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GINNA SGTR EVENT — RETRAN CALCULATIONS

by

J. H. Tessier and T. Y. C. Wei

Light Water Reactor Systems Analysis Section
Reactor Analysis and Safety Division
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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Division of Systems Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
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EXECUTIVE SUMMARY

A simulation of the Ginna Steam Generator Tube Rupture (SGTR) event of January 25, 1982 was performed utilizing the latest EPRI-released version of RETRAN, RETRAN02/MOD03, in conjunction with RETRAN02/MOD03A input decks obtained from the Institute of Nuclear Power Operations (INPO) and modified by ANL. The RETRAN02/MOD03 results agree well with the INPO RETRAN02/MOD03A calculations. A reasonable match is therefore obtained between calculations and the measured data from the actual event. Where differences between the two calculations have occurred, they can be explained in terms of code model differences between the MOD03A and the MOD03 versions and in terms of sensitivities in the INPO calibration to data. In addition, three parametrics were performed which included variations on operator actions and further equipment failure. Results of these parametric calculations demonstrated; that opportune timing in conforming to recent operator guidelines would prevent filling the disrupted steam generator solid and alleviate concerns about loading questions; that additional failures occurring in the PORV line downstream of the PORV would not necessarily lead to significant core damage; and that sufficient thermal margin exists against pressurized thermal shock conditions even if there was a further continuation of safety injection flow. Furthermore, the parametrics have contributed to the understanding of the thermal hydrualic phenomena that occurred during the actual sequence of events during the Ginna SGTR incident.

1.0 INTRODUCTION

The Institute of Nuclear Power Operations (INPO) recently performed a simulation [1] of the Ginna steam generator tube rupture (SGTR) event of January 25, 1982 [2] using an interim version of the RETRAN02/MOD02 code [3]. In this report results are presented for a comparison of calculations utilizing RETRAN02/MOD03, the most recent release of the code, and the RETRAN02/MOD03A input deck furnished by INPO with the actual event. In addition, parametric calculations were carried out varying the scenario of the actual event in terms of operator actions and mechanical failures. These served to increase the understanding of the thermal hydraulic phenomena which occurred during the incident. The three parametrics performed were:

a) A duplication of the event with a variation in the operator actions. The latest Westinghouse operator guidelines for SGTR, E-3 (July 5, 1982) were followed to determine their efficacy in preventing the ruptured generator from going solid.

b) The pressurizer PORV (pressure operated relief valve) is assumed to stick open during the operator's efforts to depressurize the primary side and the downstream block valve is presumed to concurrently fail in the open position. This examines the ability of the SI (safety injection) to maintain the system in a stable condition.

c) Finally, the safety injection (SI) system was presumed to be left on beyond the point of termination in the actual sequence of events. The quasi-steady cooldown rate in the downcomer obtained in this parametric could be of significance to pressurized thermal shock problems.

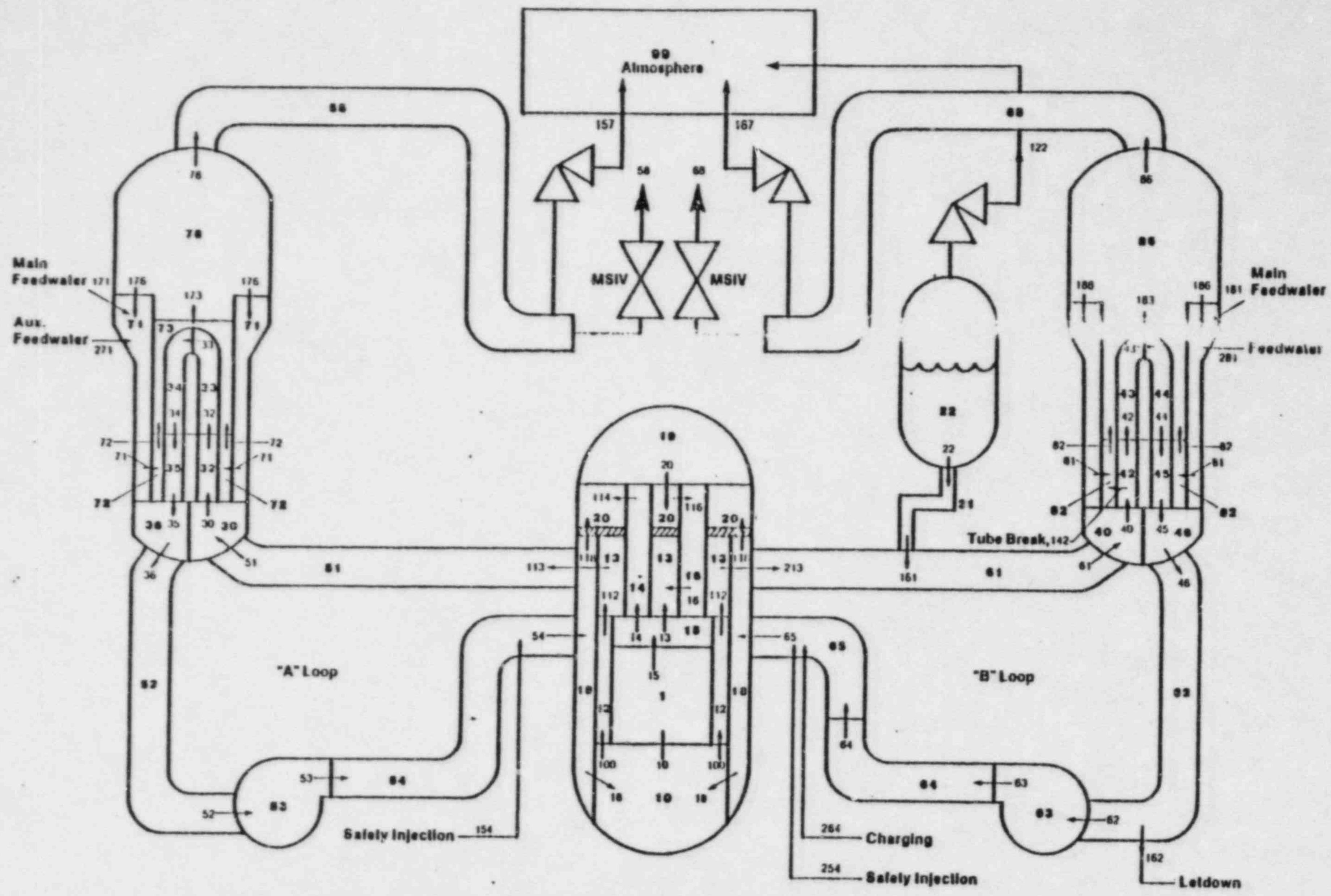
This document details the parametric calculations and discusses the results obtained, as well as those calculations performed for comparison with the INPO computations. Section 2.0 describes the INPO plant model used, the various input modifications which had to be made for successful execution and the coding changes to the RETRAN02/MOD03 source program required to correct for code deficiencies. The result of the comparison against the INPO calculations and concurrently the Ginna data are described in Section 3.0 while the parametrics are presented in Section 4.0. Conclusions are drawn in Section 5.0.

2.0 PLANT MODEL

The Ginna SGTR event started with the plant at normal operating conditions with the primary side entirely in single phase, with the exception of the two-region pressurizer which was in thermodynamic equilibrium, and a secondary side which was in two phase steaming off into the turbine. Upon tube rupture, the primary side commenced depressurization with a loss of inventory through the rupture. Flow through the rupture was choked. The secondary side of the SGB began to respond in the manner of a two region nonequilibrium pressurizer model (with a mixture level) as the steam generator began to fill up. Complications in the thermal hydraulic response were introduced because of system feedback effects with turbine load reduction, reactor scram and safety injection initiation. However, until the primary system had depressurized such that the relatively stagnant upper head region reached the saturation temperature the entire primary loop was governed by single phase hydraulics. The outsurge from the two region pressurizer can be treated by a nonequilibrium pressurizer model. Bulk flashing such as that which ultimately occurred in the upper head is also a phenomena simulated by nonequilibrium pressurizer models. Natural convection occurred on the primary side during the flashing period as the pumps coasted down and the operators initiated manual depressurization. The pressurizer rapidly refilled during the manual depressurization leading to possible nonequilibrium conditions. On the secondary side, the tubes did not uncover so dryout is not a concern and the heat transfer is the normal two phase heat transfer. While pressure measurements were available on the secondary side, data are lacking on the flow through the various valves as the ruptured B steam generator filled solid. (Data are limited in general as the Ginna plant is not instrumented as an experimental facility). With the filling of SGB, single phase choked flow through the safety relief valves from the basically incompressible volume occurred. The primary side also tended to single phase again during this period of filling SGB as the head region steam bubble began to collapse with the continuation of SI flow. Heat losses and thermodynamic nonequilibrium had to be considered during the bubble collapse. Finally, the termination of the SI did not lead to the introduction of additional thermal hydraulic phenomena not previously discussed.

Plant models developed to simulate the Ginna SGTR event have to envelope all these varied thermal hydraulic phenomena. The Institute of Nuclear Power Operations developed such a plant model using engineering judgement where necessary to compensate for the limitations of the data availability, discussed earlier, and has obtained reasonable agreement with the event after calibrations.

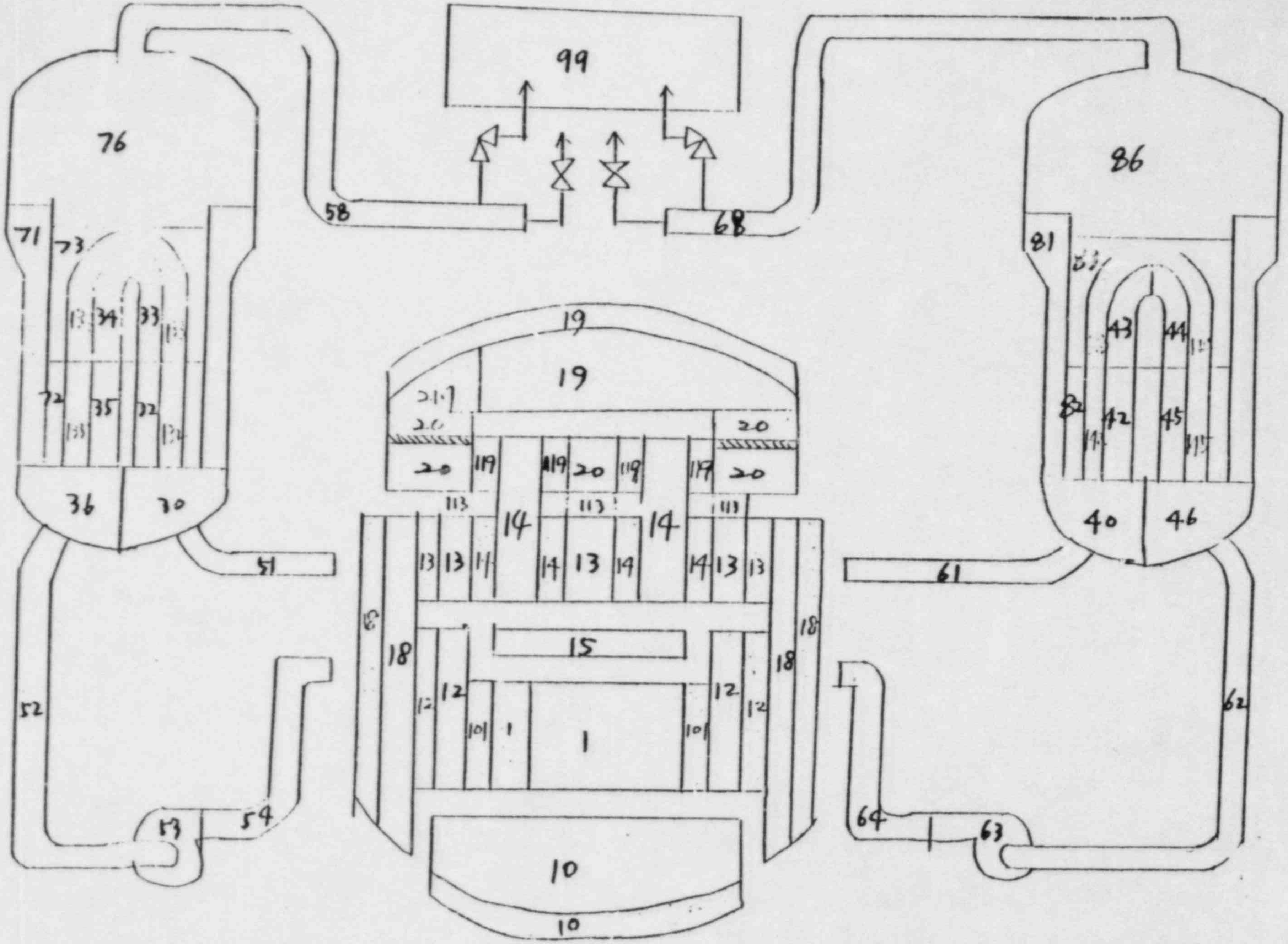
INPO provided RETRAN02/MOD03A decks (t=0 and t=42.5 minute decks) and restart information which used the volume/junction nodalization shown in Fig. 2-1. Figure 2-2 presents the heat slab nodalization used. The Fig. 2-1 nodalization was used from time = 0 to 42.5 minutes at which point renodalization was performed. The renodalization mainly took the form of moving the nonequilibrium pressurizer model from the disrupted steam generator (SGB) dome to the corresponding steam line. This apparently was done, at least in part, in order to avoid numerical problems with the nonequilibrium pressurizer model when complete filling occurred. There was also a change in the volume split between volumes 19 and 20 when complete draining or filling took place. In summary, at transient time equal to 42.5 minutes, the RETRAN model was revised by INPO to enable treating the SGB (steam generator B) steam line as a non-equilibrium pressurizer volume during the period that it filled with liquid. This was considered the most realistic modeling available and permits selection of a spray option (with condensation calculated) and variation in the inter-region heat transfer coefficient through appropriate input changes made in restart decks. It was also necessary to change the geometric model of the steam line to ensure that the liquid flowing from the steam generator, into the line, always entered the liquid region in order that excessive condensation due to homogeneous mixing of liquid and steam did not occur when the spray option was turned off. This required placement of the junction connecting the steam generator and line at an artificially low elevation, but the effect of this on the calculation is judged to be acceptably small. The impetus for incorporating the non-equilibrium model of the steam line was to provide added means for controlling the calculated pressure levels to match measurements. For further details regarding the nodalization reference should be made to the INPO draft report [1].



RETRAN Model Volumes and Junctions

Figure 2-1

Figure 2-2. RETRAN Model Heat Slabs



The aforementioned renodalization, and other INPO revisions, cannot be accomplished in a restart deck. Consequently, an entire new input deck (t = 42.5 mins deck) was required, initialized to non-steady conditions. These initial conditions were the end-point values of plant parameters calculated for the initial portion of transient (t=0 deck). It is noted that RETRAN does not permit supplying inputs for all required variables, e.g., heat conductor temperatures, surface heat fluxes and values of slip in junctions having two-phase flow. Such quantities are normally computed for steady-state conditions by iteration. However, for initialization in a dynamic condition, the steady-state initialization option is bypassed (word number 25, JSST, on card number 01000Y is set equal to 1) and quantities that are not input are computed without a steady state search. This approach is reasonable if time derivatives in the relevant equations are acceptably small. This apparently is true for this application since there is no evidence of significant discontinuities in calculated results at the time of transition to the new input deck.

In order to match the measured data with the limited instrumentation capability available INPO had to undertake fairly extensive calibration efforts. Among the more significant adjustments made are renormalizations of upper head "valve" areas, steam generator B S/R valve areas, MSIV flows and PORV areas. The INPO draft document provides additional information [1]. In general the RETRAN02/MOD03A decks obtained from INPO, once modifications were made for the RETRAN02/MOD03 version, were used "as is" to produce the results documented in this report. Minor alterations had to be made in order to affect the changes required for the three parametrics and these are discussed in the respective sections. More extensive alterations had to be made to compensate for code modifications to obtain the comparison results with the INPO computations for the actual Ginna event. These input deck alterations/code model modifications are listed below.

For the t=0 deck:

a) the input for the upper head heat slabs had to be altered to accommodate for the changes to the non-equilibrium pressurizer heat loss model;

b) the steam generator dome to bundle junction had to be forced into a zero slip calculation to avoid non-physical results. This was the original MOD03A option;

c) while it proved infeasible to compensate for the change in bubble rise model it should be noted that MOD03 normalizes the Wilson bubble rise correlation to the rise velocity computed in the steady state initializer. This was not done in MOD03A.

For the t=42.5 minutes deck:

d) the Wilson bubble rise model yielded a bubble velocity of zero except for the initial value;

e) job failures occurred when the calculation called for closure of a previously opened SRV;

f) non-physical results were calculated at the time of SI termination (two phase flow with slip in a junction connected to a non-equilibrium volume).

Each of these problems were resolved after consultation with EI personnel. Some were circumvented by changes to the input deck whereas others involved FORTRAN source changes. In addition to these problems, MOD03 stability required smaller integration time steps than those used in the INPO calculations (MOD03A). This increased the demand on computer time and caused a need for more restarts to complete these calculations.

3.0 INPO COMPARISON

In order to simulate the measured system response during the Ginna SGTR event, INPO [1] divided the transient into two parts: one from initiation of tube rupture to commencement of manual depressurization (t=42.5 minutes); the second from t=42.5 minutes to shortly after SI termination at which point the INPO computation was terminated. Discussion of the ANL comparison with INPO is therefore similarly divided into two parts. Section 3.1 details the comparison from t=0 to t=42.5 minutes and Section 3.2 concludes with the latter part of the transient from t=42.5 minutes to SI termination.

3.1 t=0 Deck

During the period from t=0, the point at which tube rupture occurs, to t=42.5 minutes when operator-initiated depressurization using the pressurizer PORV commences, the reactor system goes through a turbine load reduction phase, reactor scram and a tripping in of the safety injection (SI). Concurrently the condenser dump valves are cycled, the various feedwater pumps are manually controlled and the isolation procedures are followed by the operators in order to minimize dose rates and to reestablish control. Reference should be made to the INPO draft report for a detailed chronicle of the actual Ginna SGTR event.

The t=0 deck takes the event from the initiation of tube rupture to the time when operator initiated depressurization through the pressurizer PORV at t=42.5 minutes occurs. At this point the problem is renodalized and calculation continued with the t=42.5 minutes deck. In addition to the splicing of results made necessary by this renodalization procedure at 42.5 minutes there is an additional splice necessitated by INPO's further recalibration during the time period 112-180 seconds. With the differences in code models between RETRAN02/MOD03A, the code version used by INPO, and RETRAN02/MOD03, the version utilized by ANL to produce the results presented in this report, it can be seen that there are possibilities for discontinuities at the times of 112 seconds, 180 seconds and 42.5 minutes. While some smoothing could be rationalized and was indeed used at these junctures, the result do show some discontinuities at these points. These will be discussed in perspective. The

perspective of this document is to concentrate on the differences between the INPO calculations and the ANL computations and to understand them in terms of modifications in code models between the two code versions and also of sensitivities in the INPO calibration to the data from the original event.

Figures 3-1 to 3-26* present the comparison between the INPO and ANL results. This set of graphs represents the entire set presented in the main text of the INPO draft report. The ANL curves have been traced onto the INPO figures. Reasonable agreement has, in general, been achieved between the INPO RETRAN02/MOD03A calculation and the ANL RETRAN02/MOD03 computation. Where differences have arisen they can be attributed to three or four modifications as discussed in subsequent paragraphs.

In order to compute the narrow range pressurizer level adjusted for instrument error at nonsaturation conditions the input deck had to be altered to incorporate a stand alone control block model. An error was discovered in the transcription process which affected the adjusted level. As the control block is a stand alone model utilizing input thermal/hydraulic (T/H) conditions from the main RETRAN T/H calculation it is completely ignored by the main computation. However, Fig. 3-2 for the pressurizer level shows that the error leads to an underprediction of the instrument adjusted level.

In RETRAN02/MOD03A the nonequilibrium pressurizer model had no heat loss associated with the volume boundary. Two-sided heat slabs could be attached to the nonequilibrium volume by the user but the volume does not "see" the heat slabs. The heat slabs however do "see" the volume. This model was altered in MOD03 with the slabs and volume interchanging heat in an energy conservative manner. However, the use of an adiabatic boundary condition is now necessary on the non-pressurizer volume side of the heat slabs with Mod 03. These alterations imply that the heat transfer in the upper head region, which is modelled by INPO using the non-equilibrium pressurizer model, particularly that incurred by the "fictitious" conductor (slab 20) cannot be duplicated by the ANL MOD03 calculation. Consequences of this model

*These Figures are grouped at the end of Chapter 3.

modification can be seen in Figs. 3-1 (RCS pressure), Fig. 3-5 (vessel upper head temperatures) and Fig. 3-14 (vessel upper head steam volume). There are certainly effects on other parameters but those should be of lesser importance particularly for the parameters on the secondary side. Heat losses from the pressurizer per se are treated through the use of a control block and there are no heat slabs associated with the steam generator domes. Thus even though the non-equilibrium pressurizer model is used for the pressurizer and the steam generator domes this model alteration should not directly affect those volumes.

A modification was made to the bubble rise model between the MOD03A and the MOD03 code versions. The MOD03 model now normalizes the bubble rise velocity computed by the Wilson bubble rise correlation to the velocity computed by the initializer to obtain steady state. This may sound inconsequential as the Wilson correlation is only used in the nonequilibrium pressurizer model and pressurizers are normally initialized with no bubbles. However, INPO chose to use the non-equilibrium pressurizer model in the steam generator domes. Upon switching out of the automatic initializer and proceeding into the transient MOD03A would use a different bubble rise velocity from MOD03. The implications of this difference is an alteration in steam generator level behavior. Figures 3-11 (A SG water level), Fig. 3-12 (B SG water level) and the corresponding figure for the first 20 minutes of the transient, Fig. 3-23, show the difference in initial swell which can be attributed to this model modification. The INPO calibration of steam flow, etc., has to be recalibrated to data to account for the alteration in code model. This change should also affect steam generator pressures

Finally, the effect of the splicing discussed earlier can be seen in Figs. 3-14, and 3-24 to 3-26. To reiterate however, in general the ANL computations using RETRAN02/MOD03 compare reasonably with the INPO calculation using RETRAN02/MOD03A. Where differences do occur attribution can be made to model alterations and calibration sensitivities.

3.2 t=42 mins Deck

This portion of the calculation spans the transient time from 42.5 to 80.0 minutes after tube rupture. These results are also depicted in Figs. 3-1 through 3-20 with the initial portion discussed earlier. This latter portion of the Ginna calculation begins with initial opening of the pressurizer PORV to accelerate RCS depressurization and concludes shortly after SI termination. Throughout this period, decay heat removal and RCS cooldown occurred by continued injection of the relatively cold SI water and feed and bleed operation of the intact steam generator (SGA) as modelled in the input decks. Since there was no measure of the SG steam and feedwater flows, they were adjusted in the model to provide correspondence with measured pressure and water level data.

As the aforementioned figures show, there is excellent agreement between ANL's calculated results and those obtained by INPO. The plots for primary side parameters overlay nearly exactly, except for some of the fine structure in the oscillatory behavior of certain parameters. In addition to the agreement shown for RCS temperatures, pressures and flows, there is also essential overlap in the curves of tube rupture flow and upper head steam formation for the two sets of calculations (Figs. 3-13 and 3-14).

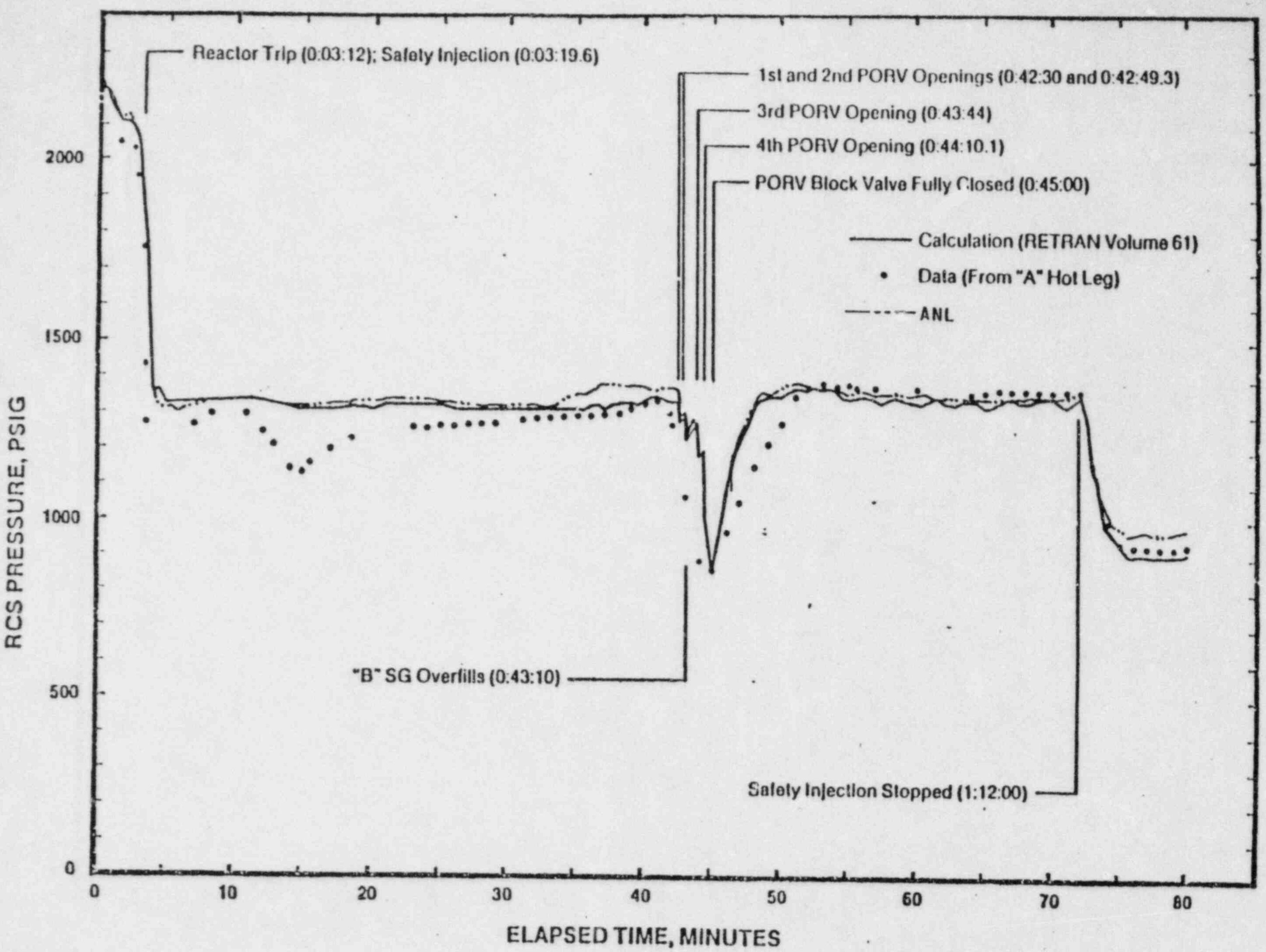
The only differences of note, albeit small, are those in parameters calculated for the faulted steam generator (SGB). INPO, in their calculations, used a complicated program for the area of the SGB safety relief valve (SRV) to control calculated pressure to match measured data. To do this, several restarts were made wherein both open and close pressure dependent trip setpoints and open and close valve area table entries were changed. Effectively, this process varied the valve area with time, opening or closing it contingent on calculated pressure levels. The trip setpoints were not the normal plant values, but were selected to be close to the running pressure levels as indicated by data. Thus, small differences in calculated pressure levels caused the valve behavior to be different in the ANL calculations.

The first departure from INPO's results is evident in Fig. 3-2 which shows three openings and closings of the SGB SRV in the time period of ~ 51 to 55 minutes whereas only a single cycle was calculated by INPO. This is also manifested in the sawtooth nature of the ANL curve of SGB pressure (Fig. 3-10) during this time period. These additional openings of the relief valve released more liquid from the system to the atmosphere causing a slight delay (~ 1.4 minutes) in the calculated time to completely fill the B steam line with liquid as shown in Fig. 3-19.

After the secondary side became solid, the calculated SGB pressure levels became very sensitive to the flow resistance out of the system at the SRV. In the INPO calculations, the valve area, and hence its resistance, was varied often using pressure trips and area tables in the manner described above, attempting to match pressure data. Using INPO's trip setpoints and valve area tables in the ANL calculations proved unworkable because the timing for switching between valve opening and closing modes is critical to obtaining the proper areas versus time. This timing of actuating the open/close trips is not deducible from the reported results and even small calculated pressure differences soon resulted in erroneous trip times and attendant valve area values, and subsequent large deviations in pressure levels. It was necessary, therefore, to change the method of programming the valve area with time in the ANL calculations. The INPO curve of valve area shown in Fig. 3-20 was used to derive numerical values which were entered in an appropriate RETRAN table and the trips were changed such that after t = 55.42 minutes all areas were obtained from that table. Also, the time at which the sudden large area increase occurs was delayed to be coincident with the later time to fill the steam line with liquid as calculated by ANL. This approach, while not precise, yields levels of agreement with INPO's results considered adequate as examplified by the comparisons of SGB pressure levels and SRV flow rates shown in Figs. 3-10 and 3-20 respectively.

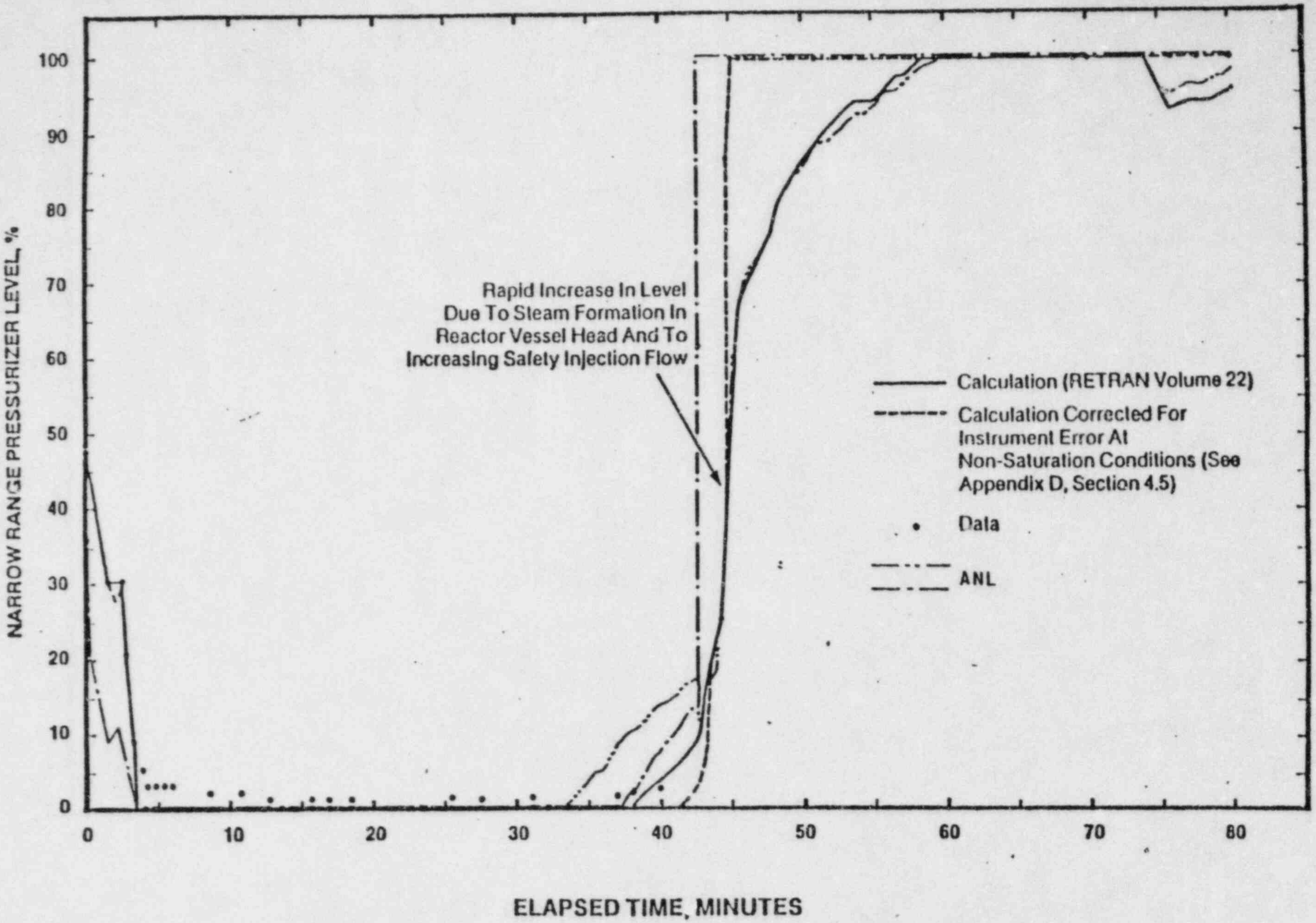
In summary, the results obtained by ANL for this phase of the Ginna event using RETRAN02 Mod 03 also show excellent agreement with and confirm INPO's earlier calculations. Although some differences were encountered, they are not sufficient to negate this conclusion. Also, the available measured data is represented quite well by the calculated system responses lending

credence to INPO's overall conclusions regarding the transient plant status following the actual steam generator tube rupture event.

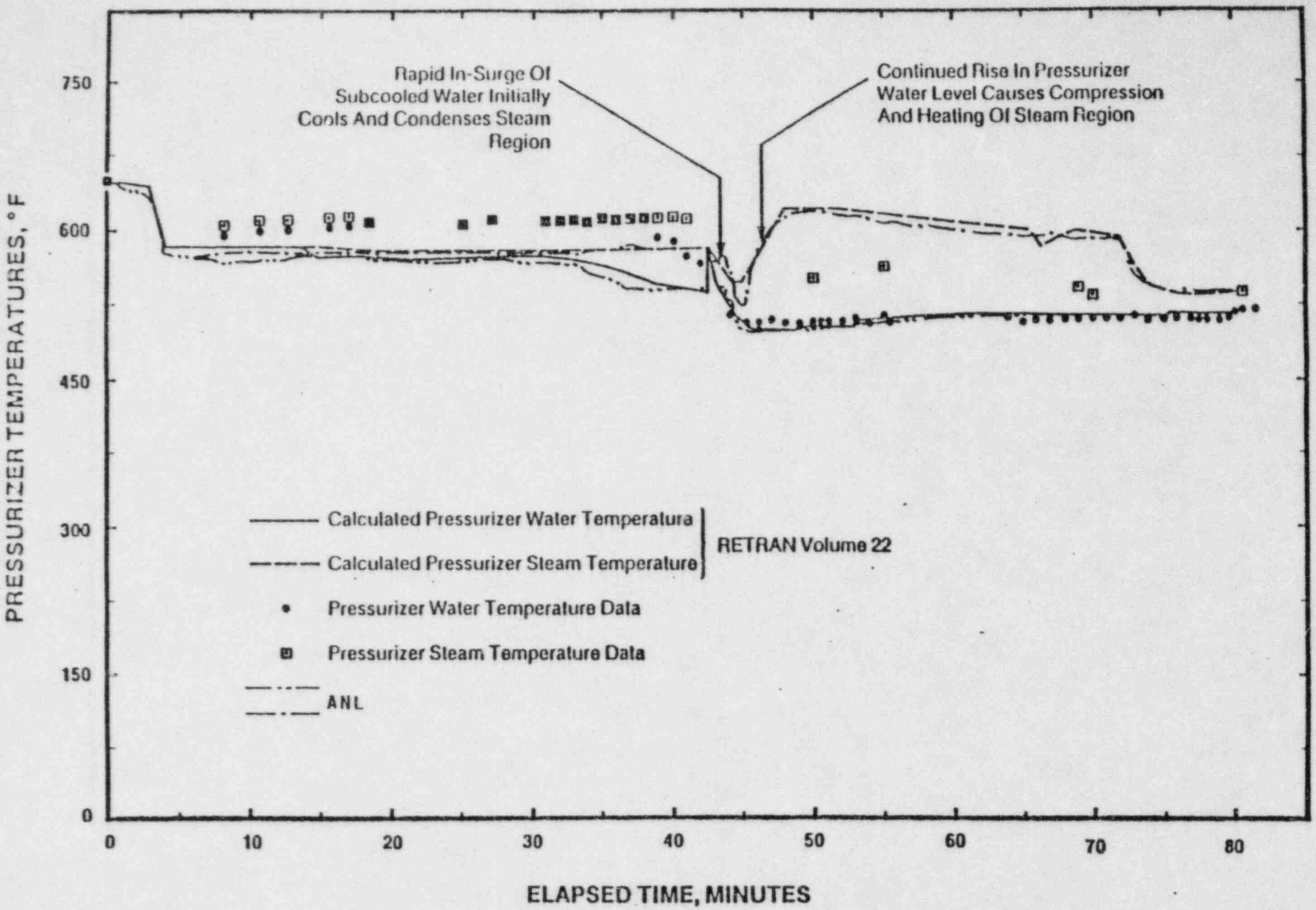


RCS Pressure vs. Time

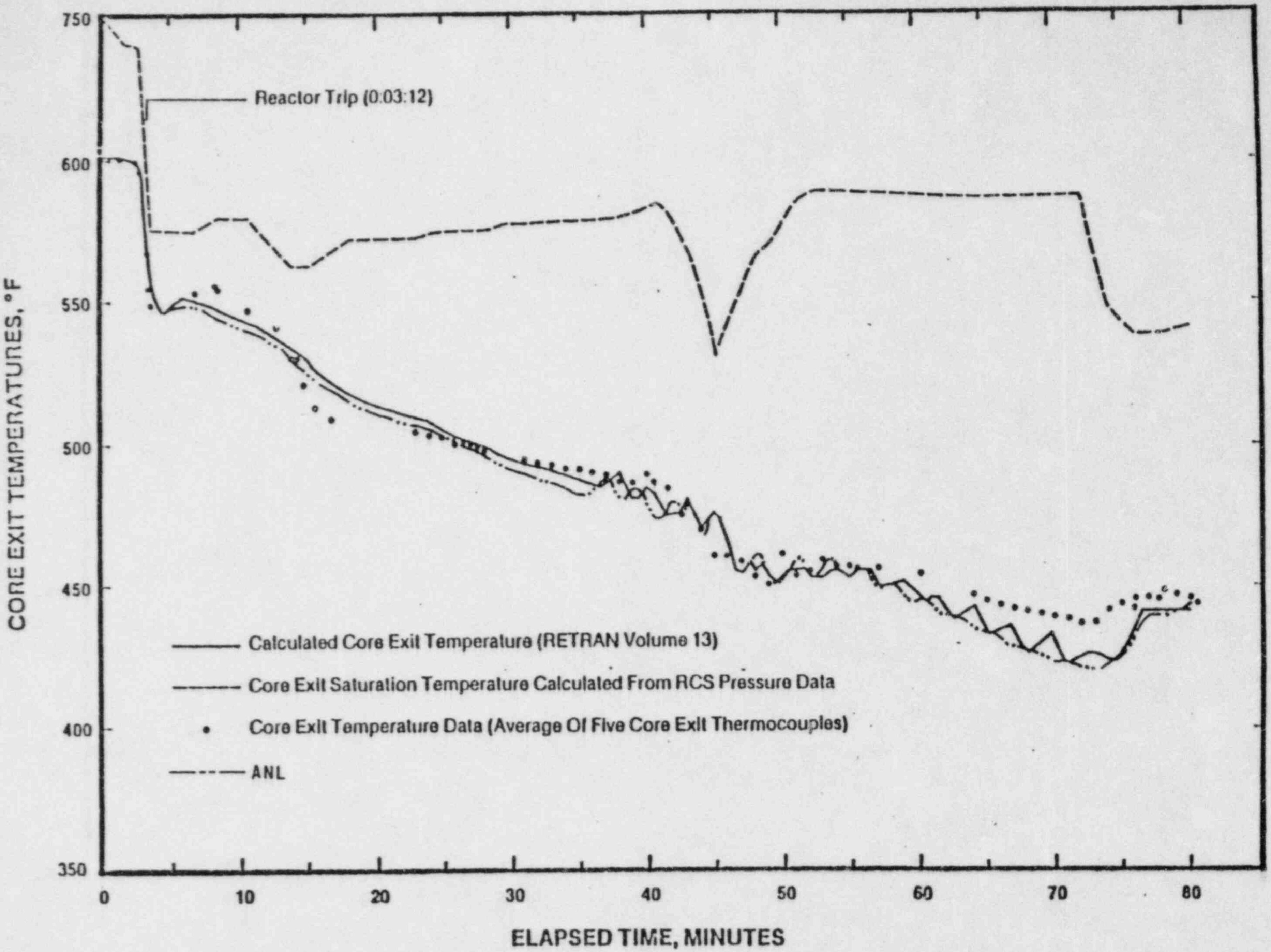
Figure 3-1



Pressurizer Level vs. Time Figure 3-2

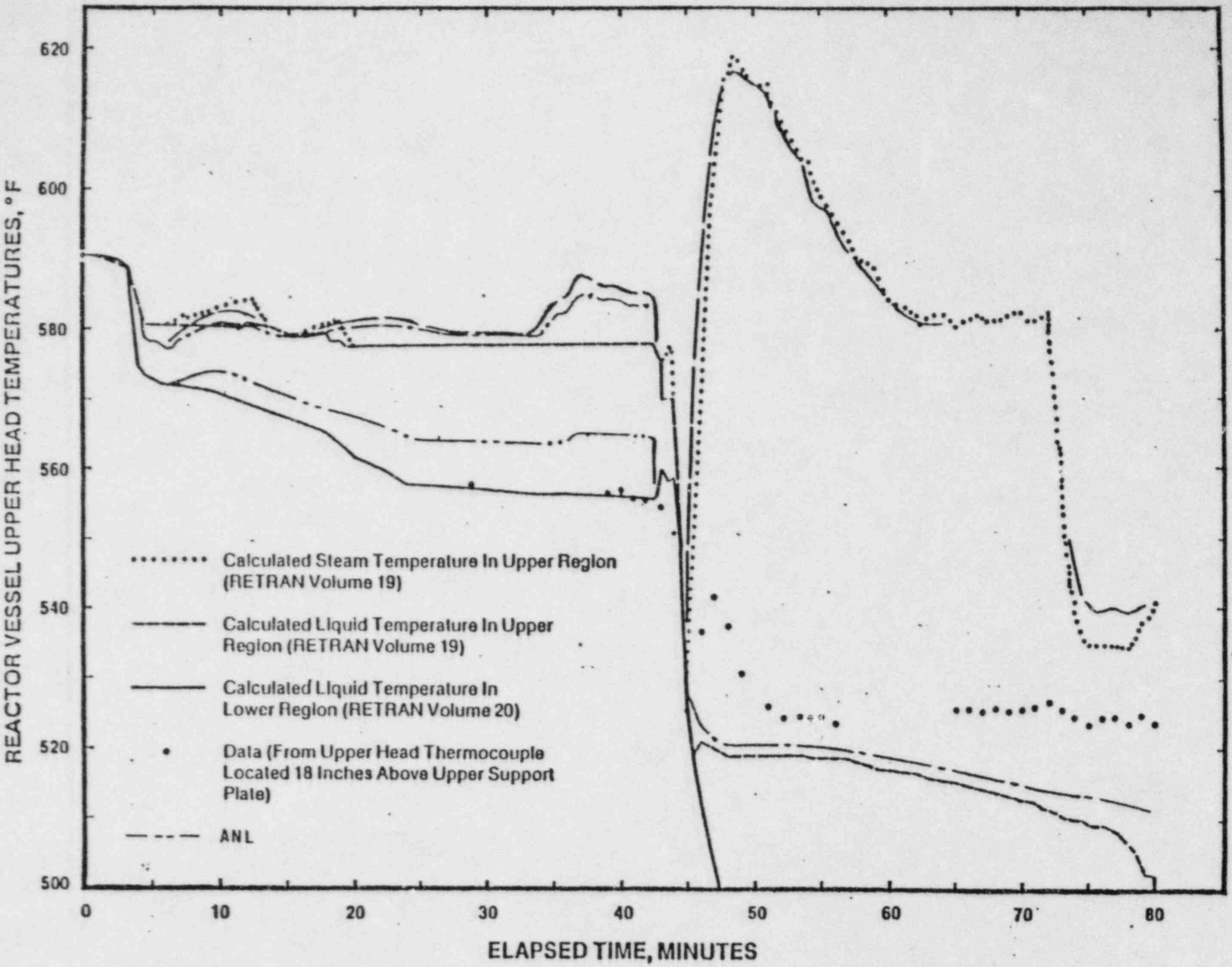


Pressurizer Temperatures vs. Time **Figure 3-3**



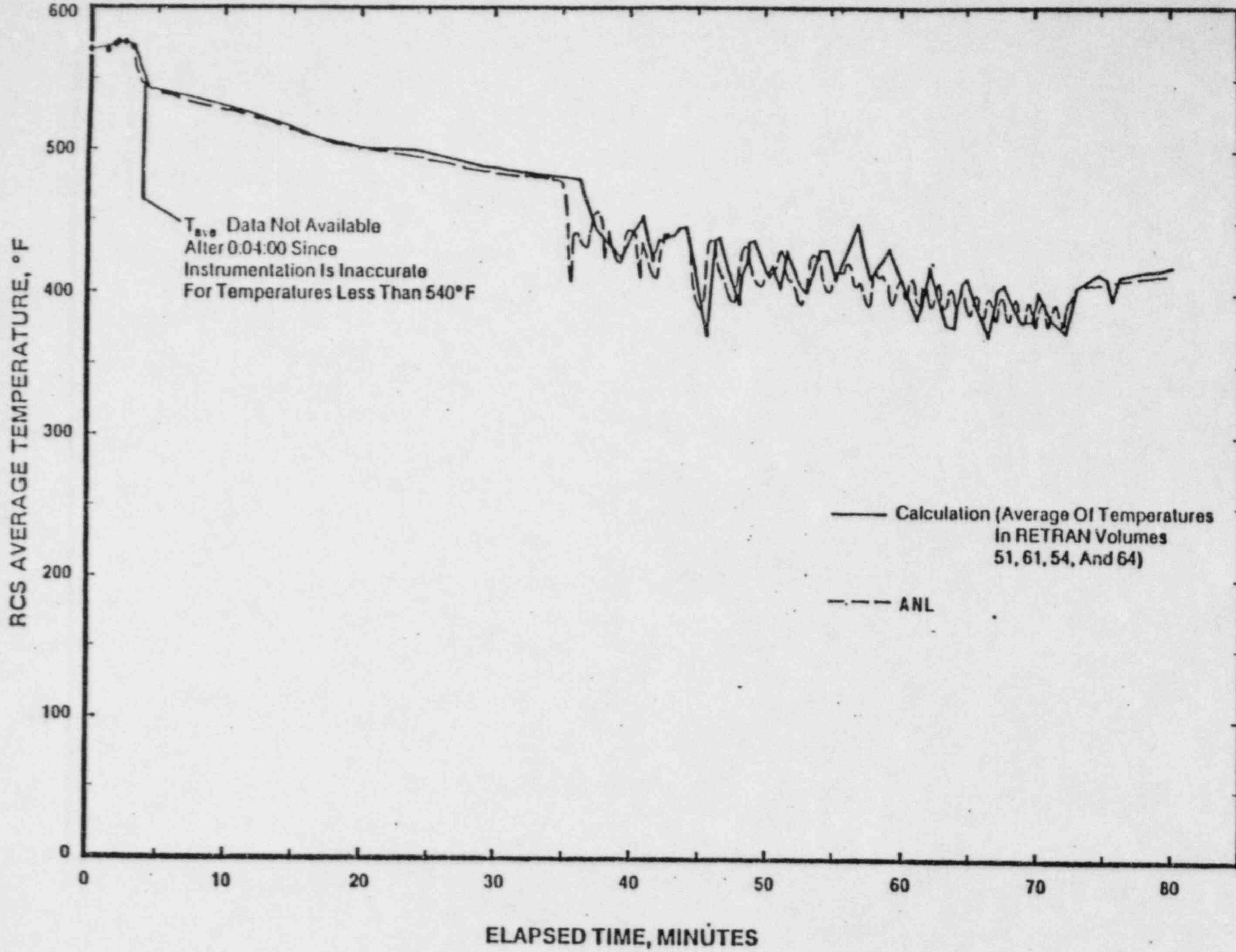
Core Exit Temperatures vs. Time

Figure 3-4



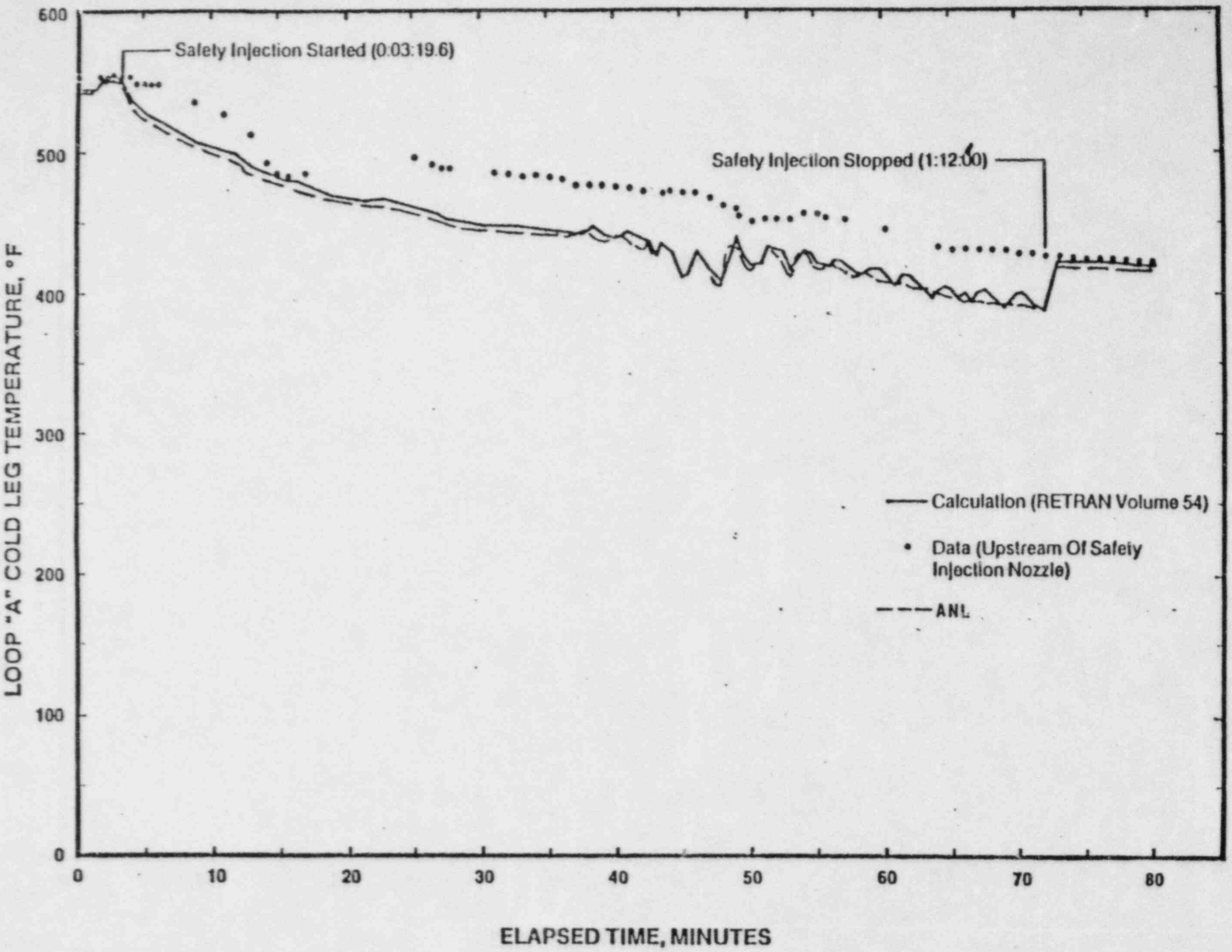
Reactor Vessel Upper Head Temperatures vs. Time

Figure 3-5



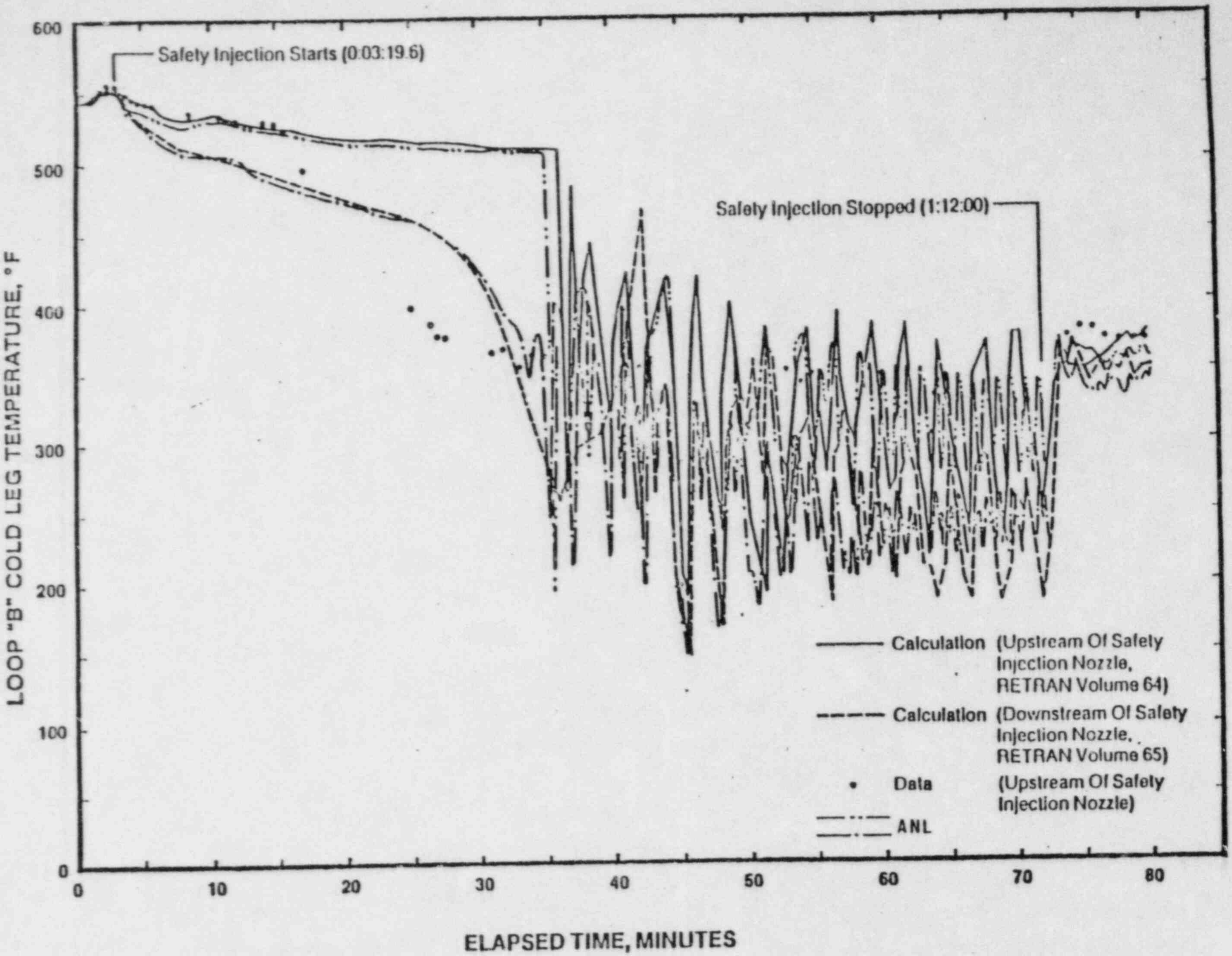
RCS Average Temperature vs. Time

Figure 3-6



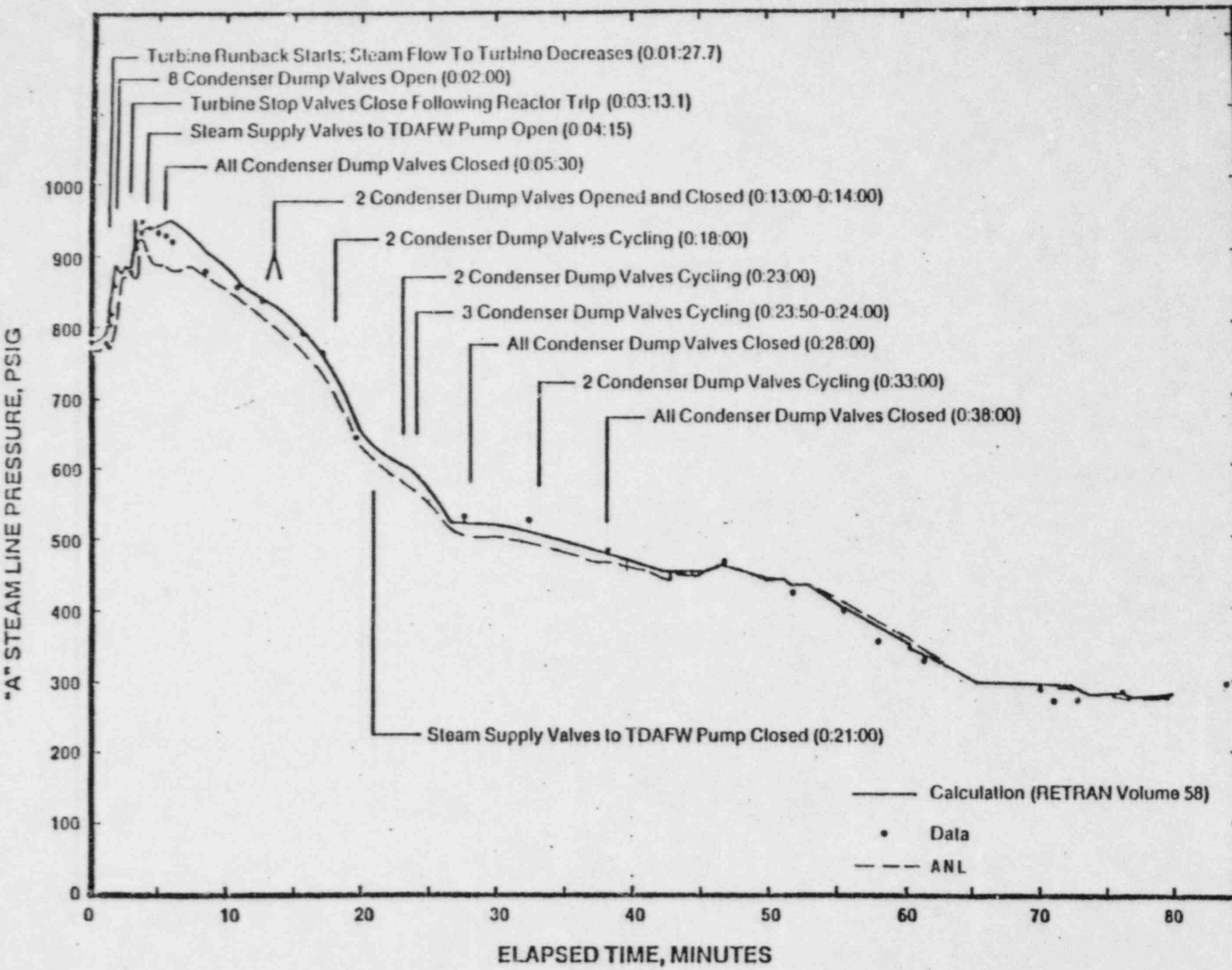
Loop "A" Cold Leg Temperature vs. Time

Figure 3-7



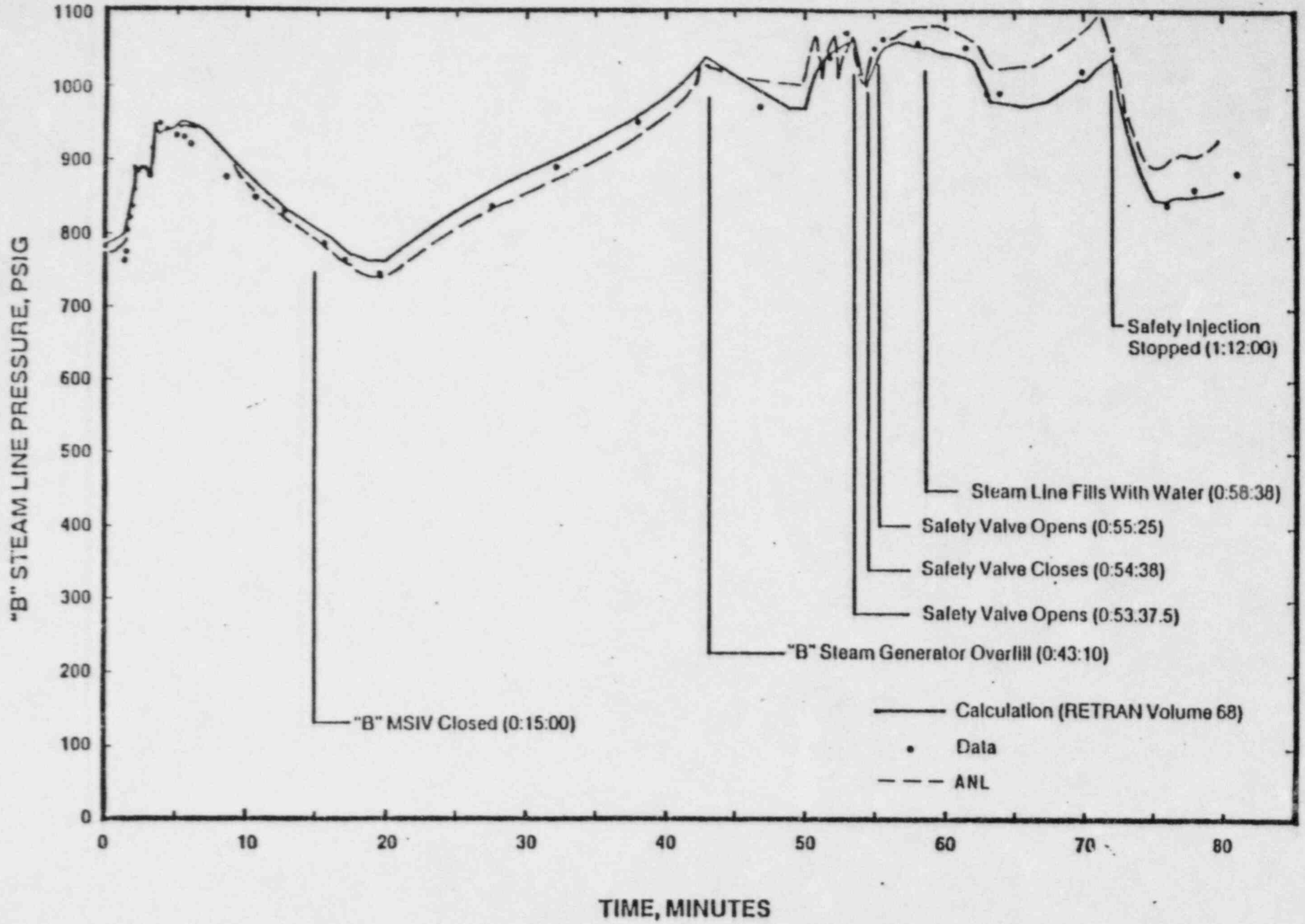
Loop "B" Cold Leg Temperature vs. Time

Figure 3-8



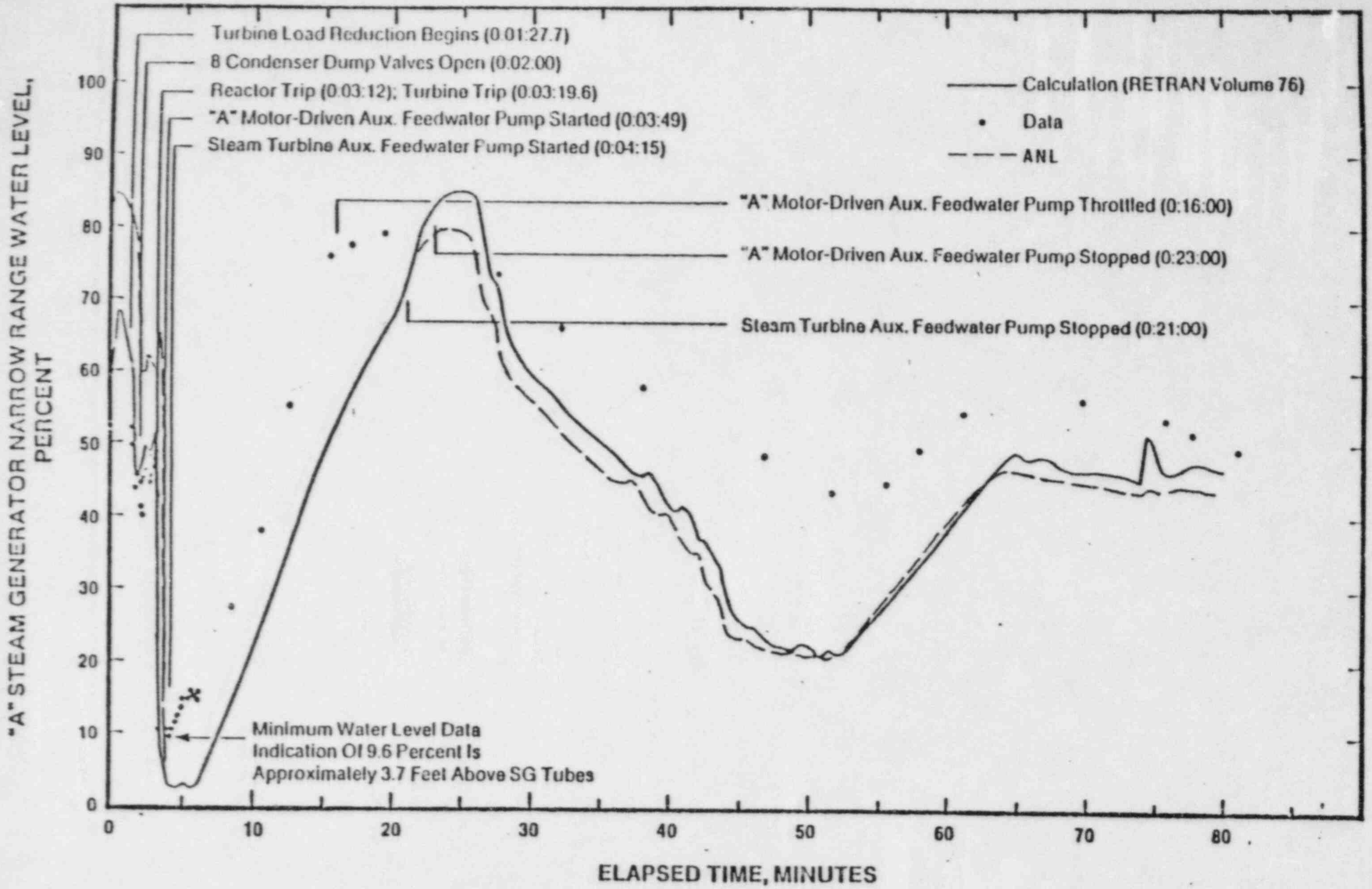
"A" Steam Line Pressure vs. Time

Figure 3-9



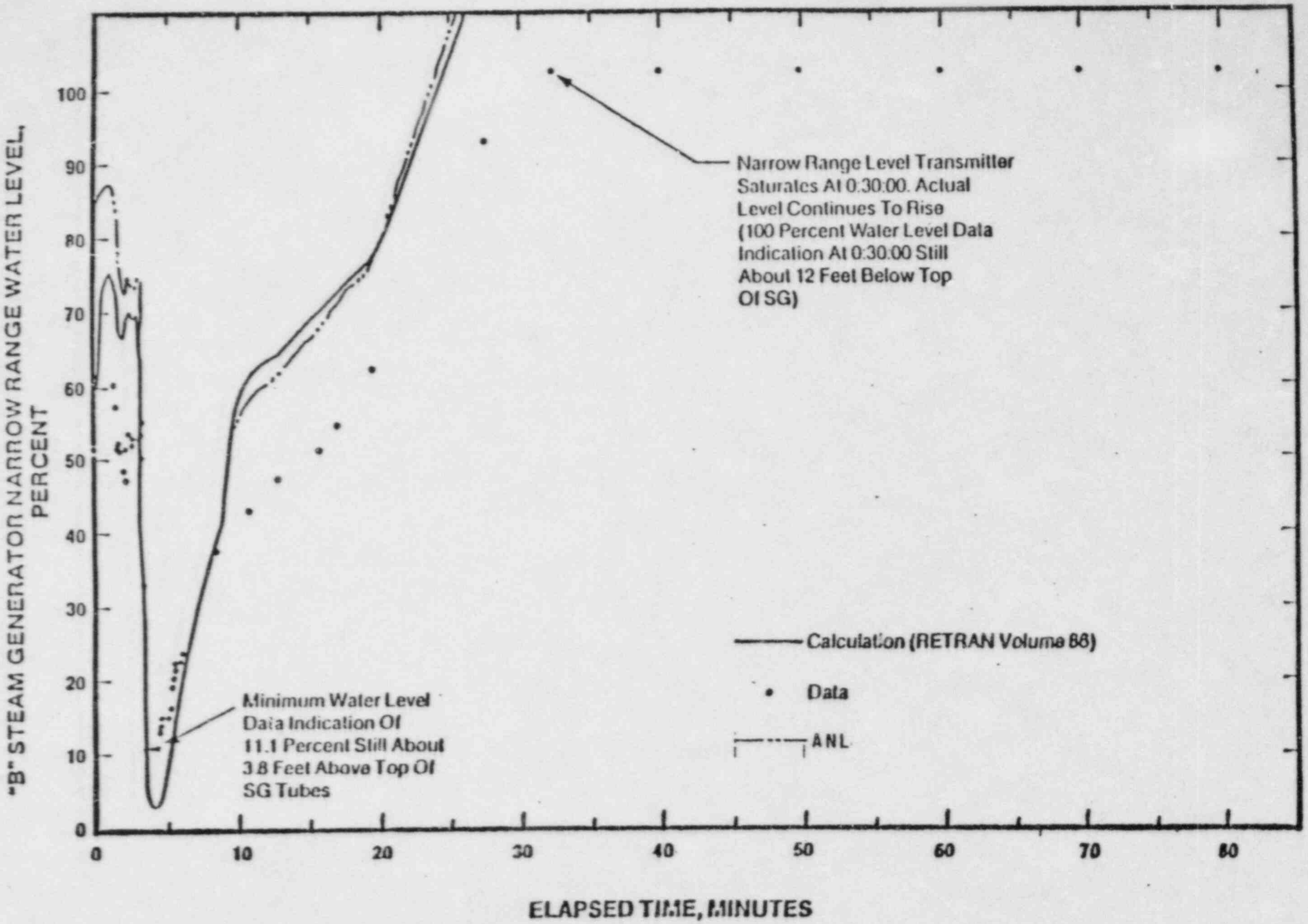
"B" Steam Line Pressure vs. Time

Figure 3-10



"A" Steam Generator Narrow Range Water Level vs. Time

Figure 3-11



"B" Steam Generator Narrow Range Level vs. Time

Figure 3-12

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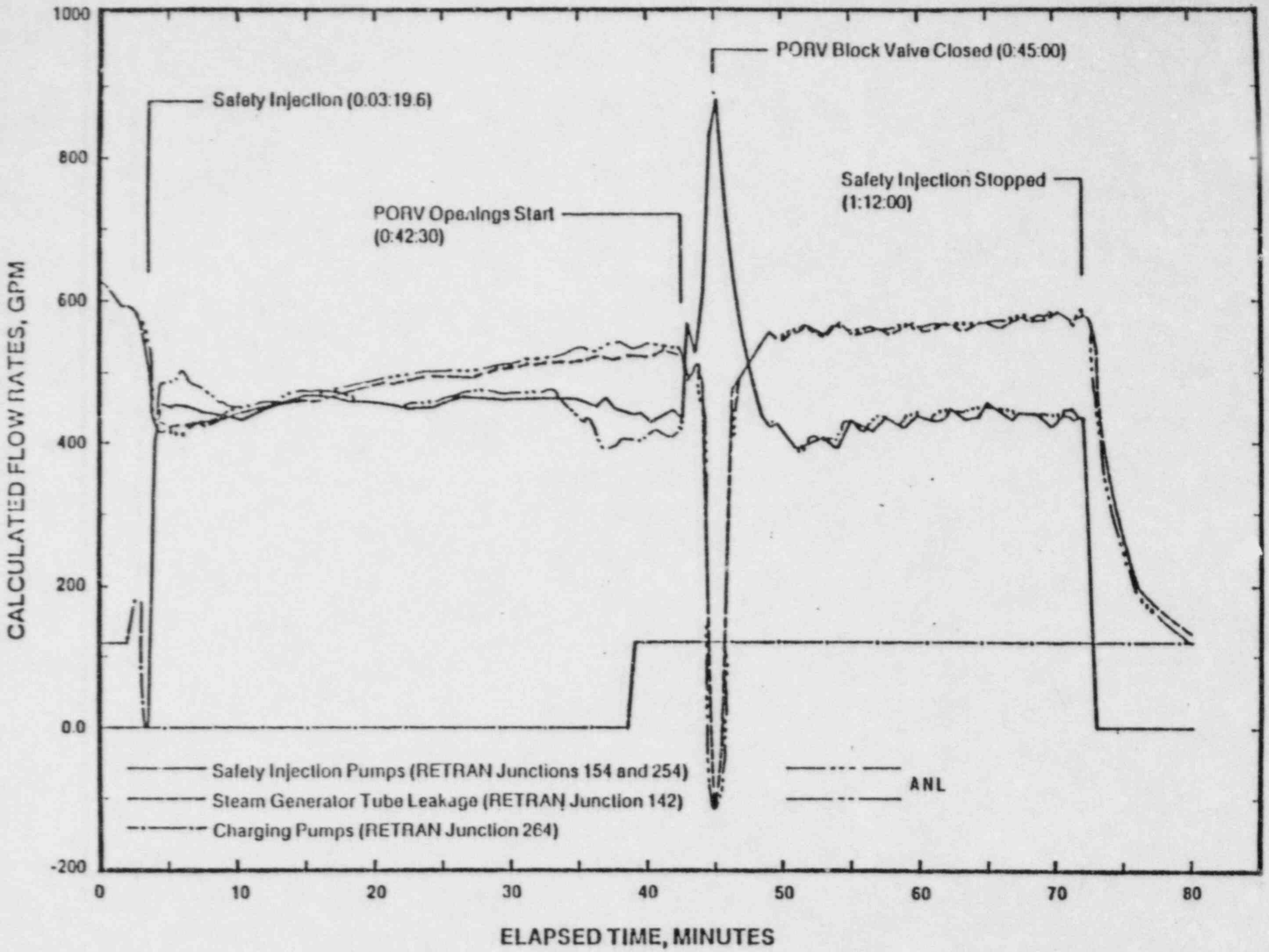
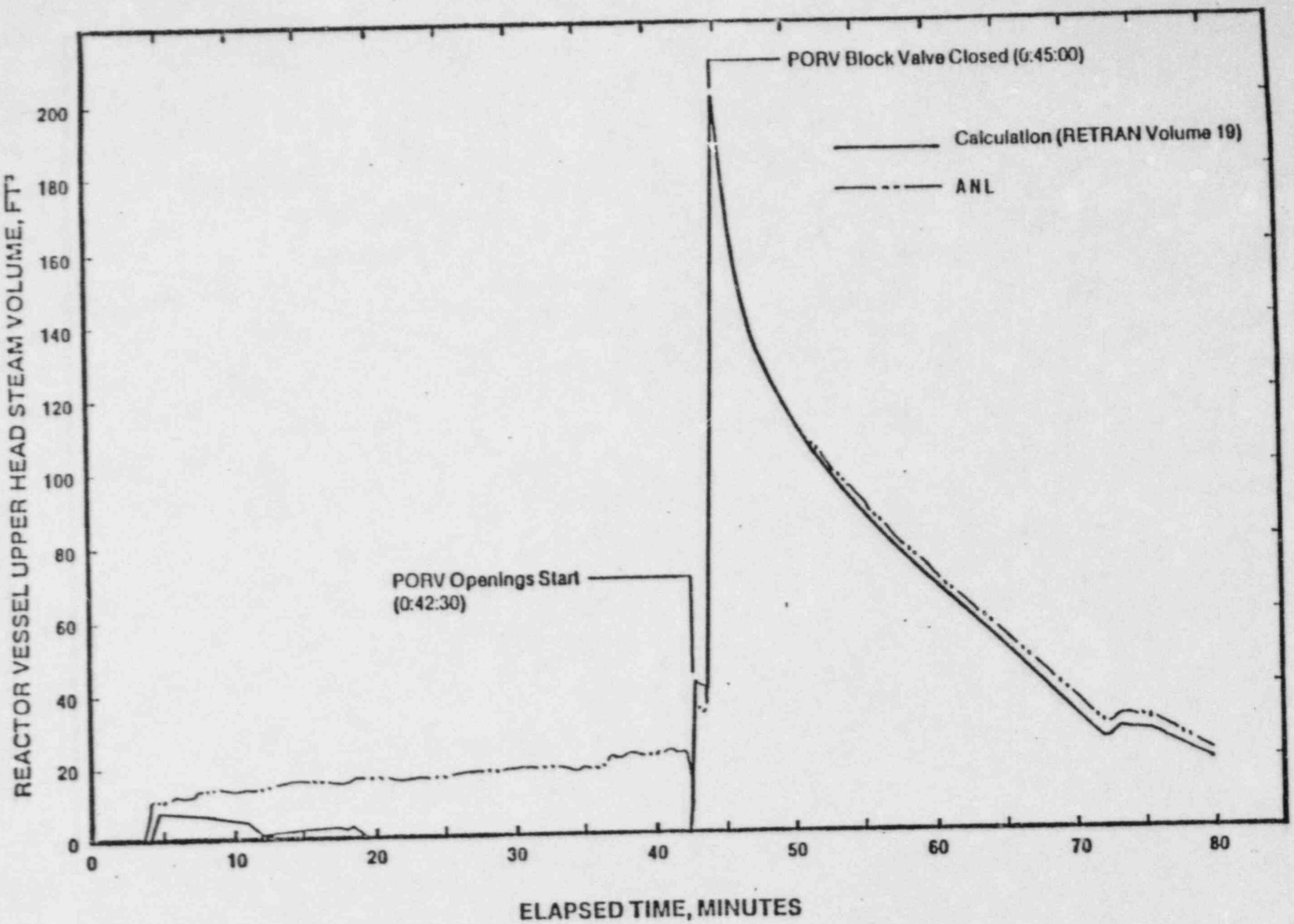
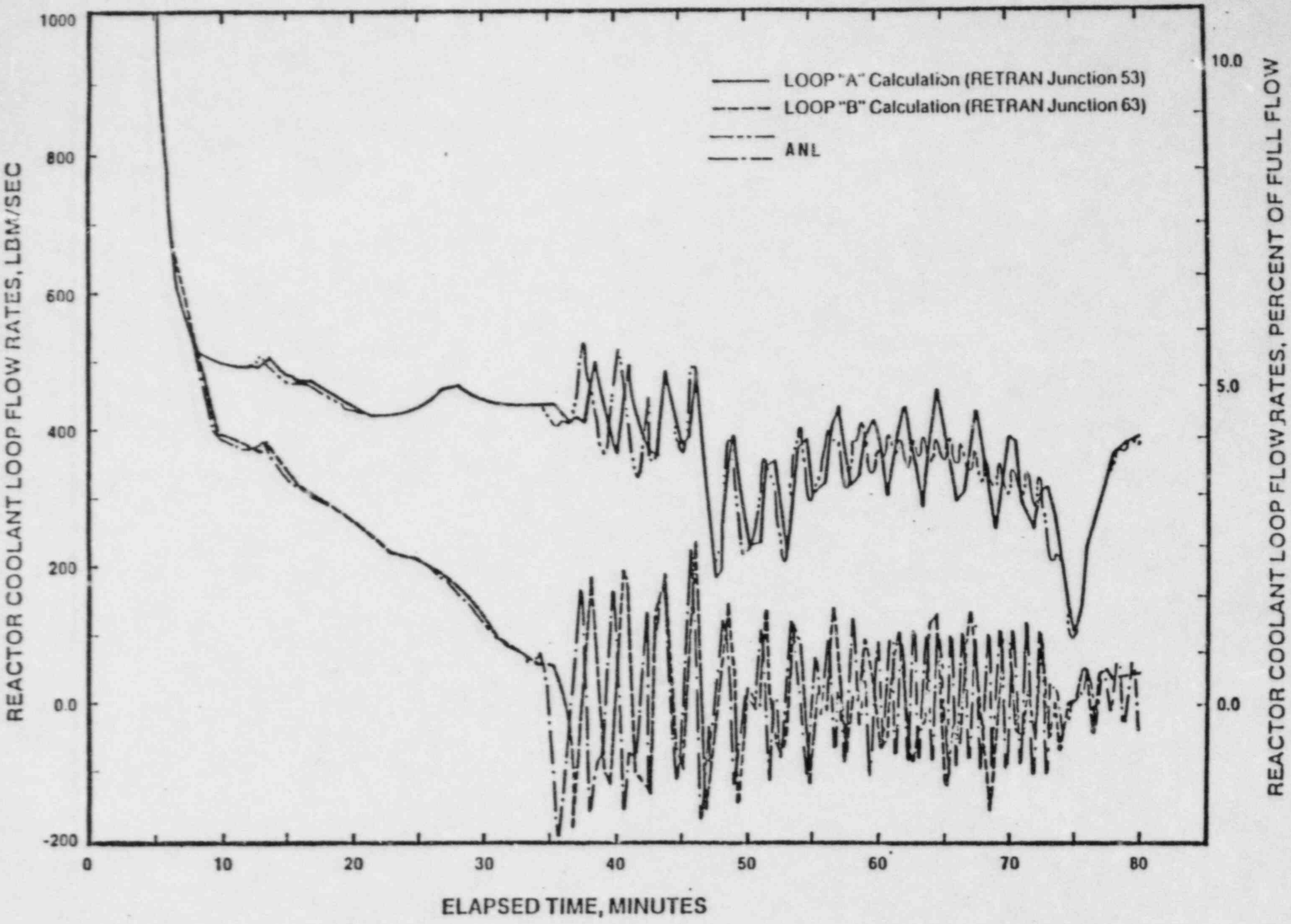


Steam Generator Tube Rupture, Safety Injection, And Charging Flow vs. Time Figure 3-13



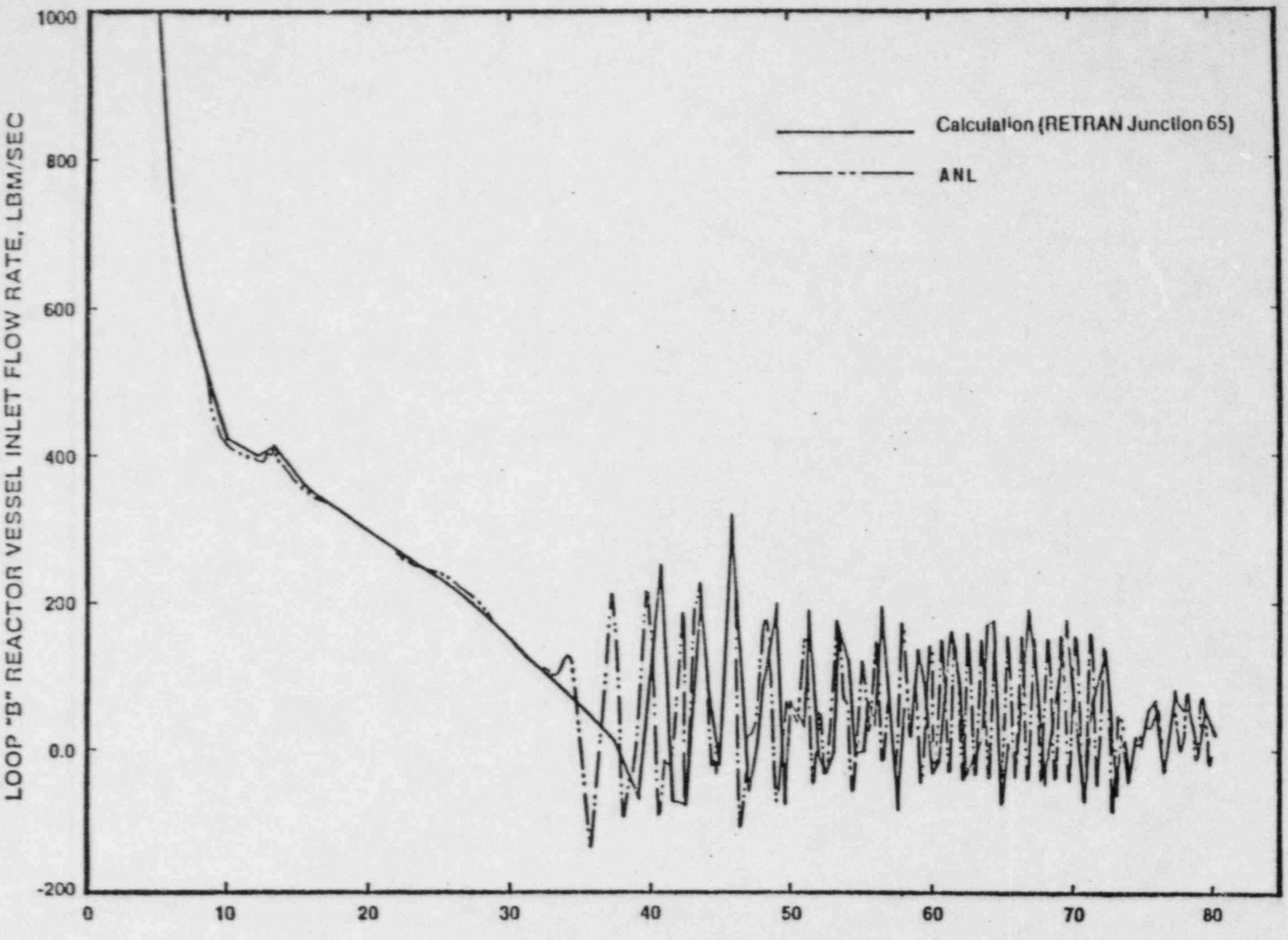
Reactor Vessel Upper Head Steam Volume vs. Time

Figure 3-14



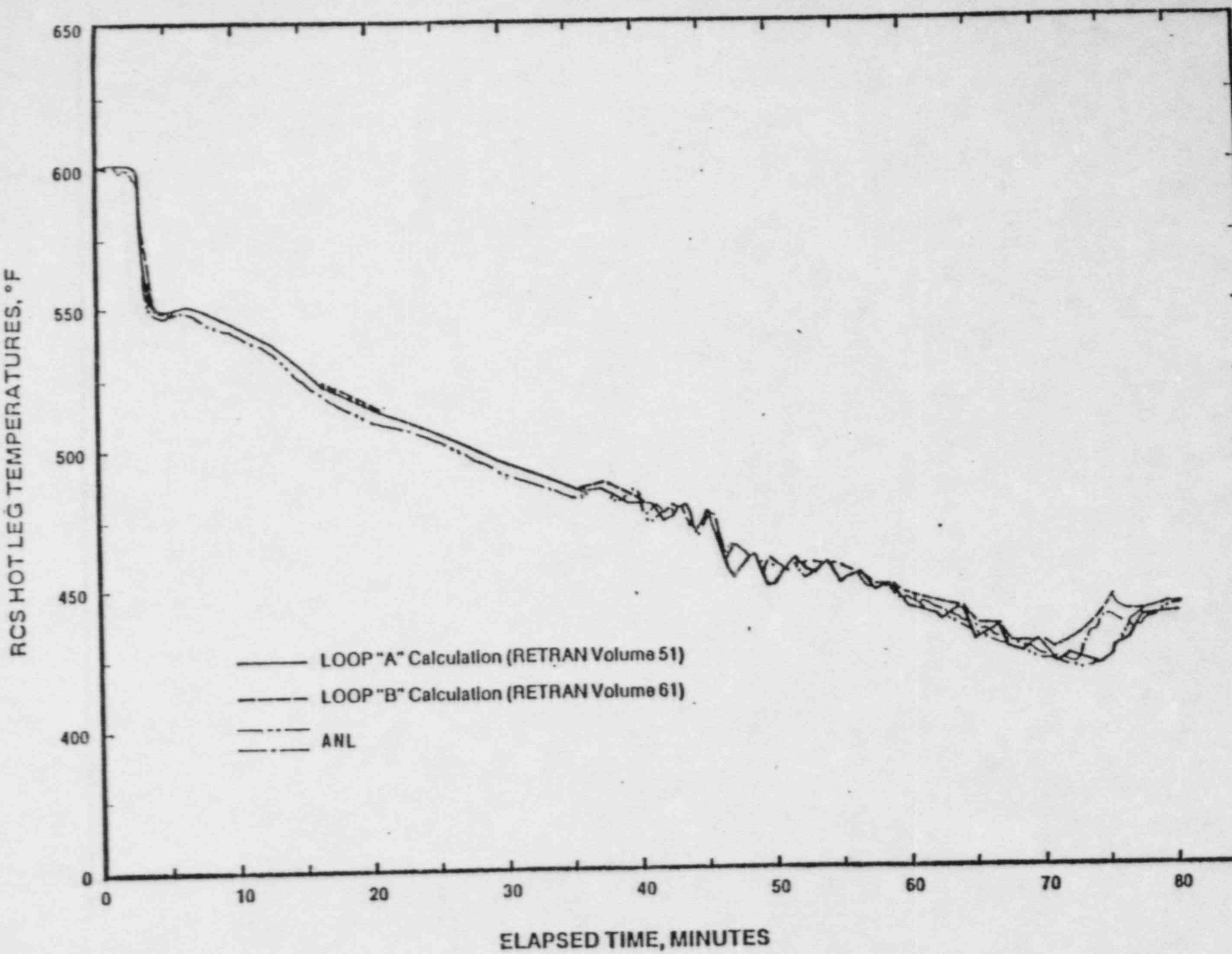
Reactor Coolant Loop Flow Rates vs. Time

Figure 3-15



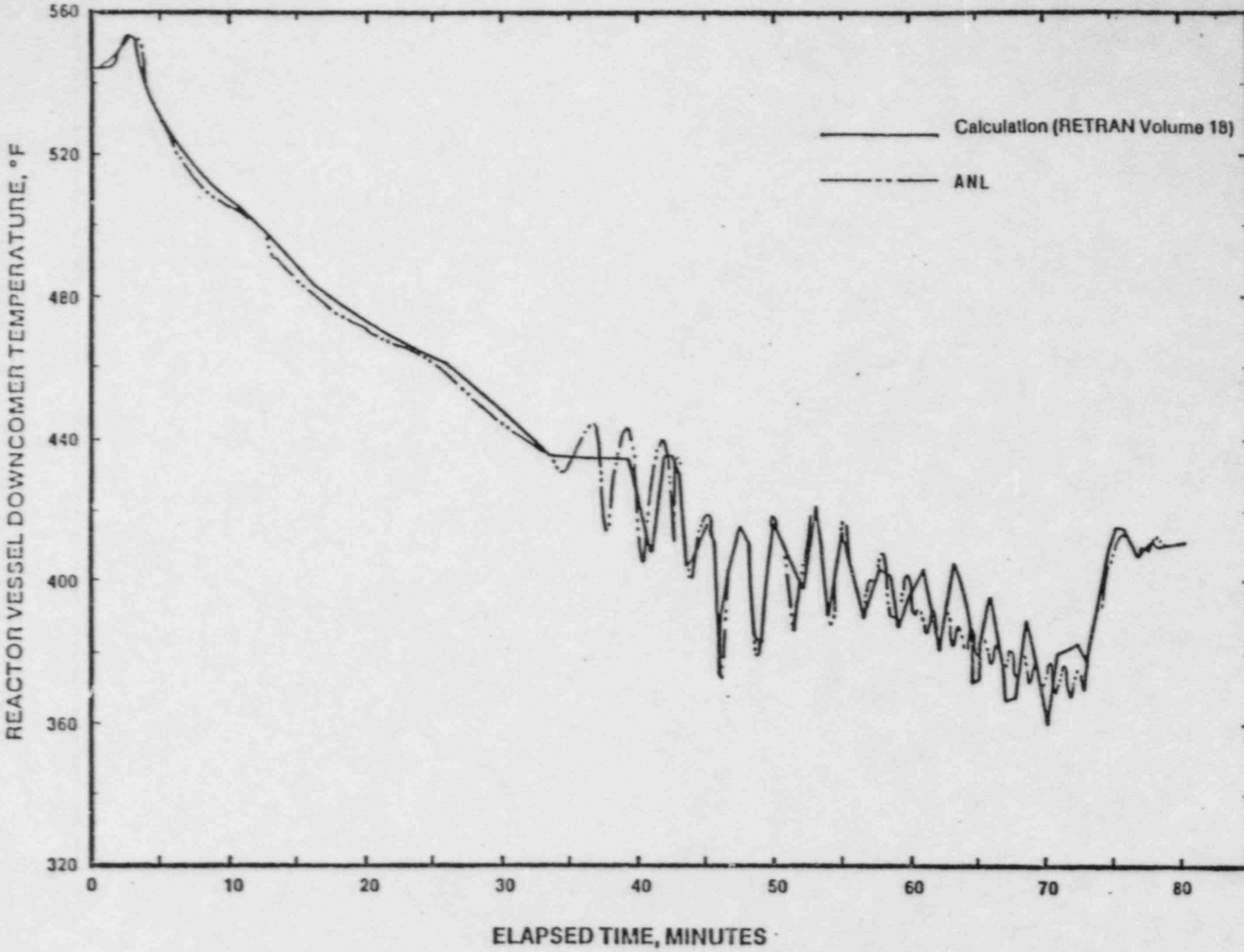
Loop "B" Reactor Vessel Inlet Flow Rate vs. Time

Figure 3-16



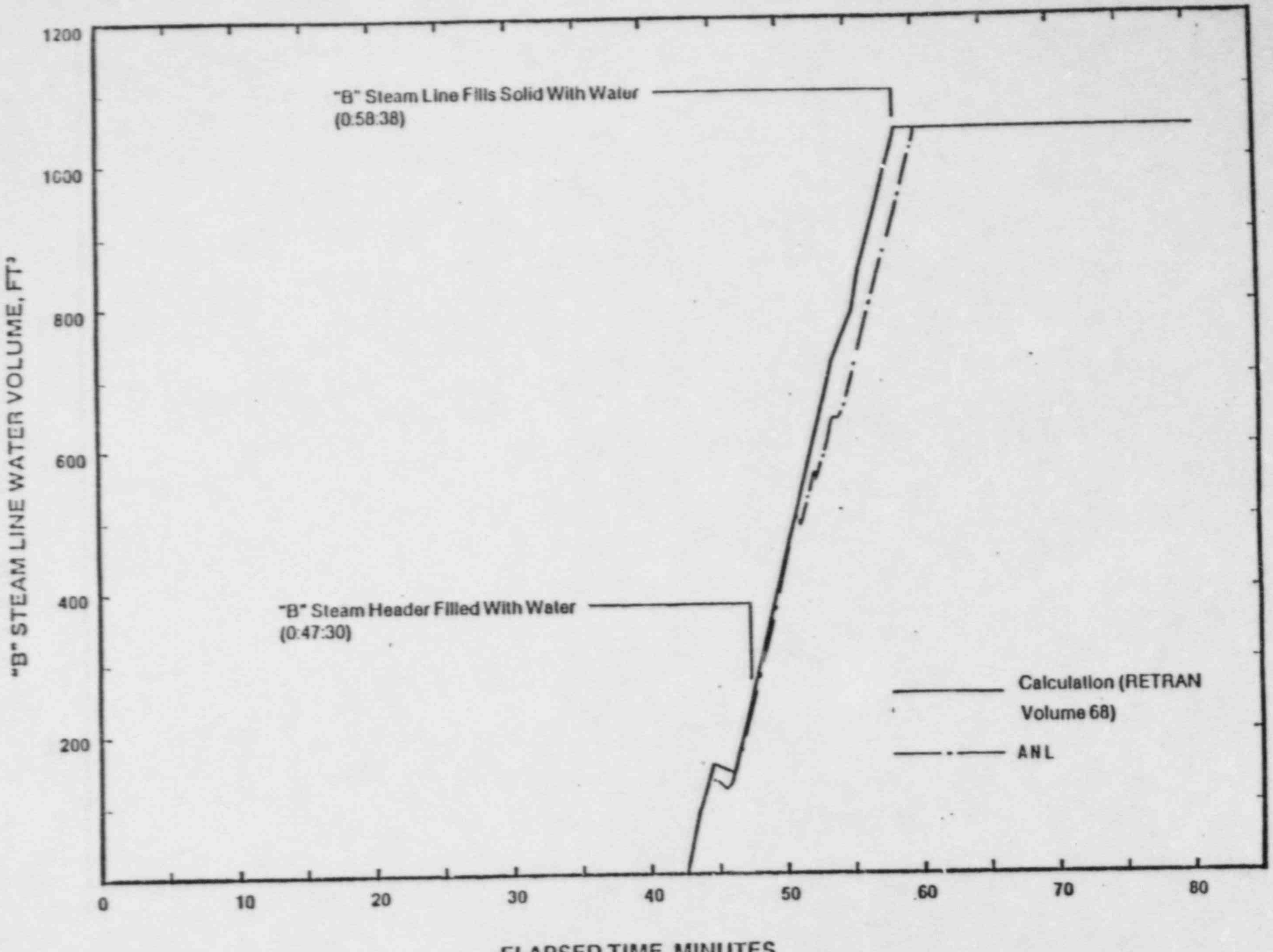
RCS Hot Leg Temperatures vs. Time

Figure 3-17



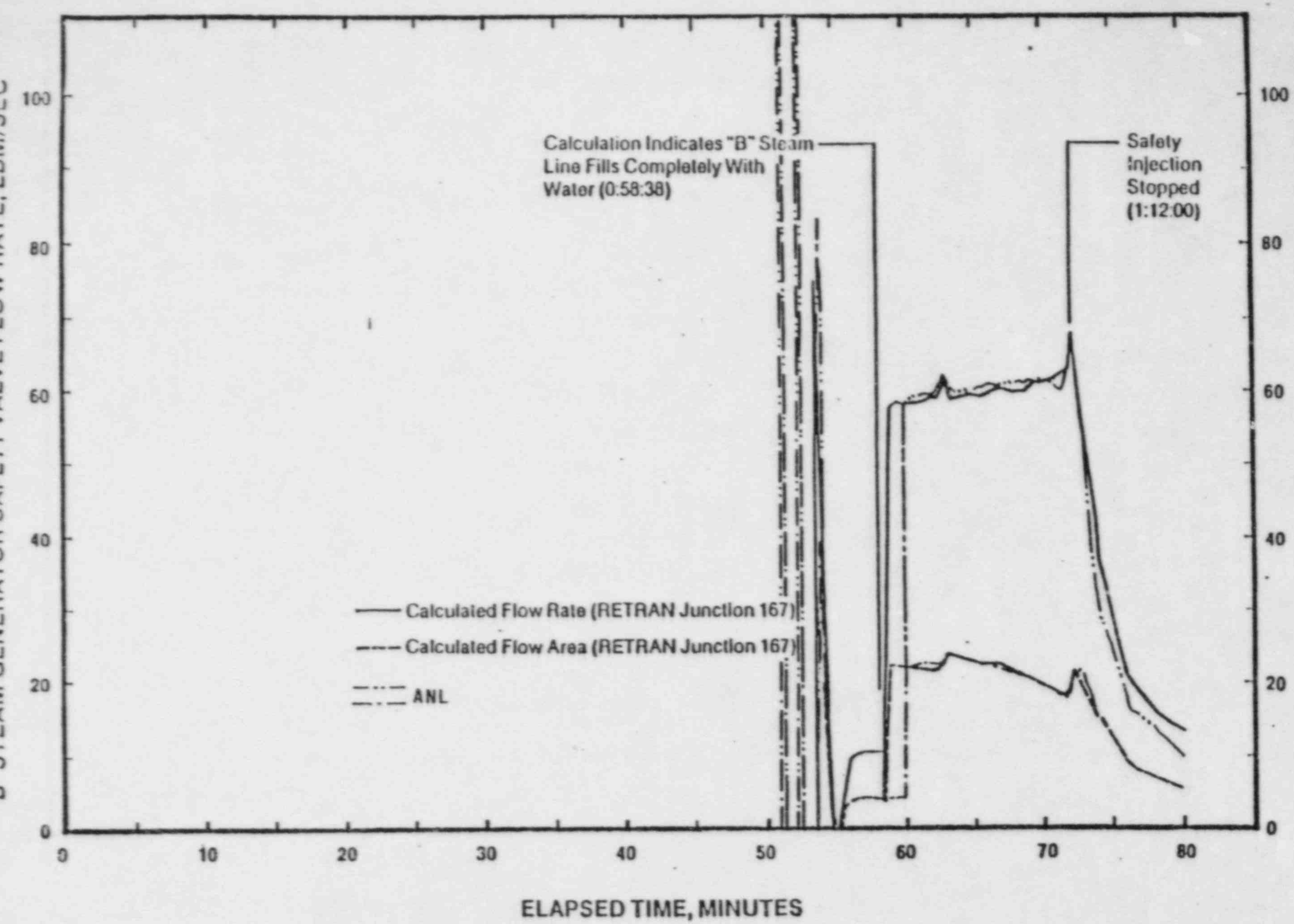
Reactor Vessel Downcomer Temperature vs. Time

Figure 3-18

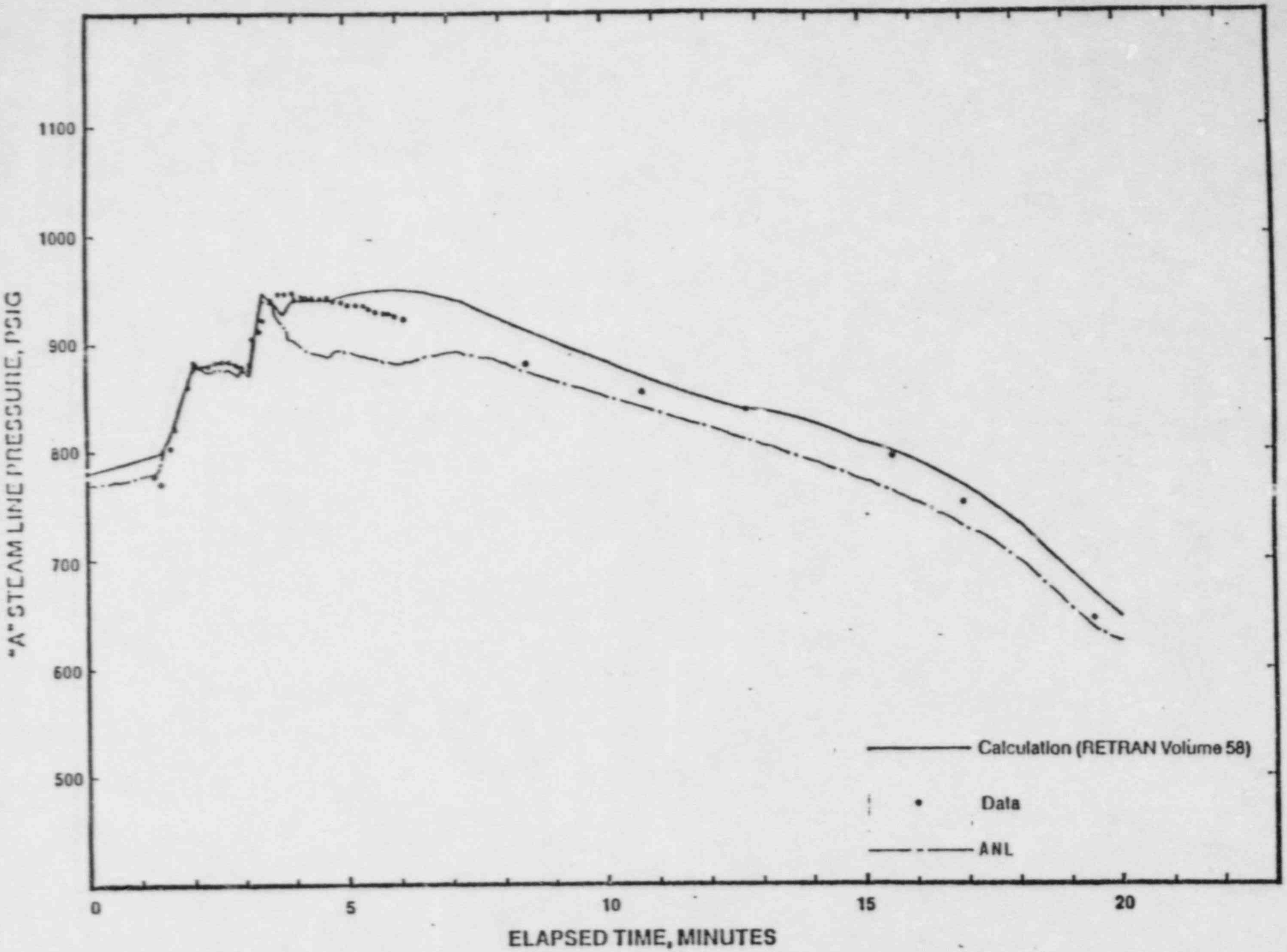


"B" Steam Line Water Volume vs. Time

Figure 3-19

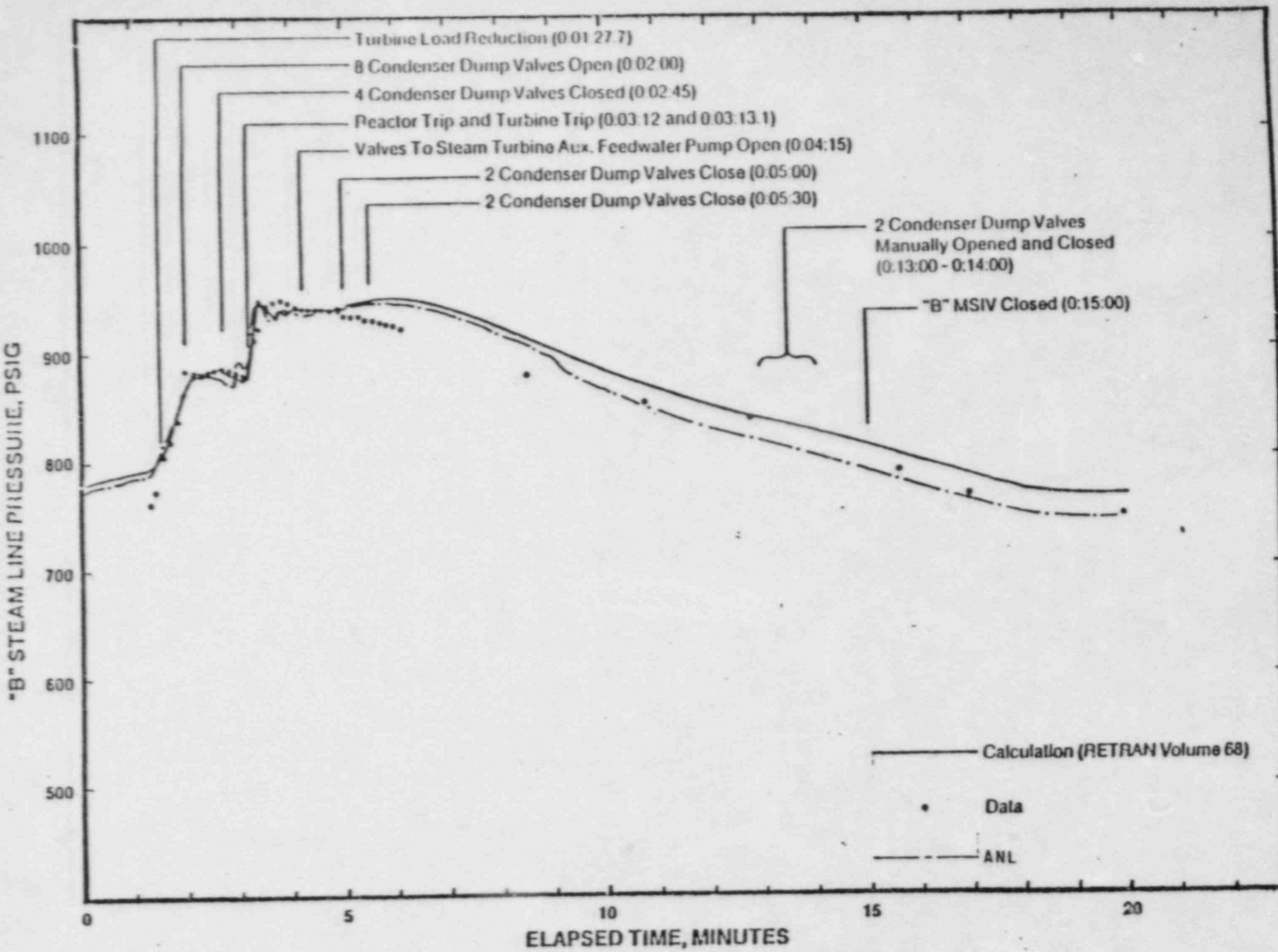


"B" Steam Generator Safety Valve Flow Rate And Flow Area vs. Time   Figure 3-20



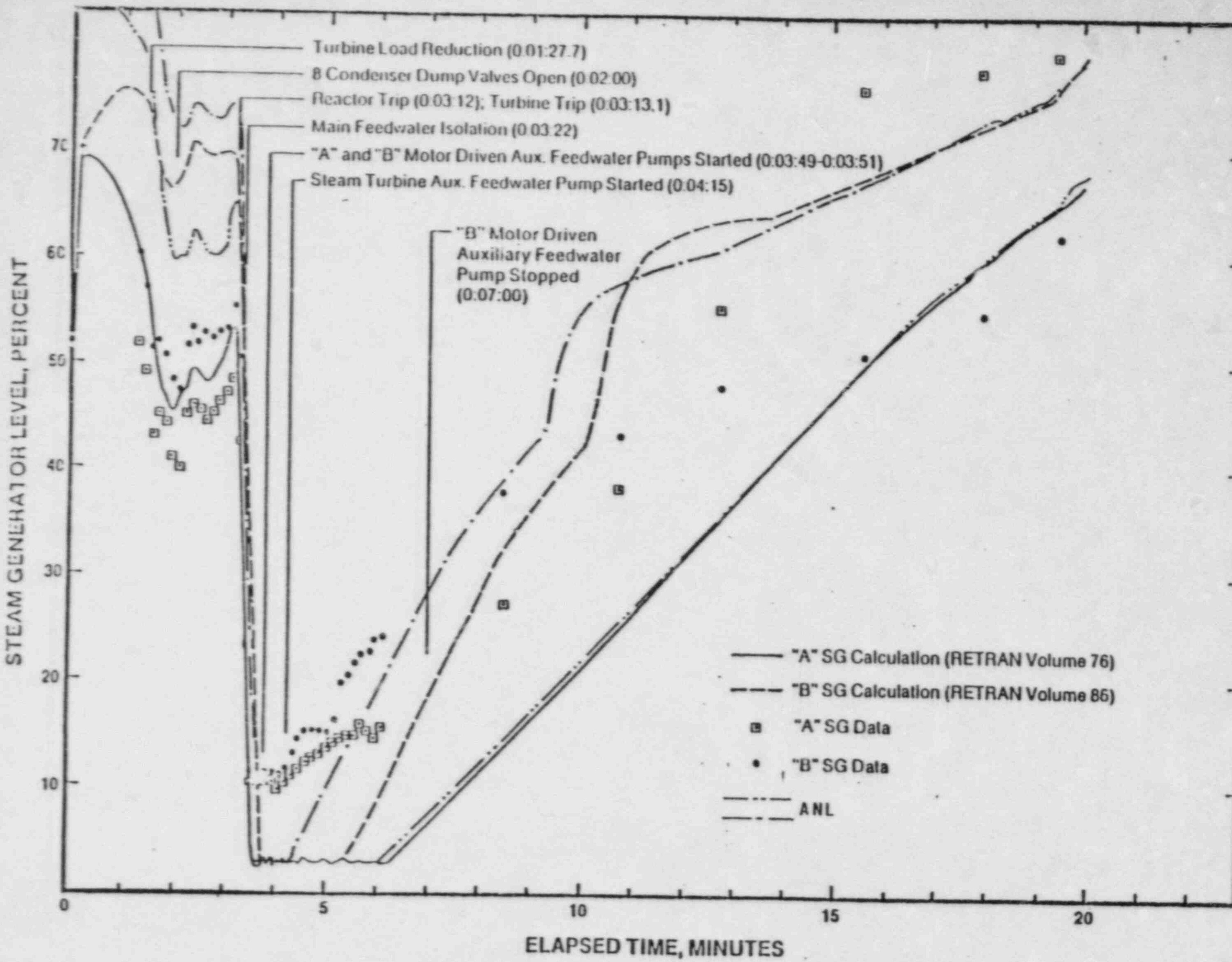
"A" Steam Line Pressure vs. Time (0-20 Minutes)

Figure 2-21



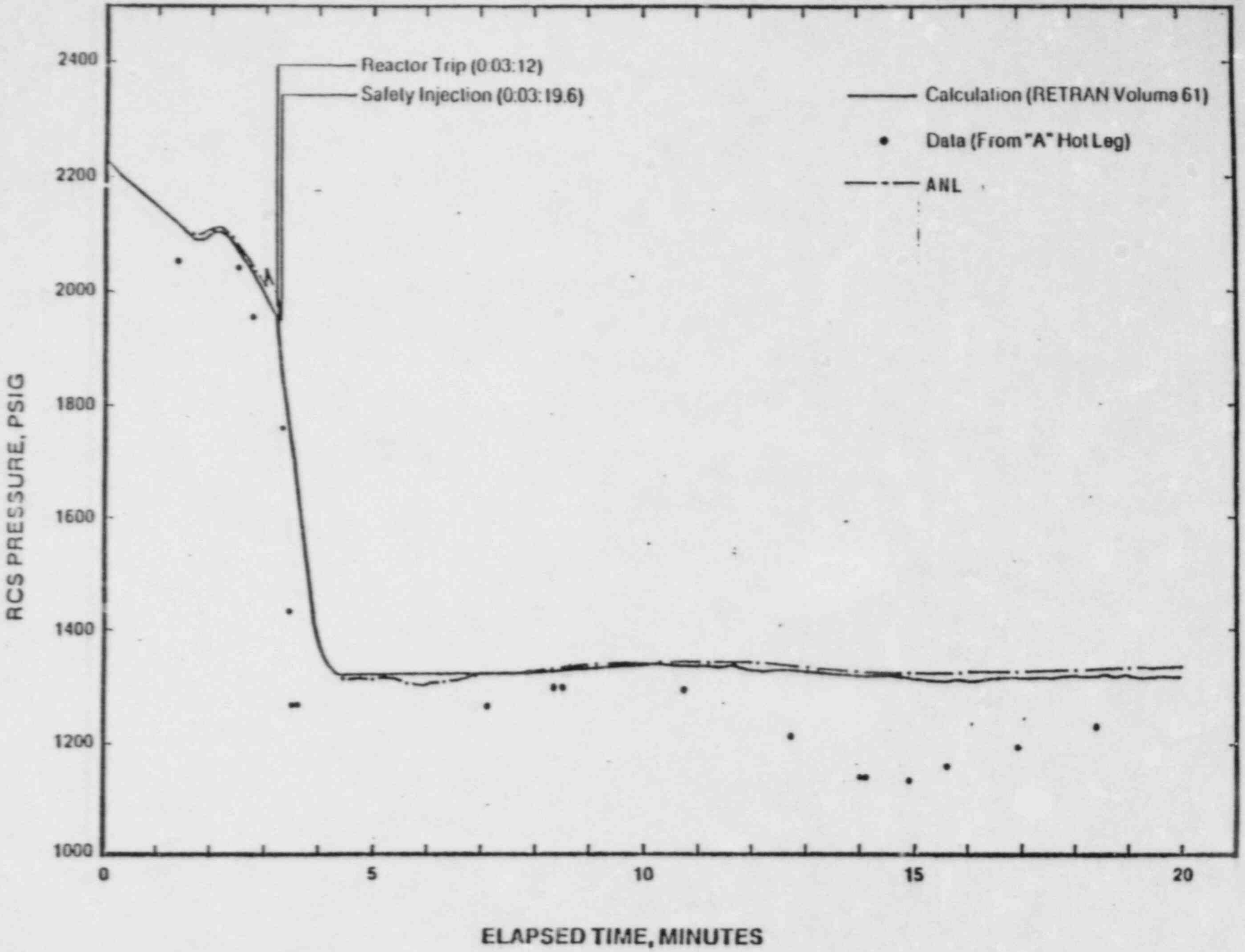
"B" Steam Line Pressure vs. Time (0-20 Minutes)

Figure 3-22



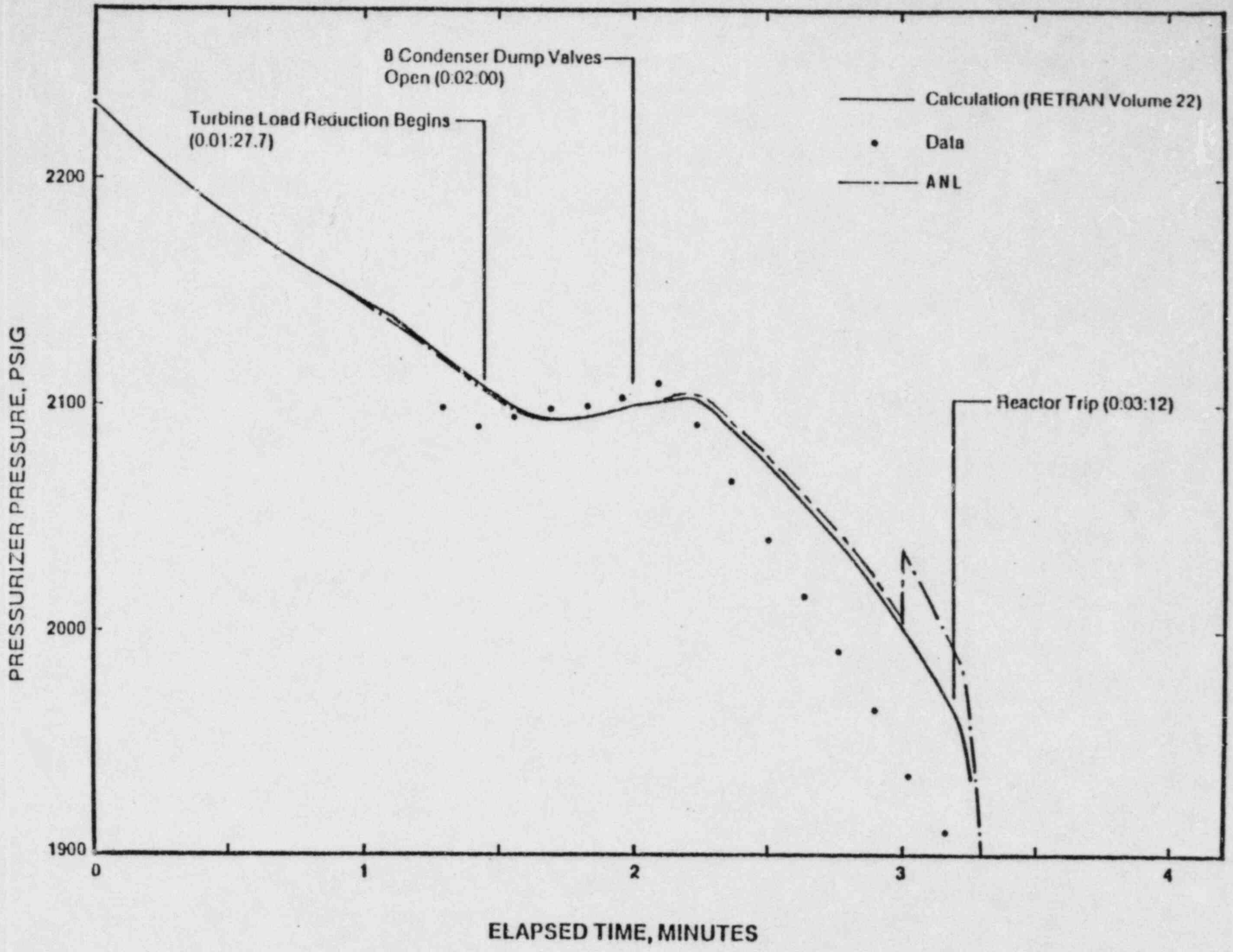
"A" And "B" Steam Generator Levels vs. Time (0 - 20 Minutes)

Figure 3-23



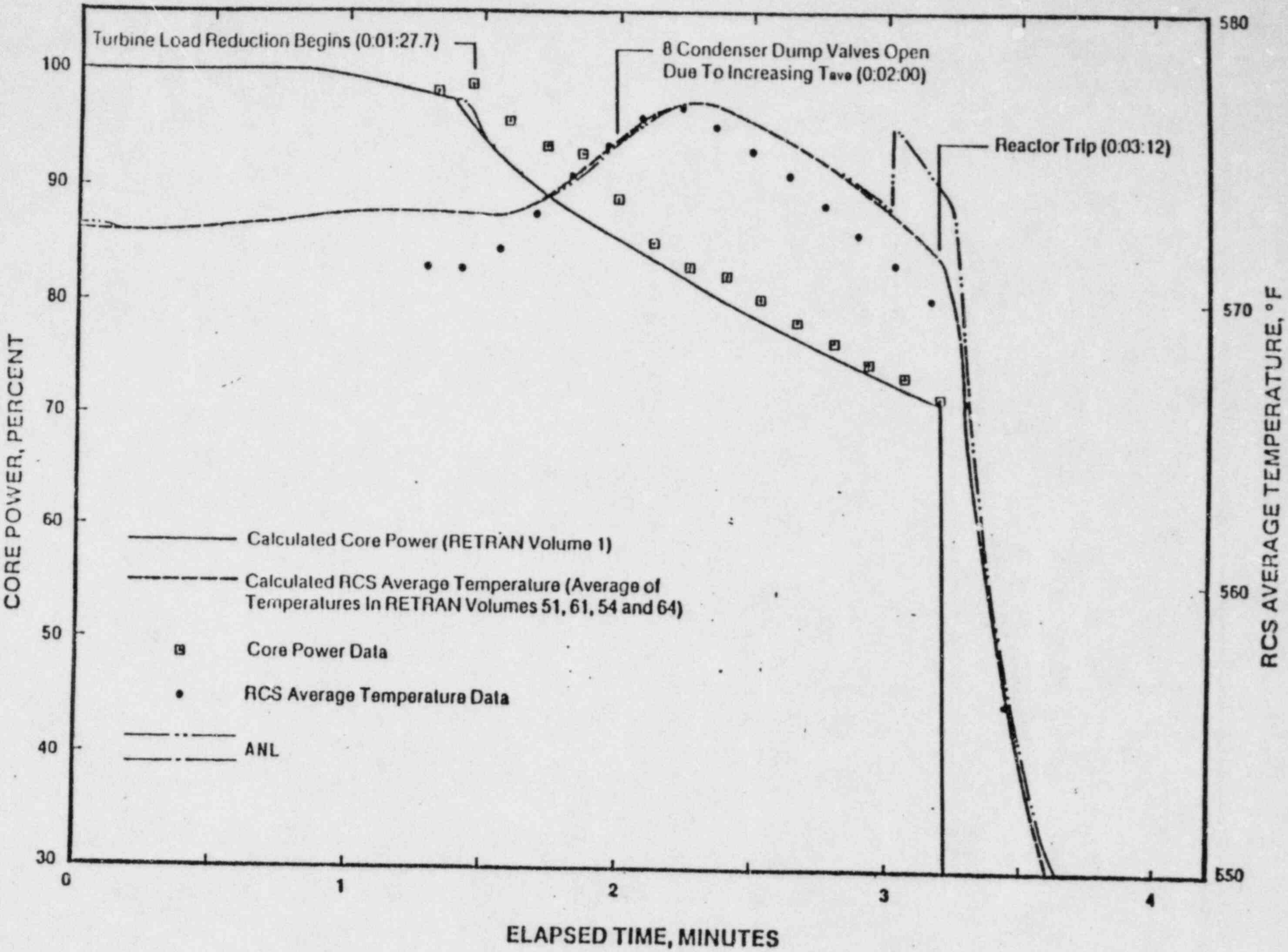
RCS Pressure vs. Time (0 - 20 Minutes)

Figure 3-24



Pressurizer Pressure vs. Time (0 - 4 Minutes)

Figure 3-25



Core Power And RCS Average Temperature vs. Time (0 - 4 Minutes)

4.0 PARAMETRICS

Three parametric analyses were conducted in order to further clarify the understanding of the thermal hydraulic phenomena which occurred during the actual sequence of events. As discussed earlier these are: simulation of the most recent Westinghouse operator guidelines; further mechanical failure as embodied in the assumption of a non-responding block valve; and a variation in operator action with continuation of the SI flow. It should be understood that for the most part the tables/control blocks, made necessary by the INPO calibration to the data obtained from the actual event, were not altered for these parametrics. Section 4.1 details the operator guidelines computation while Sections 4.2 and 4.3 describe the stuck open PORV and SI continuation parametrics, respectively.

4.1 Westinghouse Operator Guidelines

Upon comparison of the actual event sequence of the Ginna SGTR incident to the latest Westinghouse Operator Guidelines for SGTR (E-3, July 5, 1982) it is to be concluded that the operator actions conformed with the E-3 guidelines. However, implementation of certain procedures were delayed enough to negate actions taken to prevent the ruptured steam generator from going solid. For this parametric the event was reanalyzed using the INPO t=0 deck with operator initiated PORV depressurization moved up, by approximately 20 minutes, to 25 minutes after initiation of the transient. Judging from the chronicle of the Ginna event, this scenario should provide sufficient time for operator action and, in addition, all Westinghouse guideline conditions for depressurization had been met at this time. Modifications were made to the deck to follow the latest Westinghouse operator guideline from the point of pressurizer PORV depressurization and continuing on until the calculation was terminated at the time when the operator could energize the pressurizer heaters and reestablish pressure control. No alterations were made to most of the numerous tables/control blocks made necessary by INPO's calibration to the actual Ginna event. Chief among these assumptions during the relevant phase of the incident is the use of upper head "valve" area tables to simulate time dependent form factors. As these areas are held constant during the period of interest to the parametric considered here, the utilization of the tables

could be justified on the grounds of simplification of a multidimensional geometrical problem. In addition, no modifications were made to the tables/control blocks for the behavior of the A steam generator (intact generator) which implies that the procedure for utlizing the intact generator as a heat sink follows that of the Ginna event exactly. Modifications, however, had to be made to the pressurizer heater control blocks.

Figures 4.1-1 to 4.1-7 illustrate the system response. While the transient is plotted for the period from 1000 seconds after tube rupture on, attention should be focused upon the PORV depressurization and post-PORV depressurization period, namely from 1500 seconds to 2400 seconds. The event sequence of the period prior to 1410 seconds corresponds exactly to that of the actual Ginna incident. Table 4.1-1 shows the timing of the various operator actions required by the guidelines.

Table 4.1-1. Operator Actions

Event	Time (secs)
Charging on	1410
Open pressurizer PORV	1500
Cycle PORV	Same sequence as in actual Ginna incident. Just moved initiation up to 1500 seconds.
SI flow termination	1720-1765 (45 second ramp)
Letdown established	1850

The reactor coolant system (RCS) pressure trace, Fig. 4.1-1, shows the four operator initiated pressurizer PORV openings, the PORV block valve closing and the termination of the safety injection. Figure 4.1-2 depicts the collapse of the upper head bubble with the closure of the PORV block valve. Volume 20, the volume directly below volume 19 and physically a part of the reactor vessel upper head region, also undergoes flashing and further complete steam bubbles collapse during this period. This volume was modeled as a homogeneous volume by INPNO. As volume 19 does not completely empty, this is a physically consistent picture. The narrow range pressurizer level instrumentation (uncorrected for non-saturation conditions), Fig. 4.1-3, shows a refilling to ~ 65% where a leveling off takes place. At this point the operator could re-energize the heaters and reestablish pressure control. From Figs. 4.1-1 and 4.1-4 it can be seen that a quasi-steady state has been reached with the tube leakage flow now of a negligible proportion and the difference in primary to secondary pressure attributable mainly to the hydrostatic head of a ruptured steam generator filled with substantially more liquid than at normal operating conditions. However, Fig. 4.1-5 which is the narrow range B generator level, shows that there is considerable margin to filling up the ruptured generator. In the context of filling the steam dome, a narrow-range measurement of ~ 210% is to be considered as full. The core exit temperature of Fig. 4.1-6 shows that the degree of subcooling has decreased to 10°F, but that is due to the use of the actual operator actions for steam generator A (intact generator) during the actual event. Figure 4.1-7 depicts the A generator pressure which is an indicator of the generator behavior and can be compared with Fig. 3-9.

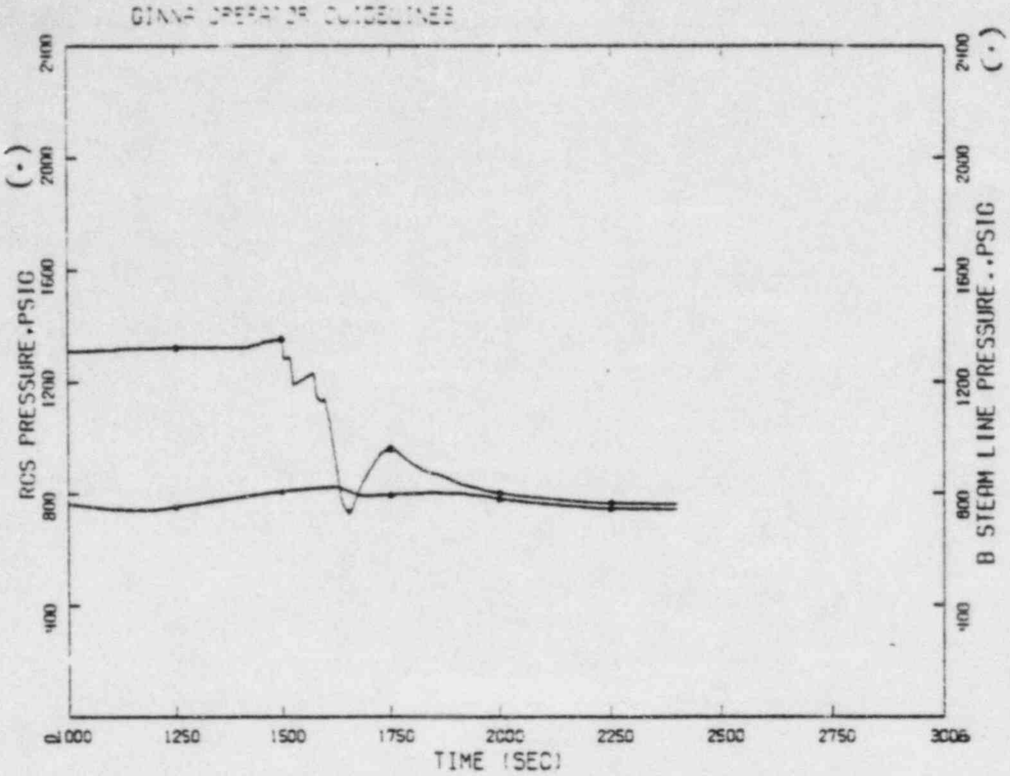


Figure 4.1-1. RCS Vol. 61 Pressure/SGB Vol. 68 Pressure.

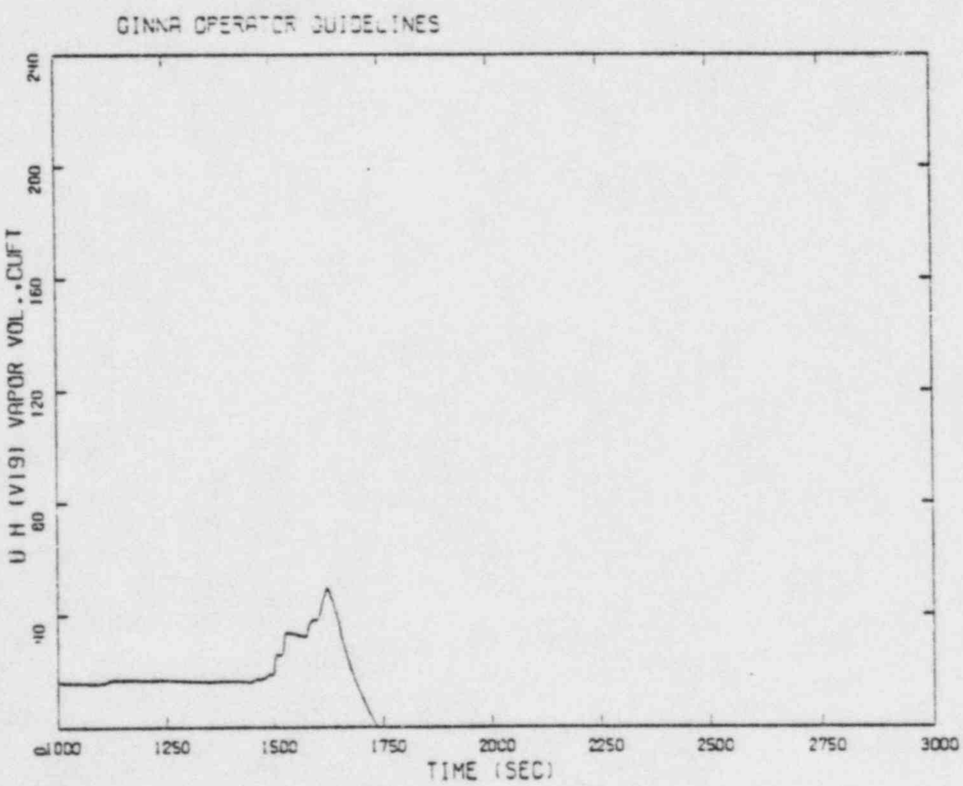


Figure 4.1-2. Upper Head Vol. 19 Vapor Volume.

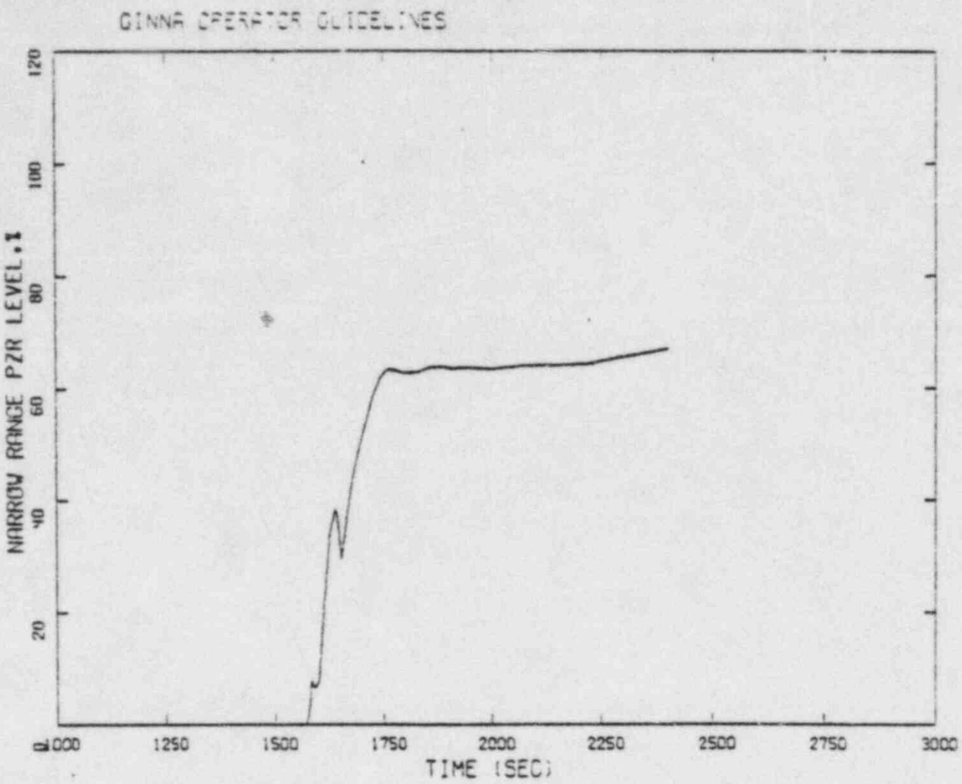


Figure 4.1-3. Narrow Range Pressurizer Level.

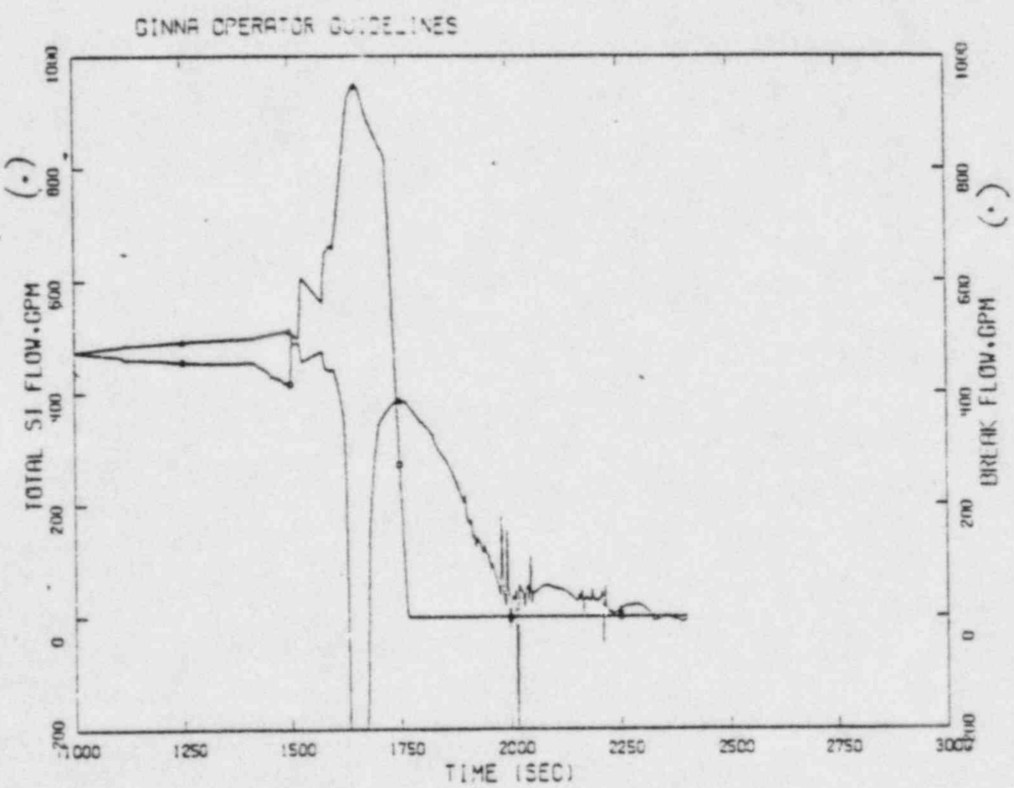


Figure 4.1-4. SI Flow/Break Flow.

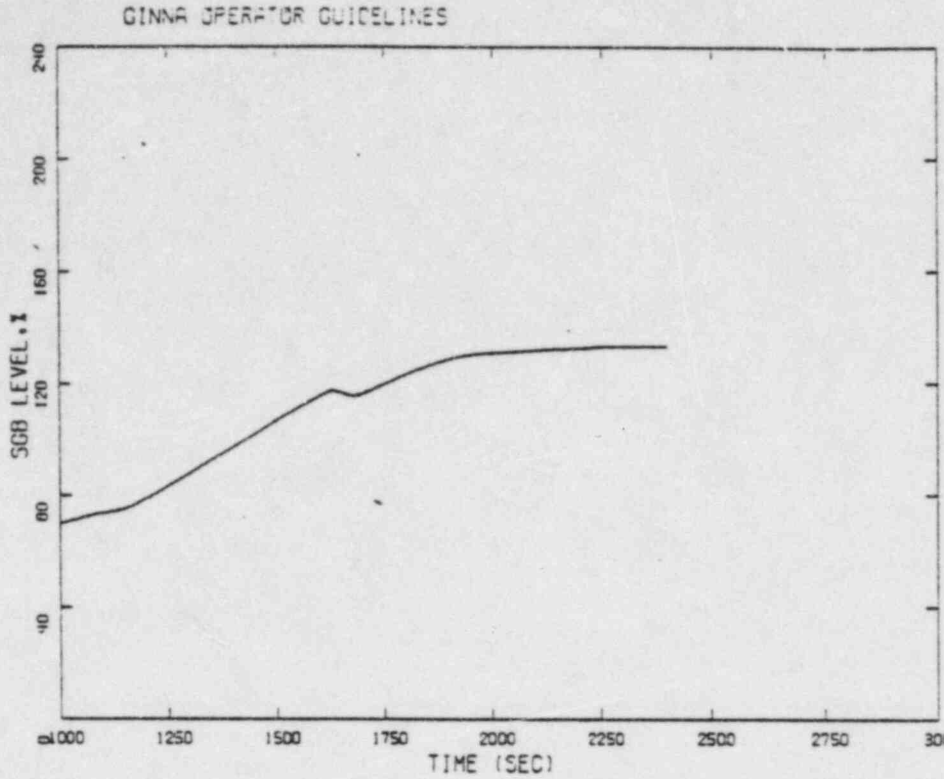


Figure 4.1-5. SGB Level.

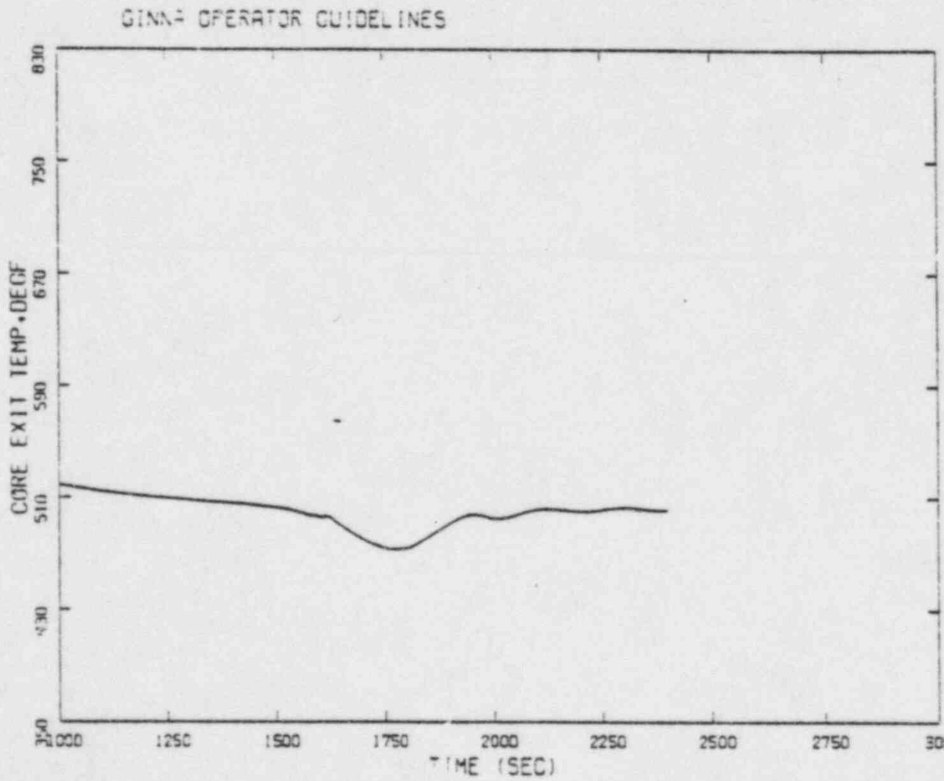


Figure 4.1-6. Core Exit Vol. 13 Temperature.

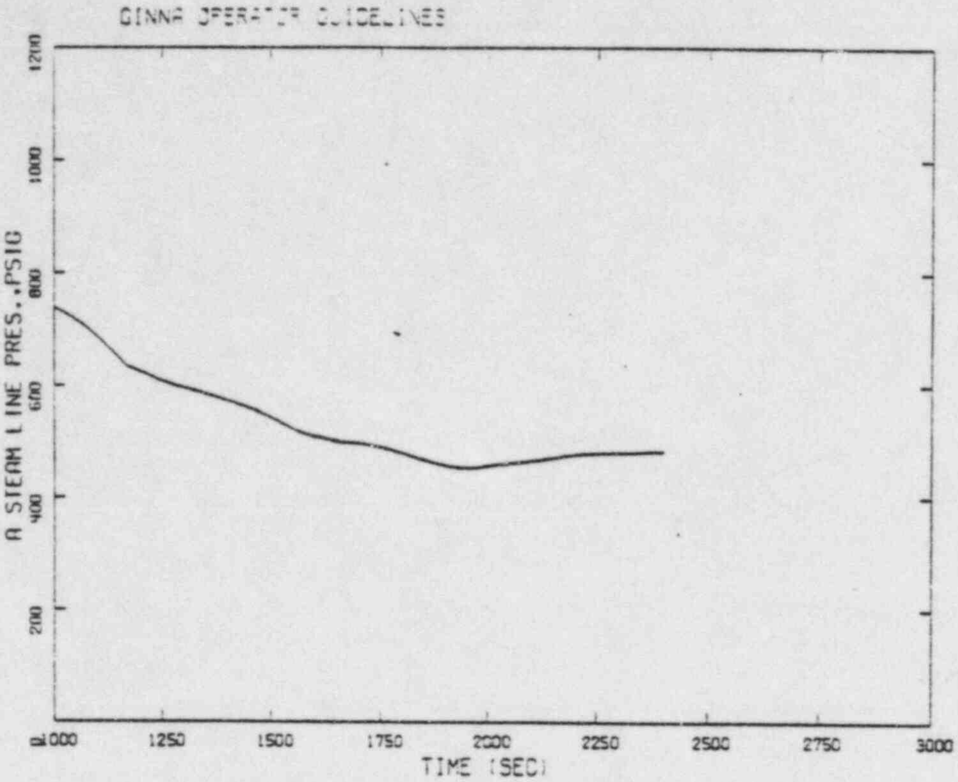


Figure 4.1-7. SGA Vol. 58 Pressure.

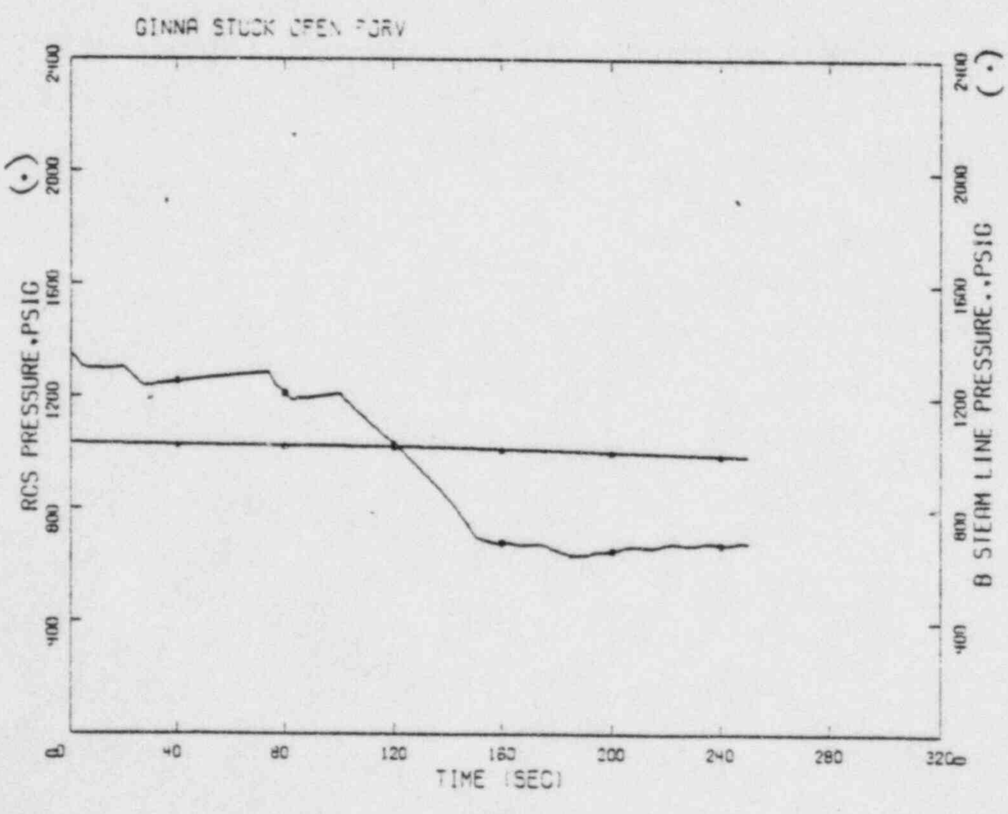


Figure 4.2-1. RCS Vol. 61 Pressure/SGB Vol. 68 Pressure.

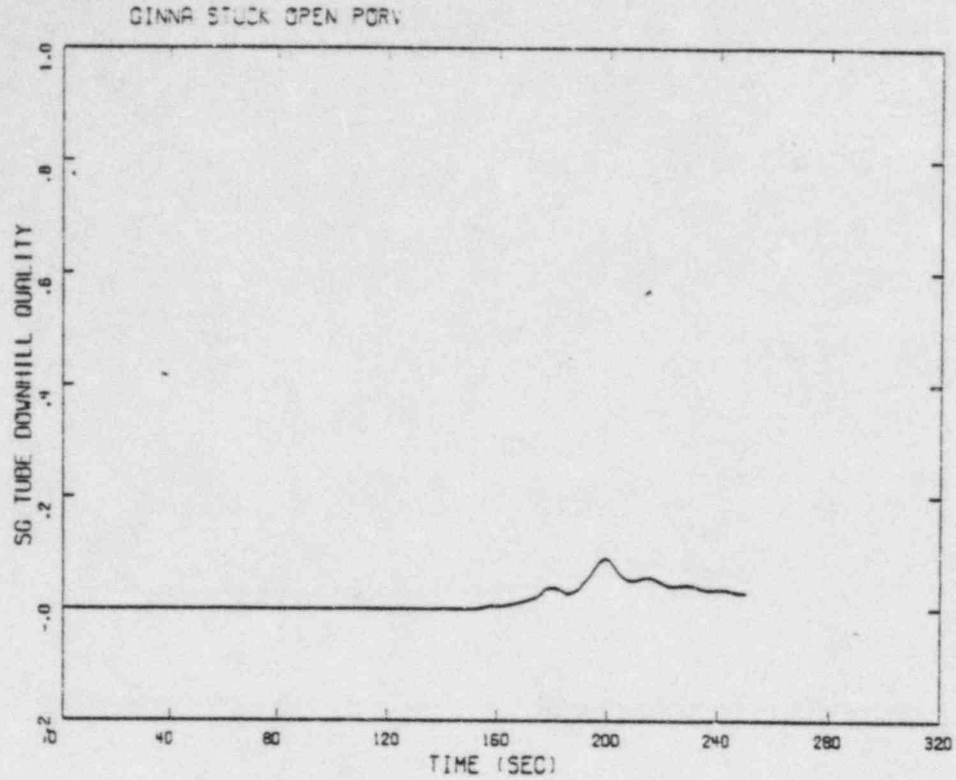


Figure 4.2-2. SG Tube Downhill Vol. 44 Quality.

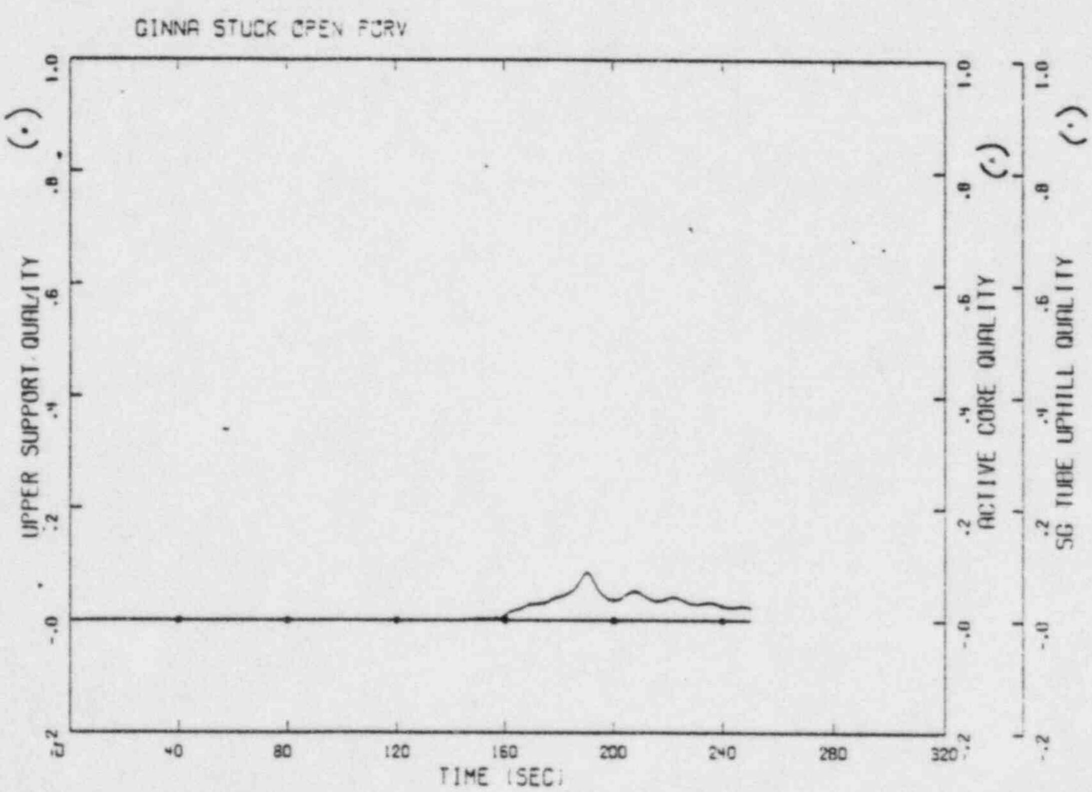


Figure 4.2-3. Upper Support Vol. 20/Active Core Vol. 1/SG Tube, Vol. 43 Quality.

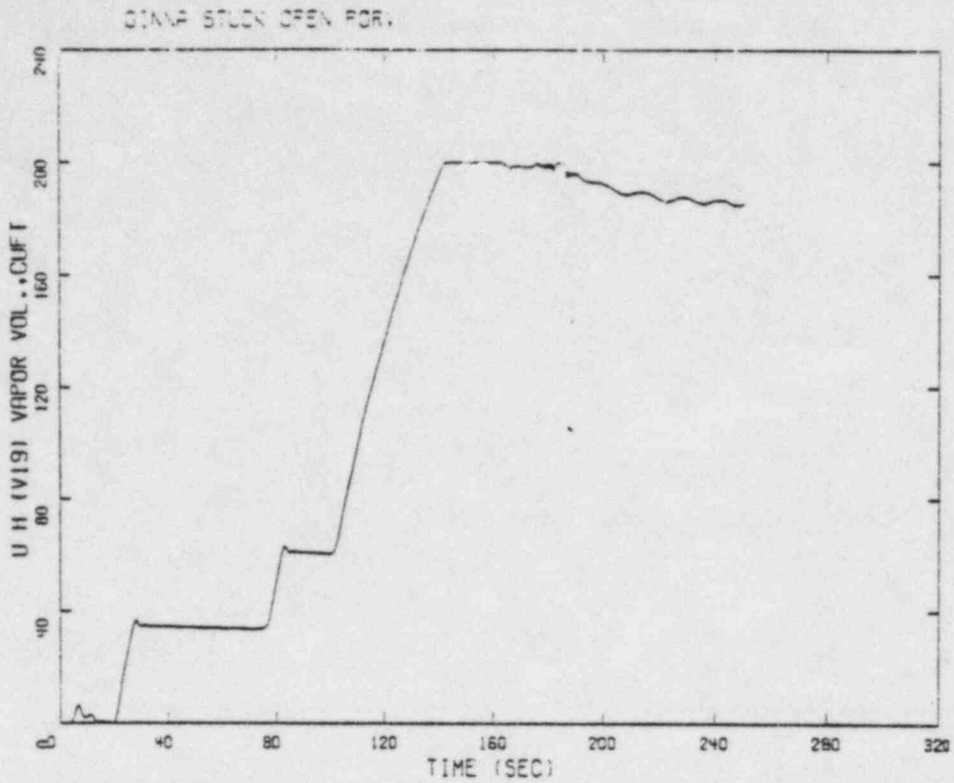


Figure 4.2-4. Upper Head Vol. 19 Vapor Volume.

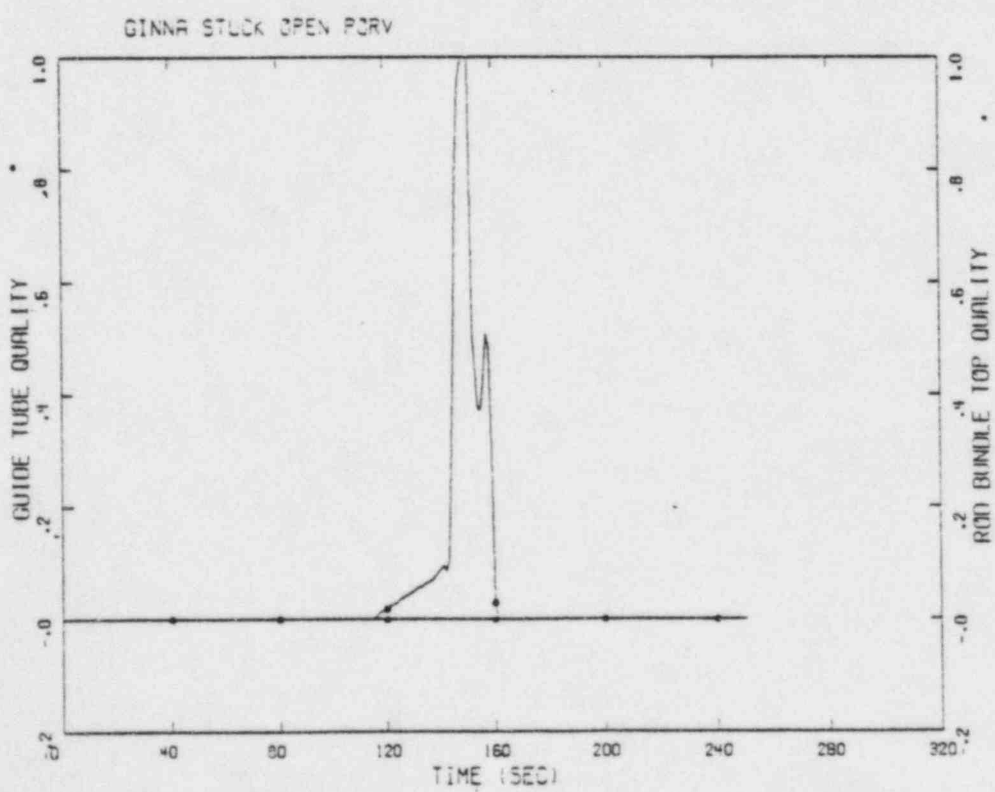


Figure 4.2-5. Guide Tube Vol. 14/Bundle Top Vol. 15 Quality.

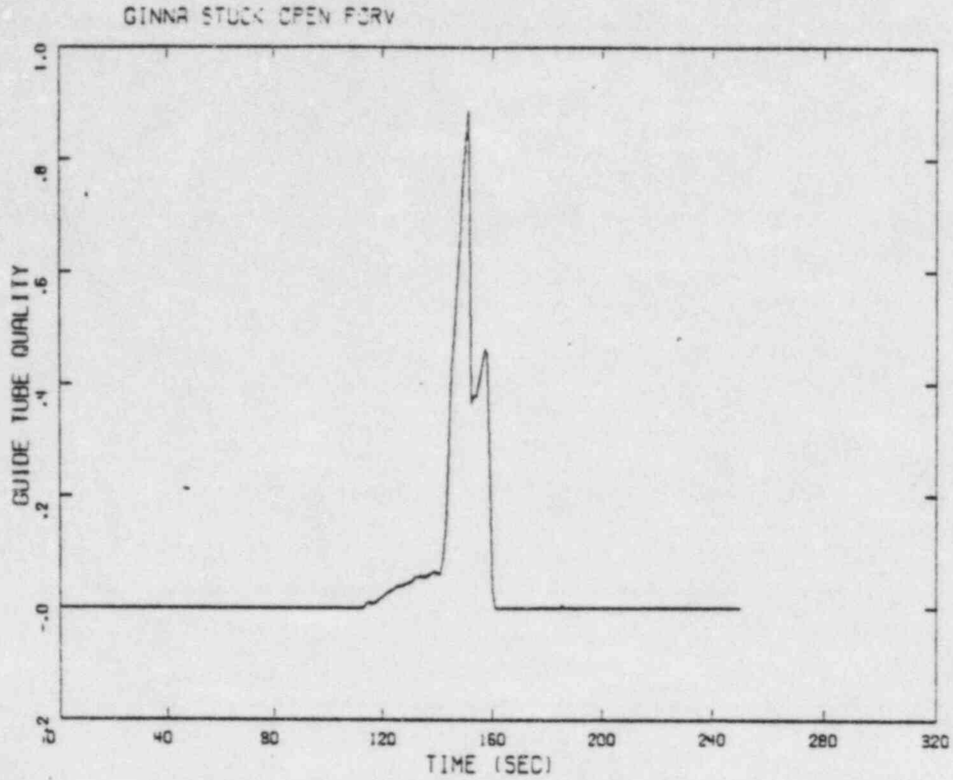


Figure 4.2-6. Guide Tube Vol. 16 Quality.

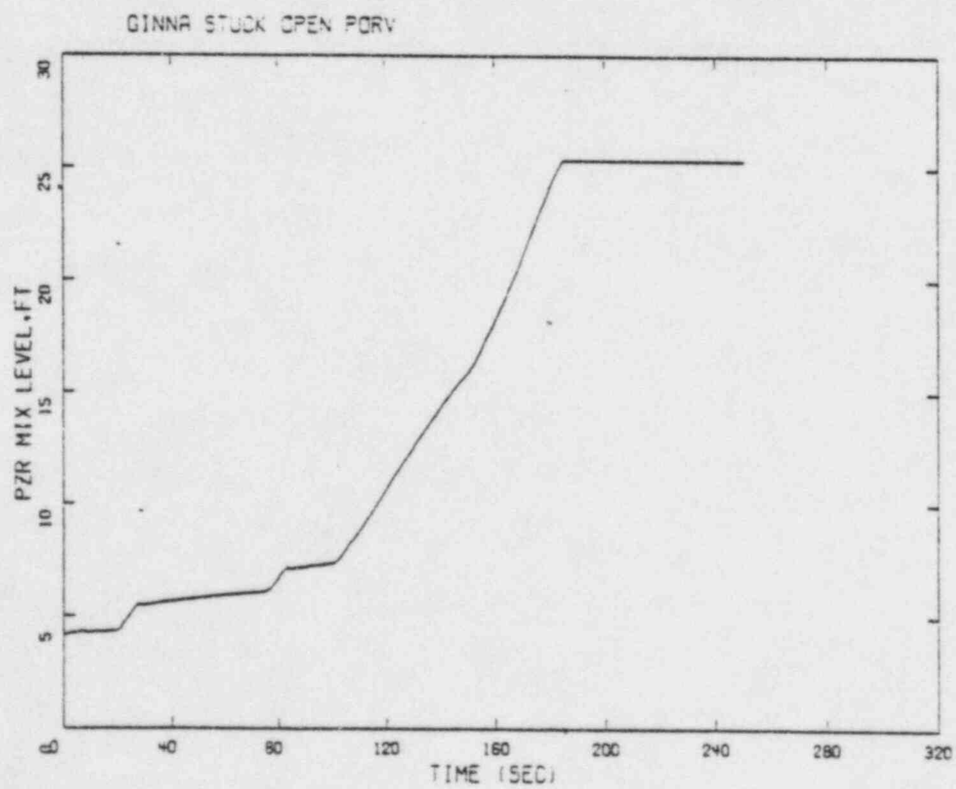


Figure 4.2-7. Pressurizer Mixture Level.

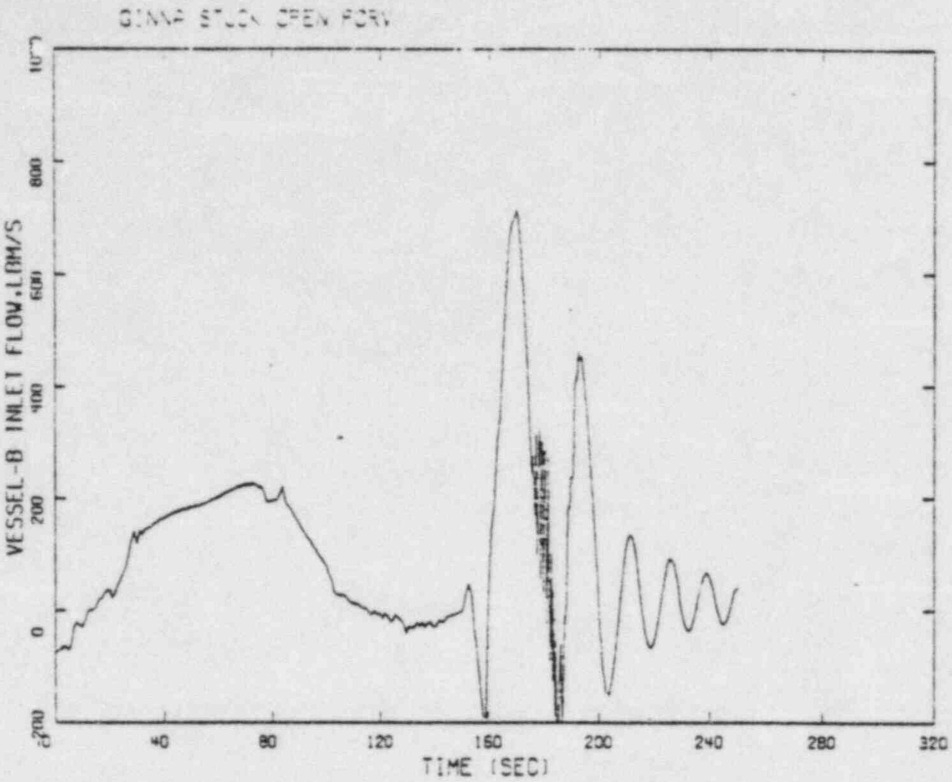


Figure 4.2-8. Vessel B Inlet Jun 65 Flow.

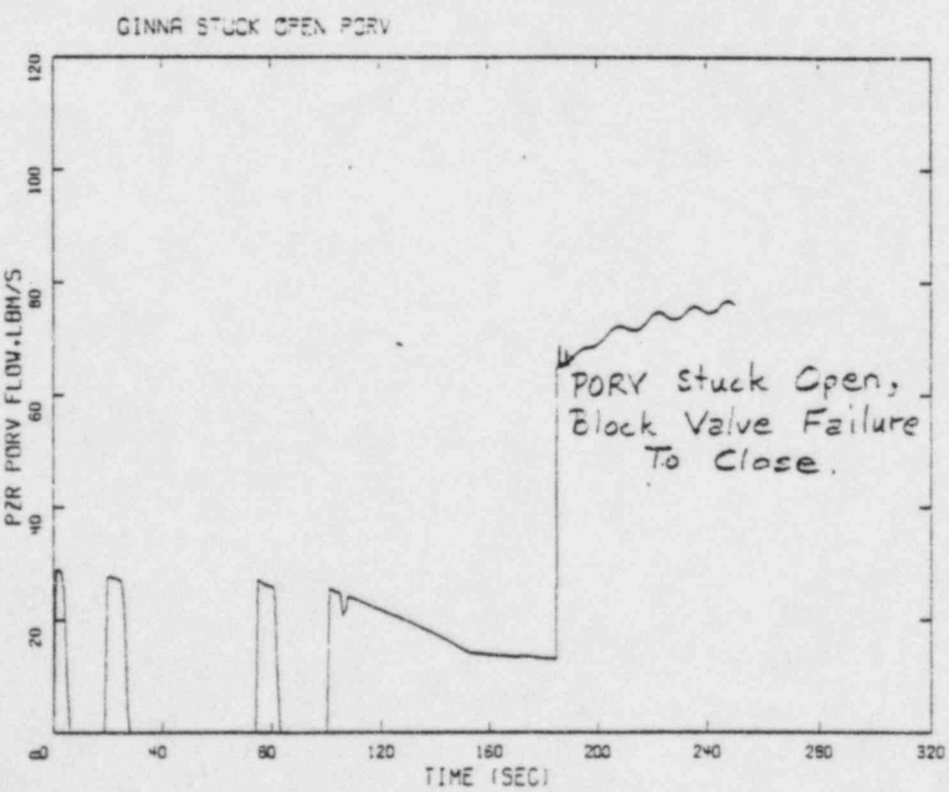


Figure 4.2-9. Pressurizer PORV Flow.

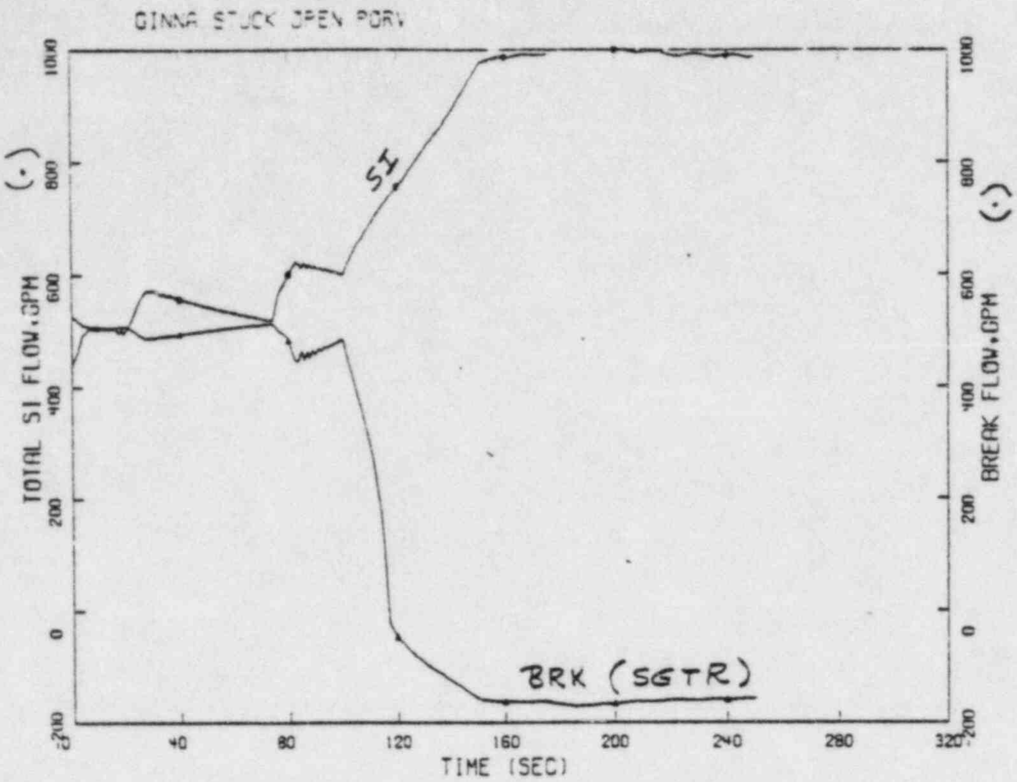


Figure 4.2-10. SI Flow/Break Flow.

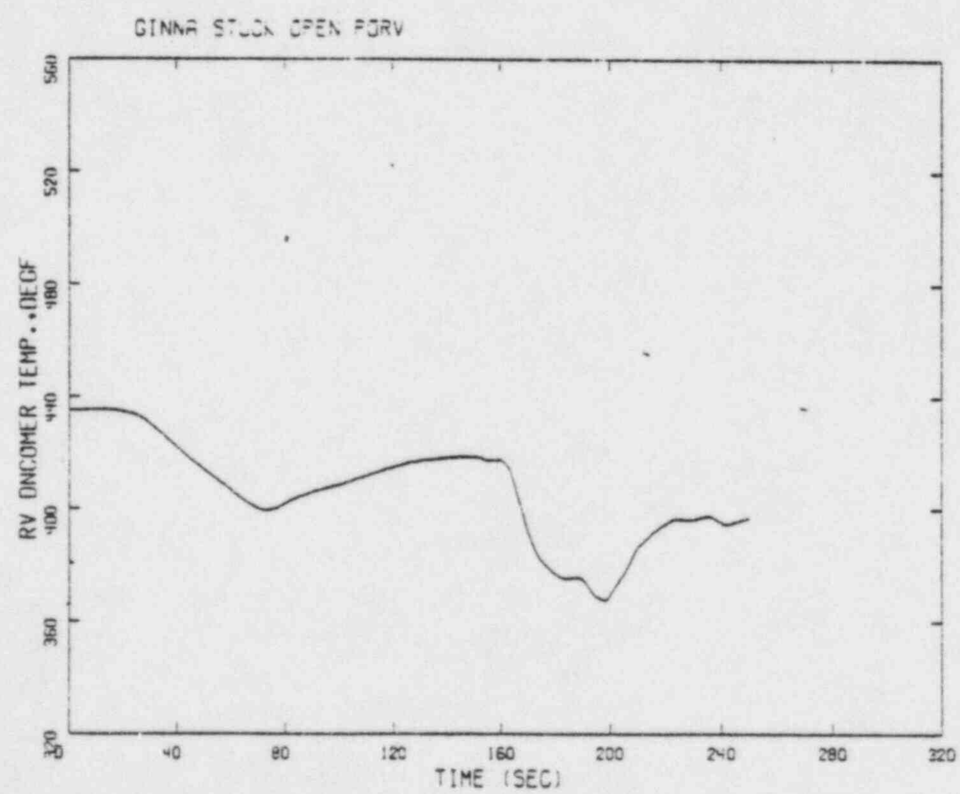


Figure 4.2-11. Vessel Downcomer Vol. 18 Temperature.

4.2 Stuck Open PORV

For this parametric it is postulated that when the pressurizer PORV failed to close as it did during the actual event, the downstream block valve failed in the fully open position. Thus unlike the actual series of events, and additional small break LOCA compounds the original SGTR. At this point the Westinghouse SGTR operator guidelines call for the LOCA guidelines. In this parametric the SGTR guidelines are followed to this time and then the calcuation is continued without any operator intervention until primary side recovery commences. The computation indicates that this occurs within a few minutes at which point the analysis is terminated with the level in the vessel slowly rising. Figures 4.2-1 to 4.2-11 show the response of the system. The calculation is performed with the t=42.5 minutes INPO deck so the time origin is t=42.5 minutes of the actual Ginna event when the operator initiated system depressurization actions in order to reduce the primary/secondary pressure differential. The only change made to the deck were alterations to the PORV area cards. None of the numerous time tables were modified which imples, for example, that the intact generator (SGA) is operated exactly the way that it was during the Ginna event. The RCS pressure trace (volume 61), Fig. 4.2-1, graphs the four pressurizer PORV openings and then at ~ 150 seconds a plateauing which can be attributed to flashing in the tubes (volumes 43 and 44) of the disrupted steam generator (SGB). Figures 4.2-2 and 4.2-3 for the steam generator tube quality confirms this. With isolation the disrupted generator becomes a region of low flow and stagnation. Reverse heat transfer across the disrupted generator tubes tends to hold the pressure up. Figures 4.2-4 to 4.2-6 show that the head (volume 19) had completely voided at ~ 160 seconds and that flashing in the guide tubes (volumes 14 and 16) had already commenced. The flashing in the steam generator tubes and the vessel head region causes the pressurizer to fill as depicted in Fig. 4.2-7 at ~ 185 seconds. With the filling the PORV (junction 122) begins to discharge single-phase liquid. There is an initial surge out of the generator into the vessel with the flashing, as can be seen in Fig. 4.2-8, but more stable conditions occur within minutes with the SI/charging flow dominating over the PORV flow from a now solid pressurizer. Figures 4.2-9 and 4.2-10 show these flows. Reverse leakage occurs in the disrupted generator and the level in the vessel head is slowly rising. As can be seen in Figs. 4.2-5 to 4.2-6 the guide tubes had

refilled at ~ 160 seconds when the SG tubes had commenced flashing. From Figs. 4.2-3 and 4.2-5 it can be concluded that no significant core uncovery occurred during this period and the calculation was terminated at this point. The minimum downcomer temperature in Fig. 4.2-11 appears to have decreased by approximately 10°F during this period as compared to the actual Ginna event. Further simulation may require renodalization to compute bubble collapse in the disrupted steam generator tubes and modifications regarding the tables and control blocks for the intact steam generator behavior.

4.3 Failure to Terminate HPI

For this case, ANL reanalyzed the Ginna event, as it occurred, but in this analysis it was assumed that the operator did not terminate operation of the high pressure injection system (HPI). The objective of this analysis was to determine the consequences in the primary and secondary systems when failing to terminate HPI.

In the actual Ginna event, the operator secured the HPI pumps at one hour and twelve minutes after tube rupture. This was considerably later than permitted by procedures with the delay largely attributed to operator's concern for potential core uncovering due to upper head void formation. Moreover, if RCS depressurization had begun when the governing criteria were met it is likely that filling the "B" steam generator and line would have been avoided (see Section 4.1). All of this is to say that there actually was a significant time delay in HPI termination during the Ginna tube rupture event which was accommodated without serious consequences to the plant. Securing the HPI pumps reduced the ongoing cooldown of the RCS and the discharge of radioactive water to the environment; some release continued because the charging pumps remained on and their flow exceeded the letdown rate. INPO estimated that the "B" generator SRV did not completely close until three hours and two minutes after tube rupture; at the time their analysis was concluded at one hour and twenty minutes, the estimated mass of water released through this SRV was 64,000 lbm.

For the purposes of this analysis, INPO's modeling of operator control of the intact steam generator (feed and bleed) was unaltered; the only changes made in the model were to inhibit tripping the HPI and to maintain a constant flow area for SGB SRV equal to that assumed when HPI was terminated in the actual event. The calculation was continued for eight minutes beyond actual HPI termination to show the response trends in the primary and secondary systems. The salient results of this calculation are shown in Figs. 4.3-1 through 4.3-6. These graphs begin at the time the INPO deck was re-nodalized; all results up to the time actual HPI termination occurred are identical to the original calculation.

As shown by comparing Figs. 4.3-1 and 4.3-2, primary system (RCS) pressure remains above that of SGB by approximately 300 psi. This pressure differential maintains flow through the ruptured tube into SGB and attendant release through its SRV. These flow rates are nearly steady at approximately 600 gpm, essentially the sum of HPI and charging flow rates. Figures 4.3-3 and 4.3-4 depict these continued flows.

The sustained injection of relatively cold HPI water into the RCS causes its moderate rate of cooldown to continue. For example, examination of the calculated fluid temperature in the reactor vessel downcomer, as shown in Fig. 4.3-5, gives an estimated rate of approximately -70°F/hour during the end period of the calculation; the results also indicate a slow reduction in cooldown rate as anticipated. Based upon a limiting acceptable rate of -100°F/hour, these results show that a certain thermal margin still exists even for continued operation of the HPI system. It is also noted that the continued injection of cold water causes eventual elimination of the upper head void at the time the calculation is ended as shown in Fig. 4.3-6.

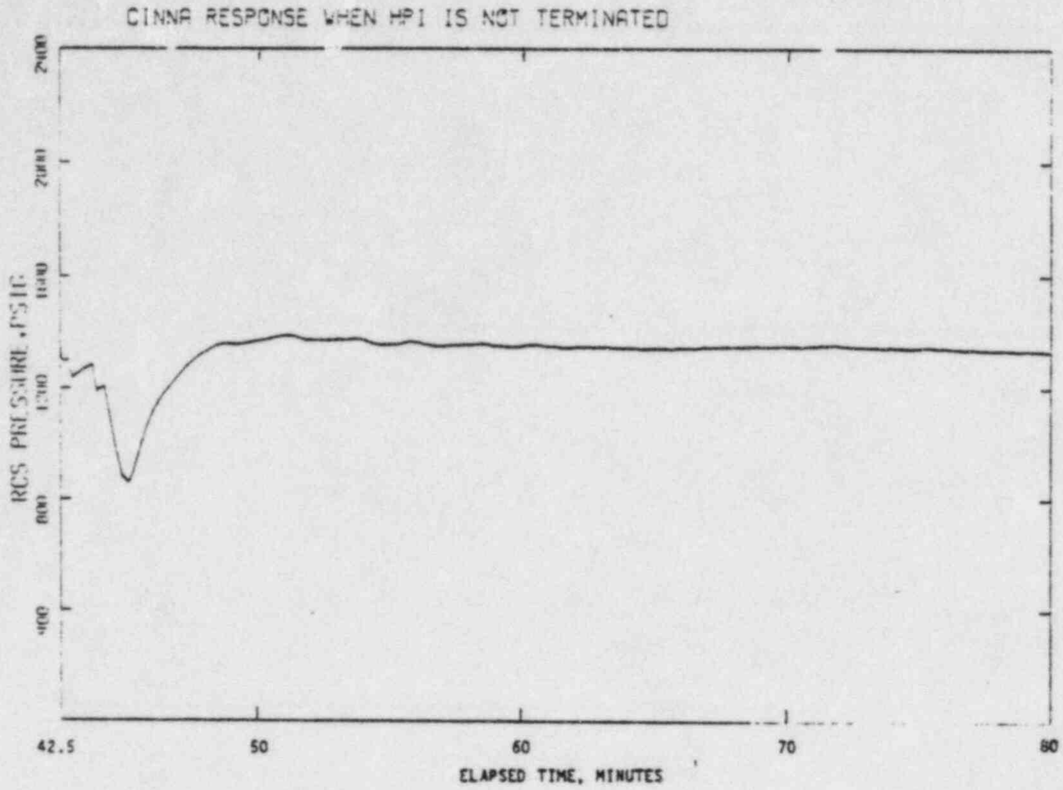


FIGURE 4.3-1. RCS (VOL. 61) PRESSURE.

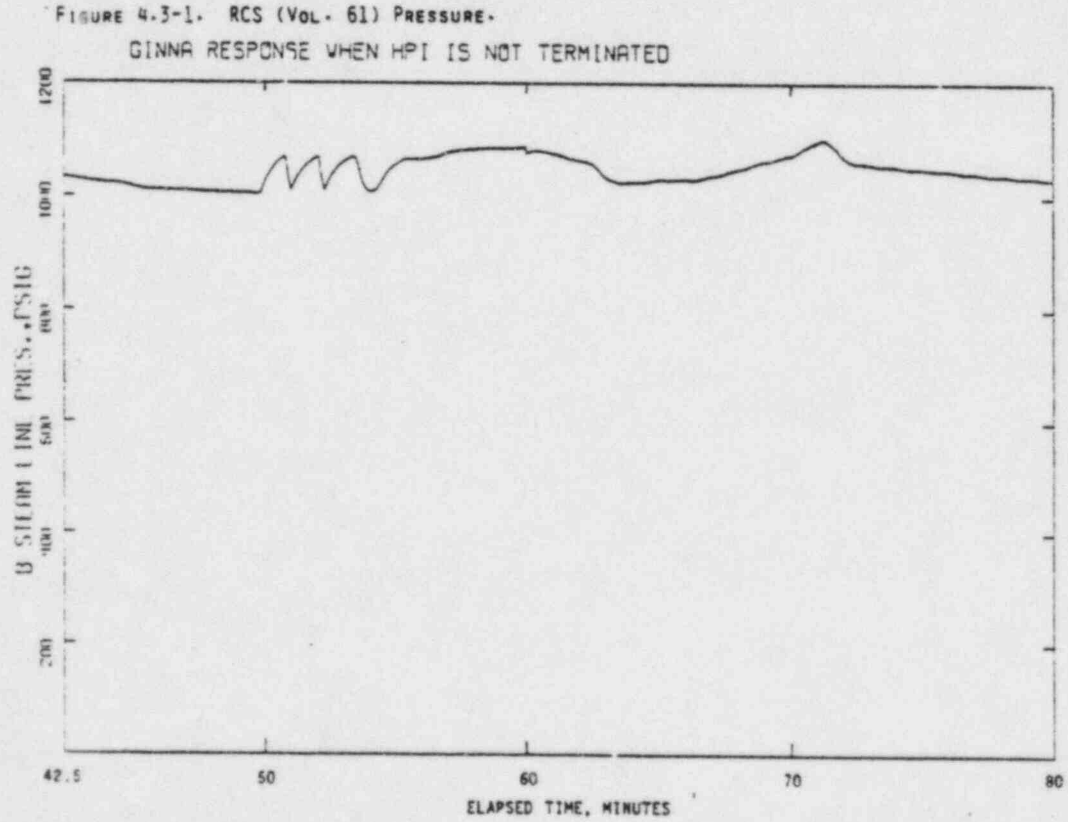


FIGURE 4.3-2. "B" STEAM LINE (VOL. 68) PRESSURE.

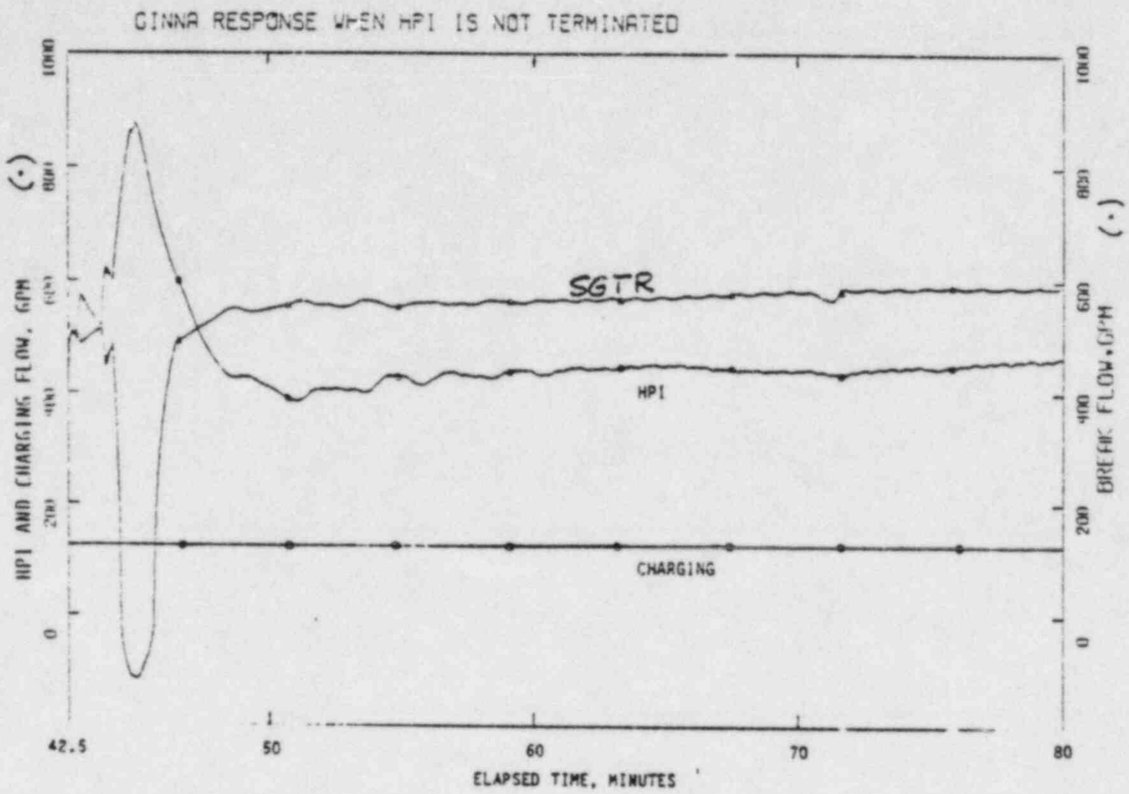


Figure 4.3-3. HPI, Charging and Tube Rupture Flow Rates.

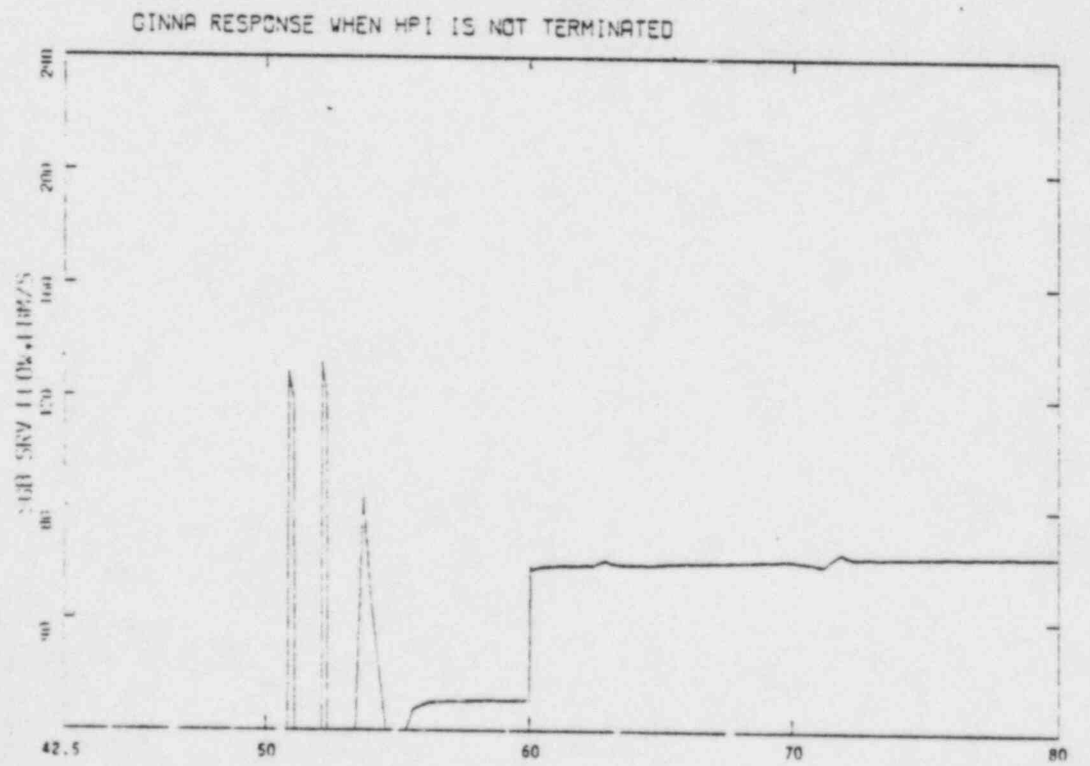


Figure 4.3-4. "B" Steam Generator Safety Valve Flow.

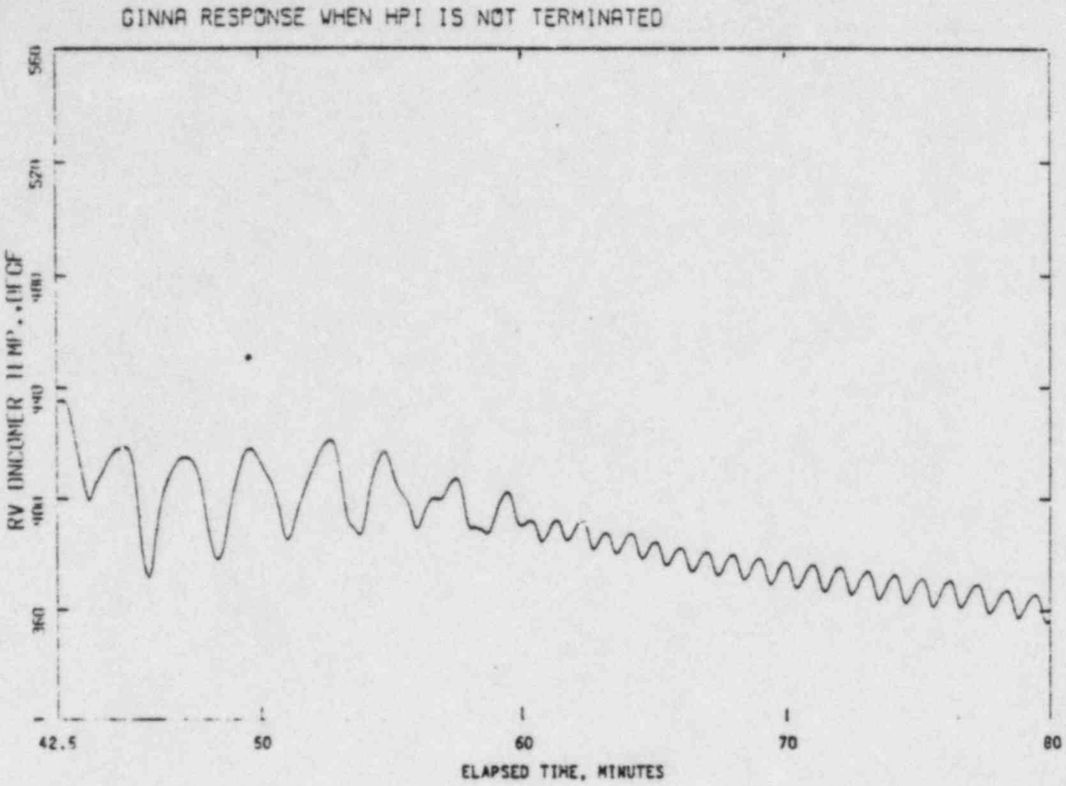


Figure 4.3-5. Reactor Vessel Downcomer (Vol. 18) Temperature.

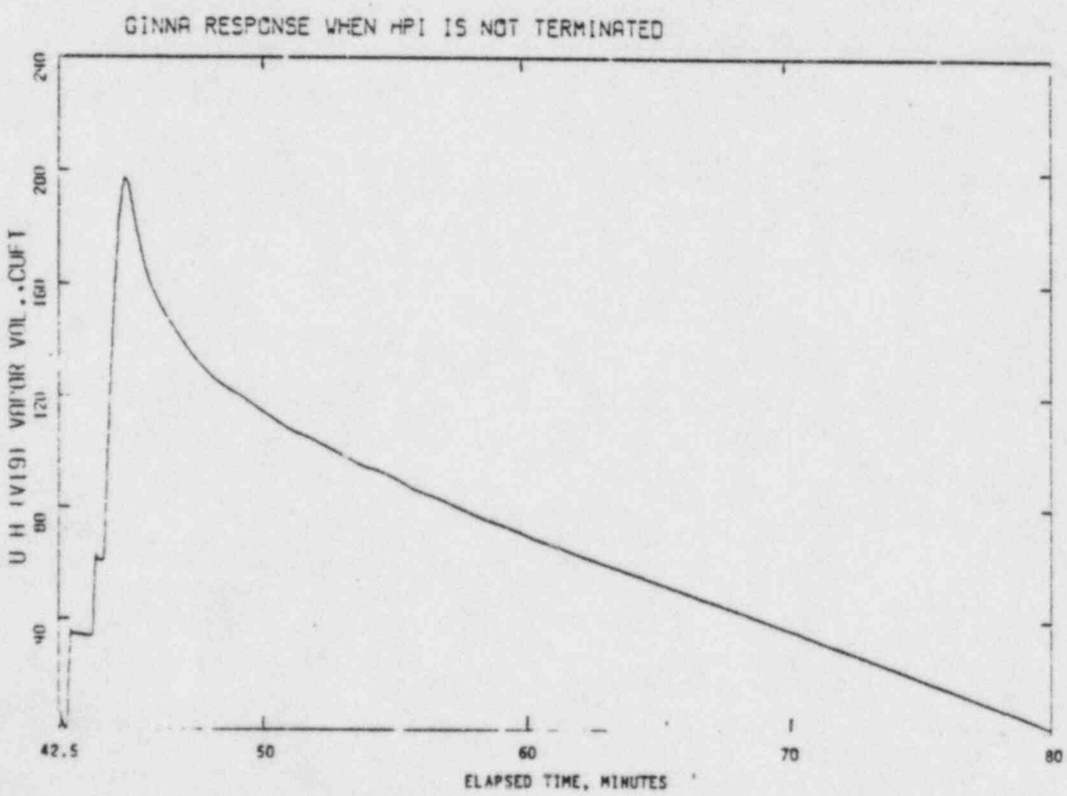


Figure 4.3-6. Reactor Vessel Upper Head (Vol. 19) Steam Volume.

5.0 SUMMARY AND CONCLUSION

A simulation of the Ginna SGTR event of January 25, 1982 was performed utilizing the latest EPRI release version of RETRAN [3], RETRAN02/MOD03 in conjunction with RETRAN02/MOD03A input decks obtained from INPO and modified by ANL. The RETRAN02/MOD03 results agree well with the INPO RETRAN02/MOD03A calculations [1]. A reasonable match is therefore obtained between calculations and measured data from the actual event [2]. Where differences have occurred, they can be explained in terms of code model differences between the MOD03A and the MOD03 versions and in terms of sensitivities in the INPO calibration to data. In order to match the limited thermal hydraulic data available, INPO had to resort to a number of calibration adjustments, based on engineering judgement, in its application of the thermal hydraulic models available in the RETRAN02 code. To further the understanding of the thermal hydraulic phenomena which occurred during the actual event, three additional parametric calculations were performed which included variations on operator actions and further equipment failure. The three parametrics performed demonstrated; that opportune timing in conforming to recent operator guidelines would prevent the filling the disrupted steam generator solid and alleviate concerns about loading questions; that failures in the PORV line downstream of the PORV would not necessarily lead to significant core damage and; that sufficient thermal margin exists for pressurized thermal shock situations even if there was a further continuation of safety injection flow. While there is a significant out-of-containment loss of ECCS inventory in the third parametric where SI flow was not terminated, timely conformance to the operator guidelines, as evidenced by the first parametric, would prevent such a condition from occurring. The parametrics have contributed to the understanding of the thermal hydrualic phenomena that occurred during the actual sequence of events during the Ginna SGTR incident.

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ANL staff who provided input to this report were J. H. Tessier and T. Y. C. Wei, authors; and K. Rank and M. Mehaffey, report preparation.

References

1. INPO staff, "Thermal-Hydraulic Analysis of Ginna Steam Generator Tube Rupture Event," Institute of Nuclear Power Operations draft report (September 1983).

2. NRC staff, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," Nuclear Regulatory Commission Report, NUREG-0909 (April 1982).

3. J. H. McFadden et al., "RETRAN-02 A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," Electric Power Research Institute Report, EPRI NP-1850-CCM (May 1981).