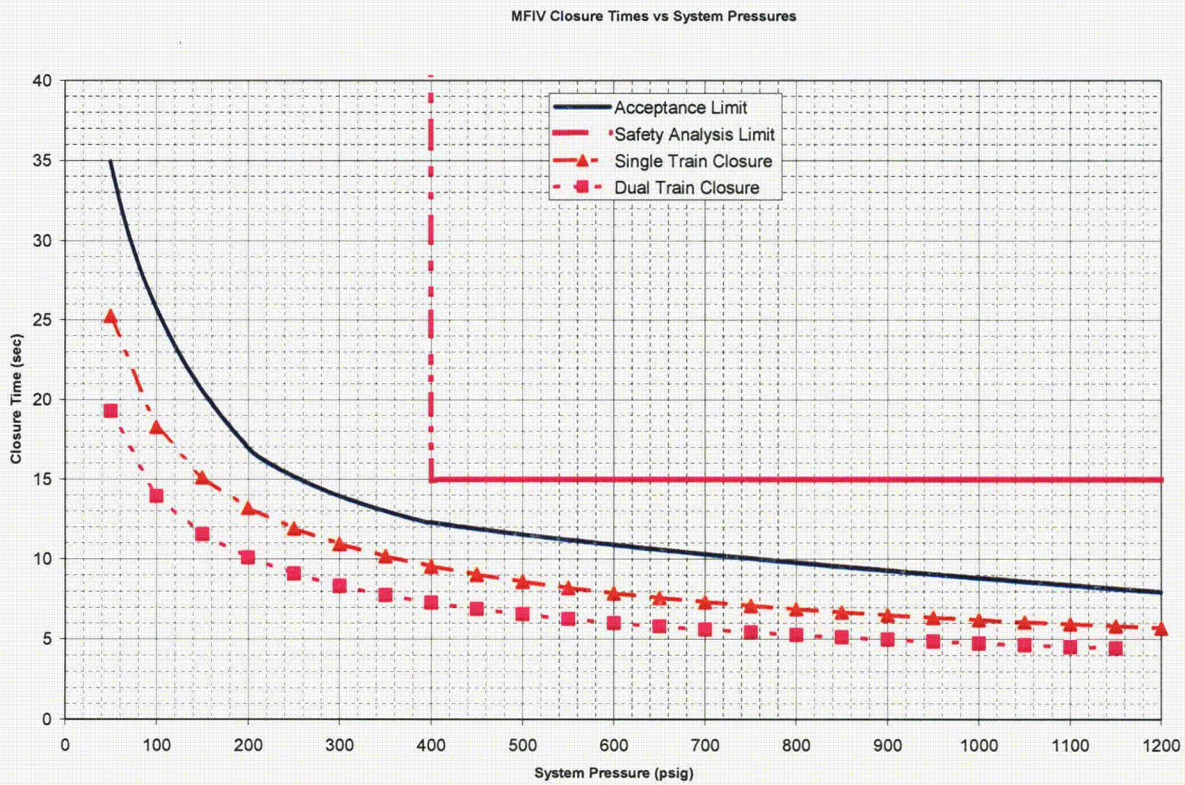

Figure 2-25: MFIV Performance Curves

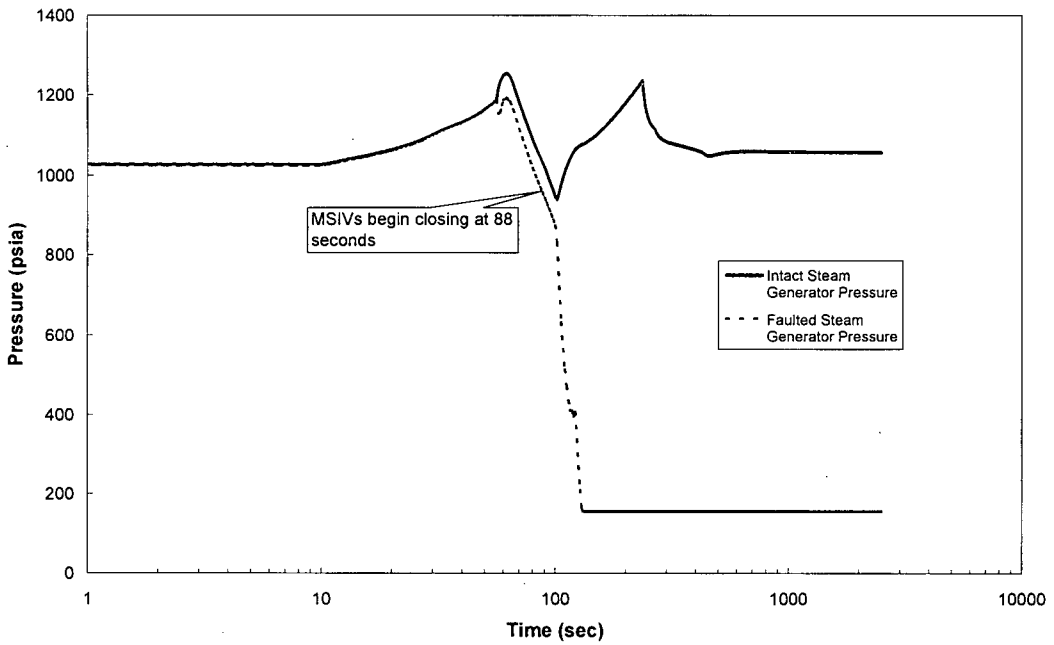

Figure 2-26: FLB: Intact and Faulted Steam Generator Pressures vs. Time

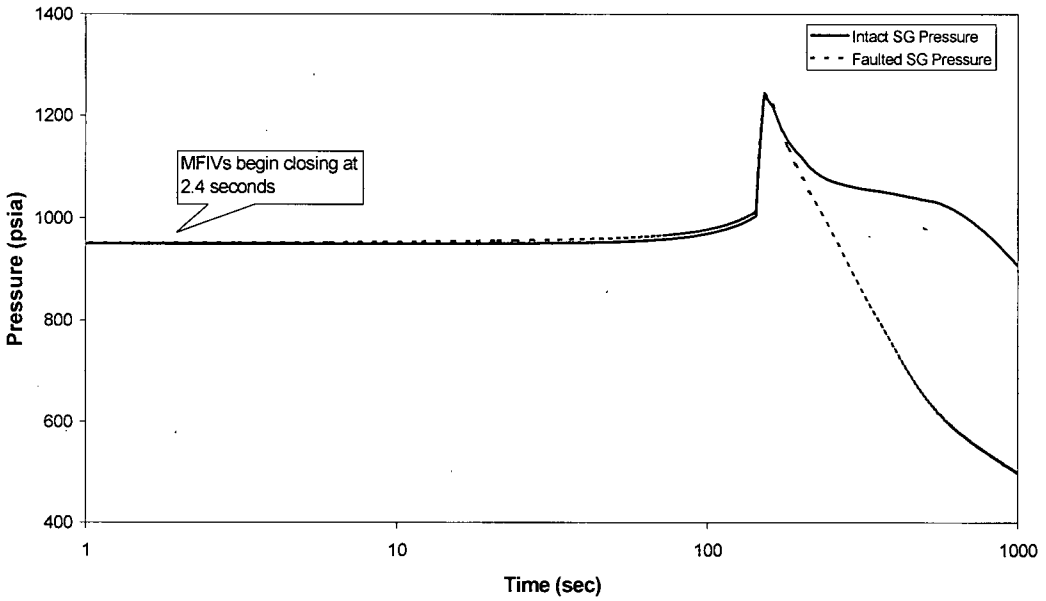

Figure 2-27: SGTR with Stuck Open ARV: Intact and Faulted Steam Generator Pressures vs. Time

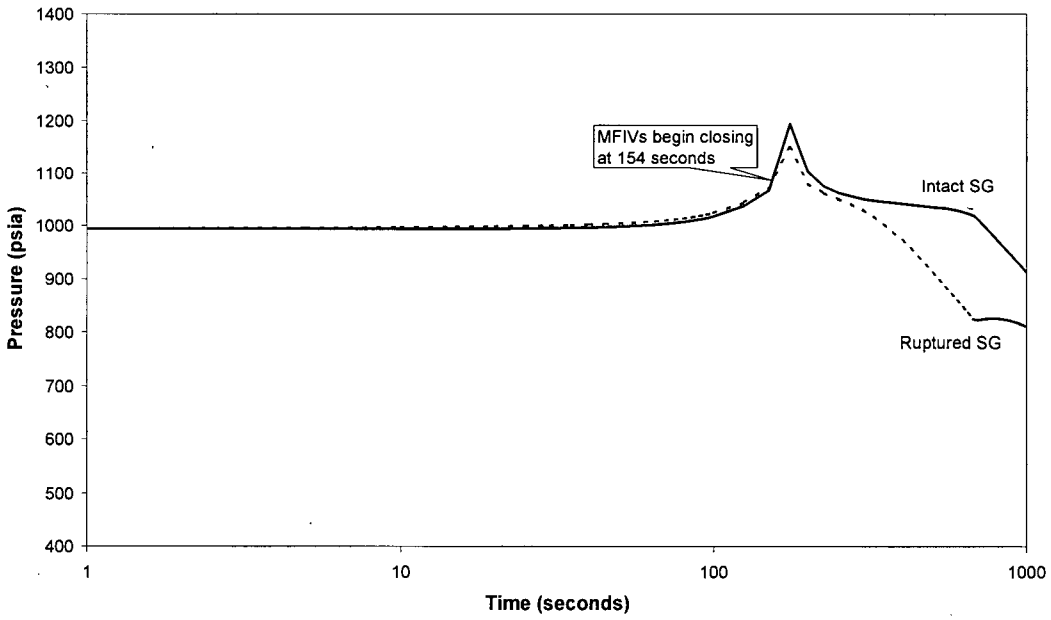

Figure 2-28: SGTR with Forced Overfill: Intact and Faulted Steam Generator Pressures vs Time

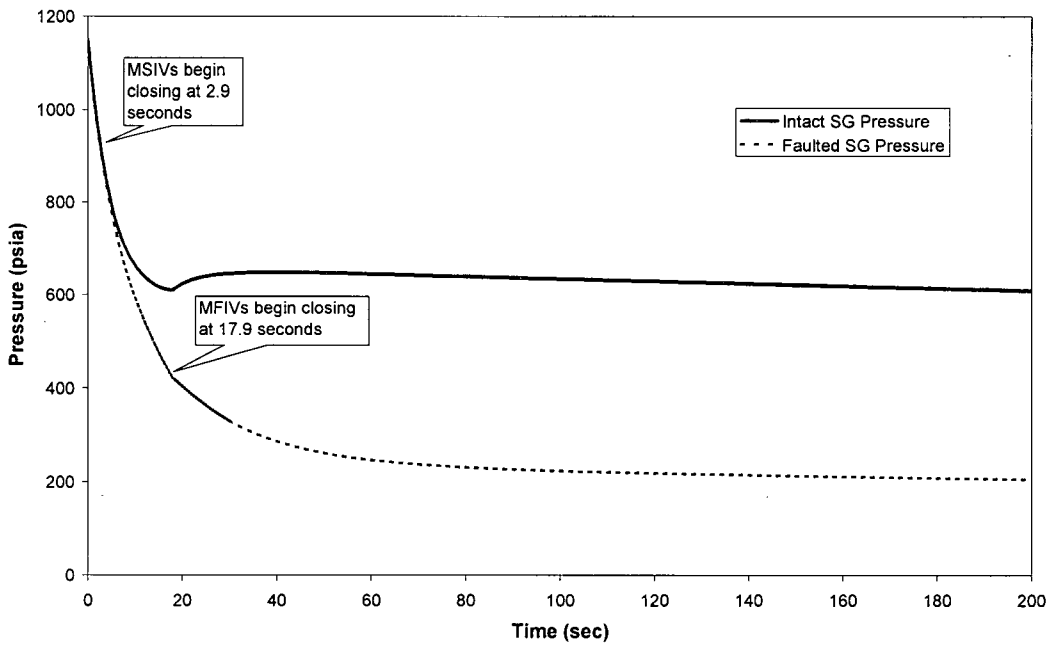

Figure 2-29: SLB: Intact and Faulted Steam Generator Pressures vs Time

Valve Performance Test Data – Item (2)

The flow characteristics of a replacement MSIV/MFIV, as provided by the vendor, are presented in Figure 2-30. The blue line (BLD) is the recorded flow area during a full scale SLB simulation. Note: The Trend Line (red) is derived from the recorded flow area, but is not pertinent to the re-analyses.

The replacement valve(s) provides a quick closing characteristic, i.e., large changes in effective flow area for small changes in stem travel. It should be noted that this closure characteristic allows a lesser amount of flow to pass through the valve during the stroke closing process than the valve closure patterns (linear closure or step closure with time delay) assumed in the re-analyses for the FLB, SGTR, and SLB events. For example, a 50 % stroke time, super-imposing a linear valve closure pattern line on Figure 2-30, would result in a 50 % flow area reduction, compared to the recorded flow area reduction of approximately 90 %.

This test information confirms the adequacy of the conservative assumption of the linear valve closure pattern used in the re-analyses.

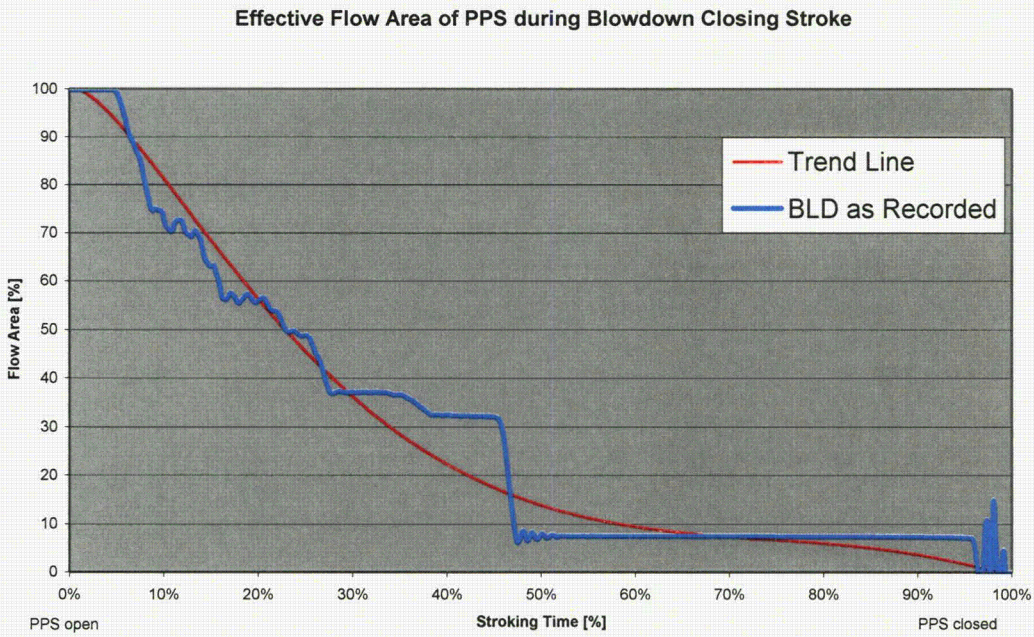

Figure 2-30: Effective Flow Area of the MSIV/MFIV vs Stroke Time during Blowdown Closing Stroke

3. *The response to RAI-5 referenced the Westinghouse Safety Analysis Standard (SAS) 12.0 to support for the re-analyses changes of the initial values of the RCS flow and pressure assumed in the AOR. List the NRC safety evaluation that approved the SAS 12.0. If SAS 12.0 is not an NRC-approved document, provide SAS 12.0 for the NRC staff to review and approve. Alternatively, provide the results of a plant-specific sensitive analysis for the limiting SLB event to demonstrate that the changes of the initial RCS flow and pressure result in a highest return-to-power level and a lowest minimum DNBR, are thus conservative.*

   The response also indicated that for assumption #9, the auxiliary feedwater (AFW) flow of 2020 gpm total to the faulted SG assumed in the AOR is changed to a total flow of 2059.75 with 1360.13 gpm to the faulted SG and 699.62 to the intact SGs. Discuss the effects of the change of the AFW flow on the results of the SLB reanalysis. In addition, discuss the changes of the method, computer codes and assumptions used to determine the AFW flow during an SLB event, and justify that the changes are applicable to the pressure and temperature ranges for the limiting SLB conditions.

Response: The Westinghouse Safety Analysis Standards (SAS) describe the general method for performing specific transient analyses and are advantageously consulted for guidance as they include, among other things, the pertinent effect on the analysis of a specific transient for the resolution of various licensee issues. The SAS is a guidance document that is not NRC approved. However, the WCNOC response to this question relies on the alternative approach of providing plant-specific analysis results to demonstrate the conservatism of the initial condition changes.

At the Hot-Zero Power (HZP) initial conditions, the initial RCS pressure is assumed to be equal to the nominal RCS pressure. For a SLB, the depressurization is driven by the break and is not significantly affected by the initial conditions. Plant-specific sensitivity studies have shown that the core power, inlet temperatures and pressures are virtually identical to the re-analysis case if the initial pressure of 2200 psia assumed in the AOR is assumed for the initial pressure. In addition, an RCS flow value corresponding to the Thermal Deign Flow (TDF) is assumed for the HZP SLB re-analysis. An increase in the peak heat flux associated with the assumption of a flow higher than TDF would be offset by crediting the higher flow rate in the DNBR calculation.

The preceding discussion is further illustrated by Figure 3-1, which presents a plant-specific comparison of the core power transients with differing initial RCS pressures and flows. The associated DNBR results presented in Table 3-1 reveal that the SLB re-analysis with a nominal initial pressure and TDF actually yields a slightly lower minimum DNBR, as compared with the analysis with AOR initial conditions (RCS flow and pressure), and is therefore conservative.

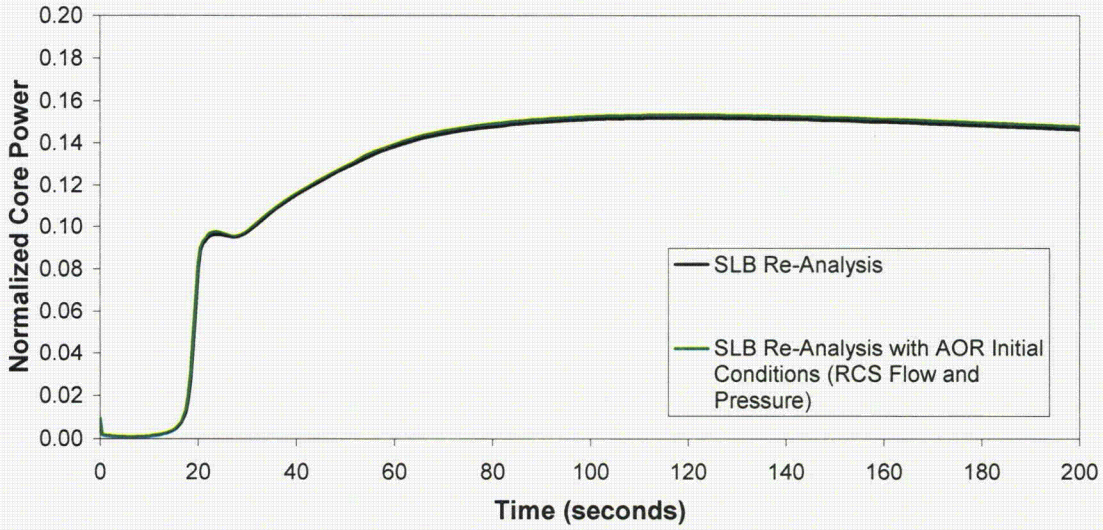

Figure 3-1: Comparison of Core Power Response to a SLB Event with Different Initial Conditions

Table 3-1: Comparison of the Calculated Minimum DNBR Following a SLB Event with Different Initial Conditions

SLB: Initial Conditions	Mininum DNBR (Design Limit is 1.50)
SLB Re-Analysis	1.652
SLB Re-Analysis with AOR Initial Conditions (RCS Flow and Pressure)	1.656

The auxiliary feedwater (AFW) delivered to the faulted SG directly impacts the cooldown of the primary system during a SLB transient. The maximum AFW flow to the faulted SG tends to maximize the extraction rate of the RCS. Therefore, the maximum possible flowrate, including flow from all motor-driven and turbine-driven AFW pumps was conservatively assumed in the current AOR. Specifically, the AFW flow of 2020 gpm total, representing the approximate total capability of the AFW System, to the faulted SG and 0 gpm to the intact SGs were assumed in the current AOR. Note that this maximum flowrate comprises 500 gpm from each motor-driven AFW pump and 1000 gpm from the turbine-driven AFW pump. However, a realistic yet conservative assumption on the AFW flowrate reflecting the actual operational capabilities of the AFW System, is used in the SLB re-analysis in order to offset a portion of the penalties associated with the significantly longer stroke time for the replacement MSIVs.

Figure 3-2 and Table 3-2 below show the effects of the AFW change on the core power transient and the minimum DNBR following a postulated major SLB. As expected, the higher AFW flow to the faulted SG assumed in the AOR indeed results in a slightly higher return-to-power and consequently the calculated DNBR margin to the safety analysis limit is somewhat reduced.

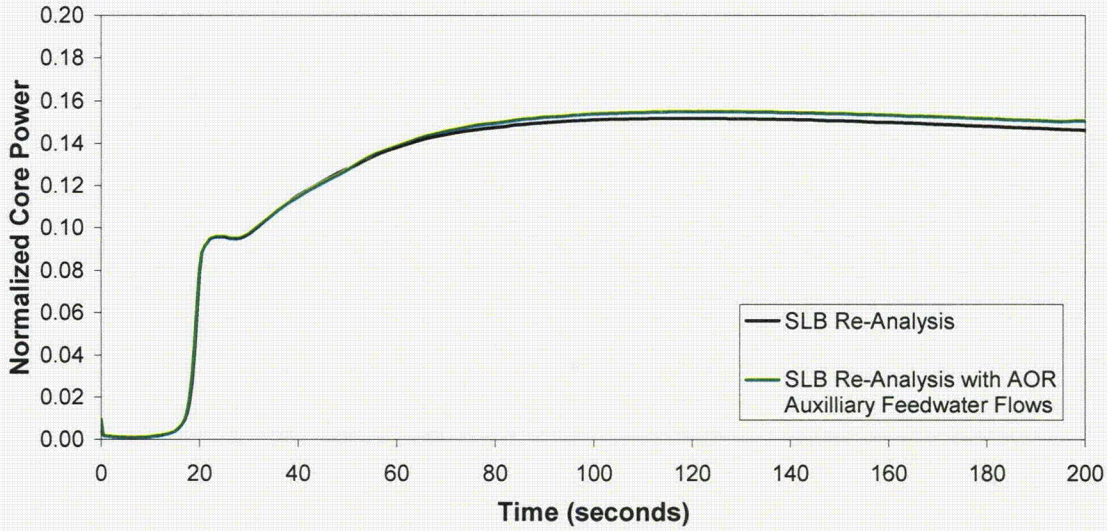

Figure 3-2: Comparison of the Calculated Minimum DNBR Following a SLB Event with Different AFW Flow Assumptions

Table 3-2: Comparison of the Calculated Minimum DNBR Following a SLB Event with Different AFW Flow Assumptions

SLB: Auxiliary Feedwater	Mininum DNBR (Design Limit is 1.50)
SLB Re-Analysis	1.652
SLB Re-Analysis with AOR Auxiliary Feedwater Flows	1.631

The AFW flow used in the SLB core response re-analysis is determined based on a fluid piping network model developed for the AFW System using the computer program "FLOW1200" [Reference 1]. The model accounts for the as-built piping configuration including various hydraulic components such as pumps, valves and pipe fittings. Note that the piping network model for the AFW System used in the re-analysis has been previously utilized in calculations to determine AFW flow for safety analyses. The major assumptions and modeling considerations are discussed as follows:

- In addition to the limiting single failure of one high-head safety injection train assumed in the SLB core response analysis, failure of the AFW control device is conservatively considered in the AFW flow calculation in order to maximize the AFW flow to the faulted SG and support the excess cooldown of the primary system during the transient. Specifically, the discharge flow control device installed on the AFW header common to both the motor-driven and turbine-driven AFW pumps is assumed to fail to perform its safety function. This device regulates the AFW flow from the motor-driven AFW pump to the faulted SG such that a minimum flow would be delivered to the intact SG from both the motor-driven and turbine-driven AFW pumps. Without the flow control device, all the

flow from the motor-driven AFW pump would go to the faulted SG and only the flow from the turbine-driven AFW pump would be delivered to the intact SGs.

- AFW flow is assumed to take suction from the condensate storage tank (CST).
- The mini-recirculation flowrate of 75 gpm for each motor-driven AFW pump and 145 gpm for the turbine-driven AFW pump are modeled as a constant external demand at the respective pump discharge junction node.
- The certified AFW pump characteristic curves (i.e., developed head vs. capacity) are used in the calculation.

Based on the assumptions and modeling techniques described above, the AFW flowrates in terms of SG pressures are calculated using the computer code FLOW1200 and the results are summarized as follows:

Steam Generator Pressure	**AFW Flow to Faulted SG (gpm)**	**AFW Flow to Intact SGs (gpm)**
All SGs @ 750 psia	933.76	925.10
Intact SGs @ 1200 psia, Faulted SG @ 14.7 psia	1360.13	699.62
Intact SGs @ 1200 psia, Faulted SG @ 400 psia	1160.43	789.65
Intact SGs @ 1200 psia, Faulted SG @ 800 psia	926.02	833.39

As shown in the above table, a total AFW flow of 2059.75 gpm, consisting of 1360.13 gpm to the faulted SG and 699.62 gpm to the intact SGs, used in the re-analysis corresponds to the situation whereby the intact SGs are maintained at 1200 psia and the faulted SG is depressurized to 14.7 psia.

As evident from Figure 3-3 below, for the pressure transient resulting from a limiting SLB event, the pressures of the intact SGs and the faulted SG remain above 600 psia and 200 psia, respectively. This means that the actual AFW flow delivered to the faulted SG is less than that assumed in the re-analysis and consequently the effect of the excess cooldown of the primary system would be less pronounced. Thus, the changes of the AFW flow are justified since the values are conservative and applicable to the pressure ranges for the limiting SLB conditions.

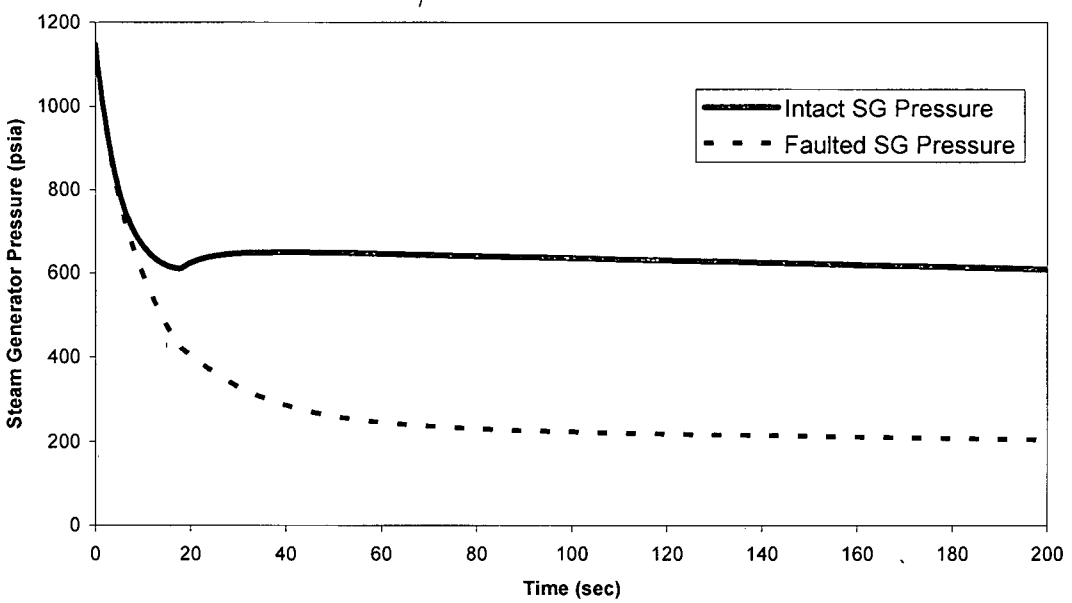

Figure 3-3: HZP Steam Line Break Transient with Offsite Power Double-Ended Rupture – Steam Generator Pressures vs Time

References:

1. Don J. Wood, User's Manual "Computer Analysis of Flow in Pipe Networks Including Extended Period Simulation", University of Kentucky, 1980.

4. *The response to RAI-3, RAI-4, and RAI-7 discussed the changes of the computer code, RCS flow, and operator times used in the SGTR re-analysis. The NRC staff compared the results of the re-analysis and the AOR for the SGTR events and found that the results of the re-analysis for the SGTR SG overfill case (Figures 4.5-1 through 4.5-10) are similar to those of the AOR (Figures 15.6-3a through 15.6.3j in the Updated Safety Analysis Report (USAR) for Wolf Creek, Revision 16). However, the NRC staff found that the results of the re-analysis for the SGTR with a stuck-open ARV case (Table 4.5-2 and Figures 4.5-11 through 4.5-20) are significantly different from those of the AOR (Tabled 15.6-1 and Figures 15.6-3k through 15.6-3t in the USAR, Revision 7) in both times of sequence of events, and magnitude of the perturbation for plant parameters including the pressurizer pressure, RCS temperatures, SG pressure and temperature, AFW flow, SG break flow and faulted SG water volume. Discuss the differences in the results of the re-analysis and AOR for the SG overfill case and the stuck-open ARV case identified above.*

Response: A review of the SGTR with Forced Overfill event with pre-accident iodine spike scenario dose consequence re-analysis results, as presented in Section 4.11 letter ET 07-0004, dated March 14, 2007, reveals that the primary radionuclide contribution (about 82% of the total thyroid dose) is from the flashed break flow released through the direct release path. This

flashed break flow provides an indication of the primary dose consequence contribution for the re-analyzed SGTR event scenarios.

The break flows presented in Figures 4.5-9 and 4.5-19 of letter ET 07-0004, for the SGTR with Forced Overfill event and for the SGTR with Stuck Open Atmospheric Relief Valve (ARV) events, respectively, are determined in the analyses from the resistance limited flow for which the primary-to-secondary pressure difference is the driving force. The ruptured site is connected to two primary fluid reservoirs, i.e., SG inlet and outlet plenums, via the segmented tube. Each segment provides a substantial resistance to fluid flow including frictional losses in the tube and pressure losses at the ruptured site. Note: The total break flow through the ruptured tube is the sum of the leakage from the long and short sections of the ruptured SG tube.

The SGTR with Forced Overfill scenario integrated break flow results (Figure 2-16) at the time Safety Injection (SI) is terminated differ from the comparable AOR results by 0.8%. As the assumed operator response times in the re-analysis are unchanged from those assumed in the AOR, the revised MFIV closure characteristics are the primary source of the minor reduction in the integrated mass release from the primary to the secondary, and by extension the slightly reduced radiological consequences of the transient.

As presented in Figure 4-9, the SGTR with Stuck-Open ARV scenario re-analysis break flow results based upon the revised operator response times and the AOR MFIV closure characteristics (revised operator response time case), are similar to the results presented in Figure 4.5-19 of letter ET 07-0004. The differences basically being due to the single-effect of extending the valve closure time to a 15 second duration.

The re-analysis integrated break flow results at the time SI is terminated are reduced nearly 4.2% due to the revised MFIV closure characteristics causing a reduction in the cooldown step time duration, as compared to the results from the revised operator response time case. However, the AOR integrated break flow results at the time SI is terminated are nearly 30% less than the comparable results from the revised operator response time case.

Therefore, the change in the time sequence of events due to the conservatively increased revised operator response times assumed are the primary cause of the resulting perturbation of plant parameters, presented in the Figures 4-1 through 4-10 below, as compared to the AOR results. Note: The minimal changes due to the revised MFIV closure characteristics are demonstrated by a comparison of Figures 4-1 through 4-10 with Figures 4.5-11 through 4.5-20 of letter ET 07-0004.

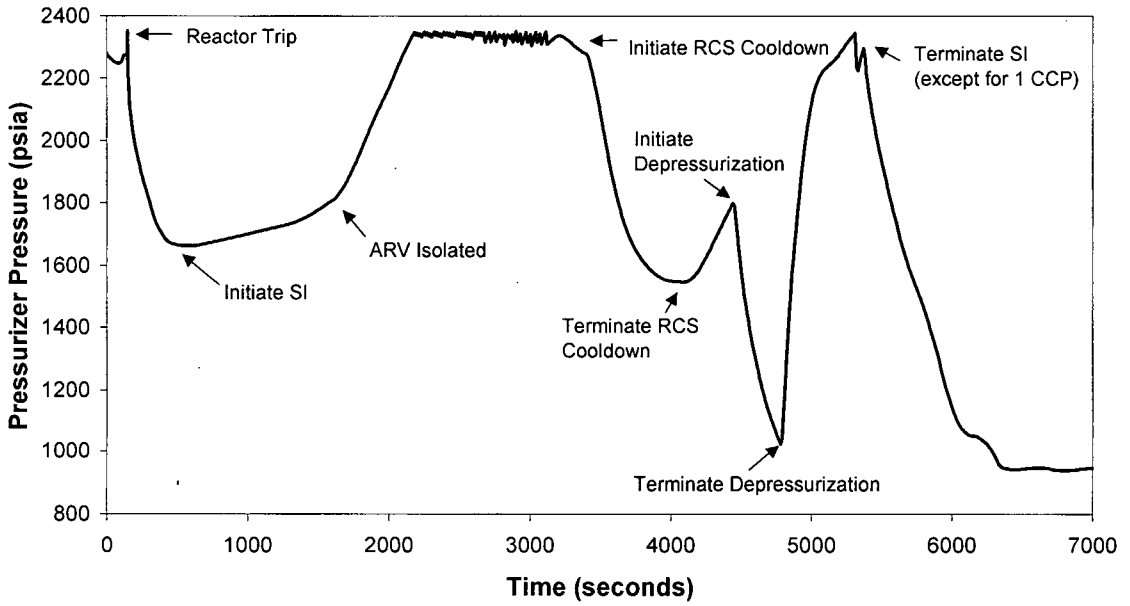

Figure 4-1: SGTR w/ Stuck-Open ARV Pressurizer Pressure

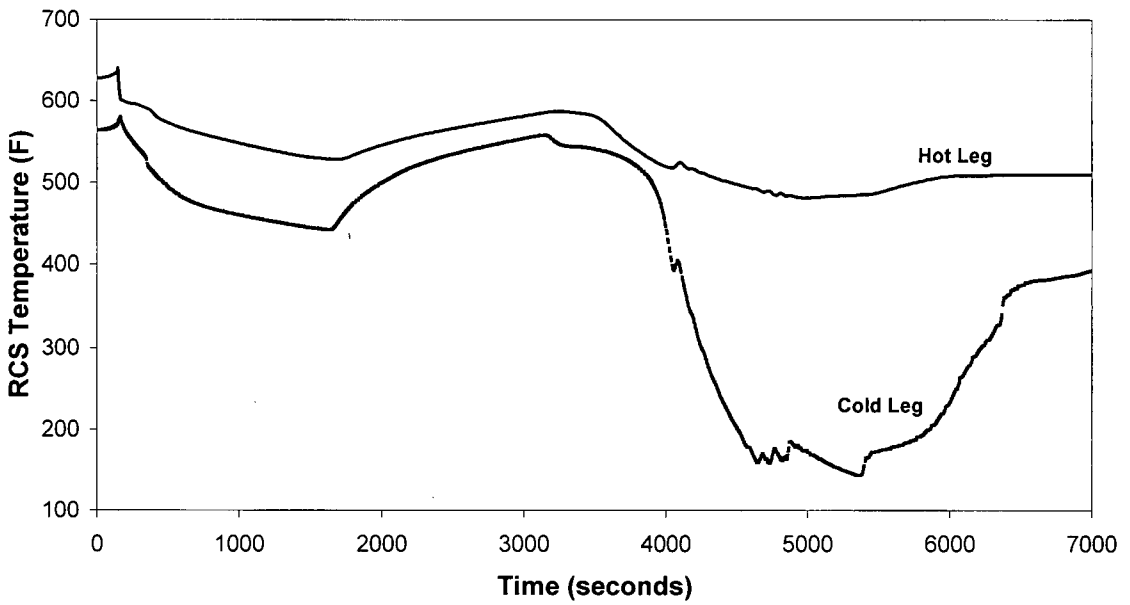

Figure 4-2: SGTR w/ Stuck-Open ARV Faulted Loop RCS Temperature

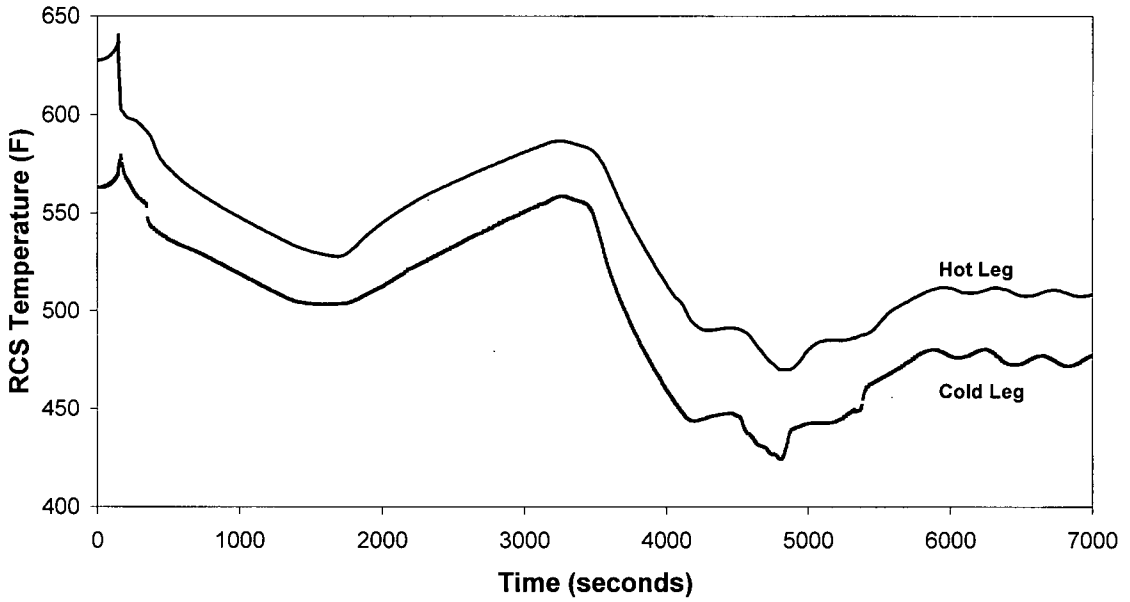

Figure 4-3: SGTR w/ Stuck-Open ARV Intact Loop RCS Temperature

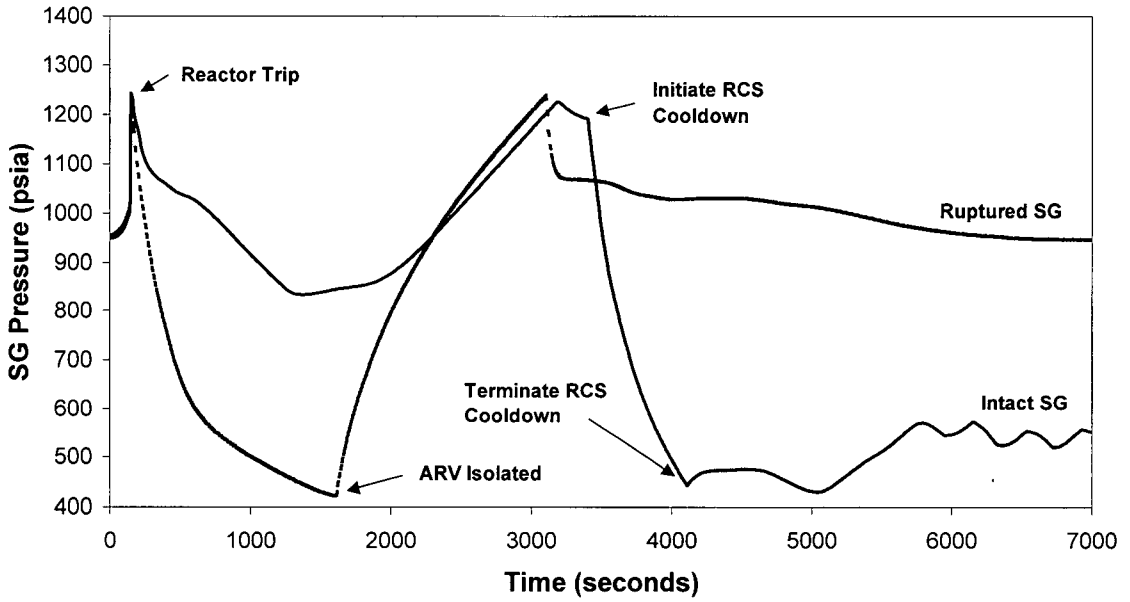

Figure 4-4: SGTR w/ Stuck-Open ARV Steam Generator Pressure

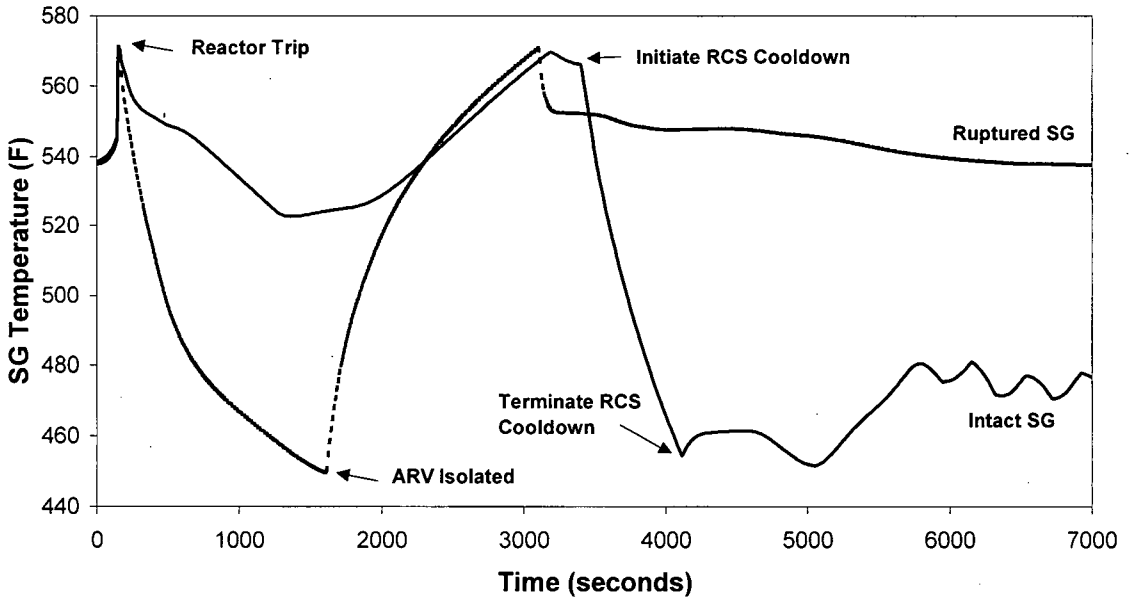

Figure 4-5: SGTR w/ Stuck-Open ARV Steam Generator Temperature

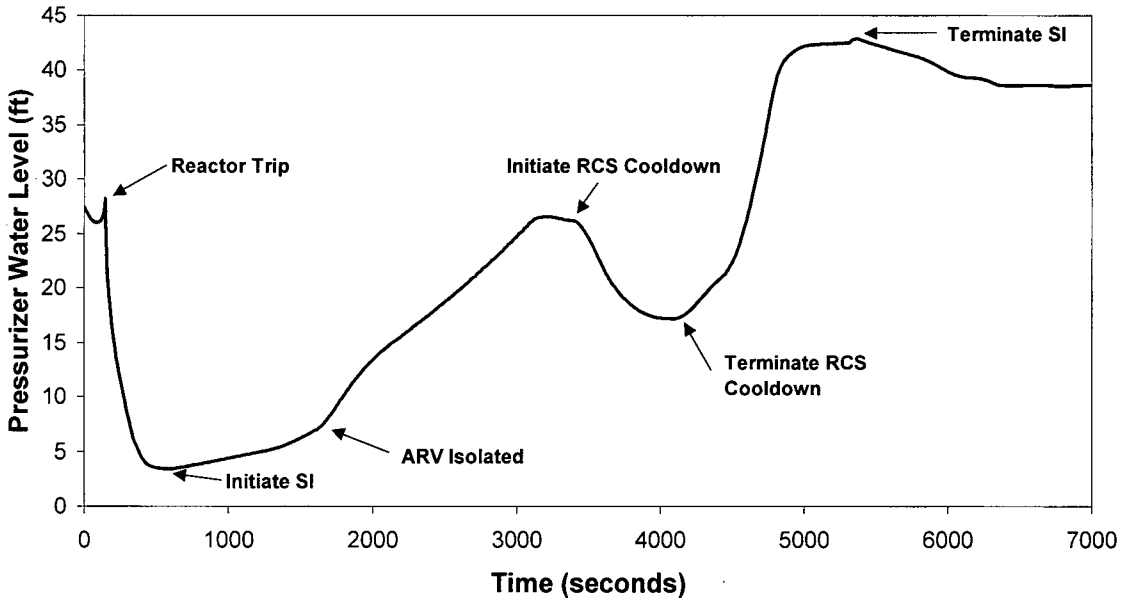

Figure 4-6: SGTR w/ Stuck-Open ARV Pressurizer Water Level

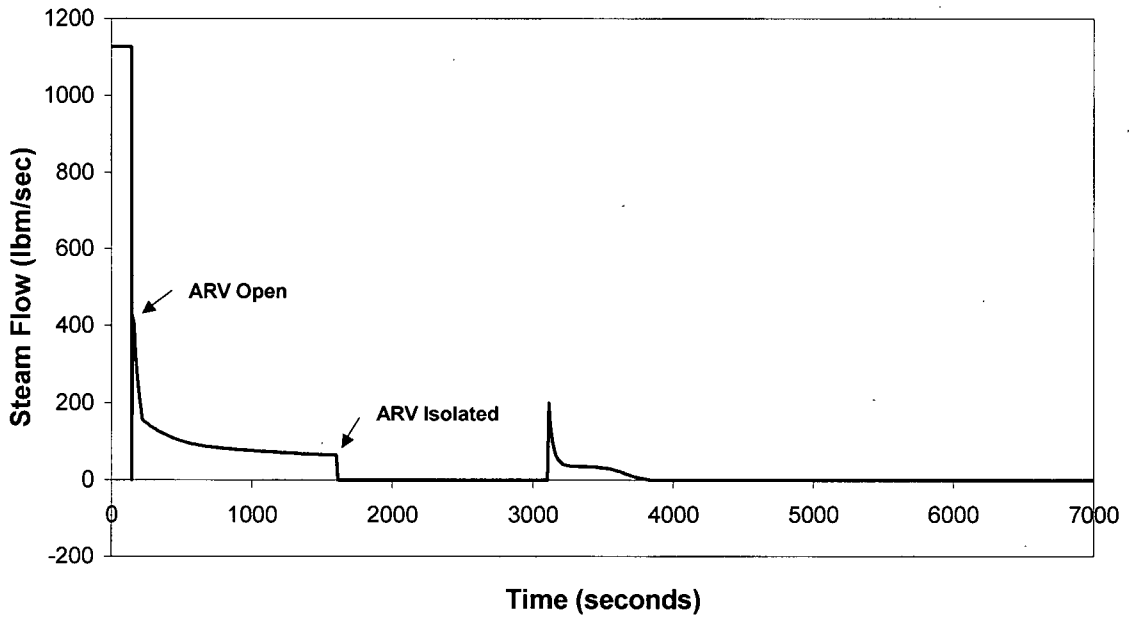

Figure 4-7: SGTR w/ Stuck-Open ARV Faulted Steam Generator Steam Flow

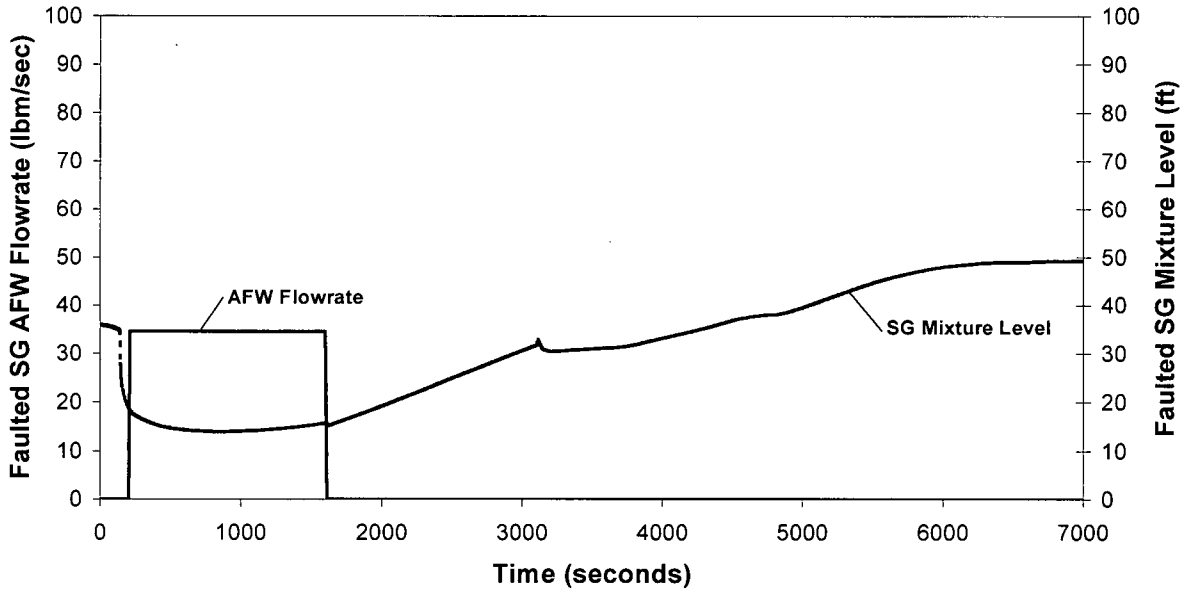

Figure 4-8: SGTR w/ Stuck-Open ARV Faulted Steam Generator Auxiliary Feedwater

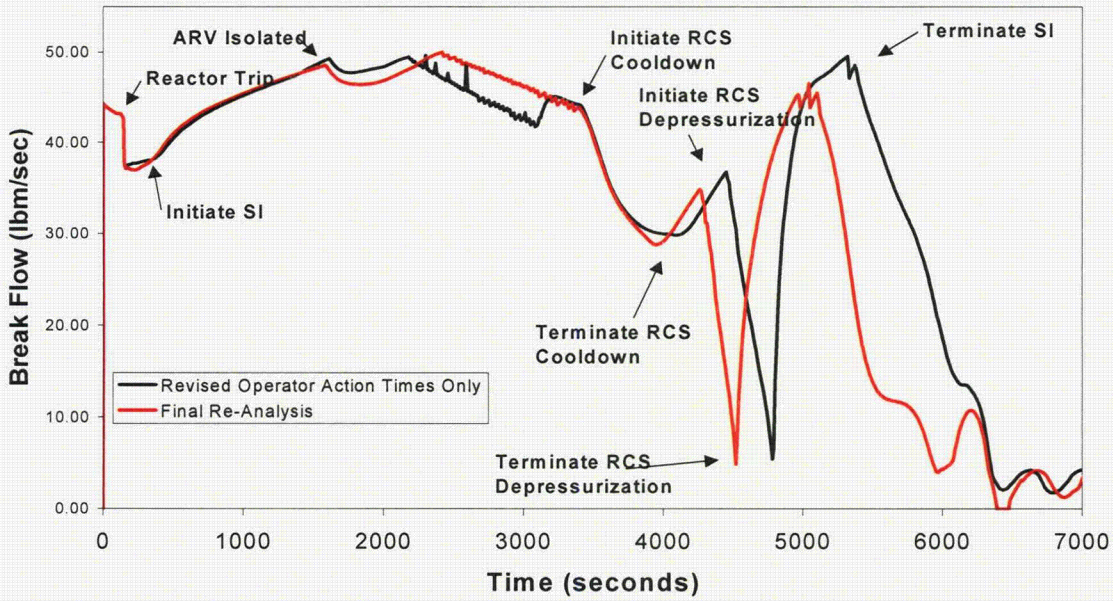

Figure 4-9: SGTR w/ Stuck-Open ARV Faulted Steam Generator Total Break Flow

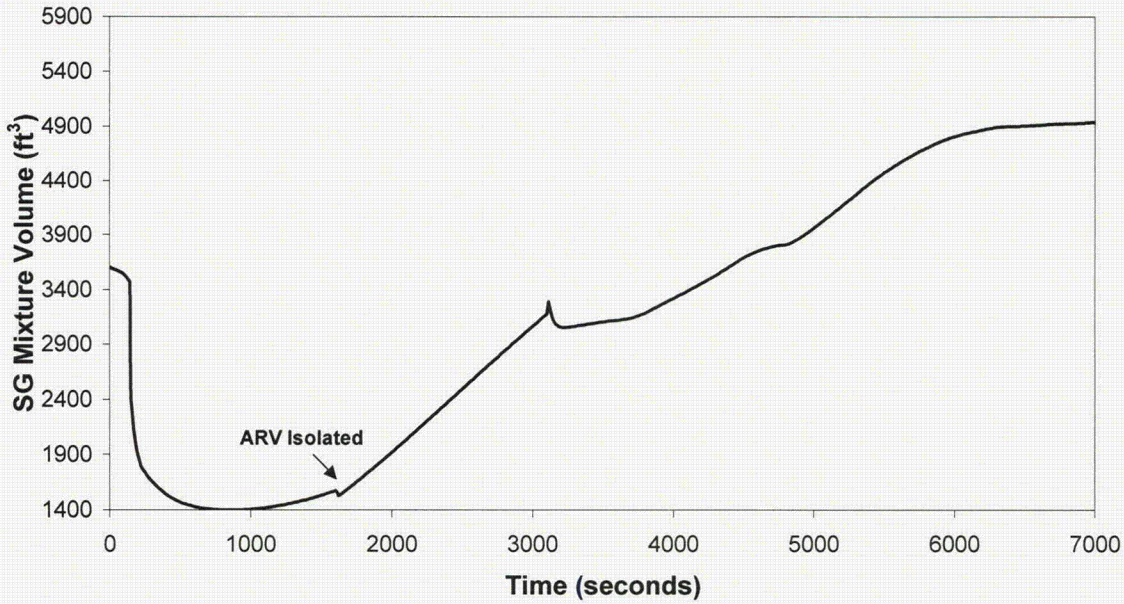

Figure 4-10: SGTR w/ Stuck-Open ARV Faulted Steam Generator Mixture Volume