APS

10 CFR 50.90

Palo Verde Nuclear
Generating Station

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Washington, DC 20555-0001

References:
1. Letter No. 102-05116-CDM/TNW/RAB, dated July 9, 2004, from C. D. Mauldin, APS, to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations in Units 1 and 3, and Associated Administrative Changes for Unit 2"
2. Technical Manual for the CENTS Code, CENPD-282-P-A, Revision 1, dated April 2004.
3. Letter No. 102-04641-CDM/RAB, dated December 21, 2001, from C. D. Mauldin, APS to U. S. Nuclear Regulatory Commission, "Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations" for Unit 2
4. Letter dated from B. M. Pham, USNRC, to G. R. Overbeck, "Palo Verde Nuclear Generating Station, Unit 2 (PVNGS-2) – Issuance of Amendment on Replacement of Steam Generators and Uprated Power Operations (TAC NO. MB3696)

Dear Sirs:

Subject: Palo Verde Nuclear Generating station (PVNGS) Units 1 and 3, Docket Nos. STN 50-528 and STN 50-530 Supplement to Request for a License Amendment to Support Replacement of Steam Generators and Uprated Power Operations in Units 1 and 3 and Associated Administrative Changes for Unit 2

This letter supplements and revises information provided in Reference 1, Attachment 4. The information provided in this submittal consists of two parts:

1. Changes in the results of the events that were described in Section 6.3 of Attachment 4 of Reference 1 as a result of an error discovered in CENTS code input (Enclosure 2).
2. Replacement information for Section 6.3.6.3.3 of Attachment 4 of Reference 1 to reflect the revised Steam Generator Tube Rupture with Loss of Offsite Power

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(SGTRLOP) analysis that corrected the error in CENTS code input (Item 1 above), and the CENTS output that is used as the criterion for steam generator fill (Enclosure 3).

In the process of verifying the PVNGS input basedeck for Reference 2, an error in the input value for the secondary (shell) side volume of the Replacement Steam Generators (RSGs) was discovered. This error necessitated a review of safety analyses prepared in support of PVNGS Units 1 and 3 Power Uprate Licensing Report (PURLR) (Reference 1) in addition to the impact review on the current operating units. During this review, a deficiency in the criterion based on a specific CENTS code output that is used for determining steam generator fill was also discovered. These two issues required a reanalysis of the Steam Generator Tube Rupture with Loss of Offsite Power (SGTRLOP) event.

The first issue was the discrepancy in secondary (shell) side volume of the RSG between the as-designed and as-built dimensions. The safety analyses that were prepared in support of Reference 3, and subsequently verified to be applicable to the request for licensing amendment for PUR of PVNGS Units 1 and 3 (Reference 1), utilized as-designed RSG dimensions. Following the as-designed configuration, some minor internals were modified in the RSGs resulting in approximately a 2% reduction in the secondary side volume of the steam generators. For all of the events with the exception of SGTRLOP event, the impact of this change on safety analyses was determined to be insignificant and not affecting the conclusions presented in References 1 and 3. However, APS concluded that, although the magnitude of this error does not affect the conclusions, it slightly changes the reported results for some accident analyses that are sensitive to the RSG volume, and is providing the updated information to the NRC for the safety evaluation of the amendment requested in Reference 1. The detailed discussion of the investigated events and the changes to results are presented in Enclosure 2 of this submittal. For the SGTRLOP event, however, the conclusions drawn in Section 6.3.6.3.3 of Attachment 4 to Reference 1 and Attachment 6 to Reference 3 were found to be invalidated by the RSG volume error, requiring a revision to the analysis.

The second issue was discovered during the review and re-analysis of the SGTRLOP event for the error described in the previous paragraph. This discrepancy involved a specific CENTS output that is tracked for determining the remaining steam space in steam generators. APS discovered that the CENTS output which indicates the remaining steam space in the generator included a portion of main steam lines, namely the section from the steam generator nozzles to the main steam isolation valves (MSIV). Thus, when the output indicated that there was still a steam space left in the steam generators, the steam generators may already have been filled, and some liquid inventory may have spilled into the main steam lines. As a result of the evaluation of this discovery, the SGTRLOP event was found to result in steam generators being overfilled, independent of the RSG volume error described earlier, contrary to the conclusions drawn in Section 6.3.6.3.3 of Attachment 4 to Reference 1 and Attachment 6 to Reference 3. Therefore, the SGTRLOP event analysis was revised to correct both errors. A detailed description of the revised SGTRLOP event analysis and the information to replace Section 6.3.6.3.3 of Reference 1 are provided in Enclosure 3.

No commitments are being made to the NRC by this letter

Should you have any questions, please contact Mr. Thomas N. Weber at (623) 393-5764.

Sincerely

David Mauldin

CDM/TNW/RAB

Enclosures:

1. Notarized Affidavit
2. Evaluation of the Error Identified in the RSG Shell Side Volume Calculations and the Results of the Review of Safety Analyses Performed in Support of PVNGS Power Uprate
3. Reasons for, and the Results of, the Revised Postulated Steam Generator Tube Rupture with Loss of Offsite Power (SGTRLOP) Event Analysis

Attachment: Revisions to Reference 1, Attachment 4, Section 6.3.6.3.3

cc:	B. S. Mallet	NRC Region IV Regional Administrator
	M. B. Fields	NRC NRR Project Manager
	G. G. Warnick	NRC Senior Resident Inspector for PVNGS
	A. V. Godwin	Arizona Radiation Regulatory Agency (ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 3rd Day Of June, 2005.

Susie Lynn Ergish

Notary Public

SUSIE LYNN ERGISH
Notary Public - Arizona
Maricopa County
My Comm. Expires Jul 14, 2007

Notary Commission Stamp

ENCLOSURE 2

Evaluation of the Error Identified in the RSG Shell Side Volume Calculations and the Results of the Review of Safety Analyses Performed in Support of PVNGS Power Uprate.

This enclosure describes the evaluation of the error identified in RSG shell side volume calculations and the results of the review of safety analyses performed in support of PVNGS Power Uprate.

1.0 Introduction:

The safety analyses prepared in support of PVNGS Unit 2 Power Uprate Licensing Report (PURLR), Attachment 6 to Reference 3, and subsequently verified to be applicable to the request for a license amendment in Reference 1, utilized as-designed RSG dimensions. During fabrication, the as-designed configuration for some internal components was modified in the Unit 2 RSGs, which are essentially identical to the RSGs for Units 1 and 3. These changes resulted in approximately 2% reduction (9808 ft^3 vs. 10021 ft^3) in the secondary (shell) side volume of the RSGs. In general, the volume reduction occurring in steam region of the steam generators results in compression of steam more quickly, thus affecting the rate of pressurization and level changes in the secondary system. On the other hand, the volume reduction in the liquid region results in less liquid inventory available for primary-to-secondary heat transfer, and thus causes a change in the pressure response for both primary and secondary systems. However, the impact of a 2% or less reduction on shell side volume is insignificant considering the volume calculations are performed under "cold" conditions, and the thermal expansion of steam generators under NOP/NOT conditions would totally or partially compensate the error. Nevertheless, the reported results for the events that are sensitive to the initial steam generator inventory would be slightly affected if it is conservatively assumed that thermal expansion does not take place. Therefore, the accident analyses were reviewed to determine the changes on reported values in Attachment 4, Section 6.3 of Reference 1 and Attachment 6, Section 6.3 of Reference 3.

2. 0 Review and Results:

The review of the Chapter 15 Safety Analyses presented in Section 6.3 of Attachment 4 of Reference 1 and Attachment 6 of Reference 3 can be categorized into two groups based on their sensitivity to the initial and transient steam generator inventory. For the events that are not sensitive to the initial steam generator secondary side volume, the volume reduction does not affect the results. For the events that are sensitive to the initial steam generator inventory, a reduction in the steam generator volume may result in either adverse or benign consequences. For the Increased Heat Removal by the Secondary System events that are sensitive to the initial steam generator inventory, such as Main Steam Line Break events, the reduction in steam generator volume results in benign consequences since the reduced secondary system inventory causes less cooldown in the RCS. Thus, the reported values for those events are bounding. The Decreased Heat Removal by the Secondary System events, and the CEA Ejection event that is analyzed for the RCS Peak Pressure, are also sensitive to the initial steam generator volume in both liquid and steam region for the reasons given in the previous section. Therefore, the events were evaluated with the corrected steam generator volumes to determine the dominant effect. Table 1 presents the results for the limiting Loss of Condenser Vacuum (LOCV) and Small Feedwater Line Break (SFWLB) events. The CEA Ejection event that was presented in References 1 and 3 assumed a lower initial inventory in the steam generators than that allowed by the plant protection system, thus the results that were provided previously are bounding.

TABLE 1.
RSG Differential Volume Evaluation Results

Description	Original Results		Corrected Results		Acceptance Criterion (psia)
	Peak Pressure (psia)	Time of Peak Pressure (sec)	Peak Pressure (psia)	Time of Peak Pressure (sec)	
LOCV RCS Pressure	2739	9.61	2740.5	9.63	2750
LOCV SG Pressure	1389	14.2	1388.5	13.9	1397
SFWLB RCS Pressure	2706	21.48	2707.4	21.03	2750

The FWLB, LOCV, and CEA Ejection events also establish the basis for Technical Specification 3.7.1, Main Steam Safety Valves, which limits the power level based on operable MSSVs. The evaluation performed for maximum allowable power with one or more MSSV inoperable showed that allowable power levels listed in TS Table 3.7.1-1 are still valid.

Steam Generator Tube Rupture (SGTR) events were also reviewed because of the impact of the change in the initial steam generator level for both dose and SG overfill consequences. The SGTR with LOP and a single failure of an Atmospheric Dump Valve (ADV) sticking open (SGTRLOPSF) event is the limiting SGTR event with respect to dose consequences. However, because of the reviewed and approved operator actions, such as maintaining level in the affected steam generator, the SGTRLOPSF event results are not adversely impacted. Impact on the limiting SGTR event for the steam generator overfill (SGTRLOP) is determined to be adverse since the magnitude of this error would have caused enough reduction in steam space volume in the steam generator dome to invalidate the safety analysis. In addition, during the evaluation of the SGTRLOP event, it was discovered that a deficiency in the specific CENTS code output that is used as the criterion for determining steam generator fill existed. Therefore, the SGTRLOP event was reanalyzed with respect to steam generator overfill, using the correct steam generator volume and corrected steam space indication. The new analysis is presented in Enclosure 3.

3.0 Conclusion:

The error discovered in the RSG shell side volume (2% reduction from the originally calculated design value) does not affect the conclusions drawn in PVNGS PUR submittal (Reference 1) with the exception of SGTRLOP event which is reanalyzed. The error results in minor adverse changes in the reported results for RCS peak pressure for the Increased Heat Removal by the Secondary System events, however, no event exceeds the acceptance criteria. Other events are either not sensitive to the error on steam generator secondary side volume, or the previously reported values bound the change due to the error.

ENCLOSURE 3

Reasons for, and the Results of, the Revised Postulated Steam Generator Tube Rupture with Loss of Offsite Power (SGTRLOP) Event Analysis

This enclosure describes the reasons for, and the results of, the revised postulated Steam Generator Tube Rupture with Loss of Offsite Power (SGTRLOP) event analysis.

1.0 Introduction

APS has analyzed the SGTRLOP event to address UFSAR Chapter 15, Accident Analyses licensing basis acceptance criterion for steam generator overfill to demonstrate that the liquid inventory of the steam generator does not spill into the main steam lines, thus preventing the failure of main steam lines with respect to the concerns described in Generic Letter 89-19. The SGTRLOP event was analyzed for PVNGS operation at 3990 MWt Rated Thermal Power with RSGs in support of the PVNGS Unit 2 PUR licensing amendment request (Reference 3). Subsequently, that analysis was evaluated and found to be applicable for Units 1 and 3 PUR with RSG as reported in Reference 1. The analysis demonstrated that the steam generators would not overfill during the SGTRLOP event.

During an internal review of the safety analyses described in Attachment 4 of Reference 1 to address the impact of the error in steam generator shell side volume (see Enclosure 2), APS engineers also discovered a deficiency in the criterion that is used for determining the steam generator overfill. This deficiency involved a specific CENTS output that is tracked for determining the remaining steam space in the steam generators. It was found that the CENTS output that lists the remaining steam space in the generator included the portion of the main steam lines from the steam generator nozzles to the main steam isolation valves (MSIVs). Thus, when the output indicated that there is still a steam space left in the steam generators, the steam generators would have already been filled, and some liquid would have spilled into the main steam lines. As a result of this discovery and the error in the RSG shell side volume, the SGTRLOP event was determined to result in steam generators being filled, invalidating conclusions drawn in References 1 and 3. Therefore, the SGTRLOP event analysis was revised.

The revised analysis verifies that the steam generators do not overfill, and prevention of the failure of main steam lines continues to be satisfied for operation at 3990 MWt with RSGs. The reanalysis of the SGTRLOP event is described in the following sections in detail.

2.0 Evaluation

The revised SGTRLOP analysis utilized current approved methodology. The analysis corrected the criterion for determining steam generator overfill. The steam generator shell side volume was also corrected with respect to the error described in Enclosure 2. In addition, several changes were made to the input parameters and operator actions in the revised SGTRLOP event analysis. These changes involve corrections to input parameters for identified errors, removal of discretionary conservatism from input parameters, incorporation of changes to the plant procedures and design documents.

Also, some operator action timings are changed as a result of the changes noted above, since the SGTR events credit approved operator actions whose timings are based on applicable criteria and symptoms as described in the Emergency Operating Procedure (EOP) guidelines.

The analytical changes were evaluated in accordance with 10 CFR 50.59, and were determined to not require NRC staff review and approval. The changes were determined to be not "adverse" as defined in NEI 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation. However, the transient simulation and input and assumptions of the event that were presented in Section 6.3.6.3.3 of Reference 1 (which noted that no change to the previously reviewed and approved analysis of Reference 3) are impacted, and necessitated this supplement to Reference 1. The following subsections describe the changes made to Section 6.3.6.3.3 of Reference 1 due to the revised analysis. Attachment 1 presents the replacement and added pages for Reference 1. Tables and Figures are numbered to correspond to those provide in Reference 3.

2.1 RCS Cooldown Rate

The revised analysis assumes a faster cooldown rate than the original analysis. A faster cooldown rate during the earlier phase of the mitigation is the key contributor to successful and timely primary-to-secondary leak isolation. Fundamentally, higher cooldown rates result in higher Subcooling Margin (SCM) which allows for a greater primary pressure reduction and hence zeroing out of the leak by faster equalization of the primary and secondary system pressures. The following differences in different phases of the simulation are noted:

- Start of the cooldown to the isolation of the affected SG – The revised analysis assumed a 99°F/hr cooldown rate vs. 80°F/hr assumed in the original analysis.
- SG isolation to the shutdown cooling (SDC) entry conditions – The revised analysis simulated a cooldown rate of 59.9°F/hr during the first hour, 74°F/hr in the next hour and the overall average rate of 31.6°F/hr for entire phase. For the similar periods, the original analysis simulated cooldown rates of ~ 41°F/hr during the first hour, 48°F/hr, for the next hour, and the overall average rate of 30°F/hr.

The cooldown rates utilized in the revised analysis are consistent with Technical Specification 3.4.3, RCS Pressure and Temperature (P/T) Limits, and credible operator actions based on the instructions provided in SGTR EOP and SGTR EOP technical guidelines.

2.2 Hot Leg Temperature Criterion for Steam Generator Isolation

The revised analysis changed the hot leg temperature criterion for SG isolation to 530°F from 515°F that was used in the original analysis. The original analysis selected a conservative temperature compared to the temperature specified in the EOP. This change in the revised analysis was made to simulate conditions which may aggravate

SG overfill while maintaining sufficient conservatisms with respect to EOP value of 540°F.

2.3 Instrumentation Uncertainty for Subcooling Margin

The revised analysis modified the timing of change in the containment conditions from normal to harsh affecting the instrumentation uncertainty on temperature measurement that is used by the operators to maintain the subcooling margin (SCM). The original analysis conservatively invoked a harsh containment condition at 50 minutes into the event which resulted in large instrumentation uncertainty for the temperature used for SCM criterion very early into the event. In essence, this resulted in inhibiting the necessary primary pressure reduction required to equalize primary and secondary pressures and thereby isolate the primary-to-secondary leak. The revised analysis used normal instrumentation uncertainty for the first 5 hours into the event changing to the harsh conditions afterwards. This was based on the justification that, for the same magnitude of steaming conditions, containment conditions would not become harsh before 5 hours into the event. By using normal containment conditions and associated instrumentation uncertainty for temperature used for SCM criterion for that duration, the primary-to-secondary leak was isolated much earlier in the revised analysis than the original analysis. This, in turn, resulted in less secondary system inventory thus in benign consequences with respect to SG overfill.

2.4 Initial Core Inlet Temperature

Change in initial core inlet temperature from 568°F to 566°F reflects the maximum allowed temperature (plus uncertainties) by the Technical Specification 3.4.1 as approved for PVNGS Unit 2 PUR Amendment #149 (Reference 4). The original analysis was prepared in anticipation of a higher allowable value which was later decreased in the final submittal of the Technical Specification amendment request leading to Amendment #149. The revised analysis changed the initial core inlet temperature simply to match the proposed T.S. 3.4.1 for PVNGS Units 1 and 3, and approved T.S. 3.4.1 for PVNGS Unit 2.

3.0 Conclusion

The revised analysis verified that the steam generators do not overfill during a SGTRLOP event, and prevention of failure of main steam lines continues to be satisfied for operation at 3990 MWt with RSGs. Changes affecting Attachment 4, Section 6.3.6.3.3 of Reference 1 are provided in the Attachment to this Enclosure. Tables and Figures are numbered to correspond to those provide in Reference 3.

Attachment

Revisions to Attachment 4 of Letter No. 102-05116 from APS to USNRC, Dated July 9, 2004

Section 6.3.6.3.3 Steam Generator Tube Rupture with a Loss of Offsite Power

Section 6.3.6.3.3.1 Identification of Causes and Event Description

As described in UFSAR Section 15.6.3, this transient is similar to that described in the previous section for SGTRLOP single failure with the exception of an ADV remaining open. It assumes that the plant is challenged by a SGTR. The radioactivity from the leaking SG tube mixes with the shell-side water in the affected SG. Before turbine trip, the radioactivity is transported through the turbine to the condenser where the noncondensable radioactive materials would be released via the condenser air removal pumps. Following reactor/turbine trip, the MSSVs open to control the main steam system pressure. The operator can isolate the damaged SG any time after reactor trip occurs. As a result of the LOP which occurs due to the grid instability following the turbine trip, electrical power would be unavailable. The plant would experience a loss of the following:

- turbine load,
- normal FW flow,
- forced RCS flow, and
- condenser.

With the SBCS unavailable, NSSS cooldown is accomplished by use of AFW flow and ADVs. Heat removal must be accomplished by natural circulation, resulting in a higher core outlet temperature for much of the transient. The higher core outlet temperature as well as steaming to the atmosphere via use of ADV, contributes to higher offsite doses . In addition, the affected SG may start filling up due to primary-to-secondary leakage, and secondary liquid inventory may spill into the main steam lines challenging the integrity of the main steam lines. The SGTRLOPSF, that is presented in the previous section, bounds the dose consequences of the SGTRLOP, however the SGTRLOP is the most limiting SGTR event with respect to the SG overfill. Thus, the SGTRLOP event is analyzed in order to confirm that the SG does not experience an overfill condition. The most limiting SGTRLOP event is for a leak flow equivalent to a double-ended rupture of a U-tube at full power conditions.

Section 6.3.6.3.3.2 Acceptance Criteria

The acceptance criteria for SGTR events are defined in SRP Section 15.6.3. In addition, the SGTR events should not result in SG overfill.

Section 6.3.6.3.3.3 Description of Analysis

The NSSS response to a SGTRLOP event is simulated using the CENTS code.

The input parameters and initial conditions are biased to aggravate SG overfill conditions.

Section 6.3.6.3.3.3. Transient Simulation

The system is initialized at 102% power using the most limiting initial parameters. At time equal zero, the SGTR is simulated by a break at the bottom of the tube sheet at the hot leg side. This causes the pressurizer level and pressure to decrease, letdown flow to go to minimum and the third charging pump to start. Pressurizer level reaches the low pressure level heater cut-off which de- energizes all heaters thus accelerating the primary depressurization. The CPC reactor trip occurs on approach to hot leg saturation, with a turbine trip following within one second of the reactor trip signal. A LOP occurs due to grid instability three seconds after turbine trip.

After reactor trip, stored and fission product decay heat energy must be removed by the RCS and main steam systems. In the absence of forced RCS flow, convective heat transfer out of the reactor core is supported by natural circulation. Initially, the water inventory in the SGs is used to cool down the RCS with the resultant steam released to atmosphere via the MSSVs and ADVs.

The EOPs contain instructions to help the operator manage the cooldown following a SGTR event. Accordingly, the required operator actions to mitigate the effects of the SGTR event, and bring the plant to SCS entry conditions have been simulated based on the EOP guidance and operator feedback. The timing of the operator actions in the model are based on ANS/ANSI-N58.8-1984 which specifies response times for safety related operator actions. The input parameters and initial conditions are biased to aggravate SG overfill conditions.

The major post-trip EOP analysis assumptions regarding operator actions are the following:

1. Preclude challenge to MSSVs.

The analysis assumes operator action to open the ADVs (on both SGs) to preclude a direct challenge to the MSSVs two minutes after the reactor trip. The ADVs are used due to the unavailability of the SBCS due to LOP.

2. Diagnose the event and stabilize the plant.

EOP procedures are oriented towards quickly diagnosing the event and stabilizing the RCS at a temperature that precludes a challenge to the MSSVs. The analysis assumes this diagnosis and stabilization period will take about 21.5 minutes; that is consistent with ANSI/ANS standards for this category of event. Within this period, the operator is assumed to use the ADVs (on both SGs) and the AFW system to maintain the post trip T_{cold}.

3. Cooldown the RCS before isolation of affected SG.

After the 21.5 minute diagnosis and stabilization period, the operators are assumed to cool the RCS at approximately maximum Technical Specification cooldown rate of 100°F/hr. The cooldown continues via the ADVs on both SGs until the affected T_{hot} reaches the isolation temperature per requirements of the EOPs. A conservatively lower temperature is assumed in the analysis in order to delay isolation of the affected SG. Additionally, during this period, AFW would be delivered to each SG as needed in order to maintain the level in both SGs per the requirements in the EOPs.

4. Manual MSIS.

During the cooldown phase, the operator is assumed to initiate a manual MSIS per EOP guidelines due to LOP.

5. Isolate the affected SG.

The operator is assumed to isolate the affected SG after the affected loop temperature has reached the isolation temperature of 530°F. This isolation criterion is conservative with respect to the EOP guidelines of 540°F. During the cooldown phase, primary pressure is reduced with the aid of the pressurizer head vent which may eventually result in harsh containment conditions. A SCM including the applicable instrumentation uncertainty based on the containment condition is used.

6. Cooldown the RCS.

The analysis assumes that post isolation of the affected SG, cooldown to SCS entry is conducted via feeding and steaming the unaffected SG. The affected SG level begins to approach fill condition. Primary pressure is reduced by throttling HPSI as necessary and use of pressurizer head vents. After the leak is reduced, primary to secondary pressure differential is minimized to less than 50 psid to facilitate leak isolation and to ensure that the reverse leak is kept to a minimum.

The natural circulation cooling with the unaffected loop is maintained less than 32°F/hr until the entry conditions for SCS is reached at 8 hours.

7. Maintain adequate RCS inventory, HPSI throttle criteria.

Besides maintaining adequate subcooling, the EOPs require the operator to assure adequate RCS inventory, specifically, to retain minimum specified levels in the pressurizer and the upper head prior to throttling back the HPSI flow. Accordingly, the pressurizer level in the analysis is maintained above the level required by the EOPs.

<u>Section 6.3.6.3.3.4 Input Parameters, Initial Conditions, and Assumptions</u>

There are no changes to this section.

Table 6.3-51
Parameters Used for SGTRLOP Event for 3990 Mw_t

PARAMETER	Value
Initial Core Power (% of rated)	102
Initial Core Inlet Temp (°F)	566
Initial Pressurizer Pressure (psia)	2325
Initial RCS Flow (% of design)	95
Initial Pressurizer Water Level (ft)	21.85
Initial SG Water Level (ft)	25.7
MTC ($x10^{-4}$ $\Delta\rho$/°F)	-4.0
FTC	Least negative
Kinetics	minimum β
CEA Worth at Trip–WRSO (%$\Delta\rho$)	- 8.0
Hot spot gap conductance (Btu/hr-ft^2-°F)	518
Plugged SG Tubes	0
SGTR Break Location	at the tube sheet
Single Failure	none
LOP	yes

Section 6.3.6.3.3.5 Results

Table 6.3-52 presents a sequence of events for the simulation of the SGTRLOP event. The representative behaviors of NSSS parameters of significance are presented in the Figures provided in this Attachment.

Table 6.3-52
Sequence of Events for the SGTRLOP Event - 3990 Mwt

Time (sec)	Event	Value
0	SGTR occurs.	---
43	Letdown control valve reduced to the minimum value (gpm).	35
78	Backup pressurizer heaters energized (psia).	2275
346	Third charging pump turned on.	---
414.5	Pressurizer heaters de-energized on low level in the pressurizer (%).	25
792	Reactor trip reached on CPC hot leg saturation margin reached (°F).	8
793	Trip breakers open.	---
793.6	Scram CEAs begin falling.	---
794	MSSVs open (psia).	1227
796	LOP occurs.	---
801	SG water level reaches AFAS analytical setpoint in unaffected SG (%WR).	20
811	Pressurizer pressure reaches SIAS setpoint (psia).	1837
811	SIAS generated, safety injection flow initiated.	---
812	Pressurizer empties.	---
838	Voids begin to form in the upper head.	---
847	AFW initiated to unaffected SG (gpm).	650
	SG water level reaches AFAS analytical setpoint in affected SG (%WR).	20%
---	AFW initiated to affected SG (gpm).	---
859	MSSVs close (psia).	1104
896	Voids collapsed in the upper head.	---
912	Operator opens one ADV in each SG to prevent cycling of safeties.	---
921	Pressurizer begins to refill.	---
1032	Operator takes manual control of the AFW system and feeds each SG at the rate of 325 gpm and stabilizes the plant.	---
2081	Operator initiates plant cooldown at the rate of 100 °F/hr, by adjusting the ADVs and using one auxiliary feed water pump per SG (gpm).	650
2202	Operator opens pressurizer head vents.	---

Time (sec)	Event	Value
2324	Operator initiates a manual MSIS.	---
	Operator reduces ADV flow to slow cooldown rate.	---
2575	Operator throttles back HPSI flow to maintain RCS inventory control.	---
4252	Operator isolates the affected SG, at the analytical affected loop temperature (°F).	530
5570	Affected SG dome temperature exceeds affected loop T_{hot} temperature; eliminates leak flashing in the affected SG.	---
7488	Onset of reverse heat transfer in the affected loop: T_{cold} greater than the loop T_{hot}.	---
9442	Operator opens the first unaffected SG ADV full open.	
12689	Leak Isolated. Operator action maintains RCS pressure to affected SG ΔP minimum (psid).	50
14710	Operator increase AFW to 160 gpm in the unaffected SG (gpm).	160
25864	SDC entry conditions reached in the unaffected loop (psia/°F).	Less than 395/335
25864	Minimum steam space left in the affected SG (ft3) at SDC entry conditions	957
28800	Operator activates SDC system.	8 hrs
28800	Minimum steam space left in the affected SG (ft3)	426

Section 6.3.6.3.3.6 Conclusions

For the SGTRLOP event all acceptance criteria are met. Affected SG does not fill up during the event thus the integrity of the main steam lines are not challenged. Dose consequences at the EAB and LPZ boundaries remain bounded by that documented for the SGTRLOPSF event

Figure 6.3-224
SGTRLOP Event–Core Power vs. Time
(Sheet 3 of 3)

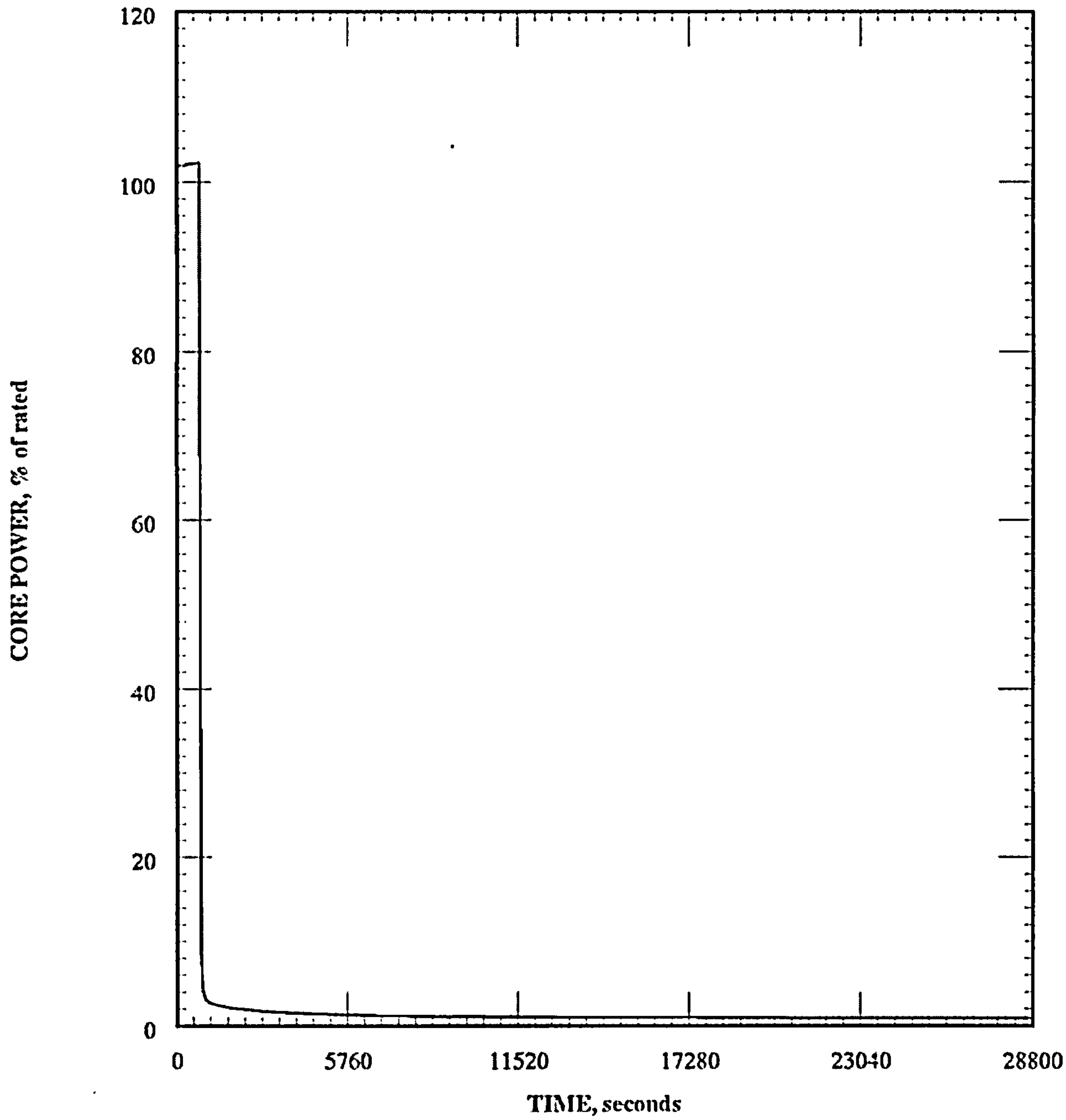


Figure 6.3-225
SGTRLOP Event–RCS Pressure vs. Time
(Sheet 3 of 3)

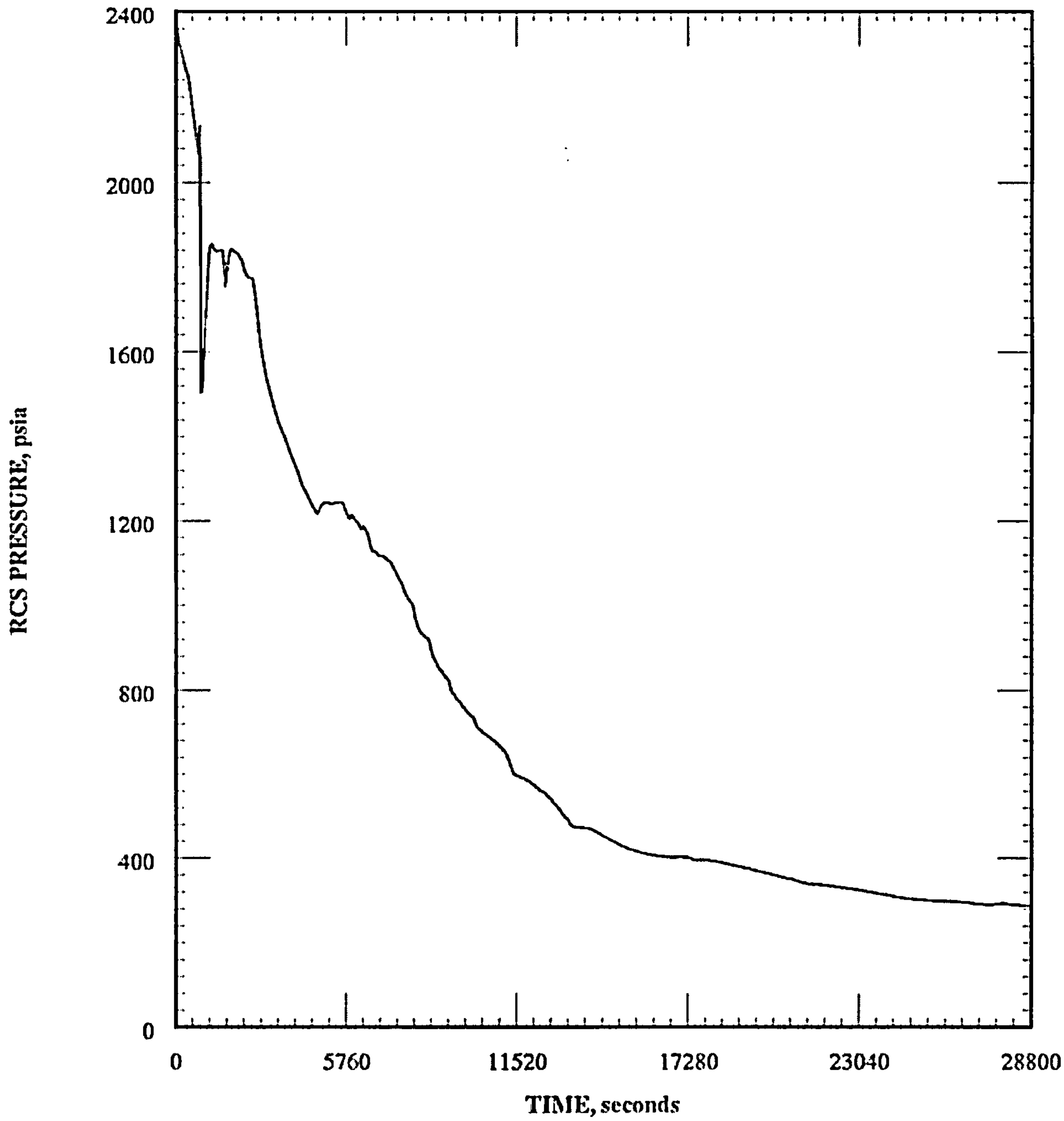


Figure 6.3-226
SGTRLOP Event–Affected Loop Coolant Temperatures vs. Time
(replaces Sheet 3 of 6)

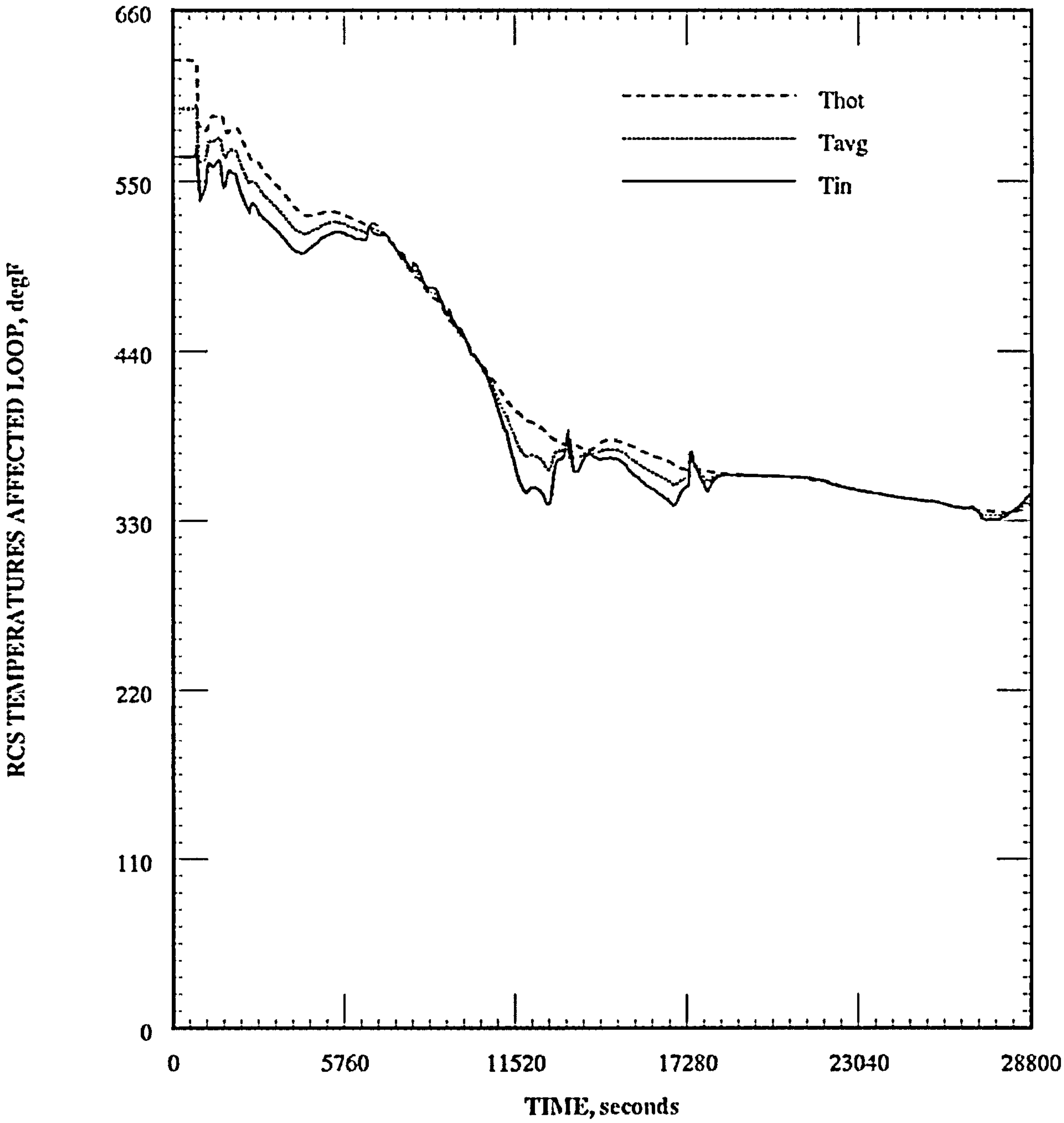


Figure 6.3-226
SGTRLOP Event–Unaffected Loop Coolant Temperatures vs. Time
(replaces Sheet 6 of 6)

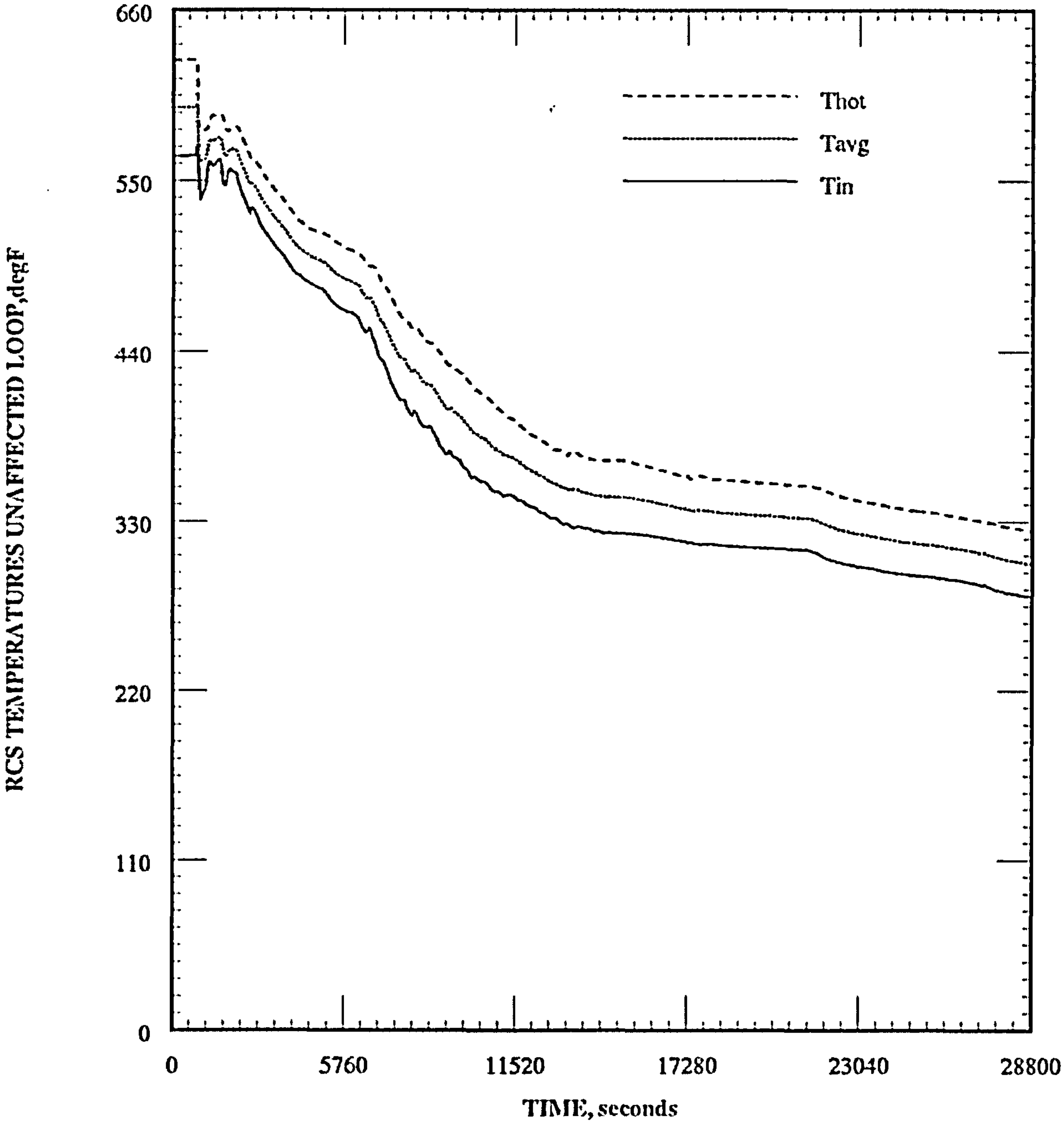


Figure 6.3-227

SGTRLOP Event–Pressurizer Liquid Volume vs. Time

(Sheet 3 of 3)

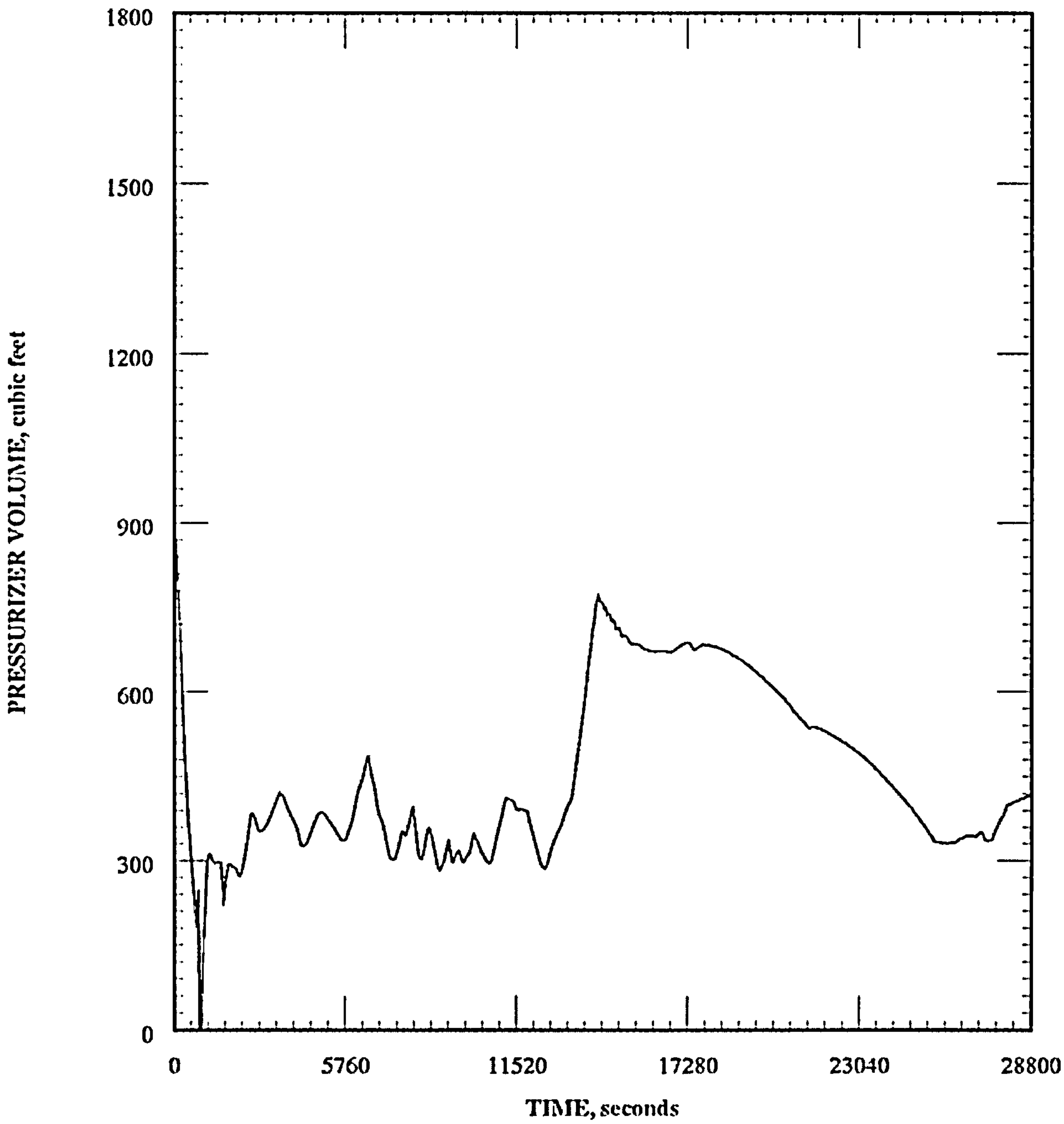


Figure 6.3-229
SGTRLOP Event–RCS Liquid Mass vs. Time
(Sheet 2 of 2)

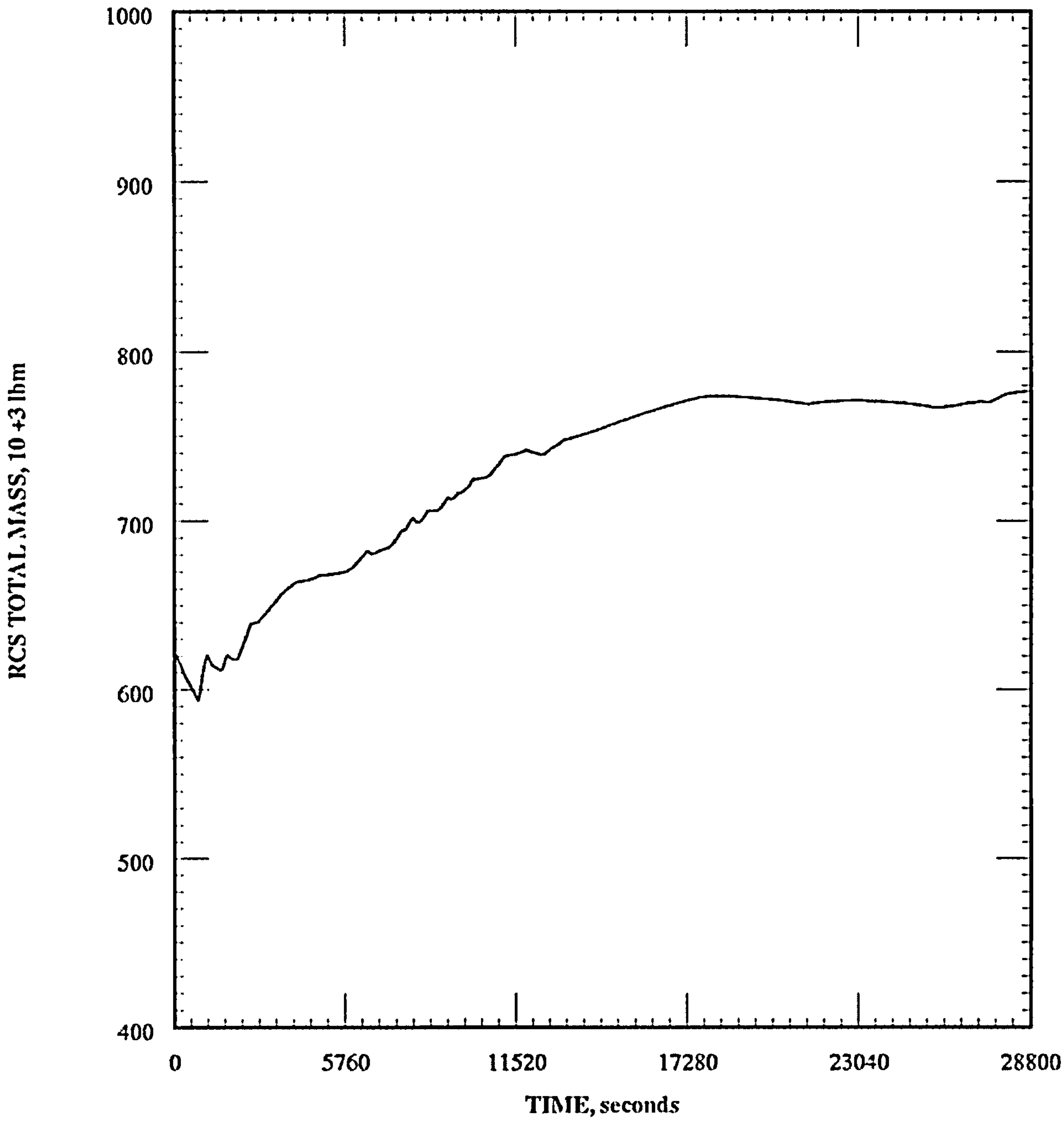


Figure 6.3-230
SGTRLOP Event–SG Pressure vs. Time
(Sheet 3 of 3)

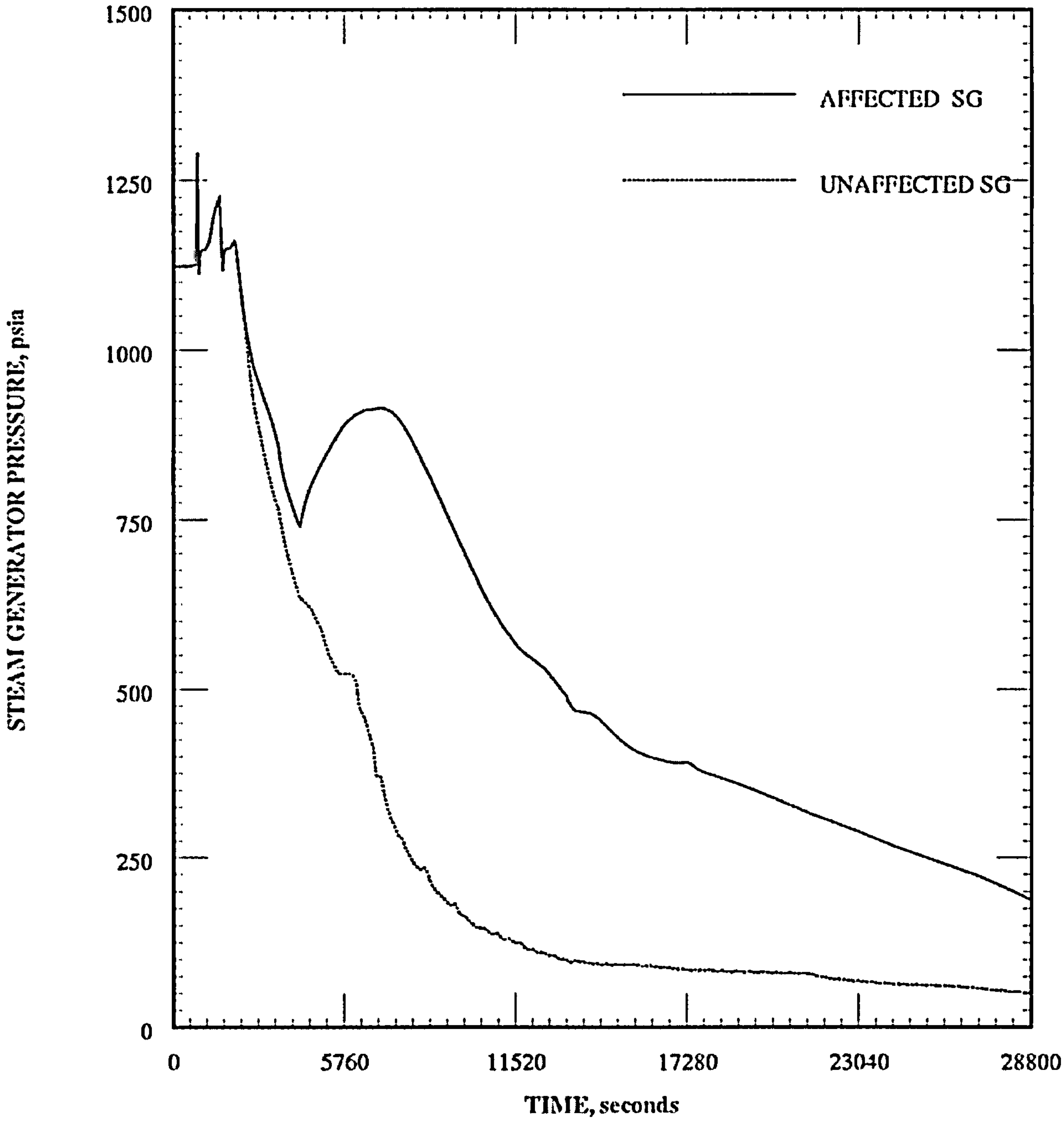


Figure 6.3-231
SGTRLOP Event–Tube Leak Rate vs. Time
(Sheet 3 of 3)

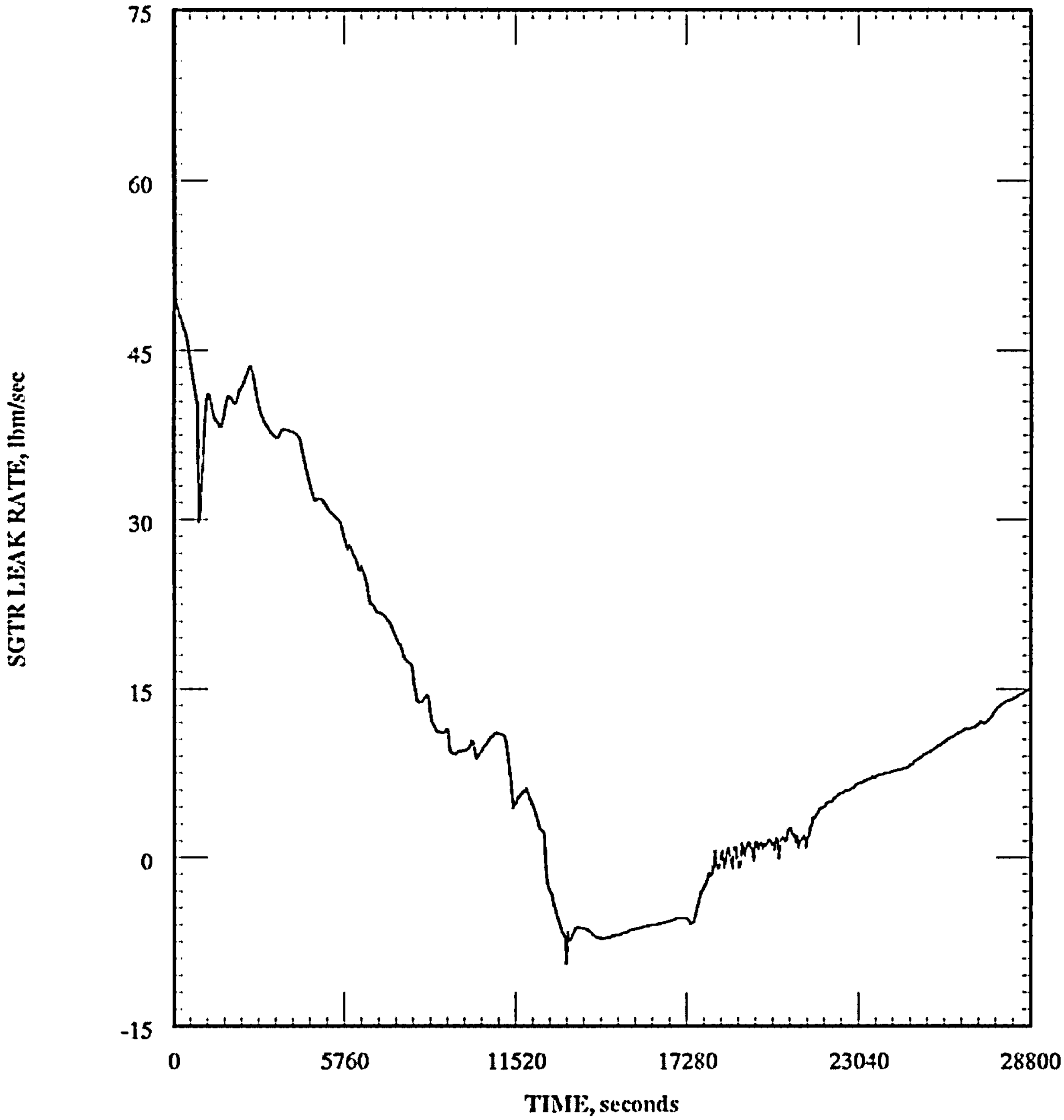


Figure 6.3-232
SGTRLOP Event–Integrated Tube Leak vs. Time
(Sheet 3 of 3)

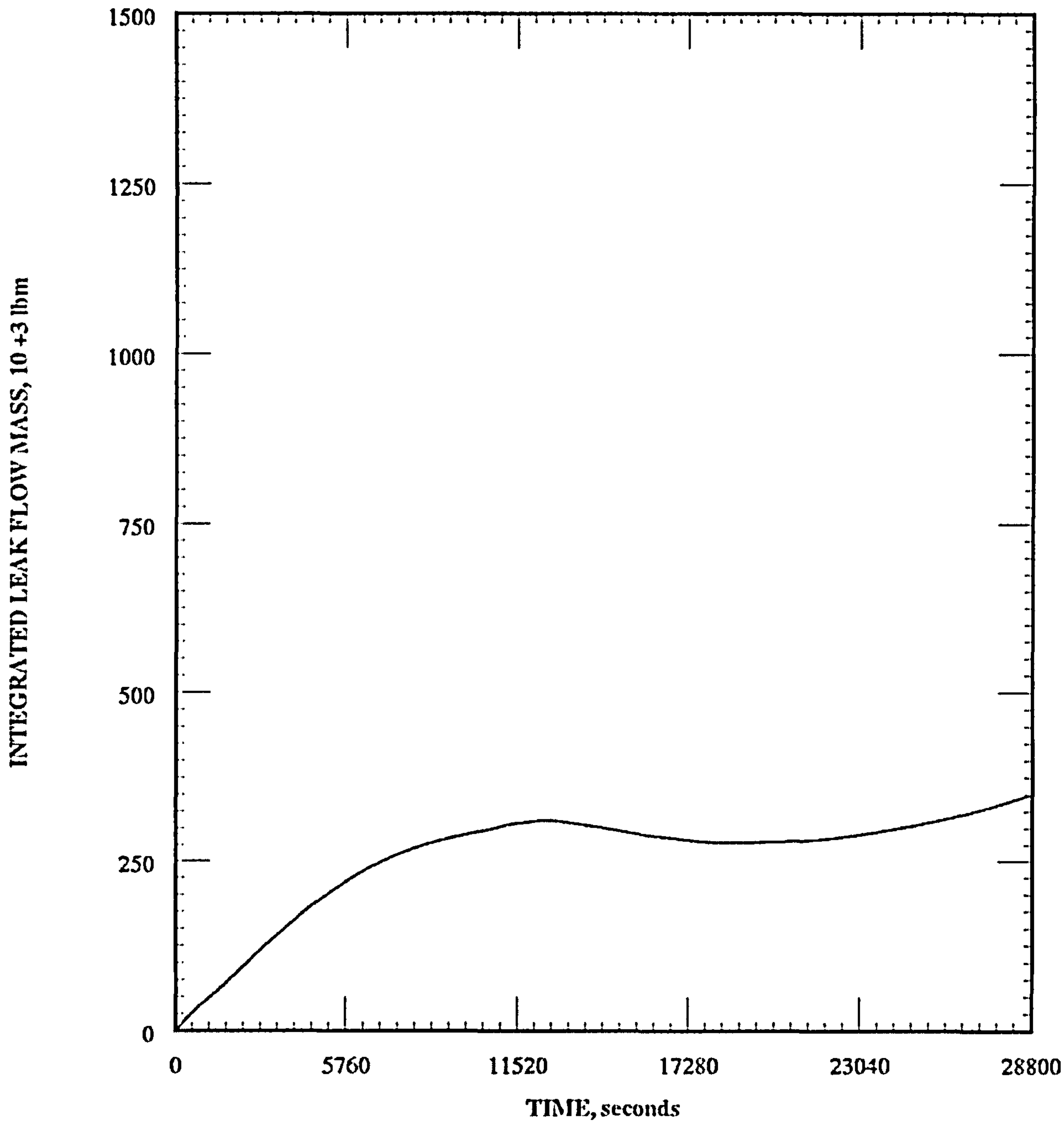


Figure 6.3-234
SGTRLOP Event–SG Liquid Inventory vs. Time
(Sheet 3 of 3)

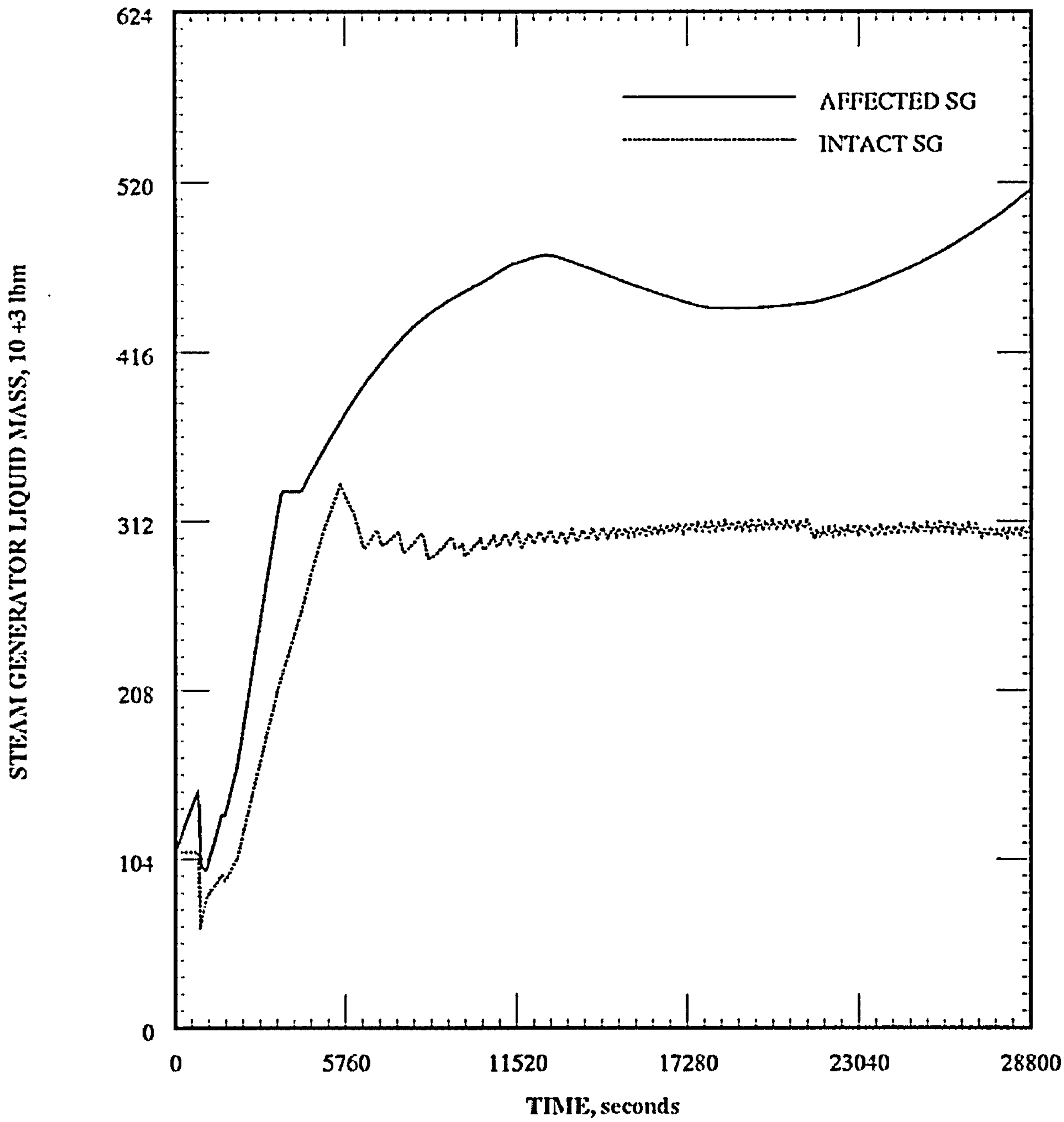


Figure 6.3-235
SGTRLOP Event–Integrated SI Flow vs. Time
(Sheet 2 of 2)

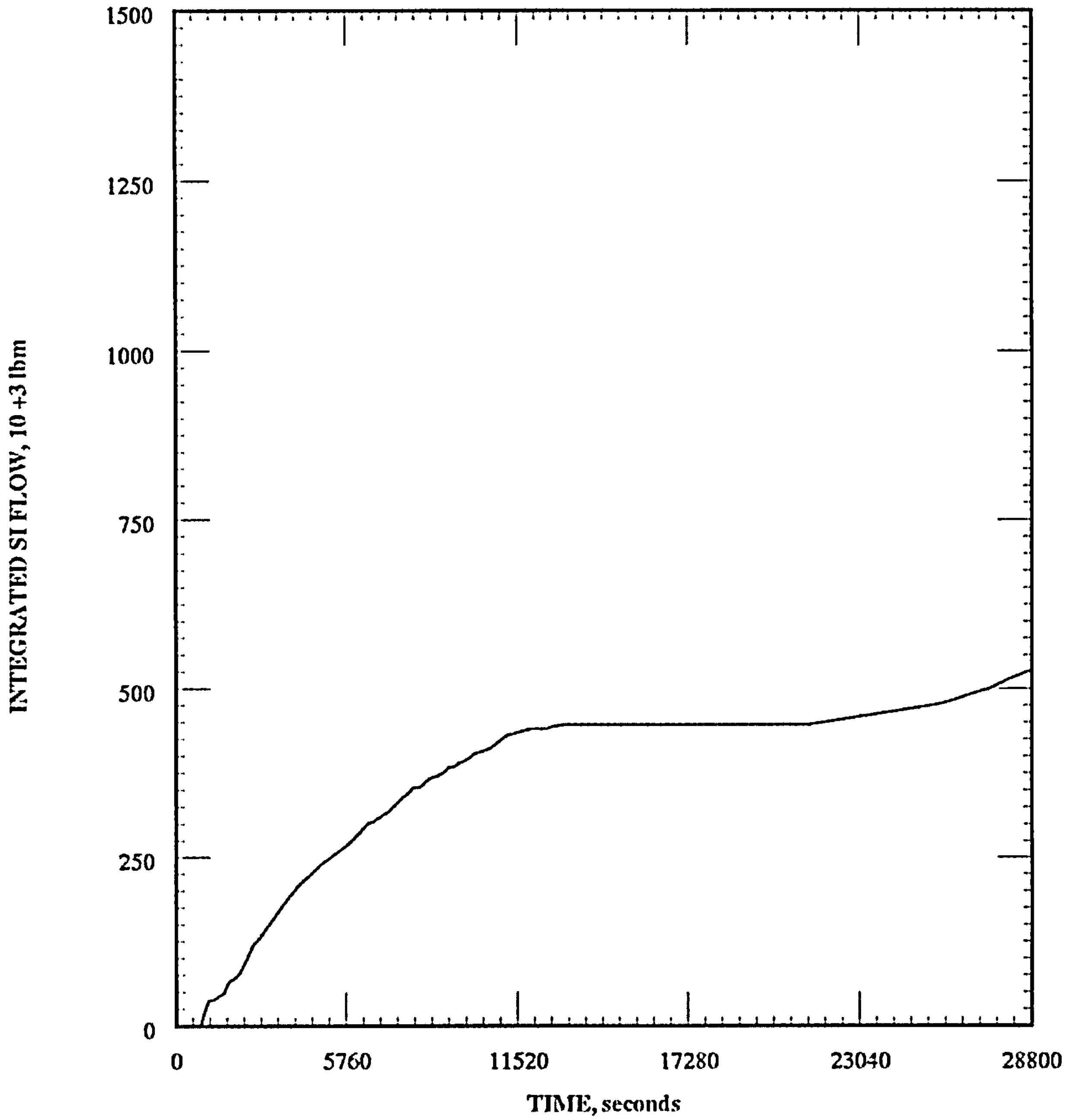


Figure 6.3-238
SGTRLOP Event–Integrated ADV Flow vs. Time
(Sheet 2 of 2)

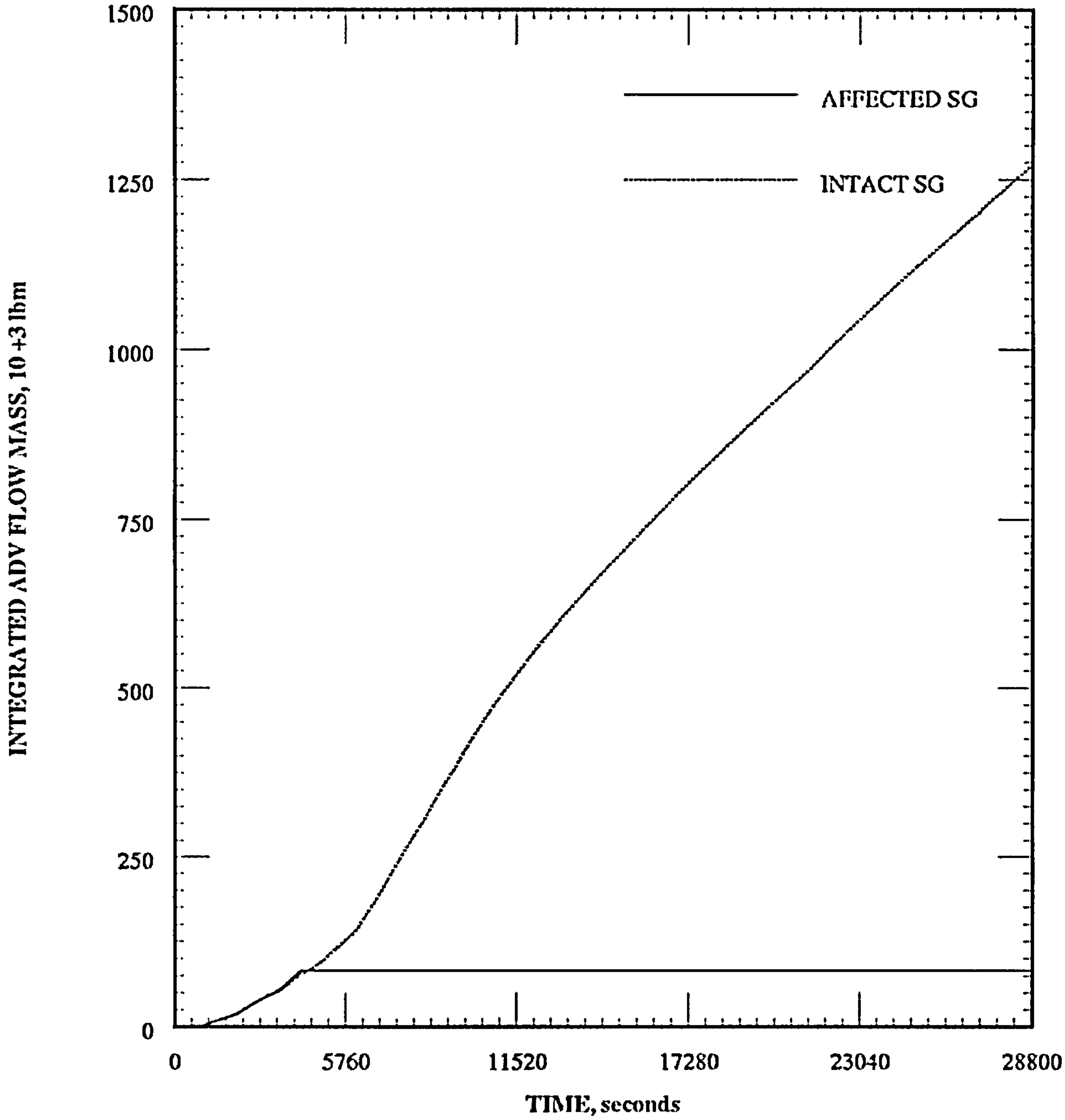


Figure 6.3-239
SGTRLOP Event–Subcooled Margin vs. Time
(Sheet 2 of 2)

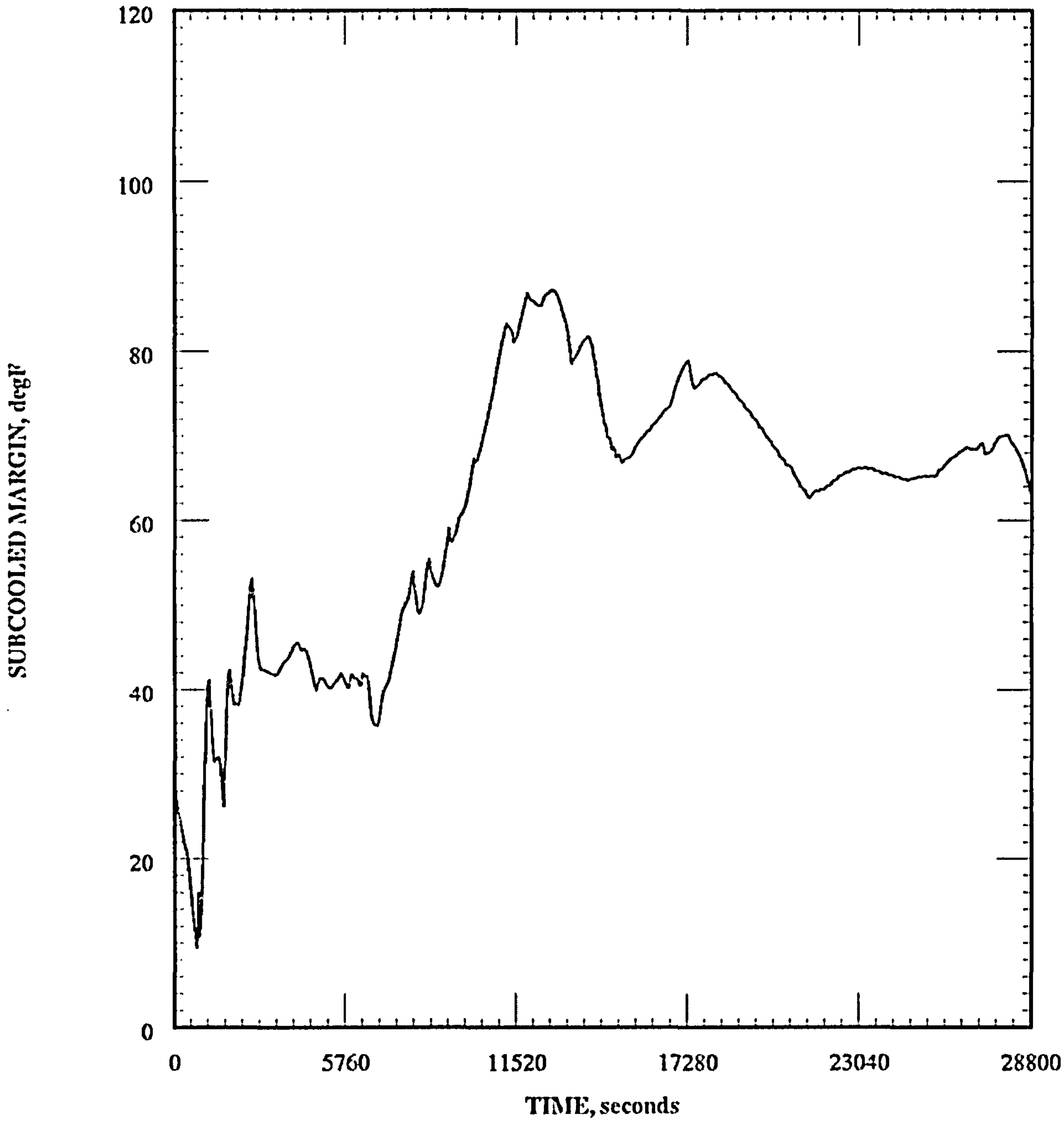
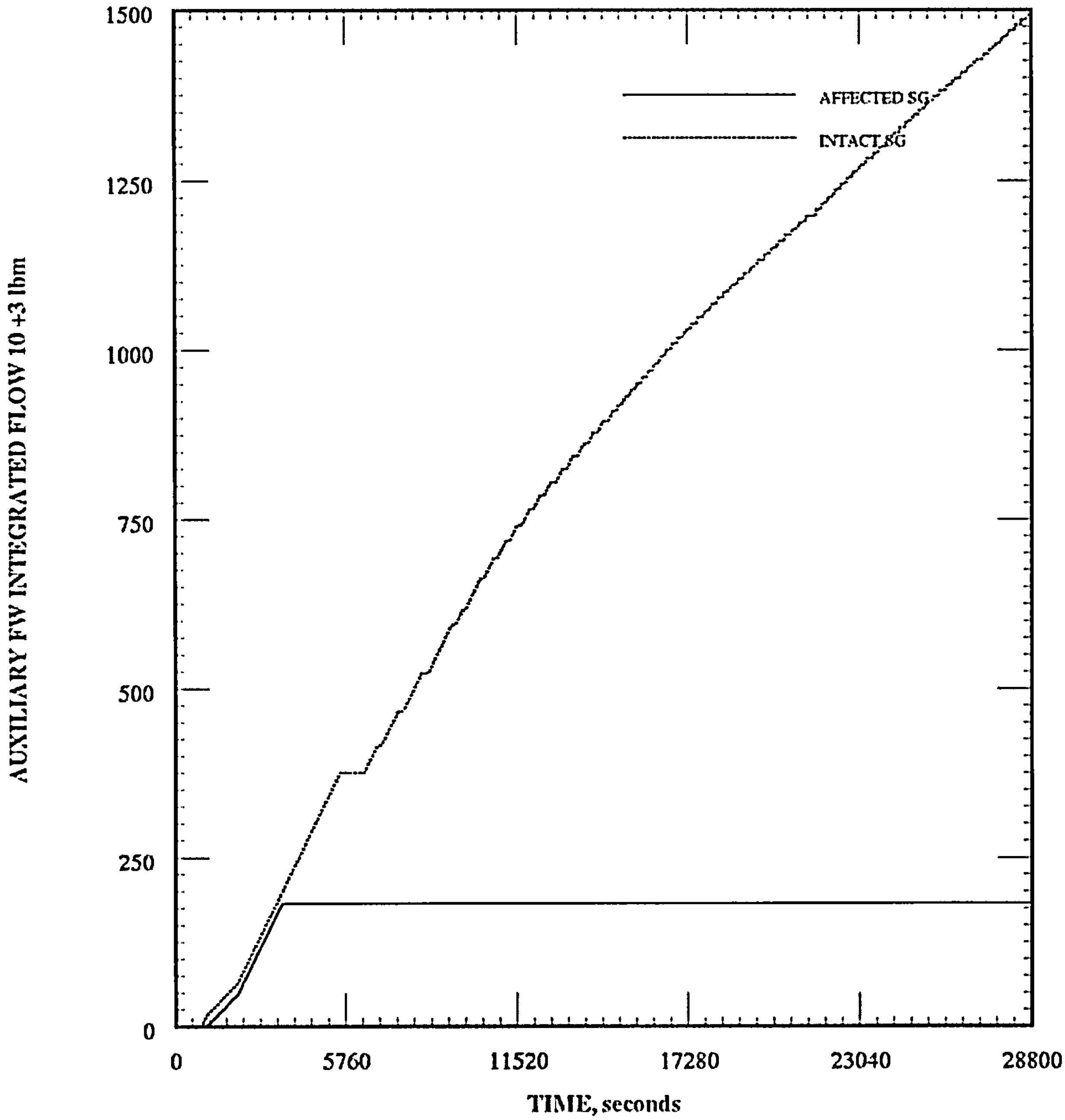


Figure 6.3-240
SGTRLOP Event–Integrated AFW Flow vs. Time



Section 6.3.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

As described in UFSAR Section 15.6.4, this event is applicable to BWRs only.

Section 6.3.6.5 Loss-of-Coolant Accidents

ECCS performance and LOCA are discussed in Section 6.1. Radiological consequences of this event are described in Section 6.4.6.3.

Section 6.3.7 Radioactive Material Release from a Subsystem or Component

This section is contained in Reference 6-1, Attachment 6. There are no changes to this section.