10 CFR 50.90

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102-06654-DCM/RKR/CJS
January 30, 2013

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2, and 3
Docket Nos. STN 50-528, 50-529, and 50-530
Response to Request for Additional Information Regarding License Amendment Request (LAR) to Revise Technical Specification 3.7.4, *Atmospheric Dump Valves (ADVs)*

Arizona Public Service Company (APS), by letter No. 102-06370, dated June 22, 2011, requested a revision of the Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Technical Specification Limiting Condition for Operation (LCO) 3.7.4, *Atmospheric Dump Valves* (ADVs) [Agencywide Documents Access and Management System (ADAMS) Accession No. ML11182A908].

By letter dated August 31, 2011 (ADAMS Accession No. ML112430084) the NRC staff issued a Request for Additional Information (RAI). By letter Nos. 102-06446 and 102-06466, dated December 9, 2011, and January 27, 2012, respectively (ADAMS Accession Nos. ML11356A088 and ML12046A649), APS submitted responses to the NRC RAI.

The NRC staff reviewed the RAI responses and determined that a regulatory audit at the PVNGS site was needed in order for the staff to complete its technical review of the LAR, which was performed November 27-29, 2012. By letter dated December 13, 2012, the NRC staff requested additional information, following the regulatory audit.

The enclosure to this letter contains the APS response to the December 13, 2012, NRC RAI. On January 24, 2013, APS requested and was granted an extension until January 30, 2013, to respond to the RAI.

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NRR

No commitments are being made to the NRC by this letter.

Should you need further information regarding this response, please contact Robert K. Roehler, Licensing Section Leader, at (623) 393-5241.

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

Executed on JANUARY 30, 2013
(date)

Sincerely,

FOR D.C. HIMS

DCM/RKR/CJS/hsc

Enclosure: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.7.4, *Atmospheric Dump Valves*

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	L. K. Gibson	NRC NRR Project Manager
	M. A. Brown	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)
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ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 3.7.4, *ATMOSPHERIC DUMP VALVES (ADVs)*

Introduction

By letter dated June 22, 2011, Arizona Public Service Company (APS), the licensee for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3, submitted a request to the U.S. Nuclear Regulatory Commission (NRC) to revise PVNGS Technical Specification (TS) 3.7.4, *Atmospheric Dump Valves (ADVs)* [Reference 1]. The proposed change would require all four ADV lines in a PVNGS unit to be operable in MODE 1 (Power Operation), MODE 2 (Startup), and MODE 3 (Hot Standby), as well as in MODE 4 (Hot Shutdown) when a Steam Generator (SG) is relied upon for heat removal. The proposed change would also modify associated TS ACTION statements to more closely conform to Standard Technical Specifications (STS) for Combustion Engineering plants [References 2, 3, and 4], while maintaining consistency with PVNGS plant-specific design attributes that are not reflected in the STS.

The NRC staff reviewed the License Amendment Request (LAR) of June 22, 2011, and determined that additional information was required in order for the staff to complete its review. By letter dated August 31, 2011, the NRC staff issued a Request for Additional Information (RAI) to APS that was comprised of three requests [Reference 5]. APS provided a response to RAI 1 on December 9, 2011, concerning proposed ADV TS ACTION statement completion times, their associated technical bases, defense-in-depth, and safety margins [Reference 6]. APS also informed the NRC staff that additional time was required to confirm several demonstration analyses that had been performed in response to the other two requests.

APS subsequently responded to RAIs 2 and 3 by providing the NRC staff with a summary description of the demonstration analyses on January 27, 2012 [Reference 7]. These analyses utilized deterministic methods and modeled postulated limiting fault accidents – including Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB), and Feedwater Line Break (FWLB) – with all four ADV lines in the affected unit assumed unavailable for a 4-hour time frame post-accident. The purpose of these analyses was to demonstrate the capability of the PVNGS units to automatically control and accommodate such accidents when ADVs were unavailable, and to identify any limitations of expected performance relative to conventional NRC acceptance criteria for these types of accident analyses. The SGTR and MSLB analyses demonstrated that a safe hot shutdown condition could be achieved and maintained without any analytical credit taken for intervention by licensed control room operators. The FWLB analysis, however, demonstrated that a single operator action – that of turning off a charging pump within 20 minutes of accident initiation – would assure safe hot shutdown and conformance to analysis acceptance criteria, by preventing a water-solid condition in the Reactor Coolant System (RCS) pressurizer and passage of water through Pressurizer Safety Valves (PSVs).

Because these demonstration analyses were predicated on a plant configuration that exceeded the single failure criterion of Appendix A to 10 CFR Part 50 [Reference 8], APS informed the NRC staff that the analyses would be controlled in accordance with internal PVNGS design control procedures but would not be incorporated into the

PVNGS Updated Final Safety Analysis Report (UFSAR) [Reference 9]. Specifically, the APS LAR of June 22, 2011, and the RAI response of January 27, 2012, stated that PVNGS had been designed such that a credible single failure involving ADV control power or instrument power could render as many as two ADVs (that is, one ADV per SG) incapable of remote operation from the control room.

The demonstration analyses, however, assumed that all four ADVs in an affected PVNGS unit would be rendered unavailable, either by a common cause failure or by multiple component failures. (This is a plant condition for which both current and proposed PVNGS TS would allow continued operation for only a 24-hour period of time, in order to provide a limited opportunity for evaluation of the condition and restoration of ADVs to service.) PVNGS analyses that consider such failures, beyond the single failure criterion, are controlled internally by APS and, where applicable, may be used to provide insight into Emergency Operating Procedure (EOP) mitigation and safety function recovery strategies. The safety analyses in the PVNGS UFSAR, on the other hand, utilize the single failure criterion of Appendix A to 10 CFR Part 50, as well as related guidance contained in NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition* [References 10, 11, and 12].

The NRC staff reviewed the RAI responses dated December 9, 2011, and January 27, 2012, and determined that a regulatory audit at the PVNGS site was needed in order for the staff to complete its technical review of the LAR. The NRC staff issued an audit plan on September 17, 2012 [Reference 13], and performed the audit during the period of November 27-29, 2012. On December 13, 2012, the NRC staff issued its regulatory audit summary as well as a supplemental RAI comprised of four requests [Reference 14].

This enclosure provides the APS response to the supplemental RAI of December 13, 2012 [Reference 14]. The four NRC staff requests are presented herein, with each one followed by the corresponding APS response.

NRC Request 1

The following questions pertain to the SGTR evaluation and supplemental analyses that are discussed in APS's response to NRC's RAI 3, provided by letter dated January 27, 2012.

a. Please provide a tabulated sequence of events for the supplemental analysis.

b. Please provide a comparison of the Henry-Fauske break flow model to the Homogenous Equilibrium Model including a plot of SG tube flow versus time that compares the design basis analysis to the supplemental analysis. Provide this comparison for a period of time that extends to the time that the ADVs are assumed to actuate in the design basis analysis and discuss the effect that any other significant differences aside from the break flow model have on the compared results.

c. Please provide plots of the following parameters as functions of time:
   i. Leak flow rate
   ii. Pressurizer pressure
   iii. Pressurizer volume
   iv. RCS subcooling
   v. SG pressure
   vi. SG level (wide range)

d. Since licensed operators are trained and expected to mitigate SGTRs, please characterize the agreement between a simulator projection of a SGTR and a CENTS[1] analysis of a SGTR, given a roughly analogous scenario executed with each tool.

e. Please describe how plant personnel mitigate a SGTR coincident with Loss of Offsite Power (LOP) event when ADVs are not available for operation from the control room. Include a discussion of applicable procedures, and provide procedure excerpts for the steps leading up to and after establishing RCS cooldown using the Steam Bypass Control System (SBCS), contingency actions (including Appendix 18 – Local ADV actuation), and main steamline isolation. Please explain how these procedures are implemented in regards to maintaining availability of required equipment to cooldown.

f. During mitigation of a SGTR, please discuss the possibility of overfilling a SG if a RCS cooldown/depressurization is delayed. Please discuss the potential for a liquid release to the environment through the Main Steam Safety Valves (MSSVs). Please describe the timing and possible delays in the operator's ability to commence a RCS cooldown/depressurization in the event the ADVs are all inoperable. Please describe defense-in-depth measures to control the liquid inventory in the steam generator until a method is available to commence a cooldown.

APS Response 1

Response 1.a

Table 1-1 in Attachment 1 of this enclosure summarizes the pertinent acceptance criteria and analysis results for the SGTR demonstration analysis that was described in the APS letter of January 27, 2012 [Reference 7]. Table 1-2 in Attachment 1 likewise provides a tabulated sequence of events for the SGTR demonstration analysis. The times delineated in Table 1-2 are specified to the nearest second.

[1] CENTS is an interactive computer code for simulation of the nuclear steam supply system and related systems. It is described in WCAP-15996.

The SGTR demonstration analysis assumed a guillotine break of a single SG U-tube, a coincident LOP following turbine trip, and all ADVs unavailable for a 4-hour time frame post-accident. Also, no operator actions were assumed to occur during this period of time. The simulation did not predict the actuation of any PSVs, but it did predict that MSSVs would automatically cycle open and closed throughout the simulation. Because the MSSVs are capable of operating for hundreds of cycles without failure as explained in References 7 and 15, the number of predicted MSSV operating cycles was determined to be acceptable. Steam generator overfill and discharge of water through the MSSVs was not predicted to occur during the simulation.

Response 1.b

The SGTR demonstration analysis utilized the CENTS computer code, which is described in Westinghouse Topical Report (TR) WCAP-15996-P-A, *Technical Description Manual for the CENTS Code* [Reference 16]. NRC staff Safety Evaluations (SEs) that document the bases for acceptance of the code for licensing applications are incorporated into the TR.

Section 4.7 of the TR states that CENTS may calculate critical (choked) flow for two-phase and subcooled liquid conditions using either of two correlations, the Homogeneous Equilibrium Model (HEM) or the Henry-Fauske (H-F) correlation. The code user specifies the model selection by setting a flag in the code input. Section 5.7 of the TR describes the details of the CENTS SGTR break flow model, and states that critical flow is calculated using the pressure at the break location rather than just the pressures in the RCS nodes that feed the break. This allows the code to model both guillotine breaks (which are fed from both the hot and cold sides of the RCS) and slot breaks (which may be fed from only one side of the RCS, depending on the size of the break). Appendix B of the TR provides tabulated values of mass flux and throat pressure, as a function of upstream stagnation pressure and enthalpy, for both of these critical flow models.

The HEM is predicated on thermodynamic equilibrium between the liquid and vapor phases in a two-phase mixture, as well as a "no slip" condition in which the phases are assumed to be so well-mixed that they have zero relative velocity with respect to each other. The HEM also assumes that there is negligible interaction between the flow and its environment, such that the effects of wall surface roughness and external heat transfer can be ignored. Under such conditions the expansion of the two-phase mixture at the break location is isentropic, and the first law of thermodynamics suggests that the mass flux is a function of the mixture's static properties at the choke point or throat. The critical mass flux may then be calculated iteratively by searching for a thermodynamic state point that both maximizes the mass flux and preserves the upstream stagnation entropy. The tabulated values in Appendix B of the TR provide the results of such iterative calculations. The CENTS code determines critical mass flux at a break location through interpolation of these tabulated values, rather than by directly performing the iterative, time-consuming calculations with the governing thermodynamic equations.

The H-F correlation, like the HEM, assumes that the mass flux is a function of the mixture's properties at the throat, that the critical mass flux can be determined by maximizing this function, and that expansion of the two-phase mixture is isentropic. The H-F correlation differs from the HEM with respect to the assumptions regarding homogenous mixing and thermodynamic equilibrium, however. Mass transfer between the phases is constrained such that the quality at the throat is equal to the upstream stagnation quality. Heat transfer during the expansion is constrained such that the liquid phase temperature at the throat is equal to that of the upstream liquid, while the vapor phase temperature is allowed to vary. Again, direct solution of the governing equations requires time-consuming iteration, so the CENTS code instead interpolates between pre-calculated, tabulated values to determine the critical mass flux.

The H-F correlation also incorporates an empirically-derived function such that, as the quality upstream of the break increases, the correlation will more closely approximate the HEM. See, for example, Figure 1-1 in Attachment 2 to this enclosure, which compares the two models for an upstream stagnation pressure of 2000 psia. The inflection point on the HEM curve at about 672 Btu/lbm corresponds to the enthalpy of saturated liquid. This also happens to be the point at which the predictions made by the two models differ the most. At higher values of enthalpy, quality increases and the curves become more similar.

For SGTR simulations that model the presence of subcooled liquid upstream of the break location, the H-F correlation will typically predict a higher break flow rate than the HEM. As Figure 1-1 suggests, however, the difference between HEM and H-F predictions will vary with the amount of subcooling, which is likewise sensitive to changes in pressure. Thus selection of the proper model will depend upon the specific purpose of a simulation, as well as how system parameters change with time throughout the simulation.

PVNGS UFSAR Section 15.6.3 describes the design basis SGTR analysis with a coincident LOP and an assumed Single Failure (SGTRLOPSF) resulting in a failed open ADV on the ruptured SG [Reference 9]. The primary purpose of that analysis is to demonstrate conformance with 10 CFR 100.11 regulatory limits for offsite radiological dose consequences as well as with General Design Criterion (GDC) 19 for dose consequences for control room personnel [References 8 and 17]. That analysis utilizes an element of methodology that maximizes flashing at the break, and thus the radioiodine release to the environment, whenever the SG U-tubes are not completely covered by water. The HEM was selected for use in the SGTRLOPSF analysis because that model reduced the break flow rate, which in turn resulted in a longer period of U-tube uncovery and higher calculated dose consequences. (The use of the HEM in the design basis UFSAR analysis is considered discretionary because it introduced conservatism into the analysis results, and because previous PVNGS licensing basis SGTR analyses, performed with the CESEC computer code, utilized the H-F correlation.)

The purpose of the SGTR analysis described in the APS letter of January 27, 2012 [Reference 7], however, was to demonstrate that SG overfill was not likely to occur for at least a 4-hour period post-accident, even without operator intervention. The H-F correlation was selected for that analysis to increase the break flow rate and thus the total mass added to the ruptured SG throughout the simulation, in order to maximize the potential for overfill.

The requested comparison of SGTR break flow rates is provided in Figure 1-2 of Attachment 2 to this enclosure, for a period of time that extends to 1 minute after ADVs are assumed to actuate in the UFSAR Section 15.6.3 SGTRLOPSF design basis analysis. Figure 1-2 shows the break flow rate for the UFSAR design basis analysis scenario, calculated with both the HEM and the H-F correlation. Figure 1-2 also shows the break flow rate for the SGTR demonstration analysis, calculated with the H-F correlation.

The demonstration analysis exhibits a relatively steady decline in break flow rate during the first few minutes of the simulation, due to depressurization of the RCS caused by the SGTR. Larger variations in break flow rate were not observed in the demonstration analysis because reactor trip was delayed for several additional minutes past the initial period shown in Figure 1-2, at which time an automatic Core Protection Calculator System (CPCS) trip occurred. In the case of the design basis analysis, however, larger variations in break flow are observed that are attributed to a variety of short-term equipment actuations, such as at the following approximate times: 100 seconds (assumed operator intervention and manual reactor trip and turbine-generator trip); 102 seconds (MSSVs cycle open for the first time); 103 seconds (LOP occurs and Reactor Coolant Pumps (RCPs) begin to coast down); 155 seconds (Auxiliary Feedwater (AFW) delivery to the intact SG, followed about 1 second later by AFW delivery to the ruptured SG); 162 seconds (MSSVs close); 220 seconds (assumed manual actuation of ADVs with one ADV failing to the full open position on the ruptured SG); 245 seconds (Safety Injection (SI) actuation); and 251 seconds (Main Steam Isolation Signal (MSIS) and main steamline isolation).

As shown in Figure 1-2, the variations in break flow rate for the design basis analysis case tend to be more pronounced when the H-F model is used. This is attributed to the sensitivity of the H-F model to the presence of subcooled liquid upstream of the break. For example, Figure 1-2 exhibits a rapid increase in the H-F break flow rate at approximately 200 seconds post-accident. Prior to this time, at approximately 162 seconds, the MSSVs had closed at a blowdown pressure that was less than their lift settings, and the temperature of the coolant in the ruptured SG's RCS cold legs was at saturation conditions corresponding to the pressure at the break location. Hence the quality of the coolant was high enough to assure that the HEM and the H-F correlation would predict similar break flow rates immediately after the MSSVs closed. Between about 162 seconds and 200 seconds post-accident, however, SG pressure increased by about 25 psi, and the coolant in the ruptured SG's cold legs became slightly subcooled relative to secondary system conditions at the break location. When this occurred, the quality of the coolant decreased and the H-F correlation predicted a

higher break flow rate than the HEM. Thermodynamic conditions at this point in time were near the inflection point shown on Figure 1-1, where the difference between HEM and H-F correlation predictions is maximized.

Figure 1-2 also shows that, when the H-F correlation was used, the break flow rate at event initiation was approximately 4 lbm/sec lower for the SGTR demonstration case than for the SGTRLOPSF case. This difference is attributed to two factors. First, the SGTRLOPSF case used the same version of CENTS that was used for the UFSAR SGTRLOPSF analysis, whereas the SGTR demonstration case used a later version of the code and a larger number of RCS nodes in the PVNGS-specific plant model. (The NRC staff has previously issued SEs for both versions of the code, so that they may be used in licensing applications.) The later version of the code also utilized the SGTR break flow model described in Section 5.7 of the CENTS TR [Reference 16], which differed from the earlier version of CENTS in terms of how the geometry associated with the broken tube was specified in the code inputs. Second, although a number of code inputs that specified the initial conditions at the start of the accident were similar for the two cases, not all inputs were identical. Thus thermodynamic conditions, such as the enthalpy of fluid in the RCS cold legs, differed between the two cases.

Response 1.c

The following figures are provided in Attachment 2 of this enclosure:

- Figure 1-3 SGTR Demonstration Analysis – SGTR Break Flow Rate vs. Time
- Figure 1-4 SGTR Demonstration Analysis – Pressurizer Pressure vs. Time
- Figure 1-5 SGTR Demonstration Analysis – Pressurizer Level vs. Time
- Figure 1-6 SGTR Demonstration Analysis – Reactor Coolant System Subcooling vs. Time
- Figure 1-7 SGTR Demonstration Analysis – Steam Generator Pressure vs. Time
- Figure 1-8 SGTR Demonstration Analysis – Wide Range Steam Generator Level vs. Time

Response 1.d

To characterize the level of agreement between the CENTS code and the PVNGS plant simulator with regard to a postulated SGTR event, a simulator analogue was created for the SGTR demonstration analysis that was described in the APS letter of January 27, 2012 [Reference 7]. As in the CENTS demonstration analysis, the simulator analogue did not take credit for operator action for 4 hours post-accident. Thus the ADVs, which may be actuated by either remote manual or local manual operator action, were not utilized during the simulation. As in the demonstration analysis, the simulator analogue also modeled a LOP following turbine trip, and all SBCS valves were rendered unavailable (including two SBCS valves that would normally be available post-LOP to

vent secondary system steam to the atmosphere). Additionally, the initial SGTR leak flow rate for the simulator analogue was matched to within about 0.2 lbm/sec of that predicted by the CENTS code for the guillotine rupture of a single SG U-tube.

The simulator analogue differed from the CENTS analysis, however, with regard to the detailed modeling of certain plant systems and components, including but not limited to SI, AFW, and the MSSVs. That is, whereas the CENTS demonstration analysis utilized limiting (typically maximum or minimum) values for modeling of temperatures, pressures, flow rates, actuation/reset setpoints, and so forth, the simulator analogue generally used the same nominal or best-estimate values that are utilized during operator training.

Despite these differences in detailed modeling, generally good agreement was achieved with regard to certain parameters of interest. For example, both the CENTS demonstration analysis and the simulator analogue showed similar long-term trends of RCS subcooling and RCS loops' ΔT (as an indicator of natural circulation flow post-LOP) during the 4-hour period. Noticeable differences were observed, however, in short-term variations superimposed on those long-term trends (for example, variations caused by MSSV cycling or AFW delivery). These differences are attributed to the selection of nominal or best-estimate values for modeling, as opposed to limiting values, as described above.

Noticeable differences were also observed in the long-term trends associated with pressurizer water level and wide range water level in the ruptured SG. Although both parameters were observed to initially drop and then later recover in both the CENTS demonstration analysis and in the simulator analogue, the CENTS analysis exhibited a larger initial drop in pressurizer level, as well as a more pronounced long-term increase in ruptured SG water level. These differences were explained by examination of the simulator analogue which revealed that, although the initial SGTR break flow rate had been reasonably matched with CENTS, the reduction in break flow rate that occurred following LOP and RCP coastdown was more pronounced in the simulator analogue.

Thus the CENTS demonstration analysis generally predicted a higher break flow rate than the simulator analogue for most of the 4-hour period. This is not unexpected because of differences in modeling between CENTS and the simulator software. For example, the simulator analogue utilizes a six-equation model in which the mass, energy, and momentum balance equations are solved separately for both the liquid and vapor phases, whereas CENTS utilizes a five-equation model in which the momentum balance equation is solved for a liquid-vapor mixture. The overall conservatism of the CENTS model and its inputs was evidenced by the fact that, at the end of the 4-hour period, the indicated wide range SG water level in the ruptured SG was approximately 98%, whereas in the simulator analogue it was approximately 73%.

Table 1-3 in Attachment 1 to this enclosure summarizes some of the similarities and differences that were observed with respect to the timing of events in the CENTS analysis and in the simulator analogue. Without operator intervention, the reactor tripped automatically in both cases on a CPCS-initiated trip in approximately 10

minutes. AFW was delivered to the intact SG seven times in both cases, with the first delivery occurring at approximately 18 to 20 minutes post-accident. Automatic AFW lockout, due to high differential pressure between the two SGs, occurred in both cases after the 3-hour mark. SG overfill was not predicted to occur in either case.

One difference that was observed in the simulator analogue involved the behavior of AFW with respect to the ruptured SG. Whereas the CENTS analysis was initiated from a high initial water level and thus did not predict the automatic delivery of AFW to the ruptured SG at any point during the simulation, the simulator analogue was initiated from a lower, nominal water level. The lower initial water level contributed to the actuation of AFW flow to the ruptured SG approximately 20 minutes post-trip. Also, because the simulator analogue utilized a higher, more realistic AFW flow rate, an actuation of AFW that occurred later in the event was sufficient to reduce pressure in the ruptured SG to the MSIS actuation setpoint. Thus the simulator analogue predicted main steamline isolation much earlier than the CENTS analysis, for which MSIS was delayed until a high level condition occurred in the ruptured SG.

Response 1.e

In the event of a SGTR from full power operating conditions, licensed operators would respond to the resulting alarms (for example, high main steamline radiation level) by entering abnormal operations procedure 40AO-9ZZ02, *Excessive RCS Leakrate* [Reference 18]. 40AO-9ZZ02 directs the operators to manually trip the plant if the leak rate exceeds the capacity of the normal charging system, (i.e., if all available charging pumps are operating, if letdown is isolated, and if pressurizer level is continuing to lower). Procedure 40AO-9ZZ02 also directs the operators to enter emergency procedure 40EP-9EO01, *Standard Post Trip Actions* [Reference 19], after they ensure the reactor is tripped.

Once in procedure 40EP-9EO01, the operators would accomplish those immediate actions that must be taken following any manual or automatic reactor trip, including verification that safety function criteria are being met, and completing any short-term contingency actions as directed by the procedure. For example, operators would assess the RCS inventory control safety function, in part, by checking to ensure that pressurizer level is both in the range of 10% to 65%, and trending as expected to the range of 33% to 53%. If pressurizer level does not meet both of these criteria, then operators would attempt to implement short-term contingency actions, pending completion of event diagnosis, by taking control of the Pressurizer Level Control System (PLCS), the charging pumps, and/or the letdown control valves.

Operators would likewise assess the RCS inventory control safety function by checking RCS subcooling and the availability of both seal injection and Nuclear Cooling Water (NCW) to the RCPs. In the event of a LOP following reactor trip and turbine trip, however, the RCPs would automatically coast down, so procedure 40EP-9EO01 would direct the operators to implement a contingency action and isolate controlled bleed-off from the RCPs.

Operators would also verify the availability of RCS heat removal in accordance with procedure 40EP-9EO01 by checking SG water levels, SG pressures, and RCS cold leg temperatures. In the event that ADVs are not available for remote operation from the control room, operators would attempt to maintain short-term RCS cold leg temperatures in the range of 560°F to 570°F by operating SBCS valves, as directed by a contingency action in the procedure. Although six SBCS valves that would normally dump steam to the main condenser would be rendered unavailable by a coincident LOP, operators would still be able to perform this step because the PVNGS plant is equipped with two additional SBCS valves that dump steam to the atmosphere. As explained in the response to Request 4 of this enclosure, these two valves would still have control power, instrument power, and pneumatic power for several hours following a LOP.

A crew of licensed operators can typically work through all of the verification activities and short-term contingency actions of procedure 40EP-9EO01 within 10 to 15 minutes, depending upon the complexity of an event, during which time all eight critical safety functions would be assessed – reactivity control; maintenance of vital auxiliaries; RCS inventory control; RCS pressure control; core heat removal; RCS heat removal; containment isolation; and containment temperature, pressure and combustible gas control.

If all acceptance criteria are met and no contingency actions have been performed, then procedure 40EP-9EO01 would direct operators to implement a procedure for an uncomplicated reactor trip. However, in the event of a SGTR with a coincident LOP, and with ADVs incapable of remote operation from the control room, at least one short-term safety function criterion would not be met and/or at least one short-term contingency action would be performed (that is, in an attempt to control RCS inventory, RCS pressure, or RCS heat removal), and operators would instead follow a diagnostic chart in 40EP-9EO01 to determine which procedure to implement next. The diagnostic chart specifically directs that, even in the presence of a coincident LOP, operators are to implement an optimized, event-specific SGTR recovery procedure rather than a more generalized safety function recovery procedure (which may otherwise be used for other, more complicated events).

Thus, for a SGTR with a coincident LOP and with ADVs incapable of remote operation from the control room, operators would enter procedure 40EP-9EO04, *Steam Generator Tube Rupture* [Reference 20], upon completion of the standard post-trip assessment and diagnostic activities. Pertinent instructions and contingency actions in procedure 40EP-9EO04 that are related to the establishment of RCS cooldown appear in Table 1-4 in Attachment 1 to this enclosure. Because of the assumed coincident LOP, Circulating Water (CW) flow to the main condenser would be disabled and Instruction number 11 would require operators to manually initiate a MSIS. Initiation of MSIS would close the Main Steam Isolation Valves (MSIVs) and thereby isolate steam flow from the SGs to the two SBCS valves that discharge to atmosphere, SGN-1007 and SGN-1008.

In addition, Instruction number 9.b requires that these SBCS valves be placed in the "off" position, and Contingency Action number 9.c.1 effectively requires that heat removal be accomplished instead by local manual operation of the ADVs (because remote manual operation of ADVs from the control room is assumed to not be available). Also, the asterisk (*) in front of Instruction number 11 allows operators to perform that step in a different sequence than the strict numerical sequence specified in the procedure, based on an assessment of plant conditions and equipment availability during an event.

On November 28, 2012, during the NRC regulatory audit, a crew of licensed PVNGS operators was tasked with implementing this SGTR procedure in a plant control room simulator. Prior to the initiation of the SGTR, the crew had been informed that all ADVs were not available for remote operation from the control room. Therefore, when the crew arrived at the instructions delineated in Table 1-4, the operators decided to implement Contingency Action number 9.c.1 and thereby establish local manual operation of the ADVs, prior to closing SBCS valves SGN-1007 and SGN-1008 (Instruction number 9.b) and prior to actuating main steamline isolation (Instruction number 11). A licensed operator training instructor served as an auxiliary operator for this simulation, and was briefed by the crew on the detailed instructions in Standard Appendix 18, *Local ADV Operation* [Reference 21], including instructions on how to equate the number of turns of the manual handwheel to valve position.

When asked about this decision after the simulation, the crew properly responded that they had decided to temporarily defer Instruction number 9.b while implementing Instruction number 9, so that they could ensure availability of adequate heat removal via local manual ADV operation before they disabled heat removal via the SBCS. In addition, if they had determined that the ADVs could not be operated locally, the crew could deviate from the SGTR emergency procedure, as permitted by the *Emergency Operating Procedure Users Guide* [Reference 22, described further in the response to Request 4.c], and could instead continue to remove heat using the SBCS valves that discharge to atmosphere, without isolating the main steamline.

Response 1.f

Overfill of a PVNGS SG, and discharge of a liquid release to the environment through the MSSVs, is not a likely scenario following a postulated SGTR. This conclusion is based on the following information, which serves to explain how defense-in-depth is afforded by plant design, plant procedures, and licensed operator training:

- A demonstration analysis of a limiting fault guillotine break of a single SGTR U-tube with the NRC-approved CENTS computer code, assuming a coincident LOP and with all ADVs inoperable (that is, beyond the design basis single failure criterion), shows that the plant design is sufficiently robust that SG overfill will not occur for at least 4 hours following event initiation, even when operator intervention is not credited in the analysis. The PVNGS SGs are larger than those installed in most other nuclear power plants, and the actuation setpoints

and design parameters associated with reactor protection, engineered safety features, and other plant components are established such that automatic equipment operation will mitigate the accident without SG overfill during this period of time. The demonstration analysis does not predict water discharge through the MSSVs.

- A plant simulator analogue of the CENTS SGTR demonstration analysis, again without any operator intervention and without utilization of any SBCS valves or any ADVs, but using nominal or best-estimate modeling of plant components, likewise indicates that the plant can accommodate and mitigate a SGTR without SG overfill for a 4-hour period post-accident. At the end of that period, the simulator analogue predicted a wide range water level in the ruptured SG that was approximately 25% of the calibrated instrument span lower than that predicted by the CENTS analysis. This suggests the existence of sufficient operating margin during normal plant operations, that would further support a conclusion that liquid discharge would not occur through the MSSVs following a SGTR.
- The PVNGS plant is equipped with eight SBCS valves, two of which would remain available post-LOP to discharge steam to the atmosphere. These valves, which are described further in the response to Request 4 in this enclosure, are normally not credited for accident mitigation in PVNGS UFSAR Chapter 15 safety analyses, nor were they credited in the CENTS demonstration analysis nor the simulator analogue described above. Licensed operators are trained to utilize these valves after a reactor trip for the purpose of fulfilling RCS heat removal safety function requirements. These valves will typically be actuated while operators are proceeding with short-term, post-trip safety function status checks and event diagnosis activities (that is, no later than about 15 minutes post-accident).
- If operators cannot transition from short-term use of the SBCS valves to long-term heat removal via the ADVs, due to all ADVs being inoperable (that is, incapable of both remote and local operation), then operators can continue to remove heat post-LOP through the two SBCS valves that discharge to atmosphere. This may be accomplished either through remote operation from the control room (up to about 2 hours post-accident as explained in the response to Request 4), or through manual operation in the turbine building. Depending on the specific conditions and sequence of events that operators may observe following a SGTR, long-term use of these SBCS valves may require deviation from the current emergency procedures direction (authorized in the *EOP Users Guide*, as described in the response to Request 4), and perhaps an override of main steamline isolation to provide a flow path from the secondary side of the SGs to these valves. Activation of the emergency response organization, including the Emergency Operations Facility (EOF) and Technical Support Center (TSC), is anticipated to occur within 2 hours post-accident as specified in PVNGS emergency plan. Thus, it is anticipated that additional support would be

available to the operating crew around the time that local manual operation of these valves may be necessary.

- The SGTR emergency procedure, 40EP-9EO04, provides detailed instructions and contingency actions for operators to cool down, isolate, and control water level in the ruptured SG. With regard to SG water level control, the procedure includes steps that provide operators with three options for controlling long-term level: by back-flow to the RCS; by draining the SG to the main condenser through the SG blowdown system; or by steaming through the main steamline isolation bypass valves. Operators routinely train with this emergency procedure.

NRC Request 2

The following questions pertain to the MSLB evaluation and supplemental analysis that is discussed in APS's response to NRC's RAI 3, provided by letter dated January 27, 2012.

a. Please provide a table of results that includes key system attributes such as peak RCS pressure, maximum and minimum pressurizer level, and the sequence of events. Identify the acceptance criteria for pressurizer fill and for water entrainment in the pressurizer safety valve (PSV) effluent.

b. Please provide plots of the following parameters as functions of time. The NRC staff acknowledges that some plots may need to be truncated if a stable or quasi-stable condition is reached. If this is done, please describe the system behavior for the period that is not included.

    i. Instantaneous SI flow

    ii. Total SI flow

    iii. Pressurizer pressure

    iv. Pressurizer volume

    v. Subcooling margin

    vi. Steam generator pressure

    vii. Steam generator level

APS Response 2

Response 2.a

Table 2-1 in Attachment 1 of this enclosure summarizes the pertinent acceptance criteria and analysis results for the MSLB demonstration analysis that was described in the APS letter of January 27, 2012 [Reference 7]. Table 2-2 in Attachment 1 likewise provides a tabulated sequence of events for the MSLB demonstration analysis. The times delineated in Table 2-2 are specified to the nearest second.

The MSLB demonstration analysis assumed a guillotine break of a main steamline inside containment, a coincident LOP, and all ADVs unavailable for a 4-hour time frame post-accident. Also, no operator actions were assumed to occur during this period of time. The initial SG water level was set higher than the normal operating band to increase the RCS cooldown that would result from the MSLB. This also increased the possibility of a return-to-power condition due to the effects of reactivity feedbacks. A small, transient return-to-power was predicted to occur at about 6 minutes post-accident; however, the simulation did not predict any fuel clad damage nor fuel centerline melting.

After the initial RCS cooldown, the system heated up again due to decay heat and the loss of heat transfer through the faulted SG. The PSVs actuated several times during the simulation to relieve pressure in the RCS. Additionally, the simulation predicted that secondary system MSSVs would repeatedly cycle open and closed to remove heat. Because the MSSVs are capable of operating for hundreds of cycles without failure as explained in References 7 and 15, the number of predicted MSSV operating cycles was determined to be acceptable.

Although RCS subcooling remained acceptable throughout the simulation, it did achieve a minimum transient value of about 4°F at approximately 1 hour and 40 minutes into the sequence of events. This occurred as a result of an actuation of MSSVs, which closed at a blowdown pressure less than the lift setting of the valves, followed about one second later by actuation of AFW delivery. Because both of these near-simultaneous actuations served to reduce SG pressure in the intact SG, and because thermal coupling with the RCS likewise caused RCS pressure to decrease, hot fluid in the outlet plenum and hot legs did not have sufficient time to pass through those regions before saturation was closely approached. This combined effect can be observed in the RCS subcooling figure provided in the response to Request 2.b below.

The acceptance criteria pertaining to pressurizer fill and water entrainment in the PSV effluent are the same as those used in the FWLB safety analysis described in UFSAR Section 15.2.8 [Reference 9]. Specifically, in order to prevent the pressurizer from going water-solid, the maximum pressurizer water volume should at no time exceed 1800 ft^3 during the accident simulation. In addition, to prevent water entrainment in the PSV effluent, the pressurizer water volume should not exceed 1697 ft^3 when the PSVs are open and discharging steam from the pressurizer. Table 2-1 in Attachment 1 of this enclosure shows that these criteria were met throughout the MSLB demonstration analysis.

Response 2.b

The following figures are provided in Attachment 2 of this enclosure:

- Figure 2-1 MSLB Demonstration Analysis – Instantaneous Safety Injection Flow Rate vs. Time
- Figure 2-2 MSLB Demonstration Analysis – Total Delivered Safety Injection Flow vs. Time

- Figure 2-3 MSLB Demonstration Analysis – Pressurizer Pressure vs. Time
- Figure 2-4 MSLB Demonstration Analysis – Pressurizer Water Volume vs. Time
- Figure 2-5 MSLB Demonstration Analysis – Reactor Coolant System Subcooling vs. Time
- Figure 2-6 MSLB Demonstration Analysis – Short-Term Steam Generator Pressure vs. Time
- Figure 2-7 MSLB Demonstration Analysis – Long-Term Steam Generator Pressure vs. Time
- Figure 2-8 MSLB Demonstration Analysis – Short-Term Wide Range Steam Generator Level vs. Time
- Figure 2-9 MSLB Demonstration Analysis – Long-Term Wide Range Steam Generator Level vs. Time
- Figure 2-10 MSLB Demonstration Analysis – Short-Term Reactivity Change vs. Time
- Figure 2-11 MSLB Demonstration Analysis – Long-Term Reactivity Change vs. Time

NRC Request 3

The following questions pertain to the FWLB evaluation and supplemental analysis that is discussed in APS's response to NRC's RAI 3, by letter dated January 27, 2012.

a. Please provide a table of results that includes key system attributes such as peak RCS pressure, maximum and minimum pressurizer level, and the sequence of events. Also include relevant acceptance criteria.

b. Please provide plots of the following parameters as functions of time. The NRC staff acknowledges that some plots may need to be truncated if a stable or quasi-stable condition is reached.

   i. Instantaneous charging flow

   ii. Total charging flow

   iii. Pressurizer pressure

   iv. Pressurizer volume

   v. Steam generator pressure

   vi. Steam generator level

c. Please discuss how, in the context of an FWLB event and associated control room procedures, operators will ensure that the charging pumps are secured

independently of taking action to open ADVs slightly. Also, please identify how procedures differ from the supplemental analysis assumptions, and explain why this is the case. For example, while the analysis assumes that charging pumps are secured within 20 minutes, operators may attempt to control pressurizer level and pressure using auxiliary spray from the charging pumps. Please explain such deviations between procedures and analysis.

d. In a control room simulator scenario where a FWLB occurs with a coincident LOP, and ADVs are unavailable, please characterize the pressurizer level and pressure response that an operator may see.

APS Response 3

Response 3.a

Table 3-1 in Attachment 1 of this enclosure summarizes the pertinent acceptance criteria and analysis results for the FWLB demonstration analysis that was described in the APS letter of January 27, 2012 [Reference 7]. Table 3-2 in Attachment 1 likewise provides a tabulated sequence of events for the FWLB demonstration analysis. The times delineated in Table 3-2 are specified to the nearest second.

The FWLB demonstration analysis modeled a rupture of main feedwater system piping with a coincident LOP, and with all ADVs unavailable for a 4-hour time frame post-accident. To maximize the potential for pressurizer fill during the resulting RCS heatup, an automatic restart of a charging pump following LOP and SIAS was incorporated into the simulation. The only operator action that was included in the event sequence was an action to turn off that charging pump at 20 minutes post-accident.

The PSVs actuated once during the simulation to relieve pressure in the RCS. MSSVs, however, repeatedly cycled open and closed during the simulation to remove heat. Because the MSSVs are capable of operating for hundreds of cycles without failure as explained in References 7 and 15, the number of predicted MSSV operating cycles was determined to be acceptable.

The acceptance criteria pertaining to pressurizer fill and water entrainment in the PSV effluent are the same as those used in the FWLB safety analysis described in UFSAR Section 15.2.8 [Reference 9], as described in the response to Request 2.a above. Additionally, because FWLBs may result in a more rapid RCS pressurization than other postulated design basis events, FWLB simulations are checked against the parameters of post-Three Mile Island PSV operability tests, as explained in Table 3-1 of Attachment 1. It was concluded that the predicted PSV response in the FWLB demonstration analysis was bounded by the pertinent PSV test data.

Response 3.b

The following figures are provided in Attachment 2 of this enclosure:

- Figure 3-1 FWLB Demonstration Analysis – Short-Term Instantaneous Charging Flow Rate vs. Time

- Figure 3-2 FWLB Demonstration Analysis – Long-Term Instantaneous Charging Flow Rate vs. Time
- Figure 3-3 FWLB Demonstration Analysis – Total Delivered Charging Flow vs. Time
- Figure 3-4 FWLB Demonstration Analysis – Pressurizer Pressure vs. Time
- Figure 3-5 FWLB Demonstration Analysis – Pressurizer Water Volume vs. Time
- Figure 3-6 FWLB Demonstration Analysis – Short-Term Steam Generator Pressure vs. Time
- Figure 3-7 FWLB Demonstration Analysis – Long-Term Steam Generator Pressure vs. Time
- Figure 3-8 FWLB Demonstration Analysis – Short-Term Wide Range Steam Generator Level vs. Time
- Figure 3-9 FWLB Demonstration Analysis – Long-Term Wide Range Steam Generator Level vs. Time
- Figure 3-10 FWLB Demonstration Analysis – Short-Term Steam Generator Water Mass vs. Time
- Figure 3-11 FWLB Demonstration Analysis – Long-Term Steam Generator Water Mass vs. Time

The last two figures, pertaining to SG water mass, are provided to demonstrate how the intact SG recovers from a FWLB.

Response 3.c

A limiting fault FWLB, such as the one described in the PVNGS demonstration analysis, is anticipated to result in an automatic reactor trip. Operators would therefore respond to the event by first entering emergency procedure 40EP-9EO01, *Standard Post Trip Actions* [Reference 19], and by performing required safety function status checks and diagnostics in a manner similar to that described above for the SGTR event (see the response to Request 1.e). In the context of a FWLB, the pertinent safety functions that relate to charging pump operation are RCS inventory control and RCS pressure control.

Short-term RCS inventory control is assured, in part, by checking that pressurizer level is both in the range of 10% to 65%, and trending as expected to the range of 33% to 53%. If pressurizer level does not meet both of these criteria, then operators would attempt to implement short-term contingency actions, pending completion of event diagnosis, by taking control of the PLCS, the charging pumps, and/or the letdown control valves. Likewise, short-term RCS pressure control is assured, in part, by checking that pressurizer pressure is both in the range of 1837 psia to 2285 psia, and trending as expected to the range of 2225 psia to 2275 psia. If pressurizer pressure does not meet both of these criteria, then operators would attempt to implement short-

term contingency actions by taking control of the Pressurizer Pressure Control System (PPCS), pressurizer heaters, and/or pressurizer spray.

In the event of a limiting fault FWLB that results in both an increase in pressurizer level as well as an increase in pressurizer pressure, control room operators would comply with 40EP-9EO01 by exercising judgment regarding how best to take short-term control of the charging pumps. On one hand, pressurizer level would begin to trend upward outside the desired band, so operators may decide to turn off charging to assist with level control. On the other hand, pressurizer pressure would also begin to trend upward outside the desired band, so operators may decide to turn on a charging pump to provide auxiliary pressurizer spray to assist with pressure control.

During the NRC regulatory audit of November 27-29, 2012, a crew of PVNGS licensed operators were tasked with this scenario in one of the control room simulators. The crew responded properly to the plant conditions by taking control of charging within just a few minutes of event initiation, and operating a single pump intermittently to control pressurizer pressure with auxiliary spray. Pressurizer level did increase to a value outside the band specified in 40EP-9EO01 (see the response to Request 3.d below), but the indicated level remained on-scale and the crew's control of pressurizer pressure prevented repeated cycling of the PSVs.

Additionally, the short-term non-conformance with the pressurizer level criteria in 40EP-9EO01 became pertinent to the event diagnosis (See also the response to Request 1.e, which notes that a reactor trip is determined to be uncomplicated only if all safety function acceptance criteria of 40EP-9EO01 are met. There is no requirement that all acceptance criteria be met while executing the instructions and contingency actions of that procedure.). Upon completion of event diagnosis in accordance with 40EP-9EO01, the crew properly decided to enter the optimized response procedure for FWLBs, 40EP-9EO05, *Excess Steam Demand* [Reference 23].

It should be noted that there is no definitive requirement in either 40EP-9EO01 or 40EP-9EO05 that mandates that charging be secured and left off for the event duration.

The FWLB demonstration analysis differs from the control room simulator exercise of November 28, 2012, in two important respects, as follows:

- The FWLB demonstration analysis utilized deterministic methods similar to those utilized for PVNGS UFSAR Chapter 15 safety analyses. None of these PVNGS safety analyses credit auxiliary pressurizer spray for RCS pressure control and event mitigation because its associated equipment is not considered to be fully qualified for safety analysis applications, as indicated by Reference 24. Auxiliary pressurizer spray is considered a highly reliable system, however, and licensed operators routinely train on use of the system as evidenced during simulator exercise that was performed during the NRC regulatory audit.
- The purpose of the deterministic analysis was to demonstrate whether the plant could automatically control or accommodate a FWLB, or whether operator intervention would be required for the analysis to meet all pertinent acceptance

criteria. Once it was decided that operator action would be required, the analysis was performed in such a manner that only a minimum number of possible actions would be modeled. Also, as explained in the preceding bullet, auxiliary pressurizer spray was not credited in the analysis, so the analysis did not model manual, intermittent operation of a charging pump. Given the potential for pressurizer overfill, it was concluded that turning off a charging pump would assure conformance to all pertinent acceptance criteria, provided operator action occurred no later than 20 minutes post-accident (that is, selection of this time frame maximized the predicted pressurizer water level during the CENTS simulation while meeting acceptance criteria). Turning off the charging pump may not be consistent with the 40EP-9EO01 contingency actions for RCS pressure control as described in the bullet above, but it is consistent with the 40EP-9EO01 contingency actions for RCS level control, which also allow the operator to take manual control of charging.

Response 3.d

During the control room simulator exercise that was performed during the NRC regulatory audit, licensed operators utilized PVNGS emergency procedures to respond to a FWLB with a coincident LOP. The operators were informed prior to the exercise that ADVs were unavailable, and thus no attempt was made to use them.

The scenario involved a FWLB inside containment, which resulted in a rapid increase in containment pressure and an automatic reactor trip. High containment pressure also resulted in a Safety Injection Actuation Signal (SIAS), Containment Isolation Signal (CIAS), Containment Spray Actuation Signal (CSAS), and Main Steam Isolation Signal (MSIS). Post-trip, the LOP resulted in the loss of the PPCS and PLCS, and coastdown of the RCPs. The MSIS actuation also resulted in the loss of the SBCS, so heat removal from the secondary system was accomplished by automatic cycling of the MSSVs.

Pressurizer pressure and level began to increase as the RCS heated up. A charging pump was automatically load sequenced to the diesel generator as a result of the LOP with a coincident SIAS signal, thereby adding inventory to the RCS. The LOP also resulted in a loss of NCW, which resulted in a loss of letdown from the RCS.

Operators responded to the event by assessing plant conditions and the status of safety functions in accordance with 40EP-9EO01. As explained in the response to Request 3.c, the operators decided to take manual control of charging for the purpose of RCS pressure control through use of the auxiliary pressurizer spray system. Pressurizer pressure was reduced and maintained within the band specified in 40EP-9EO01, thereby preventing repeated cycling of PSVs.

Pressurizer level continued to rise during performance of 40EP-9EO01, and eventually exceeded the band specified in that procedure. Upon completion of the standard post-trip actions, the indicated pressurizer level remained on-scale at approximately 77%, and the operators diagnosed the event as a FWLB. The crew then made the

appropriate decision to enter the optimized response procedure for that event, 40EP-9EO05, as noted above in the response to Request 3.c.

For comparative purposes, the FWLB demonstration analysis predicted that pressurizer level would exceed about 90%, at about the same time that the operators in the simulator observed a level of approximately 77%.

NRC Request 4

The RAI response refers to the use of the non-safety-related SBCS valves as a defense-in-depth measure while the plant is in the 24-hour condition statement with all four ADVs inoperable and a design basis event such as an SGTRLOP occurs. The Final Safety Analysis Report (FSAR) describes two SBCS valves as atmospheric relief valves with the same relief capacity as an ADV. However, the atmospheric SBCS valves are non-safety-related valves located in the turbine building, downstream of the MSIVs.

a. Please describe why the 1007/1008 SBCS valves would be available as a defense-in-depth measure when there are no ADVs available to mitigate accidents and transients described in the FSAR. Please include a description of the means to control operation of the valve (i.e., control power, medium to physically reposition valve, and remote/local/manual capabilities).

b. Since the atmospheric SBCS valves are downstream of the MSIVs, please describe whether the SBCS valves will be available during transients and accidents or if the SBCS valves can be made available through operator actions in a reasonably short period of time to provide accident mitigation and whether sufficient flow can be achieved through these normal/alternate/bypass lines.

c. In the event the ADVs are not available during an accident, the atmospheric SBCS valves may be used to conduct a RCS cooldown. Please describe how operators will execute steps in the SGTR procedures related to use of the SBCS valves and closing the MSIVs, and achieving the use of the SBCS valves with the MSIVs closed in FWLB and MSLB procedures.

d. Please determine whether any measures are necessary or in place to assure that the 1007/1008 SBCS valves are available for accident mitigation prior to entering Technical Specification Condition statement for all four ADVs not available.

APS Response 4

Response 4.a

Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,* [Reference 25], Section 2.2.1, *Defense-in-Depth*, provides examples of the elements of the defense-in-depth philosophy for Technical Specifications. While the proposed LAR is not a risk-informed submittal, the proposed change to the TS 3.7.4, *Atmospheric Dump Valves,* Action B, for potentially all ADVs inoperable and a completion time of 24 hours, remains consistent with the defense-in-depth elements provided in the regulatory guide.

The defense-in-depth philosophy has traditionally been applied in reactor design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena, which (because they are unknown or unforeseen) neither the PRA nor traditional analyses reflect. Consistency with the defense-in-depth philosophy is maintained by the proposed LAR.

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved (i.e., the proposed change in the TS does not significantly change the balance among these principles of prevention and mitigation) to the extent that such balance is needed to meet the acceptance criteria of the specific design-basis accidents and transients. The APS proposed LAR completion time of 24 hours is unchanged from the current completion time and the PVNGS facility design remains robust in the design features to provide for accident mitigation, as will be described further in this response.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system. Specifically, appropriate restrictions will be in place to preclude simultaneous equipment outages that would erode the principles of redundancy and diversity, as described further in the response to Request 4.d in this enclosure and in the proposed TS Bases, which states, in part: "Entry into Condition B for all four ADV lines simultaneously inoperable is not intended for voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable."

- Defenses against potential common-cause failures (CCFs) are maintained and the potential for introduction of new CCF mechanisms are assessed. The proposed TS change does not change the existing LCO Condition completion

time of 24 hours, so there are no anticipated operational changes associated with a change in a CT that could introduce any new CCF modes not previously considered. The relevant element of the existing TS Bases, unchanged by this proposed LAR, continues to indicate that the SBCS can be used to support the 24 hour completion time. Specifically, the TS Bases states: "The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass Control System and MSSVs, and the low probability of an event occurring during this period that requires the ADV lines."

- Defenses against human errors are maintained. The proposed TS change does not propose any operational changes associated with a change in a CT that could change the expected operator response or introduce any new human errors not previously considered, such as a change from performing maintenance during shutdown to performing maintenance at power, when different personnel and different activities may be involved. The 24 hour completion time remains unchanged. As will be described further in the response to Request 4.c in this enclosure, enhancements to existing emergency operating procedures are planned, to coincide with implementation of the approved amendment, that would clarify operational alternatives for SGTR and Excess Steam Demand events.

- The intent of the plant's design criteria is maintained. The proposed TS change does not alter the plant's design criteria. The change establishes more restrictive LCO controls for the ADVs, consistent with the PVNGS existing design.

The following describes why the 1007 / 1008 SBCVs would be available as a defense-in-depth measure when there are no ADVs available to mitigate accidents and transients described in the UFSAR. Included is a description of the means to control operation of the valve (i.e., control power, medium to physically reposition valve, and remote / local / manual capabilities).

Various redundant power supplies support the SBCS system from the control room. Specifically, control power is provided by DC distribution panel (NKN-D42) which feeds 3 Auxiliary Relay Cabinets; ZAN-C01, C02 and C03. NKN-D42 is fed from DC bus NKN–M45, which, in addition to the battery, can be aligned to the E, EF and E1 charger. This design provides redundancy in DC power for the SBCS.

NNN-D11 is an uninterruptible instrument 120 VAC power supply to the SBCS control system with a non-class and class power supply (NHN-M13 and PHA-M31), NNN-D12 also feeds the SBCS control system with an uninterruptible 120 VAC power supply (PHB-M32 and NHN-M10). This design provides redundancy in 120 VAC power for the SBCS, including power from the standby diesel generators, in the event of a LOP.

In the event that SBCS valves 1007 and 1008 are not operable from the control room boards, procedural guidance (41/42/43OP-1/2/3OP01, *Manual Operation of Air Operated Valves*, Appendix B) exists for local manual operation.

Both the ADV and SBCS valves use similar actuators, but somewhat different controls. Manual operation of the SBCS valves is accomplished in a similar manner as the ADVs. Air is manually isolated, and an equalizing valve on the positioner is taken to manual. A manual override shaft, using a clevis, connects to an actuator shaft. A manual handwheel is then utilized to open and close the valve as needed.

Access to the SBCS valves 1007 and 1008 is unobstructed, and located in the turbine building, separate from the MSSVs in the Main Steam Support Structure (MSSS). This location provides easier and safer access for operating personnel for remote manual operation of the valves, as compared to the ADVs that are located in the MSSS.

Both the SBCS valves and the ADVs are electro-pneumatic, 12-inch globe drag valves. A single SBCS valve is capable of passing slightly more than 10 percent steam flow. An ADV can pass 1.47×10^6 lbm/hr and a SBCS atmospheric relief valve can pass 1.36×10^6 lbm/hr.

Similar to the ADVs, 19 turns would be approximately 10 percent open. Since the plant is shutdown, utilizing the MSIV bypass valves, as compared to the MSIVs, would be sufficient due to the lower steam flowrate needed to control decay heat.

The worst case steam flow path consists of flow through one MSIV bypass line and subsequently through the two SBCS valves that release to atmosphere. This flow path will be sufficient to cooldown the plant after approximately 40 minutes. The potential to rely upon this heat removal flow path, in the unlikely scenario of all 4 ADVs being inoperable, is appropriate because the plant has been shown to be capable of maintaining a safe hot standby condition for 4 hours following a SGTRLOP with all ADVs inoperable, without operator intervention, as described in Reference 7.

Based upon the non-class low pressure nitrogen storage capacity for each PVNGS unit, the operators are expected to have sufficient pneumatic power to remotely operate the MSIV bypass valves and the two SBCS valves that release to atmosphere, in the control room. The operators are expected to be provided a minimum of 5.4 hours of pneumatic power. This bounds the control power that is limited to 2 hours of operation by batteries. The flow capacity of the low pressure vaporizer is also expected to be satisfactory.

Relying on battery backup, both the SBCS valves and MSIV bypass valves will have a minimum of two hours of control power in the control room. This is believed to be an acceptable amount of time to station an operator to remotely manually control the required valves 1007 and 1008, if needed.

Response 4.b

The SBCS valves 1007 and 1008 would remain available during transients and accidents that do not result in an MSIS. For those events where an MSIS is initiated, the MSIV bypass valves SG-169 and SG-183 could be opened.

Currently Unit 2 has new actuators on SG-169 and SG-183 and can be opened from the control room without additional operator actions. For Units 1 and 3, there is currently an administrative clearance (electrical only), that would need to have a single disconnect switch closed, to allow operation from the control room. Unit 1 is scheduled to receive new actuators for the MSIV bypass valves in the Spring 2013 outage, and Unit 3 in the Fall 2013 outage, which will make Units 1 and 3 the same as Unit 2, such that the MSIV bypass valves SG-169 and SG-183 could be opened from the control room, with no additional operator actions required outside the control room.

Since control power is available in the control room and remote manual operation options exist, as described in the response to Request 4.a of this enclosure, ample time exists to open the MSIV bypass valves to permit use of the SBCS valves 1007 and 1008, following completion of the standard post trip actions (SPTAs), that typically take about 15 minutes to perform.

Response 4.c

As demonstrated during the NRC audit, the PVNGS operations crews are cognizant of the various emergency operating procedures and are capable of responding to the various events, using existing procedures, even for the unlikely scenario of not having any of the ADVs available. As a result of the NRC audit and operating experience in the simulator, as well as the finalization of the demonstration analysis, enhancements to the EOPs are being planned to coincide with implementation of the proposed amendment, if approved. The specific enhancements are described in this response, with the discussion of each of the relevant procedures.

During the period immediately following reactor trip, the operations crew performs the standard post-trip actions (SPTAs) in accordance with emergency operating procedure 40EP-9EO01, *Standard Post Trip Actions,* which requires that the safety functions acceptance criteria be addressed in a specified hierarchy. The RCS heat removal guidance is provided in Section 3, *Instructions / Contingency Actions*, number 7.b, which includes the following:

> Check that T_c is 560 - 570°F.

This SPTA includes contingency actions if T_c is out of the acceptance band to restore T_c to the specified band using SBCS or ADVs. The contingency actions also state, for

being below the acceptance band, that if MSIS has actuated and the cooldown terminates, then stabilize T_c using the ADVs. This direction and the related contingency actions are considered to be appropriate and no enhancements are planned for procedure 40EP-9EO01, *Standard Post Trip Actions.*

Following completion of the SPTAs, the operating crew would transition to procedure 40EP-9EO04, *Steam Generator Tube Rupture,* based upon the diagnostic actions flow chart of the 40EP-9EO01, *Standard Post Trip Actions* procedure, Section 4.0, *Diagnostic Actions.*

Consistent with Section 3, *Instructions / Contingency Actions*, of procedure 40EP-9EO04, *Steam Generator Tube Rupture,* operators will reach Instruction number 9.c which states:

> Commence an RCS cooldown to a T_h of less than 540°F using SBCS.

The contingency action for this step, which would be applicable for this SGTR with a LOP scenario (which makes the condenser unavailable), is to cooldown to a T_h of less than 540°F using the ADVs by one of the following:

- Operation from the Control Room
- Appendix 18, Local ADV Operation

For the particular scenario assumed in the NRC audit and this RAI, of not having any ADVs available, the operations crew could apply the following direction from procedure 40DP-9AP16, *Emergency Operating Procedure Users Guide,* Section 20.0, *Directions for Use,* step 7.d to address this SGTR EOP step:

> Emergencies may not proceed as expected. Sufficient flexibility must be provided to aid the CRS with steps that cannot be performed as written. If a step cannot be performed as written, and the CRS wants to perform the step in another manner, then he shall obtain the concurrence of the SM.

By this process, the operations crew could proceed to cooldown using SBCS valves 1007 and 1008, in lieu of the ADVs. This approach could also be used for SGTR Instruction number 29 that is related to cooling down to shutdown cooling entry conditions.

Based upon operating experience in the simulator, as well as the finalization of the demonstration analysis, enhancements to the SGTR EOP and related safety function recovery procedure (40EP-9EO09, *Functional Recovery;* success paths for heat removal, HR-1 and HR-2) are being planned to coincide with implementation of the proposed amendment, if approved.

Specifically, enhancements are planned to more clearly direct the use of the SBCS valves 1007 and 1008 if the condenser and ADVs are not available. In addition, procedure 40EP-9EO04, *Steam Generator Tube Rupture,* Instruction numbers 11, 13, 16 and 47 will be assessed to ensure clarity with regard to MSIS actuation and use of the MSIV bypass valves.

For the case of use of the SBCS valves 1007 and 1008 for feedwater and main steam line breaks, when ADVs are not available, emergency operating procedure 40EP-9EO05, *Excess Steam Demand,* Section 3.0, *Instructions / Contingency Actions,* contains similar direction to the SGTR procedure. Specifically, Instruction number 14, states:

> Steam the least affected Steam Generator using any of the following:
> - SBCS
> - ADVs from the Control Room
> - Appendix 18, Local ADV Operation

Specifically, enhancements are planned to more clearly direct the use of the SBCS valves 1007 and 1008, if the condenser and ADVs are not available. In addition, procedure 40EP-9EO05, *Excess Steam Demand,* Instruction number 47 will be assessed to ensure clarity with regard to use of the MSIV bypass valves to support cooldown to shutdown entry conditions.

Response 4.d

APS does not plan to perform elective maintenance or testing on the ADVs that would render redundant trains inoperable. Specifically, the proposed TS Bases for 3.7.4 contains the following note:

> Entry into Condition B for all four ADV lines simultaneously inoperable is not intended for voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

In addition, procedure 51DP-9OM03, *Site Scheduling*, Appendix B, is planned to be enhanced to contain a statement ensuring ADV work is to be scheduled when SBCS valves 1007 and 1008 are available and that SBCS valves 1007 and 1008 are to be scheduled only when ADVs are operable. This procedural enhancement is a defense-in-depth measure to provide additional confidence that the ADV / SBCS heat removal safety function will be available.

APS Conclusion

There would be sufficient time for the operators to diagnose and mitigate an event even if the ADVs were not available. The MSSVs would provide adequate heat removal for several hours without the use of ADVs. Therefore, the current TS completion time of 24 hours for four inoperable ADVs remains adequate.

The requested TS change would allow for continued operation of a PVNGS unit for up to 24 hours when all four ADV lines are inoperable. The proposed completion time for this condition remains the same as in the current PVNGS TSs. The demonstration analyses described herein, and in Reference 7, confirms that, should a postulated accident or transient occur while a PVNGS unit is in this condition, the unit may be maintained in a safe hot standby condition for at least four hours, without reliance on the ADVs.

There is time for operators to diagnose the event, assess plant conditions, and obtain support from emergency response personnel, if necessary, to effect one or more mitigation strategies with the aim of placing the unit in a safe cold shutdown condition. Such strategies include local manual operation of the ADVs, as specified in station Emergency Operating Procedures (EOPs), and other alternative actions that emergency response personnel may determine are appropriate for the event under consideration, including use of the SBCS from the control room and remote manual operations later in the event.

References

1. APS Letter to NRC, No. 102-06370-DCM/DFS, *Palo Verde Nuclear Generating Station (PVNGS); Units 1, 2, and 3; Docket Nos. STN 50-528, 50-529, and 50-530; Request for Amendment to Technical Specification 3.7.4, "Atmospheric Dump Valves (ADVs),"* June 22, 2011. [NRC ADAMS Accession No. ML11182A908]
2. NUREG-1432, *Standard Technical Specifications; Combustion Engineering Plants; Specifications*, Volume 1, Revision 3, June 2004. [NRC ADAMS Accession No. ML041830597]
3. NUREG-1432, *Standard Technical Specifications; Combustion Engineering Plants; Bases*, Volume 2 (Part 1 of 2), Revision 3, June 2004. [NRC ADAMS Accession No. ML041830100]
4. NUREG-1432, *Standard Technical Specifications; Combustion Engineering Plants; Bases*, Volume 2 (Part 2 of 2), Revision 3, June 2004. [NRC ADAMS Accession No. ML041830101]
5. NRC Letter to APS, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.7.4, "Atmospheric Dump Valves" (TAC Nos.*

ME6566, ME6567, and ME6568), August 31, 2011. [NRC ADAMS Accession No. ML112430084]

6. APS Letter to NRC, No. 102-06446-DCM/DFS, *Palo Verde Nuclear Generating Station (PVNGS); Units 1, 2, and 3; Docket Nos. STN 50-528, 50-529, and 50-530; Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.7.4, "Atmospheric Dump Valves (ADVs)" (TAC Nos. ME6566, ME6567, and ME6568)*, December 9, 2011. [NRC ADAMS Accession No. ML11356A088]

7. APS Letter to NRC, No. 102-06466-DCM/GAM, *Palo Verde Nuclear Generating Station (PVNGS); Units 1, 2, and 3; Docket Nos. STN 50-528, 50-529, and 50-530; Responses to Requests 2 and 3 of the Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.7.4, "Atmospheric Dump Valves,"* January 27, 2012. [NRC ADAMS Accession No. ML12046A649]

8. 10 CFR Part 50, Appendix A, *General Design Criteria for Nuclear Power Plants*, 36 FR 3256, February 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, February 12, 1976; 43 FR 50163, October 27, 1978; 51 FR 12505, April 11, 1986; 52 FR 41294, October 27, 1987; 64 FR 72002, December 23, 1999; and 72 FR 49505, August 28, 2007.

9. APS Letter to NRC, No. 102-06375-DCM/TLC, *Palo Verde Nuclear Generating Station (PVNGS); Units 1, 2, and 3; Docket Nos. STN 50-528/529/530; Updated Final Safety Analysis Report, Revision 16*, June 30, 2011.

10. NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Revision 3, Part 1 of 3, November 1978. [NRC ADAMS Accession No. ML011340072]

11. NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Revision 3, Part 2 of 3, November 1978. [NRC ADAMS Accession No. ML011340108]

12. NRC Regulatory Guide 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*, Revision 3, Part 3 of 3, November 1978. [NRC ADAMS Accession No. ML011340116]

13. Email, L. K. Gibson (NRC) to R. K. Roehler (APS), *Palo Verde Nuclear Generating Station – Request to Revise Technical Specifications Related to Atmospheric Dump Valves – Audit Plan (TAC Nos. ME6566, ME6567, and ME6568)*, September 17, 2012. [NRC ADAMS Accession No. ML12261A137]

14. NRC Letter to APS, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 – Regulatory Audit Summary and Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.7.4, "Atmospheric Dump Valves" (TAC Nos. ME6566, ME6567, and ME6568)*, December 13, 2012. [NRC ADAMS Accession No. ML12342A383]

15. APS Letter to NRC, No. 102-04847-CDM/TNW/RAB, *Palo Verde Nuclear Generating Station (PVNGS); Unit 2; Docket No. STN 50-529; Response to Request for Additional Information Regarding Steam Generator Replacement and Power Uprate License Amendment Request*, October 11, 2002. [NRC ADAMS Accession No. ML022940385]

16. Westinghouse Owners Group Letter to NRC, No. WOG-05-510, *Transmittal of Approved Topical Report: WCAP-15996-P-A, Rev. 1, Vols. 1-4 (Proprietary) and WCAP-15996-NP-A, Rev. 1, Vols. 1-4 (Non-Proprietary), "Technical Description Manual for the CENTS Code" (TAC No. MB6982) (WOG Project Task PA-ASC-0141)*, November 21, 2005. [NRC ADAMS Accession No. ML053290344]

17. 10 CFR 100.11, *Determination of Exclusion Area, Low Population Zone, and Population Center Distance*, 27 FR 3509, April 12, 1962, as amended at 31 FR 4670, March 19, 1966; 38 FR 1273, January 11, 1973; 40 FR 8793, March 3, 1975; 40 FR 26527, June 24, 1975; 53 FR 43422, October 27, 1988; 64 FR 48955, September 9, 1999; and 67 FR 67101, November 4, 2002.

18. PVNGS Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*, Revision 16, July 11, 2012.

19. PVNGS Procedure 40EP-9EO01, *Standard Post-Trip Actions*, Revision 18, August 1, 2012.

20. PVNGS Procedure 40EP-9EO04, *Steam Generator Tube Rupture*, Revision 26, August 19, 2011.

21. PVNGS Procedure 40EP-9EO10, *Standard Appendices*, Revision 74, December 12, 2012.

22. PVNGS Procedure 40DP-9AP16, *Emergency Operating Procedure Users Guide*, Revision 6, February 13, 2009.

23. PVNGS Procedure 40EP-9EO05, *Excess Steam Demand*, Revision 26, August 19, 2011.

24. APS Letter to NRC, No. 102-03278-WLS/AKK/GAM, *Palo Verde Nuclear Generating Station (PVNGS); Units 1, 2, and 3; Docket No. STN 50-528/529/530; Charging System/Auxiliary Pressurizer Spray System Commitments*, March 9, 1995.

25. NRC Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, Revision 1, May 2011.

ATTACHMENT 1

TABLES

List of Tables

Table 1-1. Acceptance Criteria and Analysis Results for the SGTR Demonstration Analysis

Parameter	Acceptance Criterion	Analysis Result
Maximum RCS Pressure	< 2750 psia (110% of 2500 psia design pressure)	< 2389 psia
Maximum SG Pressure	< 1397 psia (110% of 1270 psia design pressure)	< 1354 psia
Minimum Steam Volume in Ruptured SG	> 1257 ft^3 (SG overfill does not occur and water does not pass through main steam safety valves)	> 2433 ft^3
Thyroid Dose at Exclusion Area Boundary (2 Hours Post-Accident; PIS)[a]	< 300 Rem	< 15 Rem
Thyroid Dose at Low Population Zone (Event Duration; PIS)[a]	< 300 Rem	< 7 Rem[c]
Thyroid Dose at Exclusion Area Boundary (2 Hours Post-Accident; GIS)[b]	< 30 Rem	< 6 Rem
Thyroid Dose at Low Population Zone (Event Duration; GIS)[b]	< 30 Rem	< 5 Rem[c]

Notes: (a) Coincident Pre-Accident Iodine Spike (PIS) of 60 μCi/gm.
(b) Coincident accident-Generated Iodine Spike (GIS) with a spiking factor of 335.
(c) Event duration thyroid dose values correspond to the end of the 4-hour CENTS code simulation.

Table 1-2. Sequence of Events for the SGTR Demonstration Analysis

Time (seconds)	Event
0	SGTR occurs in SG 1 at hot side tubesheet.
33	Letdown flow decreases to the minimum programmed flow rate.
85	Pressurizer backup heaters turn on due to decreasing pressurizer pressure.
357	Third charging pump starts due to decreasing pressurizer level.
516	All pressurizer heaters turn off due to low pressurizer level heater cutoff.
612	RCS hot leg temperature reaches CPCS hot leg saturation margin trip setpoint.
615	Reactor trip occurs.
615	Turbine trip occurs.
615	Turbine admission valve closes.
615	Scram CEAs begin falling into the reactor core.
617	U-tubes in both SGs begin to uncover for the first time.
618	LOP occurs (RCPs trip; pressurizer heaters, pressurizer sprays, charging flow, and letdown flow disabled; main feedwater flow begins to ramp down).
619	First bank of MSSVs open on both SGs for the first time.
622	Second bank of MSSVs open on both SGs (only time).
624	U-tubes in both SGs begin to recover for the first time.
624	Maximum pressure reached in both SGs.
628	Main feedwater flow ceases to both SGs.
652	Second bank of MSSVs close on both SGs.
662	Pressurizer indicated level goes off-scale low for the first time.
668	Pressurizer pressure reaches SIAS setpoint; SI flow initiated automatically with 2 HPSI pumps.
682	First bank of MSSVs close on SG 2 for the first time.
685	Primary system mass flow rate begins transition to natural circulation.
687	First bank of MSSVs close on SG 1 for the first time.
736	Pressurizer indicated level begins to come back on-scale for the first time.

Table 1-2. Sequence of Events for the SGTR Demonstration Analysis (continued)

Time (seconds)	Event
762	Reactor power transitions from reactivity-based calculations to ANS decay heat curve (including 2σ uncertainty).
1,042	SG 2 water level decreases to the AFAS setpoint for the first time.
1,088	AFW delivered automatically to SG 2 as a result of AFAS, using 2 AFW pumps.
1,211	SG 2 water level increases to the AFAS reset setpoint for the first time.
1,212	AFW flow ceases automatically to SG 2 as a result of AFAS reset.
11,589	MSIS occurs due to high water level in SG 1.
11,592	MSIVs close.
11,723	AFW lockout occurs to SG 2 on high differential pressure between SGs.
14,358	Maximum water level occurs in ruptured SG.
< 14,400	AFW and MSSVs automatically operate throughout simulation to support heat removal.
14,400	End of demonstration analysis.

Table 1-3. Comparison Between CENTS SGTR Demonstration Analysis and Simulator Analogue

CENTS Demonstration Analysis	Simulator Analogue
SGTR+LOP w/o Operator Action for 4 hrs	SGTR+LOP w/o Operator Action for 4 hrs
Initial SGTR Leak Rate ~ 48.6 lbm/sec	Initial SGTR Leak Rate ~ 48.4 lbm/sec
CPCS Reactor Trip ~ 10 m 15 s	CPCS Reactor Trip ~ 9 m 44 s
AFW First Delivered to Intact SG ~ 18 m 8 s (AFW Delivered 7 Times to Intact SG)	AFW First Delivered to Intact SG ~ 20 m 28 s (AFW Delivered 7 Times to Intact SG)
No AFW Delivered to Ruptured SG	AFW First Delivered to Ruptured SG ~ 30 m 24 s (AFW Delivered 2 Times to Ruptured SG)
Main Steam Isolation ~ 3 h 13 m 9 s (High Level in Ruptured SG)	Main Steam Isolation ~ 1 h 38 m 34 s (Low Pressure in Ruptured SG)
AFW Lockout on SG ΔP ~ 3 h 15 m 23 s (Lower Pressure in Intact SG)	AF Lockout on SG ΔP ~ 3 h 25 m 44 s (Lower Pressure in Intact SG)
SG Overfill Does Not Occur	SG Overfill Does Not Occur

Table 1-4. SGTR Emergency Procedure Instructions and Contingency Actions (Excerpt Only)

Instructions	Contingency Actions
9. Perform the following:	
a. Ensure ARN-HS-19, Post Filter Mode Select Switch, is in the "THRU FILTER MODE."	
b. Select "OFF" on BOTH of the following switches: • SGN-HS-1007, Valve 7 Mode Select • SGN-HS-1008, Valve 8 Mode Select	
c. Commence an RCS cooldown to a T_h of less than 540°F using SBCS.	c.1 Cooldown to a T_h of less than 540°F using the ADVs by **ONE** of the following: • Operation from the Control Room • Appendix 18, Local ADV Operation
d. Perform Appendix 5, RCS and Pressurizer Cooldown Log.	
*10. **IF** steaming to atmosphere, **THEN** inform Radiation Protection and the RMS Technician.	
*11. **IF** CW flow to the Main Condenser is lost, **THEN** actuate MSIS.	

Table 2-1. Acceptance Criteria and Analysis Results for the MSLB Demonstration Analysis

Parameter	Acceptance Criterion	Analysis Result
Maximum RCS Pressure	< 2750 psia (110% of 2500 psia design pressure)	< 2563 psia
Maximum Pressurizer Water Volume (with Pressurizer Safety Valves open)	< 1697 ft^3 (water does not pass through Pressurizer Safety Valves)	< 1451 ft^3
Maximum Pressurizer Water Volume (with Pressurizer Safety Valves closed)	< 1800 ft^3 (water-solid condition does not occur in pressurizer)	< 1765 ft^3
Maximum SG Pressure	< 1397 psia (110% of 1270 psia design pressure)	< 1266 psia
Macbeth Minimum Departure from Nucleate Boiling Ratio (mDNBR)	> 1.30	> 2.4
Peak Linear Heat Rate	< 21 kW/ft	< 9.6 kW/ft[(a)]

Note: (a) Includes both fission power and decay heat power.

Table 2-2. Sequence of Events for the MSLB Demonstration Analysis

Time (seconds)	Event
0	Double-ended guillotine MSLB occurs on SG 1 inside containment; coincident LOP occurs and RCPs begin to coast down.
0	SG water level reaches MSIS setpoint; FWIVs close.
1	RCP shaft speed reaches CPCS auxiliary trip setpoint.
1	Reactor trip breakers open.
2	CEAs begin to fall into the core.
6	MSIVs closed; steam flow halted from SG 2.
14	AFW lockout occurs to SG 1 on high differential pressure between SGs.
20	AFW actuation and delivery to SG 2.
70	SIAS occurs on low pressurizer pressure.
79	Pressurizer empties.
88	Void begins to form in the reactor vessel upper head.
90	HPSI pump begins injecting into the RCS.
172	SI boron reaches the RCS cold legs.
251	AFAS reset occurs in SG 2 and AFW flow is terminated.
295	Maximum post-trip reactivity occurs.
341	Maximum return-to-power occurs.
341	Macbeth minimum DNBR occurs.
361	SG 1 dries out.
2,116	Maximum RCS pressure occurs.
2,190	Maximum SG pressure occurs.
2,560	Maximum pressurizer water level, with PSVs open, occurs.
13,202	Maximum pressurizer water level, with PSVs closed, occurs.
< 14,400	PSVs, AFW, and MSSVs automatically operate throughout simulation to support heat removal.
14,400	End of demonstration analysis.

Table 3-1. Acceptance Criteria and Analysis Results for the FWLB Demonstration Analysis

Parameter	Acceptance Criterion	Analysis Result
Maximum RCS Pressure	< 3000 psia (120% of 2500 psia design pressure)	< 2573 psia
Maximum Pressurizer Water Volume (with Pressurizer Safety Valves open)	< 1697 ft^3 (water does not pass through Pressurizer Safety Valves)	< 1650 ft^3
Maximum Pressurizer Water Volume (with Pressurizer Safety Valves closed)	< 1800 ft^3 (water-solid condition does not occur in pressurizer)	< 1691 ft^3
Pressurizer Pressure (with Pressurizer Safety Valves open)[(a)]	< 2697 psia	< 2451
Rate of Pressurizer Pressurization (with Pressurizer Safety Valves open)[(a)]	< 322 psi/sec	< 17 psi/sec
Maximum SG Pressure	< 1397 psia (110% of 1270 psia design pressure)	< 1367 psia

Note: (a) FWLB analysis results are compared against the parameters of PSV operability tests that were performed in response to a post-Three Mile Island accident action plan requirement, in order to ensure continued conformance to all pertinent licensing bases. The PSV operability tests and their associated parameters are described in PVNGS UFSAR Section 18.II.D.

Table 3-2. Sequence of Events for the FWLB Demonstration Analysis

Time (seconds)	Event
0	FWLB occurs on SG 1; complete loss of feedwater to both SGs.
31	Pressurizer pressure reaches reactor trip setpoint; high pressurizer pressure trip initiated.
31	SIAS, CIAS, MSIS, and CSAS signals generated.
31	PSVs open.
31	Faulted SG dries out; AFAS initiated to faulted SG.
32	Reactor trip breakers open; turbine trip occurs.
32	Maximum RCS pressure occurs.
32	Scram CEAs begin to fall into the core.
35	LOP occurs.
37	MSIVs close.
37	MSSVs open for the first time.
39	PSVs close.
43	Maximum SG pressure occurs.
44	AFW lockout to SG 1 occurs on high differential pressure between SGs.
60	MSSVs begin to close for the first time.
75	One charging pump restarts upon coincident LOP + SIAS condition.
77	AFW delivered to SG 2.
1,200	Operators take action to secure the operating charging pump.
2,761	Maximum pressurizer water volume occurs.
< 14,400	AFW and MSSVs automatically operate throughout simulation to support heat removal.
14,400	End of demonstration analysis.

ATTACHMENT 2

FIGURES

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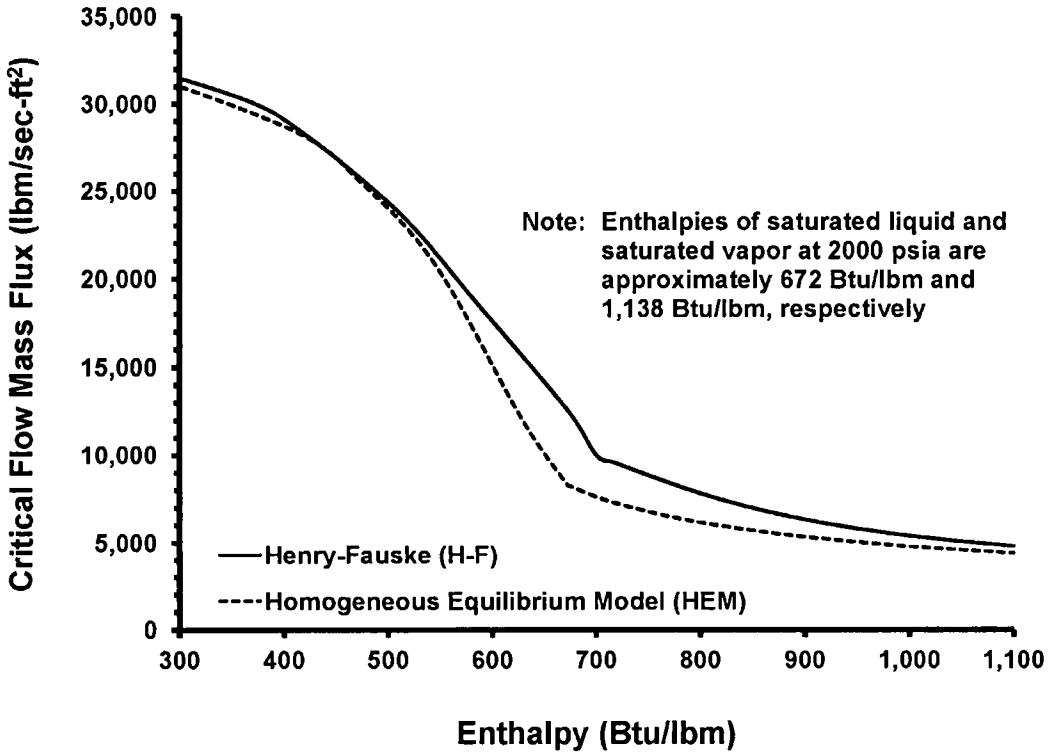

Figure 1-1. Critical Flow Mass Flux Predicted by the Henry-Fauske Correlation and the Homogeneous Equilibrium Model for an Upstream Stagnation Pressure of 2000 psia

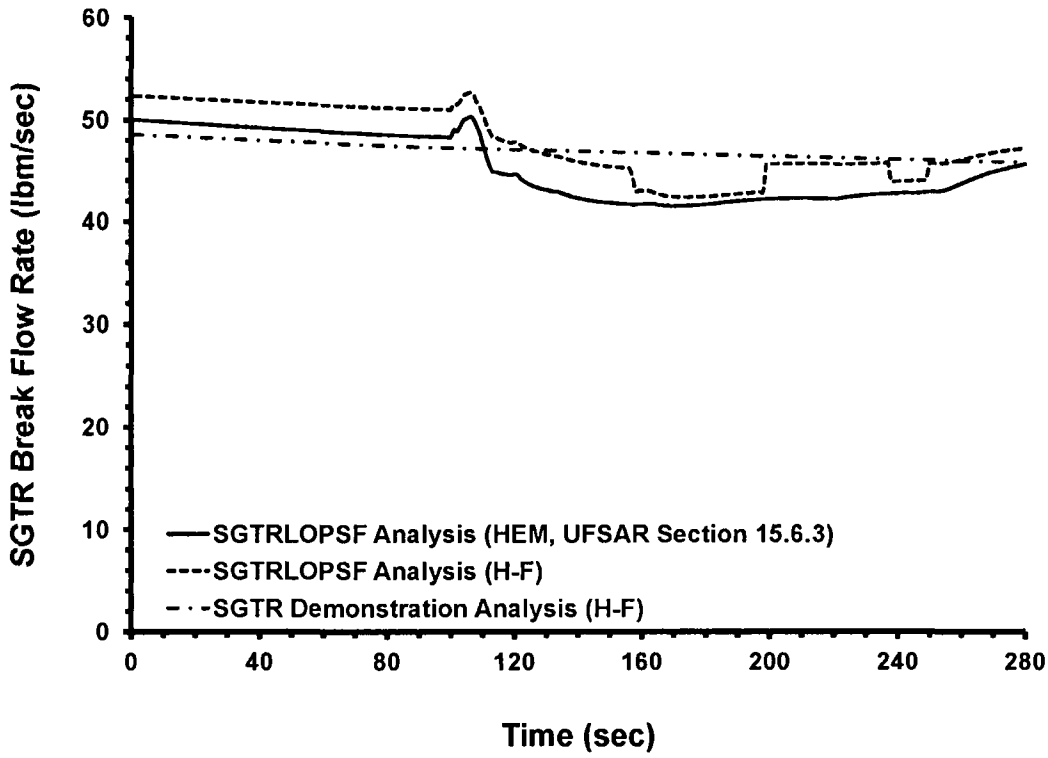

Figure 1-2. Short-Term SGTR Break Flow Rates Predicted by the Henry-Fauske Correlation and the Homogeneous Equilibrium Model

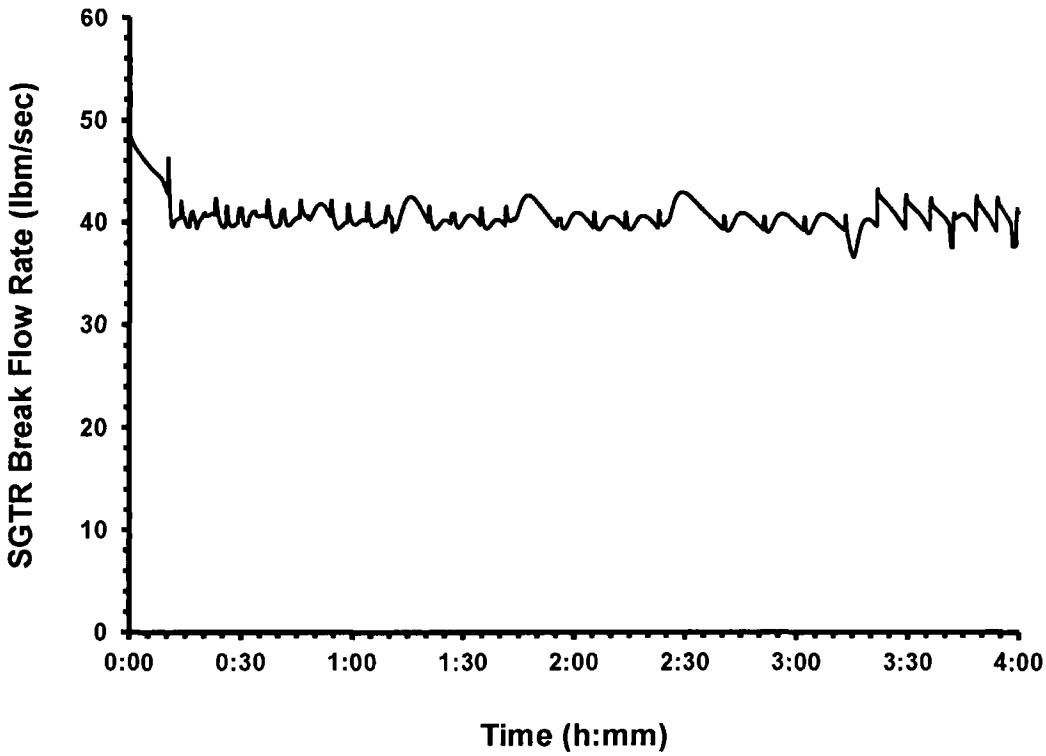

Figure 1-3. SGTR Demonstration Analysis – SGTR Break Flow Rate vs. Time

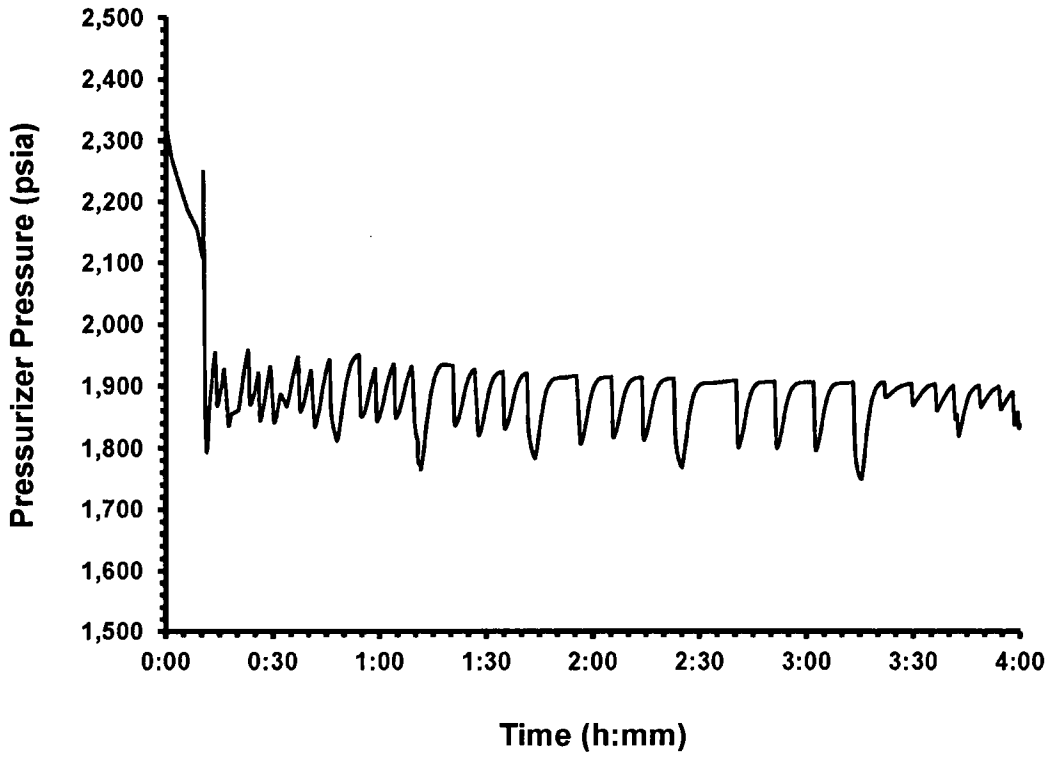

Figure 1-4. SGTR Demonstration Analysis – Pressurizer Pressure vs. Time

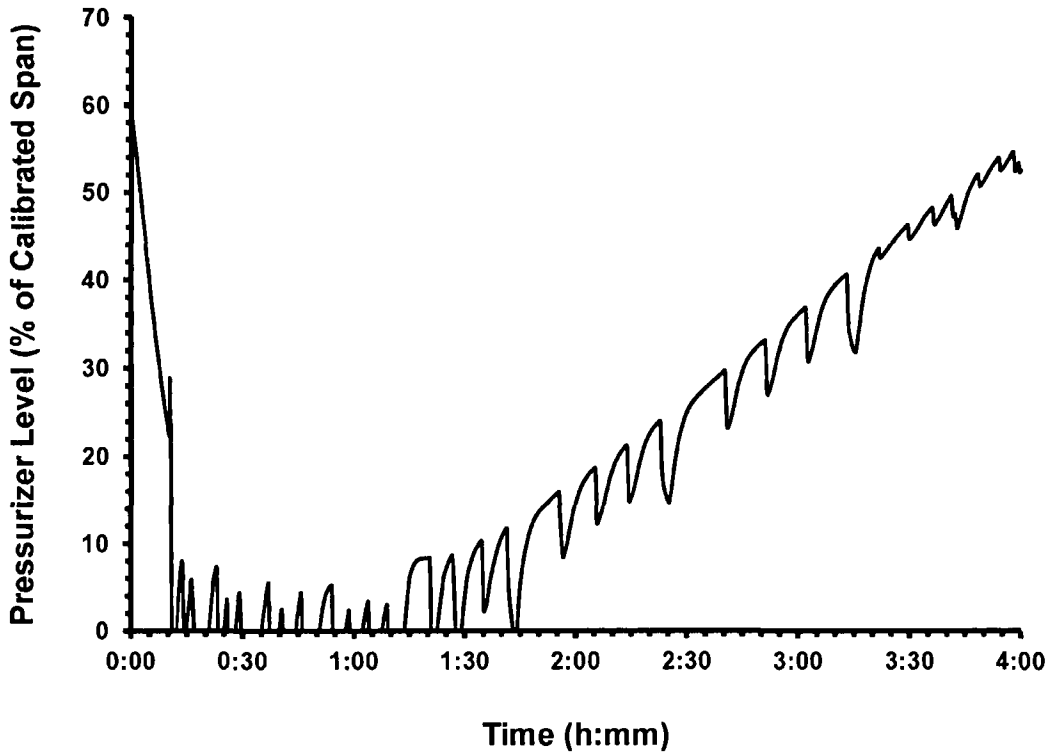

Figure 1-5. SGTR Demonstration Analysis – Pressurizer Level vs. Time

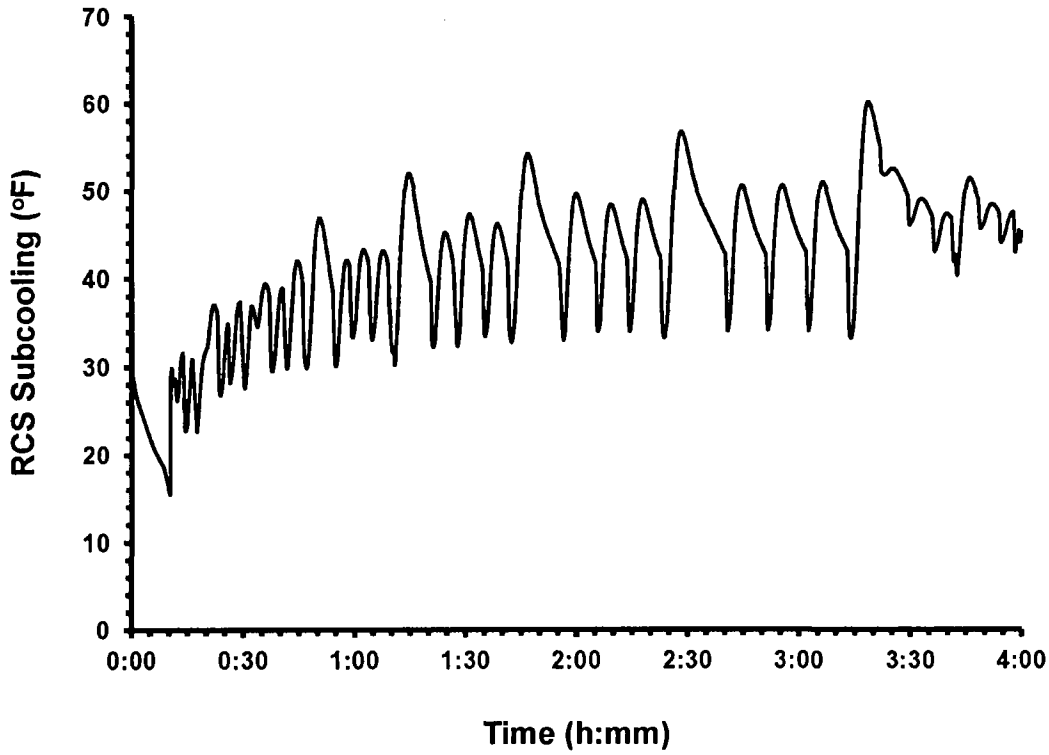

Figure 1-6. SGTR Demonstration Analysis – Reactor Coolant System Subcooling vs. Time

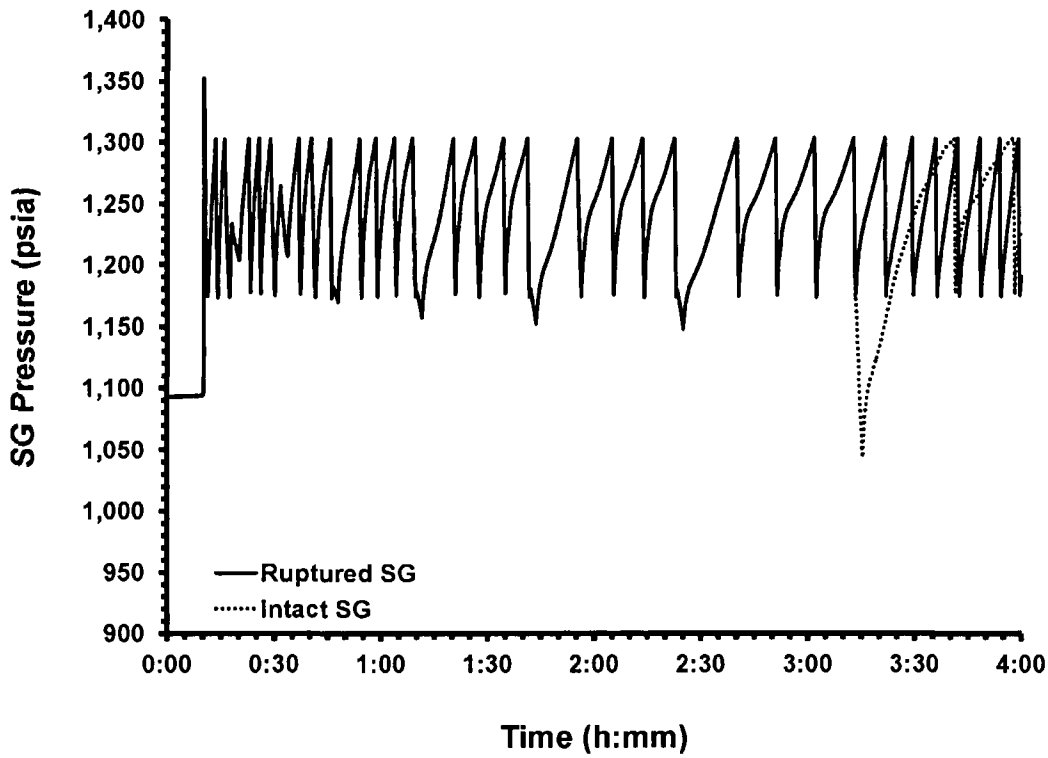

Figure 1-7. SGTR Demonstration Analysis – Steam Generator Pressure vs. Time

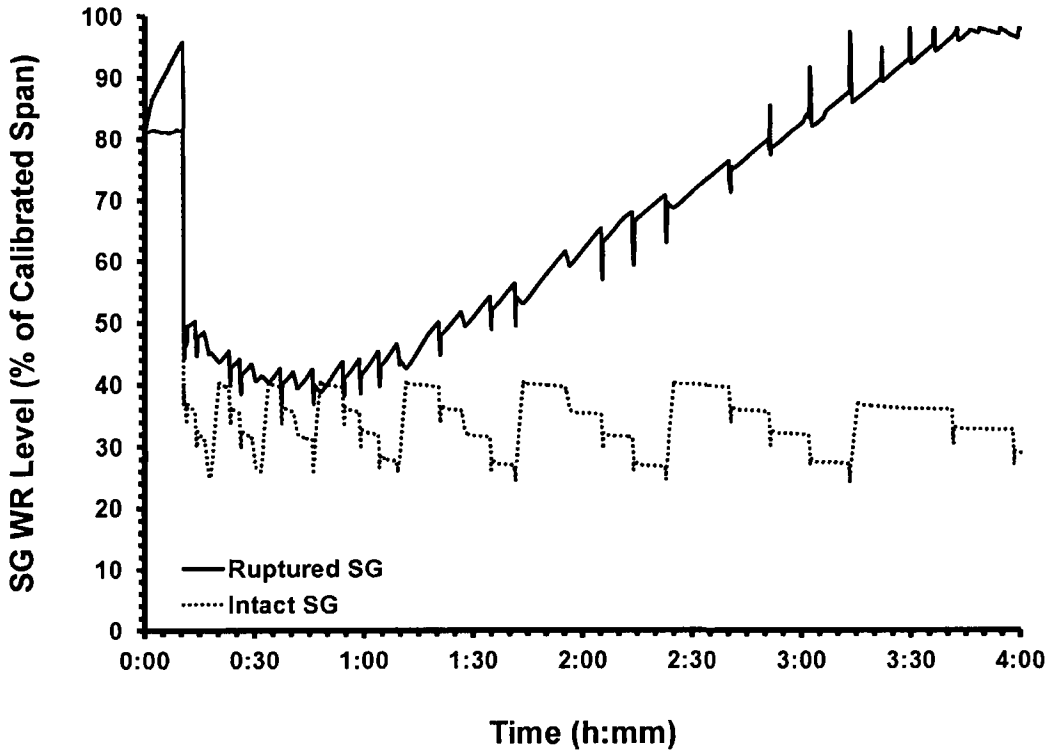

Figure 1-8. SGTR Demonstration Analysis – Wide Range Steam Generator Level vs. Time

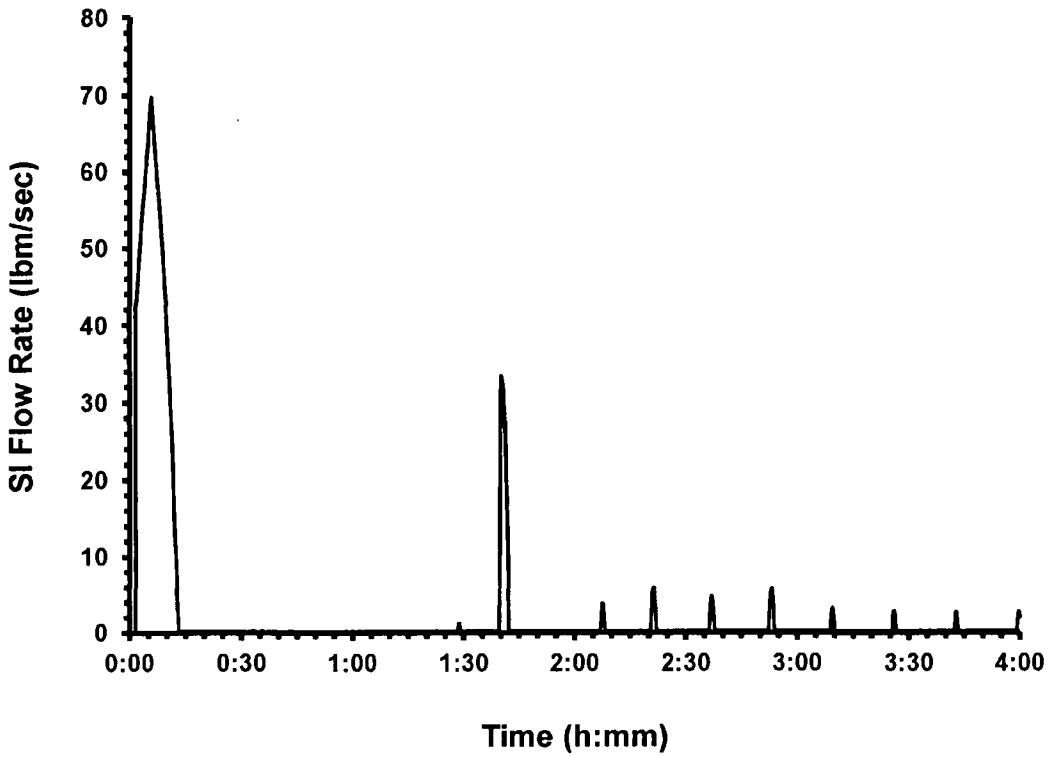

Figure 2-1. MSLB Demonstration Analysis – Instantaneous Safety Injection Flow Rate vs. Time

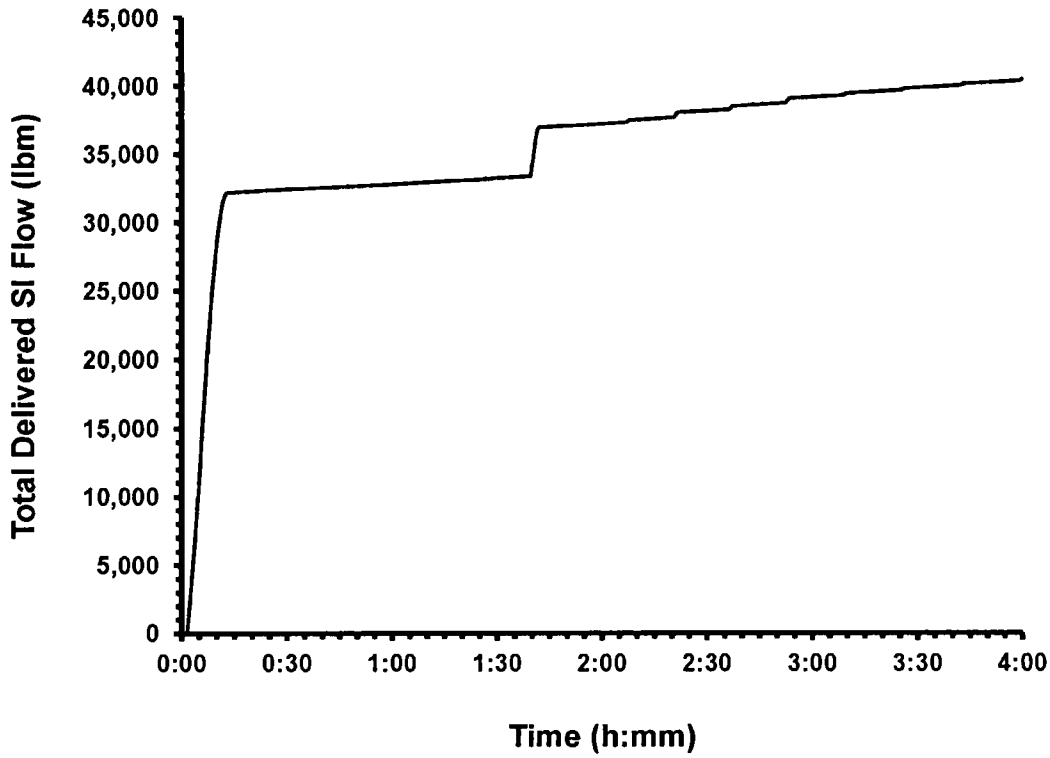

Figure 2-2. MSLB Demonstration Analysis – Total Delivered Safety Injection Flow vs. Time

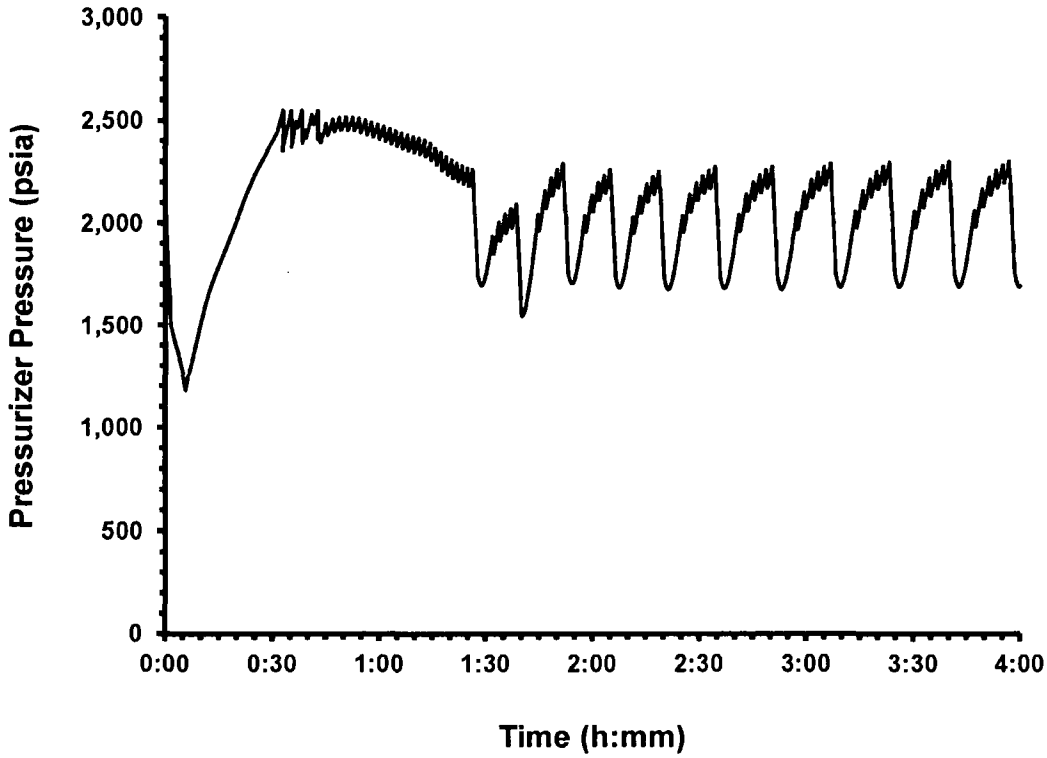

Figure 2-3. MSLB Demonstration Analysis – Pressurizer Pressure vs. Time

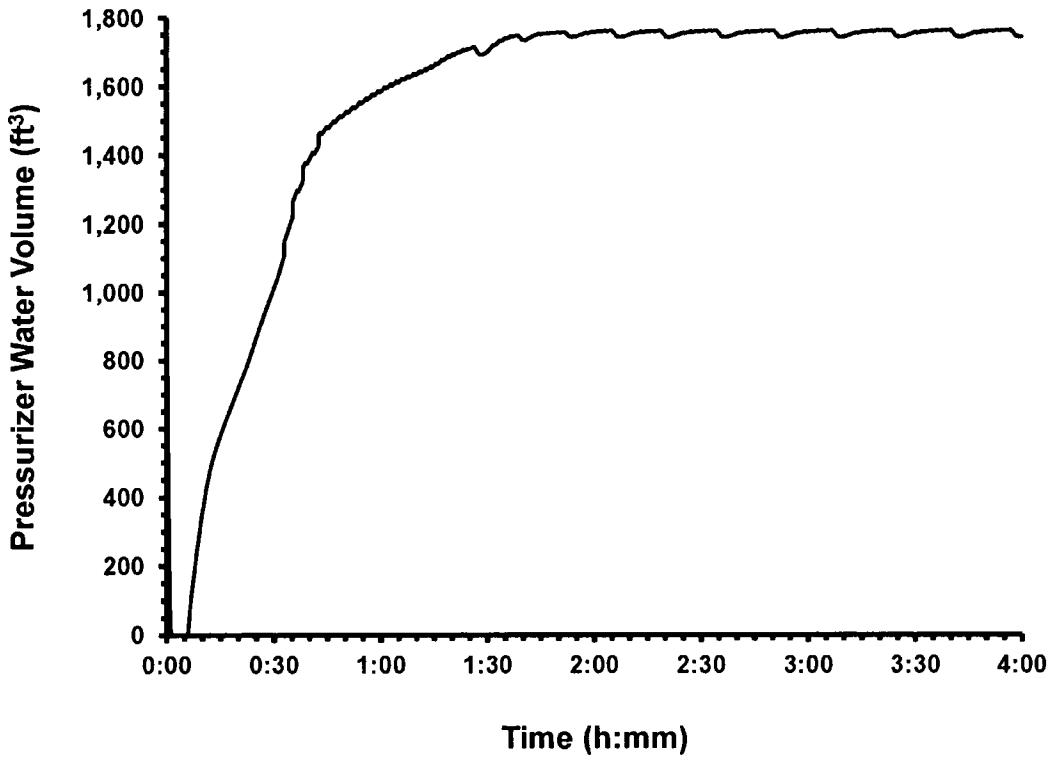

Figure 2-4. MSLB Demonstration Analysis – Pressurizer Water Volume vs. Time

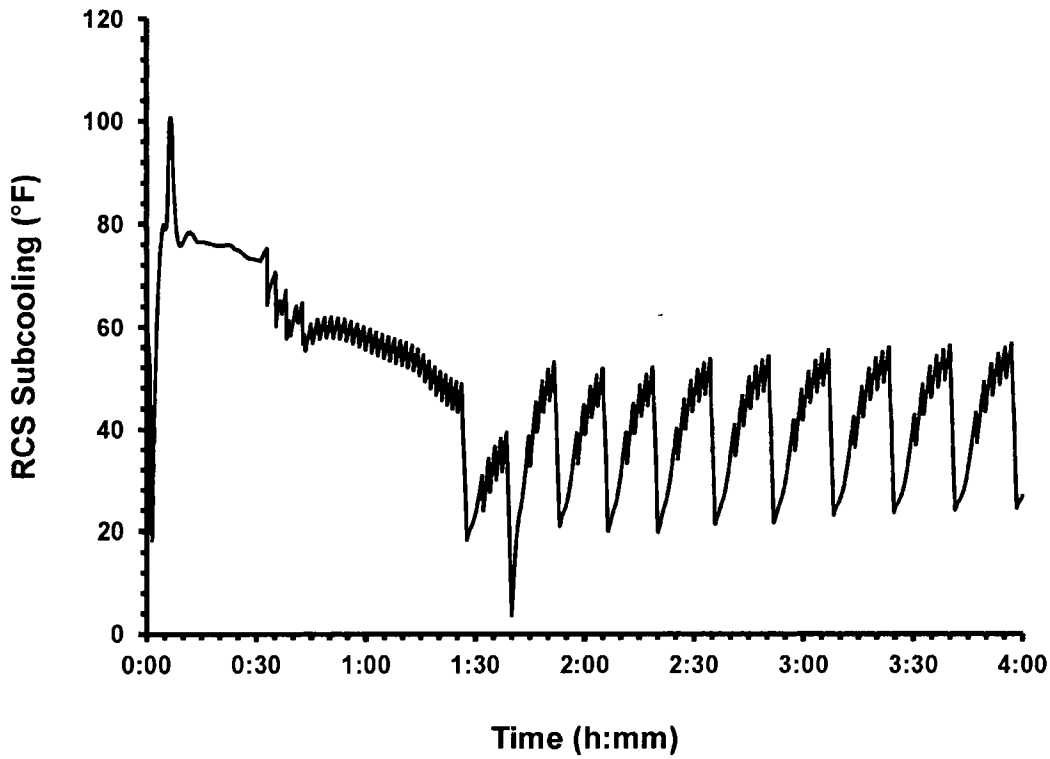

Figure 2-5. MSLB Demonstration Analysis – Reactor Coolant System Subcooling vs. Time

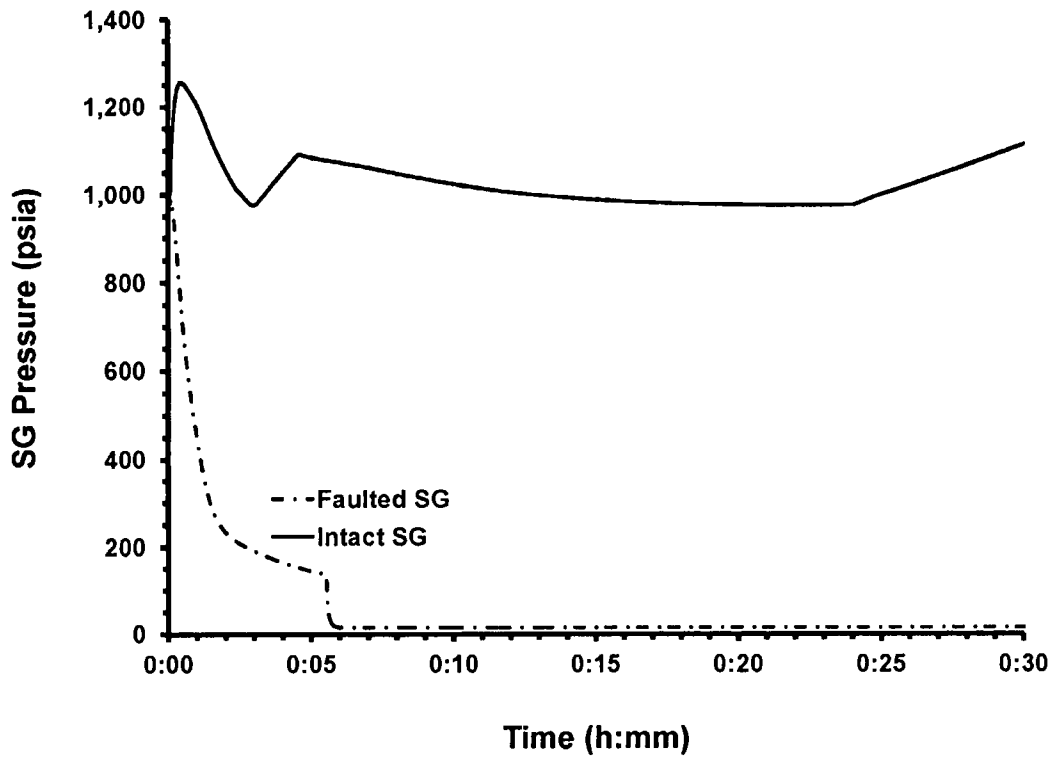

Figure 2-6. MSLB Demonstration Analysis – Short-Term Steam Generator Pressure vs. Time

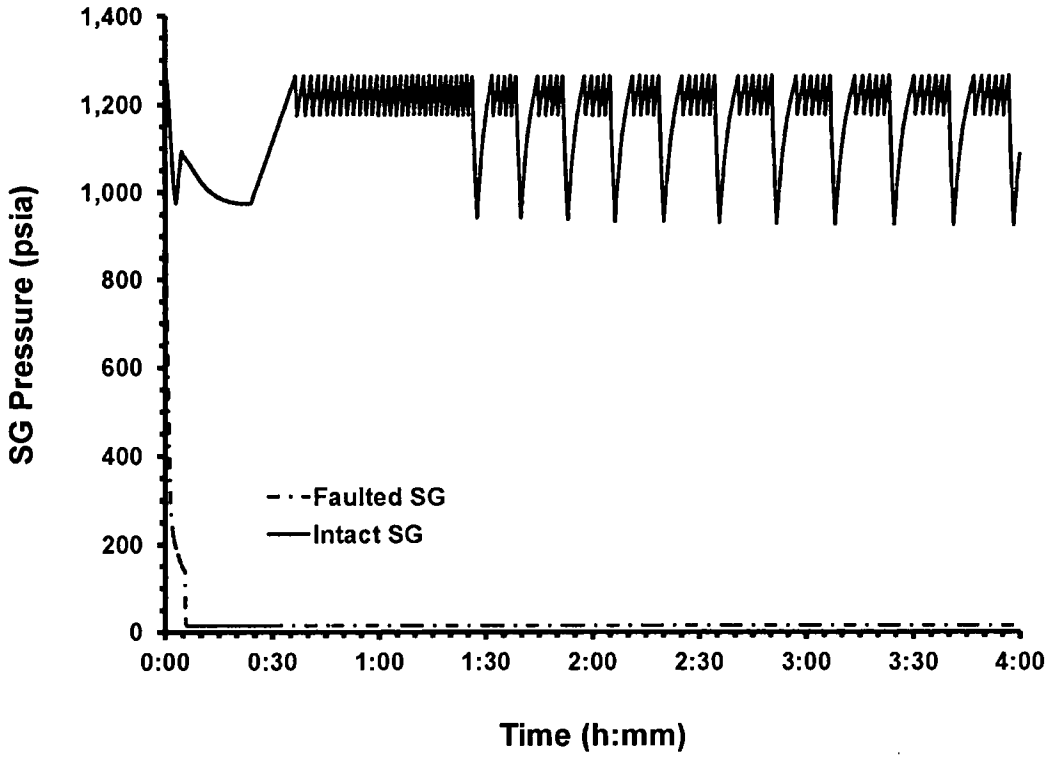

Figure 2-7. MSLB Demonstration Analysis – Long-Term Steam Generator Pressure vs. Time

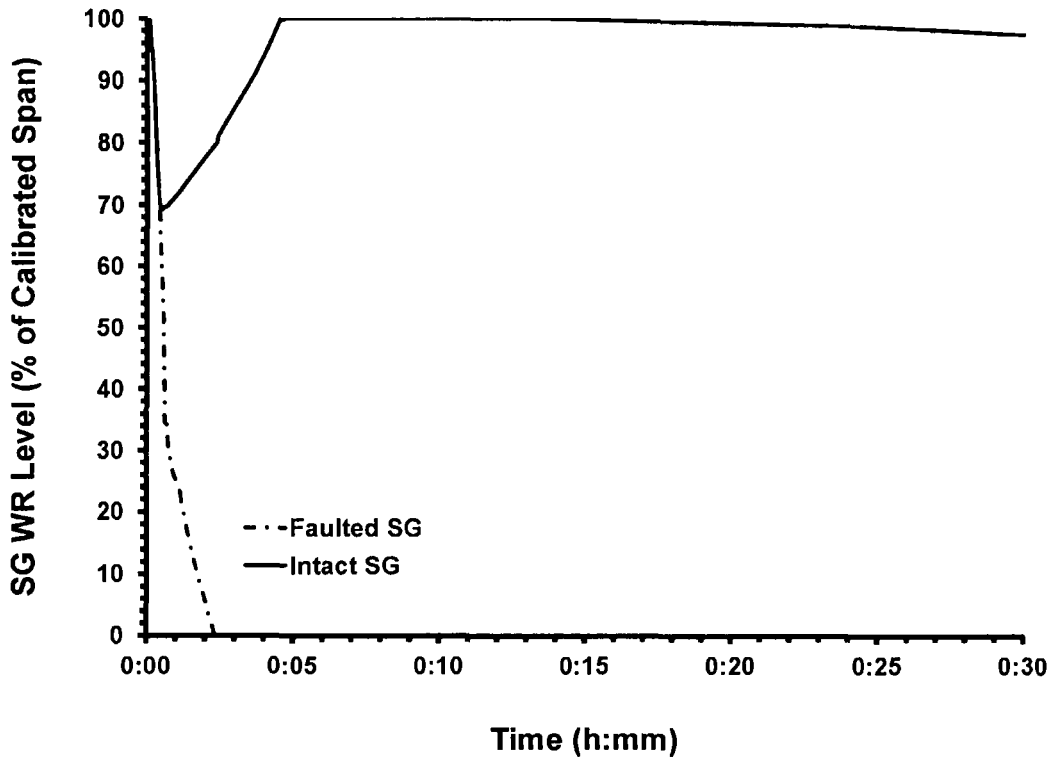

Figure 2-8. MSLB Demonstration Analysis –
Short-Term Wide Range Steam Generator Level vs. Time

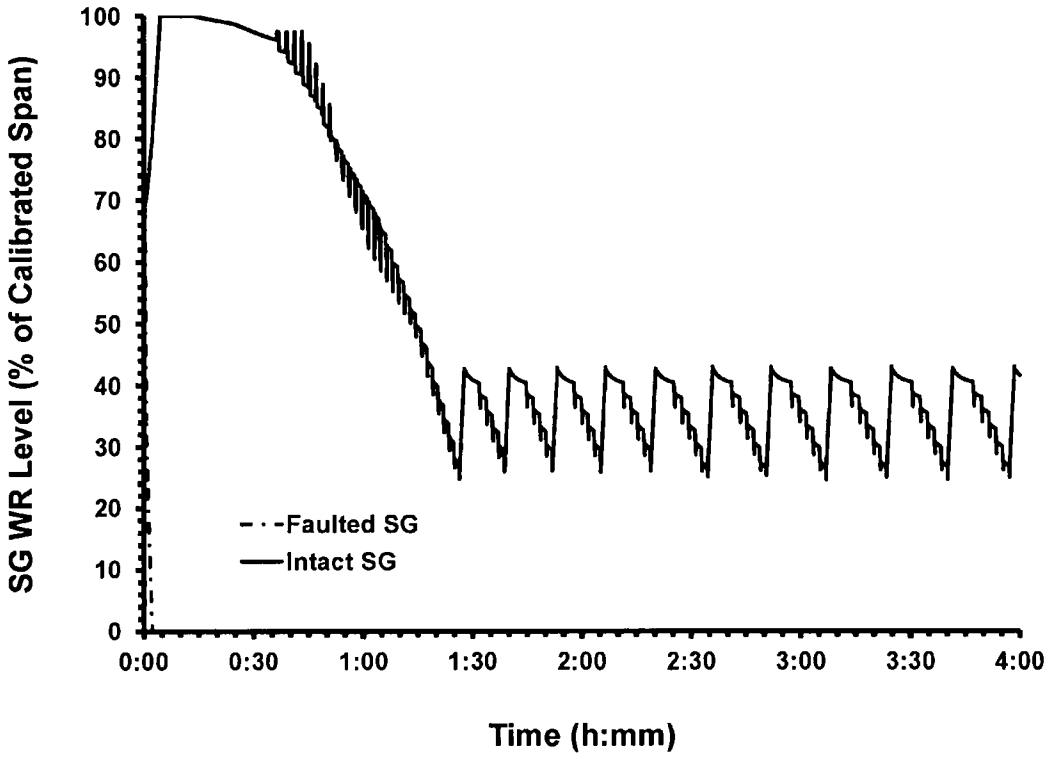

Figure 2-9. MSLB Demonstration Analysis – Long-Term Wide Range Steam Generator Level vs. Time

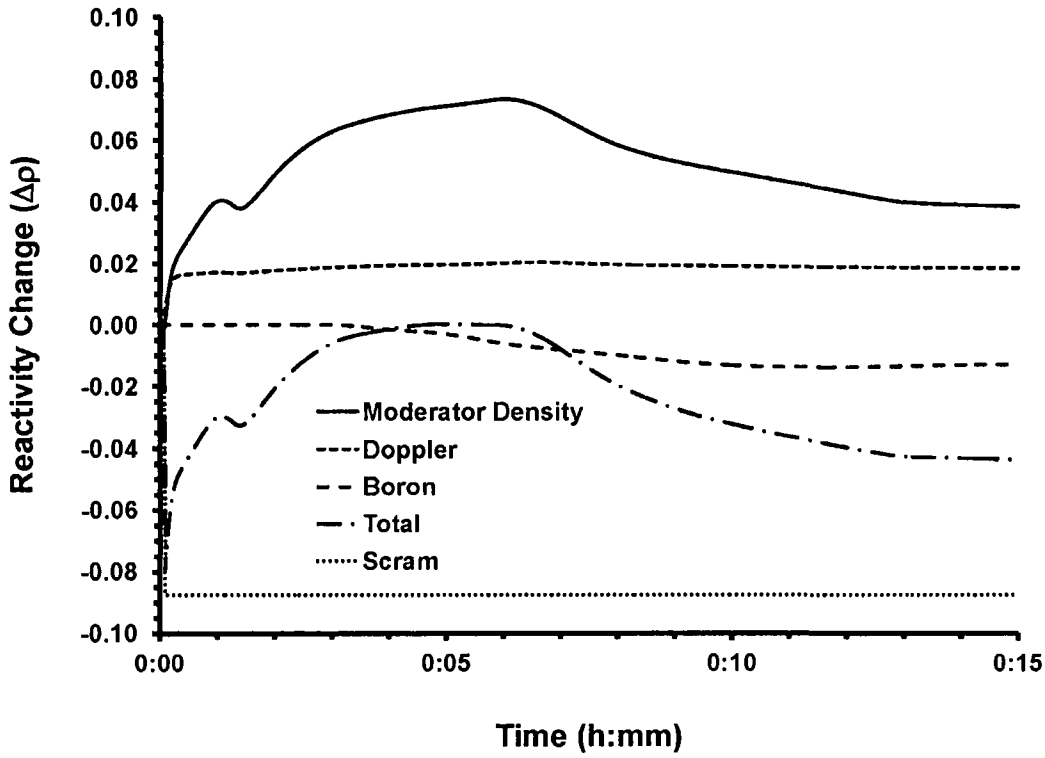

Figure 2-10. MSLB Demonstration Analysis – Short-Term Reactivity Change vs. Time

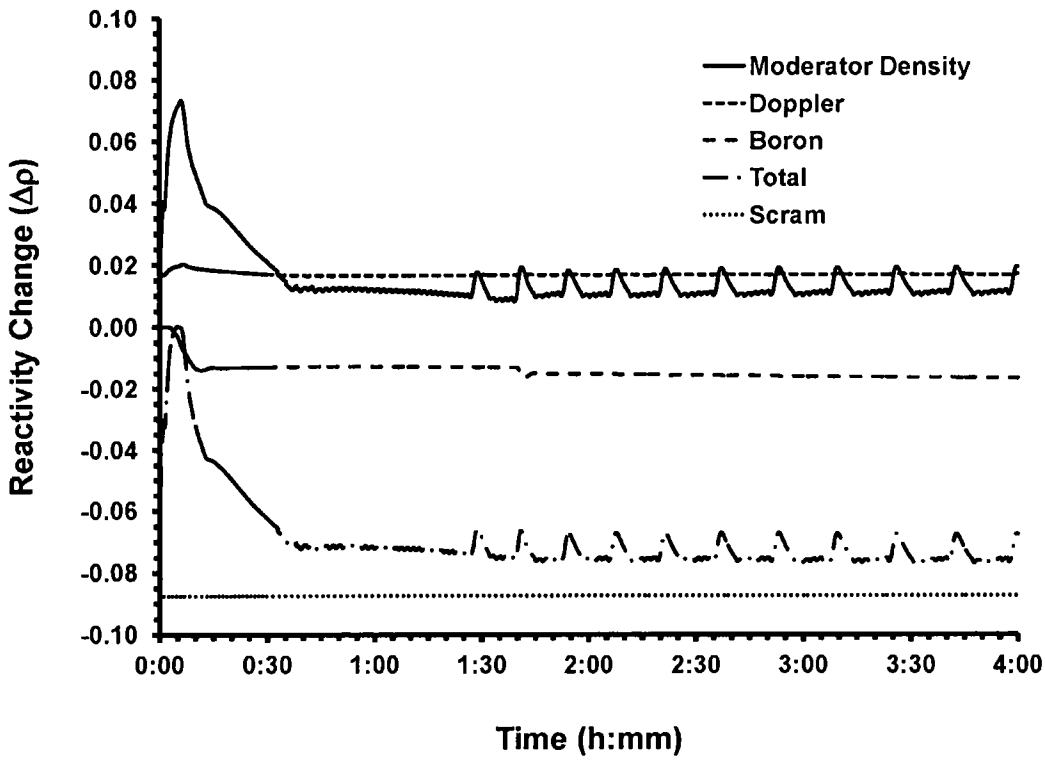

Figure 2-11. MSLB Demonstration Analysis – Long-Term Reactivity Change vs. Time

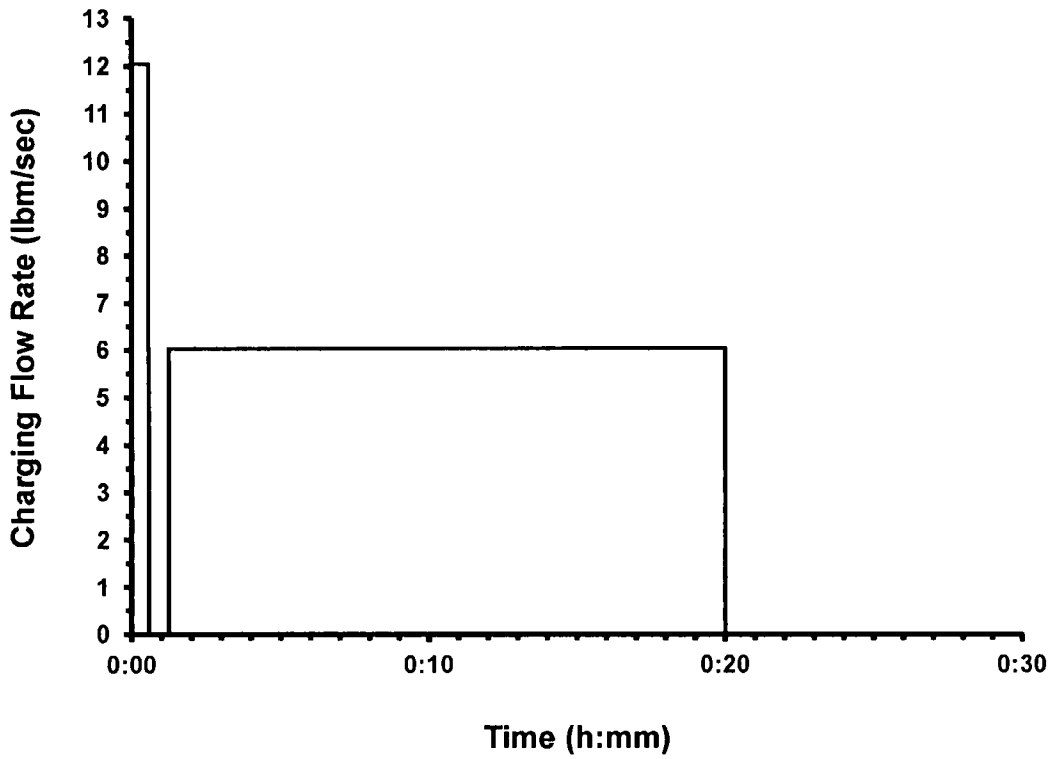

Figure 3-1. FWLB Demonstration Analysis –
Short-Term Instantaneous Charging Flow Rate vs. Time

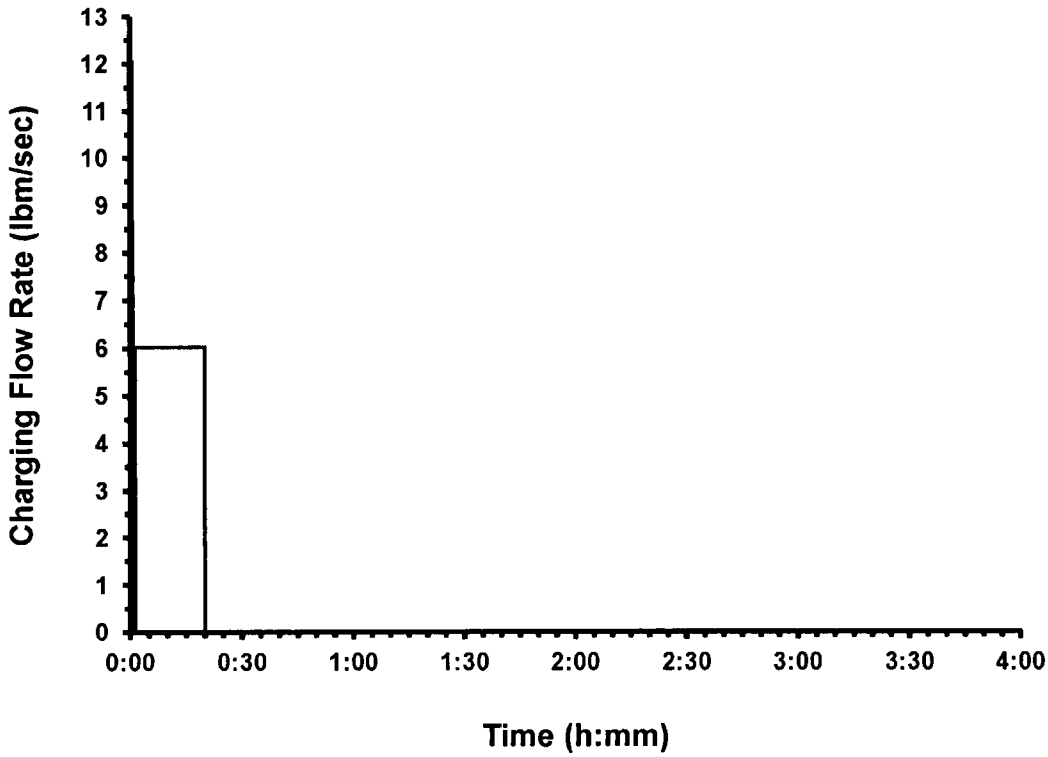

Figure 3-2. FWLB Demonstration Analysis –
Long-Term Instantaneous Charging Flow Rate vs. Time

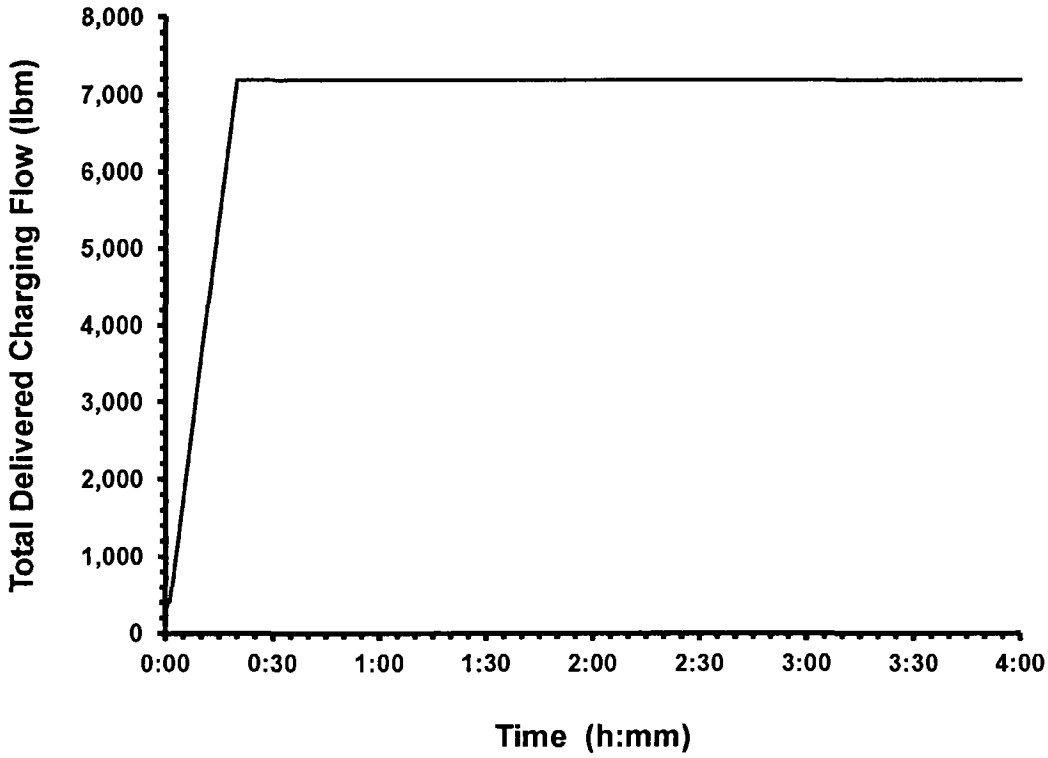

Figure 3-3. FWLB Demonstration Analysis – Total Delivered Charging Flow vs. Time

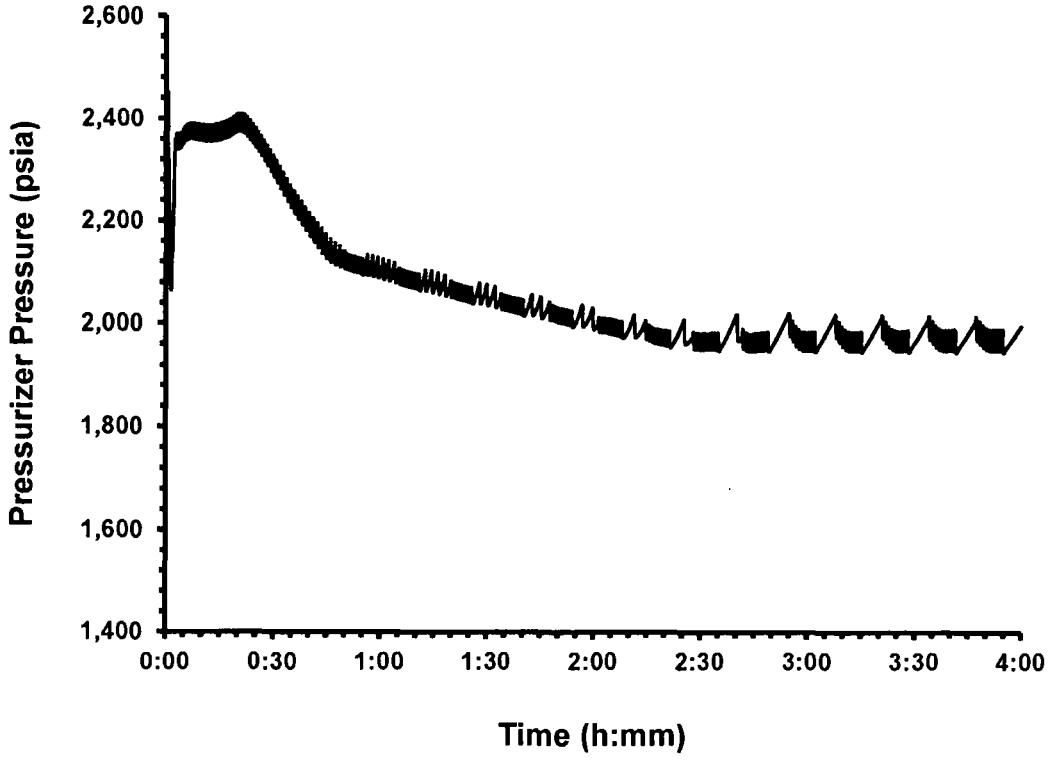

Figure 3-4. FWLB Demonstration Analysis – Pressurizer Pressure vs. Time

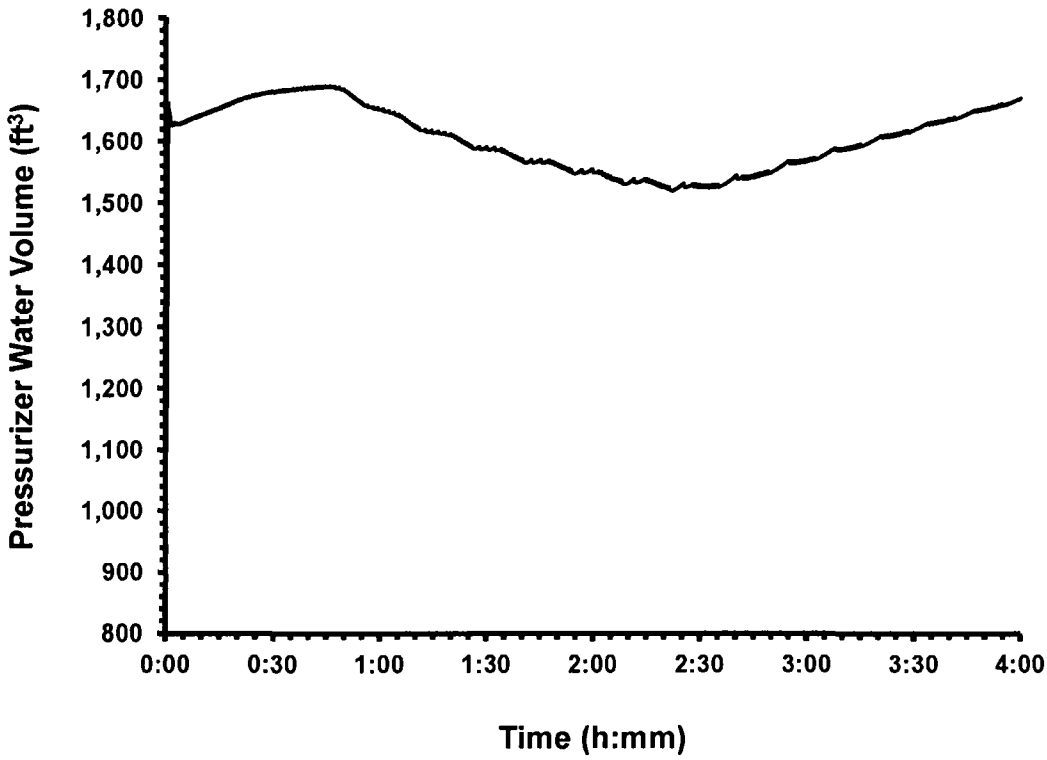

Figure 3-5. FWLB Demonstration Analysis – Pressurizer Water Volume vs. Time

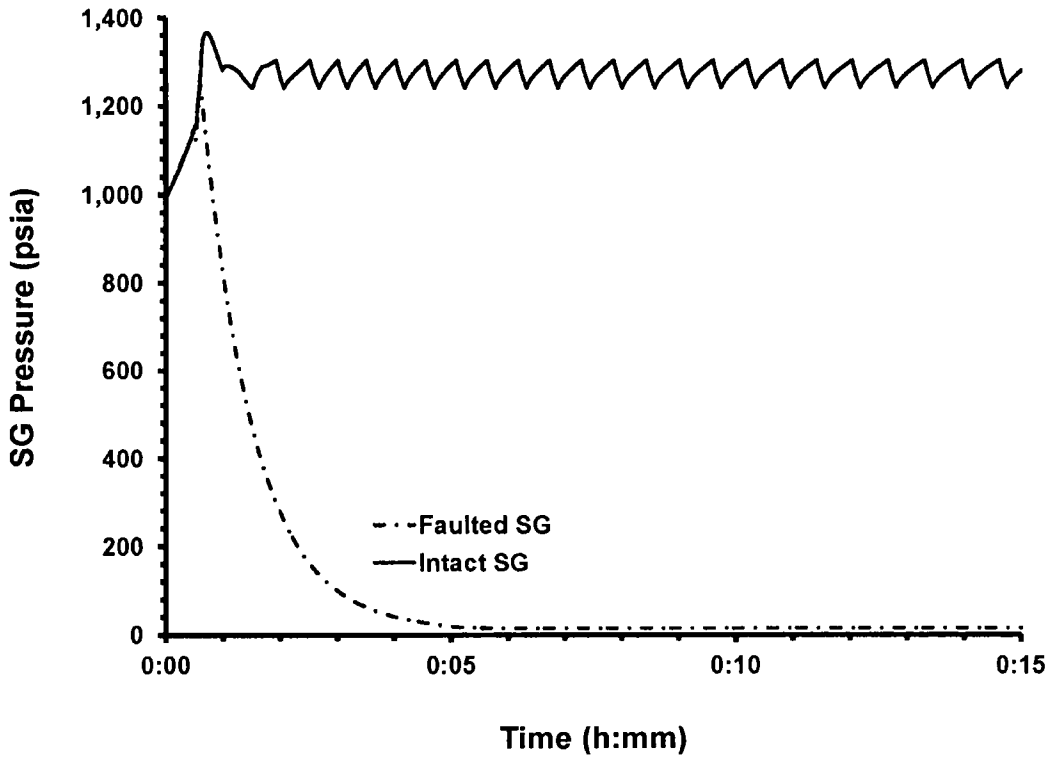

Figure 3-6. FWLB Demonstration Analysis – Short-Term Steam Generator Pressure vs. Time

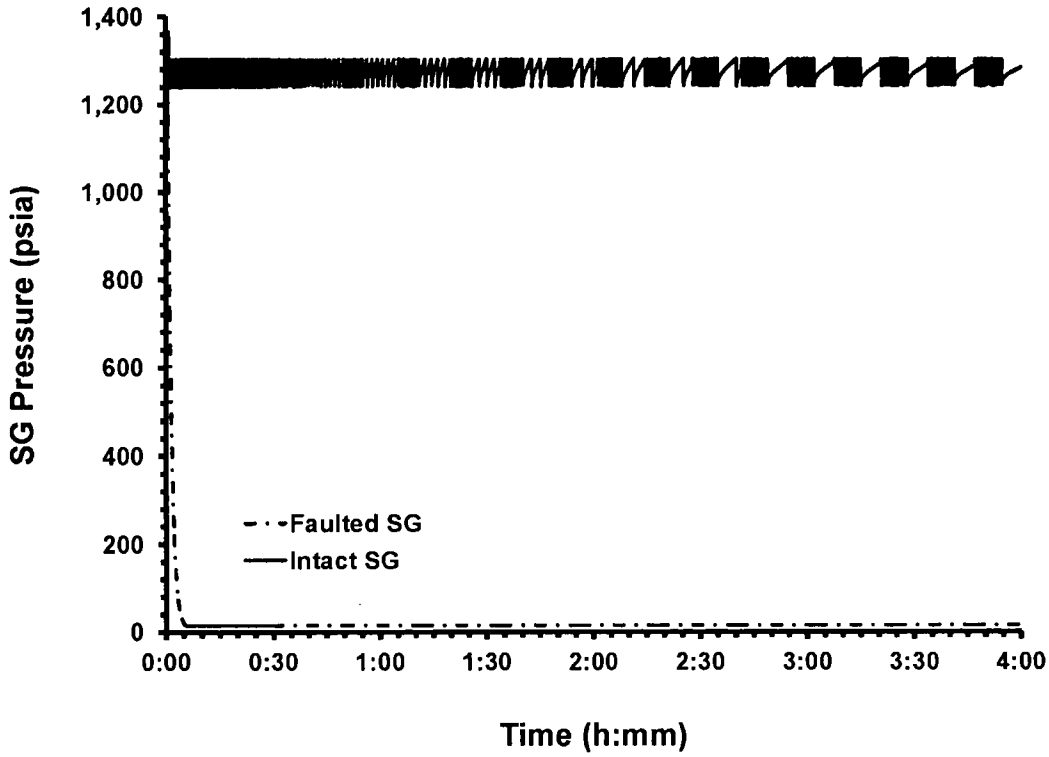

Figure 3-7. FWLB Demonstration Analysis – Long-Term Steam Generator Pressure vs. Time

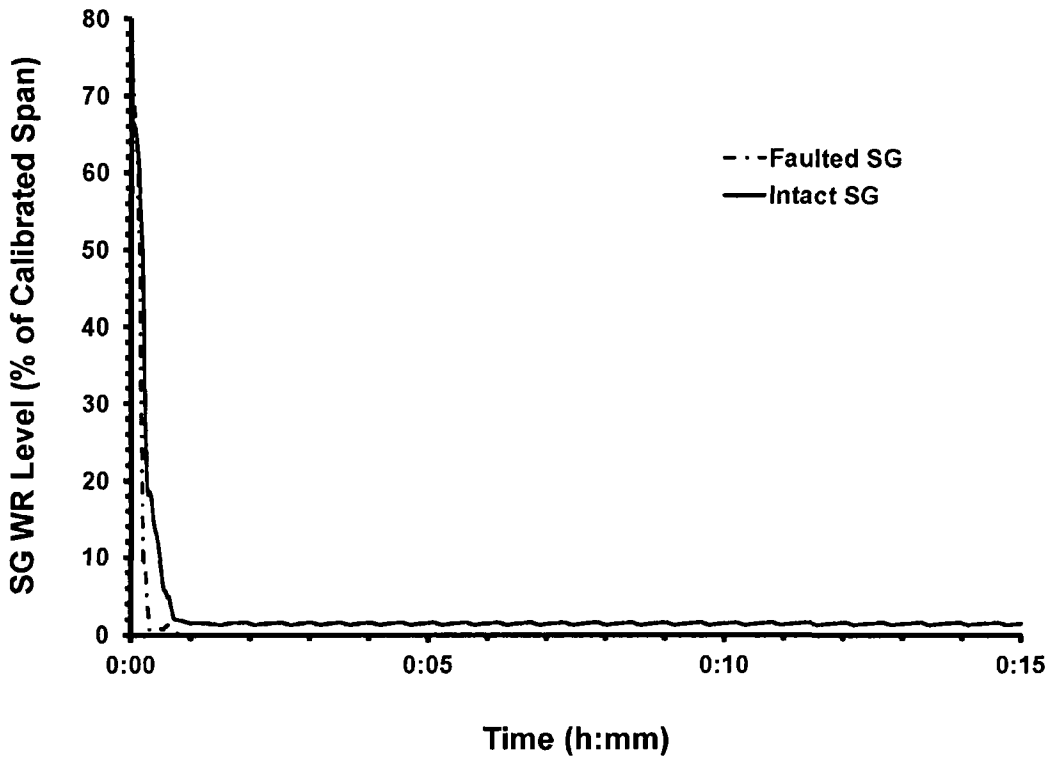

Figure 3-8. FWLB Demonstration Analysis –
Short-Term Wide Range Steam Generator Level vs. Time

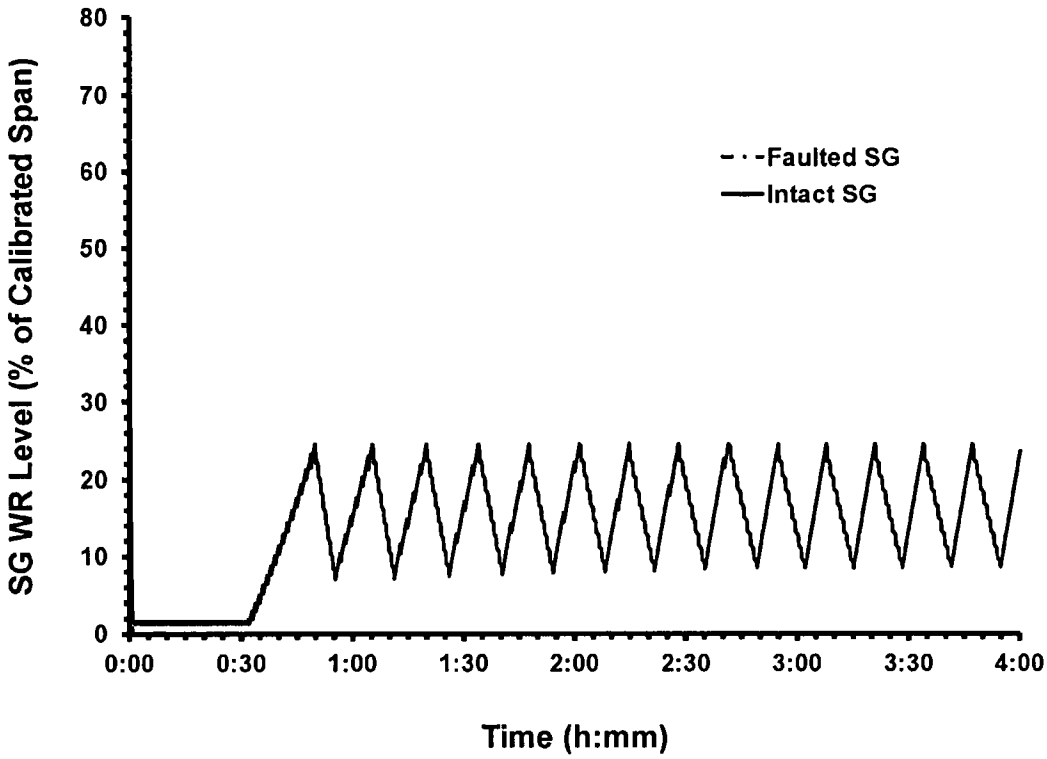

Figure 3-9. FWLB Demonstration Analysis –
Long-Term Wide Range Steam Generator Water Level vs. Time

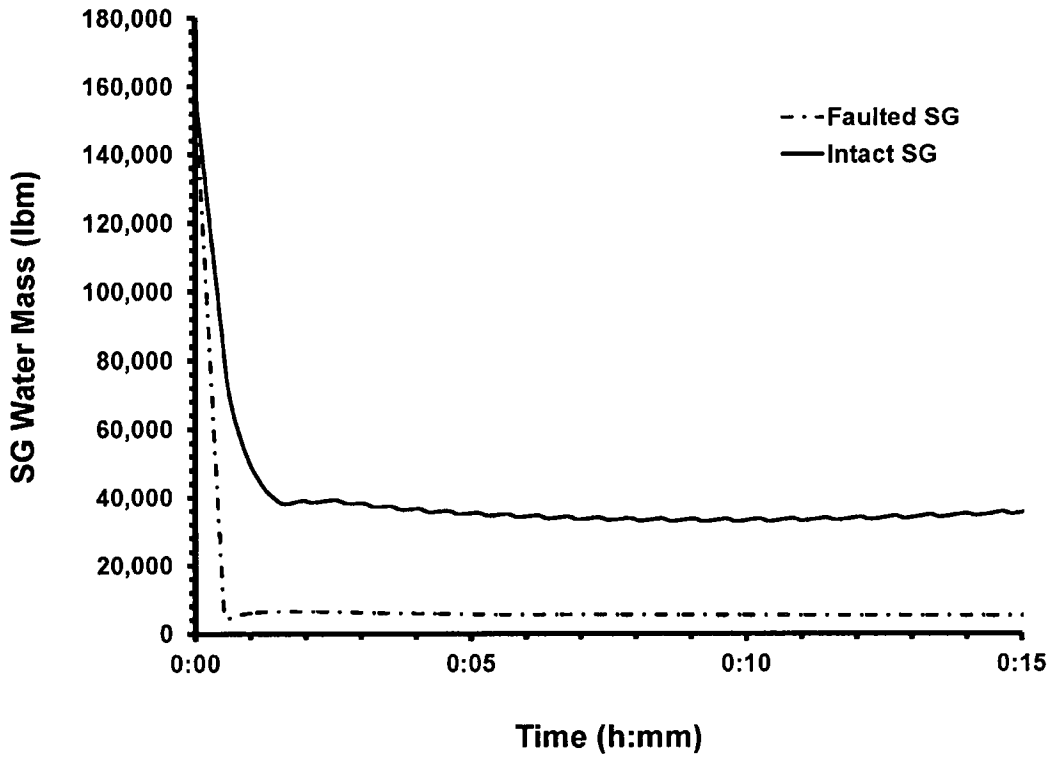

Figure 3-10. FWLB Demonstration Analysis – Short-Term Steam Generator Water Mass vs. Time

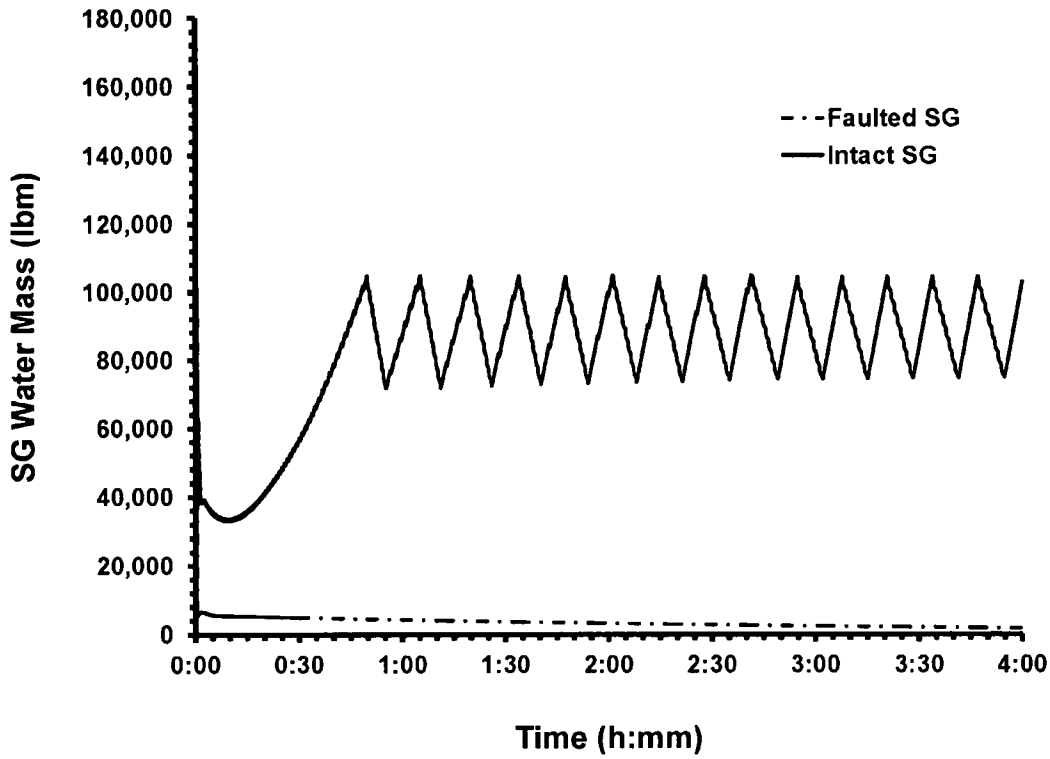

Figure 3-11. FWLB Demonstration Analysis – Long-Term Steam Generator Water Mass vs. Time