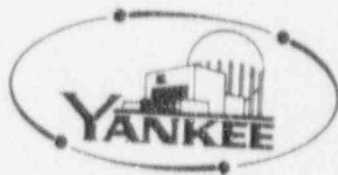


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Analysis of
A Postulated Design Basis
Steam Generator Tube Rupture
For The
Seabrook Nuclear Power Station

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ABSTRACT

This report documents an analysis of a postulated design basis Steam Generator Tube Rupture (SGTR) at the Seabrook Nuclear Power Station. The analysis has been performed with credit for operator actions in response to a SGTR.

Plant operator actions and action times assumed in the analysis were verified by extensive simulator tests. These tests were performed with plant operators trained to follow the Seabrook Station Emergency Response Procedures. The equipment required for mitigation was identified and its safety classification was verified.

A plant-specific RETRAN computer model was developed to simulate the overall plant response and operator actions. RETRAN has been previously reviewed and approved by the NRC for use in the analysis of certain transients, including SGTR. Two single failure scenarios were analyzed. One scenario included a failed closed Atmospheric Steam Dump Valve (ASDV) on an intact steam generator and resulted in the least margin to overfill of the ruptured steam generator. The second scenario included a failed open ASDV on the ruptured steam generator and resulted in the most severe radiological consequences.

Although margin to steam generator overfill has been demonstrated, a static loads analysis of flooded main steam lines has been performed as required in the NRC's Safety Evaluation Report. Main steam lines will remain intact in this highly unlikely consequence of a SGTR.

The off-site radiological doses received at the exclusion area boundary and the low population zone boundary were determined to be within the limits of 10CFR100.

ACKNOWLEDGEMENTS

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TABLE OF CONTENTS

	Page
DISCLAIMER OF RESPONSIBILITY.....	iii
ABSTRACT.....	iv
ACKNOWLEDGEMENTS.....	v
TABLE OF CONTENTS.....	vi
LIST OF FIGURES.....	ix
LIST OF TABLES.....	x
MAIN REPORT	
1.0 EXECUTIVE SUMMARY.....	1
2.0 INTRODUCTION.....	4
3.0 DESIGN BASIS STEAM GENERATOR TUBE RUPTURE SCENARIO.....	7
4.0 RETRAN MODEL OF SEABROOK.....	11
4.1 Qualification of RETRAN-02 for Use in SGTR Analysis.....	11
4.2 Description of Seabrook SGTR RETRAN Model	11
4.2.1 Description of Reactor Vessel Model.....	13
4.2.2 Description of Intact Coolant Loop Model.....	14
4.2.3 Description of the Ruptured Loop and Pressurizer Models.....	15
4.2.4 Description of SG Secondary and Main Steam Line Models.....	16
4.2.5 Description of the Tube Rupture Model.....	17
4.2.6 Description of SG Narrow Range Level Model.....	18
5.0 OPERATOR RESPONSE TIME EVALUATION.....	22
6.0 REQUIRED EQUIPMENT AND ITS SAFETY CLASS.....	31
7.0 TRANSIENT ANALYSIS OF THE CASE WITH MINIMUM MARGIN TO OVERFILL...	35
7.1 Description of Case Scenario.....	35
7.2 Initial Conditions and Conservative Assumptions.....	36
7.2.1 Initial Power Level.....	37
7.2.2 Initial RCS Pressure and Time of Reactor Trip.....	37
7.2.3 Pressurizer Water Level.....	38
7.2.4 Initial Steam Generator Secondary Mass.....	39

TABLE OF CONTENTS
(Continued)

	<u>Page</u>
7.2.5 Break Location.....	39
7.2.6 Off-Site Power Availability.....	39
7.2.7 Reactor Trip Delay.....	40
7.2.8 Turbine Trip Delay.....	40
7.2.9 Steam Generator Relief Valve Pressure Setpoint.....	40
7.2.10 Low Pressurizer Pressure Reactor Trip Setpoint.....	41
7.2.11 Pressurizer Pressure for SI Initiation.....	41
7.2.12 Safety Injection Flow Rate.....	41
7.2.13 Emergency Feedwater Flow.....	41
7.2.14 Emergency Feedwater System Delay.....	42
7.2.15 Emergency Feedwater Temperature.....	42
7.2.16 Chemical and Volume Control System and Pressurizer Heater Operation.....	43
7.2.17 Turbine Runback.....	43
7.2.18 RCP Operation.....	44
7.2.19 Decay Heat.....	44
7.2.20 Average Temperature.....	44
7.3 Transient Description.....	45
7.3.1 RCS Response.....	45
7.3.2 SG and Secondary System Response.....	49
8.0 STATIC LOAD ANALYSIS OF FLOODED MAIN STEAM LINES.....	63
9.0 TRANSIENT ANALYSIS OF THE CASE WITH THE MOST SEVERE RADIOLOGICAL CONSEQUENCES.....	64
9.1 Description of Case Scenario.....	64
9.2 Initial Conditions and Conservative Assumptions.....	66
9.2.1 Initial RCS Pressure and Time of Reactor Trip.....	66
9.2.2 Initial Steam Generator Secondary Mass.....	67
9.2.3 Reactor Trip Delay.....	67
9.2.4 Steam Generator Relief Valve Pressure Setpoint.....	68
9.2.5 Low Pressurizer Pressure Reactor Trip Setpoint.....	68
9.2.6 Pressurizer Pressure for SI Initiation.....	68
9.2.7 Emergency Feedwater System Delay and Flow Rate.....	68
9.2.8 Turbine Runback.....	69
9.3 Transient Description.....	69
9.3.1 RCS Response.....	69
9.3.2 SG and Secondary System Response.....	72
9.4 Determination of Mass Releases.....	74

TABLE OF CONTENTS
(Continued)

	<u>Page</u>
10.0 RADIOLOGICAL DOSE CALCULATIONS.....	90
10.1 Analysis, Methodology, and Assumptions.....	91
10.1.1 Source Term Calculations.....	91
10.1.2 Radioactivity Transport and Release.....	92
10.1.3 Dose Calculations.....	94
10.2 Off-Site Radiation Doses.....	96
11.0 CONCLUSIONS.....	103
12.0 REFERENCES.....	104
APPENDIX A - "GENRUP" Description.....	A-1

LIST OF FIGURES

<u>Number</u>	<u>Title</u>	<u>Page</u>
4.1	RETRAN Model of Seabrook Station - Volumes	20
4.2	RETRAN Model of Seabrook Station - Junctions	21
7.1	RCS Pressure versus Time	54
7.2	Intact Loops Hot and Cold Leg Temperatures versus Time	55
7.3	Ruptured SG Inlet & Outlet Plenum Temperatures versus Time	56
7.4	Pressurizer Level versus Time	57
7.5	SG Pressures versus Time	58
7.6	SG NR Level versus Time	59
7.7	Pressure Differential Across Break versus Time	60
7.8	Rupture Flow Rate versus Time	61
7.9	Ruptured SG Liquid Volume versus Time	62
9.1	RCS Pressure versus Time	81
9.2	Intact Loops Hot and Cold Leg Temperatures versus Time	82
9.3	Ruptured SG Inlet & Outlet Plenum Temperatures versus Time	82
9.4	Pressurizer Level versus Time	83
9.5	SG Pressures versus Time	84
9.6	SG NR Level versus Time	84
9.7	Pressure Differential Across Break versus Time	85
9.8	Rupture Flow Rate versus Time	85
9.9	Ruptured SG Mass Release Rate to Atmosphere versus Time	86
9.10	Intact SG Mass Release Rate to Atmosphere versus Time	87
9.11	Liquid Mass in Ruptured SG versus Time	88
9.12	Water Level Above Break Location in Ruptured SG versus Time	89
10.1	Flashing Fraction versus Time	102

LIST OF TABLES

<u>Number</u>	<u>Title</u>	<u>Page</u>
5.1	Operator Actions/Time Intervals for SGTR Recovery Actions	29
5.2	Operator Action Times Observed During Design-Basis Tube Rupture Simulations at Seabrook Simulator	30
6.1	Summary of Systems, Components, and Instrumentation Credited for Mitigation of the SGTR Event	32
7.1	Summary of Conservative Assumptions and Initial Conditions for Case With Minimum Margin to Overfill	51
7.2	Sequence of Events for Case With Minimum Margin to Overfill	52
9.1	Summary of Conservative Assumptions and Initial Conditions for Case With Most Severe Radiological Consequences	77
9.2	Sequence of Events for Case With Maximum Radiological Consequences	78
9.3	Mass Flows for a Design Basis SGTR Assuming Failure of the Ruptured SG ASDV Fully Open at Time of SG Isolation	80
10.1	Reactor Coolant Iodine and Noble Gas Activity	97
10.1a	Secondary Coolant Iodine Activity	97
10.2	Iodine Appearance Rates in the Reactor Coolant for a Design Basis SGTR	98
10.3	Short-Term Atmospheric Dispersion Factors and Breathing Rates for Accident Analysis	99
10.4	Isotopic Data	100
10.5	Calculated Off-Site Radiation Doses for a Design Basis SGTR Assuming a Failed Open ASDV and 30-Minute Isolation Time	101

1.0 EXECUTIVE SUMMARY

This report documents an analysis of a postulated design basis Steam Generator Tube Rupture (SGTR) at the Seabrook Nuclear Power Station. The analysis complies with the requirements of recent Nuclear Regulatory Commission (NRC) Safety Evaluation Reports (SERs) in References (5) and (6).

Section 2.0, "Introduction," provides a brief history of the SGTR issue. In response to NRC concerns about the adequacy of SGTR analyses, a subgroup of utilities operating under the Westinghouse Owners Group charter was formed in 1983. The subgroup developed a new analysis methodology to address the concerns. It was subsequently approved (References (5) and (6)), and the NRC requested utilities to provide plant-specific analyses. This report provides a plant-specific analysis for the Seabrook Nuclear Power Station. Computer codes and some assumptions employed differ from those approved, but the methods are fundamentally the same and the differences are justified in the report.

Section 3.0, "Design Basis Steam Generator Tube Rupture Scenario," describes the accident which is analyzed. The plant is assumed to suffer a guillotine severance of one steam generator tube while operating at full power. Since the Reactor Coolant System (RCS) leakage flow rate exceeds the make-up capacity of the Chemical and Volume Control System (CVCS), pressurizer level and pressure decrease. A reactor trip and safety injection will be manually or automatically initiated. A coincident loss of off-site power is assumed to occur at the time of the reactor trip. Although a variety of indications are available, operator diagnosis is assumed to be made on the basis of an uncontrollable increase of water level in the faulted steam generator. Credit is assumed for subsequent operator action using Emergency Response Procedures (ERPs) to isolate the faulted steam generator, cool down and depressurize the RCS, terminate safety injection, and terminate the RCS leakage. RCS radioactivity is assumed to be transported through the tube rupture to the fluid in the faulted steam generator. From there, it is assumed to pass to the environment via the condenser, the Atmospheric Steam Dump Valves (ASDVs), the steam supply to the turbine-driven Emergency Feedwater (EFW) Pump, or the Main Steam Safety Valves (MSSVs).

Section 4.0, "RETRAN Model of Seabrook," describes the computer model of Seabrook Station used for this analysis. The RETRAN code has been reviewed and approved by the NRC for use in the analysis of non-LOCA transients, including SGTRs. The Seabrook Station model is similar to that used by the Institute of Nuclear Power Operations (INPO) in their RETRAN analysis of the Ginna SGTR.

Section 5.0, "Operator Response Time Evaluation," describes the operator actions required to mitigate the consequences of a SGTR. The action times assumed are listed. They were derived from times observed during SGTR simulation runs on the Seabrook Station training simulator and by plant walkdown. These simulations were performed using design basis SGTR conditions and Seabrook operating crews undergoing scheduled retraining. Conservative action time values for each operator action credited in the analysis were selected from a total of six simulator runs.

Section 6.0, "Required Equipment and its Safety Class," provides a comprehensive list of systems, components, and instrumentation which are credited for mitigation of the design basis SGTR. Most equipment is nuclear safety related. Exceptions are noted and justified. The analysis considers single active failures in equipment required to mitigate the SGTR.

Section 7.0, "Transient Analysis of the Case With Minimum Margin to Overfill," discusses the first of two design basis SGTR scenarios analyzed. In this case, initial conditions and conservative assumptions were made so as to yield a minimum margin to overfill in the faulted steam generator. The most limiting single active failure was determined to be a failed closed ASDV on an intact steam generator. Results showed that steam generator overfill will not occur.

Section 8.0, "Static Load Analysis of Flooded Main Steam Lines," satisfies the requirement of the NRC's SER (Reference (6)). Section 7.0 of the report demonstrates that steam generator overfill will not occur during a design basis SGTR. Nevertheless, a static load analysis of the main steam

lines in a flooded condition was performed as required in Reference (6). The results showed that the main steam lines are structurally adequate for this condition.

Section 9.0, "Transient Analysis of the Case With the Most Severe Radiological Consequences," discusses the second of the two design basis SGTR scenarios analyzed. In this case, initial conditions and conservative assumptions were made so as to yield the most severe radiological consequences. The most limiting single active failure was determined to be a failed open ASDV on the faulted steam generator. Corrective action was assumed to be closure of the associated block valve in accordance with Emergency Response Procedure E-3, "Steam Generator Tube Rupture." Mass releases to the environment were determined for an eight-hour period following the tube rupture. These mass releases were subsequently used in evaluating the radiological consequences of the event.

Section 10.0, "Radiological Dose Calculations" provides an assessment of the radiological consequence of a design basis SGTR. The scenario includes a failed open ASDV on the faulted SG as described in Section 9.0, "Transient Analysis of the Case With the Most Severe Radiological Consequences." As required in Reference (5), Section 10.0 includes a detailed description of the explicit iodine transport models used in the analyses. Dose calculations were performed for both pre-accident and accident-initiated iodine spikes as indicated in the NRC Standard Review Plan (SRP), Section 15.6.3. Thyroid doses for the accident-initiated iodine spike are most limiting relative to the SRP criteria. The calculated doses are within the SRP guideline values.

Section 11.0, "Conclusions," is summarized here. Plant-specific thermal-hydraulic transient analysis of the Seabrook Nuclear Power Station using an NRC-accepted computer program demonstrates that SG overfill will not occur during a design basis SGTR. Nevertheless, results of a static load analysis of the main steam lines in a flooded condition show that they are structurally adequate. Radiological consequences of a design basis SGTR are within SRP guideline values.

2.0 INTRODUCTION

The analysis for a design basis SGTR accident is included (Reference (1)) in the Section 15.6.3 analyses of Final Safety Analysis Reports (FSARs). The accident which is analyzed is the complete severance of one steam generator tube which results in the leakage of reactor coolant into the secondary side of the steam generator. Based on the assumptions used previously in the FSAR analyses, it was concluded that the consequences of a design basis SGTR meet the appropriate acceptance criteria. Following the SGTR which occurred at the Ginna Plant in January, 1982, the traditional assumptions which have been used for the FSAR analysis of a design basis SGTR have been questioned. In particular, the time required for the operator to terminate the leakage into the ruptured steam generator may be longer than the 30 minutes which has been assumed in the FSAR analysis. In addition, the qualification of certain equipment which is used to mitigate a SGTR may not conform to licensing basis criteria.

The consequences of a SGTR depend largely upon the ability of the operator to take the necessary actions to terminate the primary to secondary leakage. If the leakage continues significantly beyond the 30 minutes previously assumed in the FSAR accident analysis, the secondary side of the steam generator may become filled and water may enter the steam line. If the leakage continues, the release of liquid through the secondary side relief valves to the atmosphere could result in an increase in the radiological doses. The structural integrity of the main steam lines may also be of concern due to the accumulation of water in the steam line. Thus, one important concern related to a SGTR is the possibility of steam generator overfill and the potential consequential effects.

The concern over the potential for overfill has resulted in the following three issues regarding the FSAR analysis for a SGTR: (1) the operator action time required to terminate the primary to secondary leakage following a design basis SGTR, (2) the qualification of the equipment which is assumed to be used in the SGTR recovery, and (3) the evaluation of the worst

case single active failure of equipment required for the SGTR analysis. In order to resolve these issues, a subgroup of utilities in the Westinghouse Owners Group (WOG) was formed and a program was initiated to address the issues on a generic basis.

The subgroup developed a new analysis methodology to address all the NRC concerns and applied it to a typical or "reference" plant to demonstrate compliance with the NRC's acceptance criteria for this accident (References (2), (3), and (4)).

Subsequently, the NRC approved the new analysis methodology, but with several stipulations (References (5) and (6)):

1. Each member of the WOG SGTR subgroup must submit a plant-specific analysis.
2. Since the new analysis methodology credits operator action to mitigate the consequences of a design basis SGTR, simulator demonstration runs must be performed to support the operator action times used in the plant-specific analysis.
3. The new methodology is expected to show no steam generator overfill. The subgroup also demonstrated acceptable radiation release for the "reference" plant in the unlikely event of overfill. Nevertheless, an evaluation of the structural adequacy of the main steam lines and associated supports under water filled conditions is required.
4. Systems, components, and instrumentation credited for SGTR mitigation in the plant-specific Emergency Operating Procedures (EOPs) must be listed with safety classification.
5. Plant-specific applicability of all assumptions used in the analysis must be demonstrated.

An extensive Yankee Atomic Electric Company (YAEC) and New Hampshire Yankee (NHY) program to address these stipulations has been completed, and the results are documented in this report.

The computer codes and some of the assumptions employed in this plant-specific analysis for Seabrook Station are different than those in the methodology approved by the NRC. However, the methodologies are fundamentally the same. Differences are described and justified within this report.

3.0 DESIGN BASIS STEAM GENERATOR TUBE RUPTURE SCENARIO

This report presents the results of analyses of several design basis tube rupture scenarios for Seabrook Station. The design basis tube rupture for Seabrook Station is the instantaneous guillotine rupture of a single Steam Generator (SG) U-tube. This results in two flow paths for leakage of primary system coolant into the secondary side of the ruptured SG (e.g., through the two segments of the broken tube). The evolution of a design basis tube rupture scenario assuming no single failures is described in this section. The evolution of the design basis tube rupture scenarios, including the limiting single active failures with respect to minimizing margin to SG overfill and maximizing off-site radiological doses, are described later in Sections 7.0 and 9.0.

The scenario described here assumes the plant to be initially operating at 100% Reactor Thermal Power (RTP). After the tube ruptures, the flow of coolant out of the primary system results in a gradual reduction in primary system pressure and pressurizer level, causing the charging system to increase charging flow as the deviation between the actual level and the programmed pressurizer level slowly grows.

Since the leakage flow rate exceeds the maximum make-up flow capacity of the CVCS, pressurizer level and pressure continue to decrease. The operators may manually initiate SI in response to the uncontrolled decrease in pressurizer level or they may attempt to manually reduce plant power level in preparation for a controlled shutdown. If neither action is taken, then it is likely that periodic automatic turbine runbacks will be initiated as the Reactor Coolant System (RCS) pressure decrease causes the overtemperature runback setpoint to be reached. Eventually, if no operator action is taken to initiate SI or trip the reactor, the continuing pressure decrease will result in an automatic reactor trip and SI actuation on low RCS pressure. A coincident loss of off-site power is assumed to occur at the time of the reactor trip.

The reactor trip results in a turbine trip and the closing of the turbine stop valves, bottling up the SG secondary side. Pressure in the main steam lines will then rise until the SG Atmospheric Steam Dump Valves (ASDVs)

and Main Steam Safety Valves (MSSVs) open. The MSSVs will reseal shortly after the trip, but the ASDVs will remain open until RCS loop temperatures are reduced to below the saturation temperature of the SG secondary side at the ASDV setpoint. Condenser steam dumps are not available due to the loss of condenser vacuum resulting from the loss of off-site power.

The combined action of the MSSVs, ASDVs, and the post-trip reduction in core power causes the RCS fluid temperatures to decrease, shrinking the RCS liquid volume, resulting in decreasing RCS pressure and pressurizer level.

The loss of off-site power results in the loss of motive power for the Reactor Coolant Pumps (RCPs) which then coast down, resulting in decreasing RCS flow rates until natural circulation is established. The loss of power also causes a direct actuation of the Emergency Feedwater (EFW) pumps.

Safety Injection (SI) actuation causes the SI pumps to start and the Centrifugal Charging Pumps (CCPs) discharge to be redirected from normal charging to cold leg emergency core cooling injection. It also results in the tripping of the steam-driven Main Feedwater Pumps (MFWPs) and closure of the main feedwater isolation and control valves.

Immediately after the trip, the Unit Shift Supervisor will refer to the E-0, "REACTOR TRIP OR SAFETY INJECTION," Emergency Response Procedure (ERP). The E-0 procedure directs the operator to verify the automatic actions discussed above and provides guidance for placing the plant in a stable condition and diagnosing the cause of the reactor trip. The design basis tube rupture scenario assumes that diagnosis of the event is based on observation by the operators of an uncontrolled increase in the level of the ruptured SG using SG Narrow-Range (NR) level instrumentation. No credit is taken for operation of the Radiation Data Management System (RDMS), since the applicable portions of the RDMS are not safety related.

Emergency Core Cooling System (ECCS) flow causes the RCS pressure and pressurizer level to increase very gradually, while the RCS temperatures drop slightly due to the falloff in core decay heat level and cold temperature of the injected water.

Once the event has been diagnosed as a tube rupture, the Unit Shift Supervisor will switch to implementing the event-specific ERP E-3, "STEAM GENERATOR TUBE RUPTURE." E-3 directs the operator to isolate steam flow from the ruptured SG and terminate EFW flow to that SG. As discussed later in Section 5.0, the analysis assumes that these actions will be completed at the time the ruptured SG level returns to 33% NR after the reactor trip. The operator is then directed to open the intact SG ASDVs to begin cooling the RCS. The end point of the cooldown is a target average core exit temperature which is a function of the ruptured SG pressure. The purpose of the cooldown is to establish adequate subcooling to allow RCS pressure to subsequently be reduced to the ruptured SG pressure without causing steam voids to form in the reactor core region.

The cooldown proceeds until the average core exit temperature specified in the E-3 procedure is reached. The operator then controls the intact SG ASDVs to maintain this temperature.

The cooldown results in significant shrinkage of the volume of the RCS coolant inventory, causing a further reduction in RCS pressure and pressurizer level. Since there is very little coolant flow into the Reactor Vessel (RV) upper head region under natural circulation cooling conditions, the water and metal in this region remain hot during the cooldown and a vapor bubble may form in the upper head when the RCS pressure drops below the saturation pressure of the fluid in this region.

After terminating the cooldown, the procedure directs the operator to open one pressurizer Power-Operated Relief Valve (PORV) to reduce RCS pressure below the ruptured SG pressure. As a result, any steam bubble in the RV upper head region may increase in size as more of the fluid in that region flashes to steam. The increased size of the steam bubble may cause a relatively rapid displacement of coolant from the upper head into the pressurizer, raising pressurizer level during the depressurization.

The venting of the pressurizer steam space results in RCS pressure decreasing rapidly until the operator closes the PORV, based on achievement of one of three PORV closure criteria specified in the E-3 procedure.

1. RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 5% of scale; or
2. Pressurizer level is greater than 75% of scale; or
3. RCS subcooling based on core exit thermocouples is less than 40°F.

The RCS pressure may still be above the ruptured SG pressure when the PORV is closed if the second or third closure criteria is met prior to RCS pressure falling below the SG pressure.

After closing the PORV, the E-3 procedure directs the operator to perform several steps to terminate SI flow and restore normal charging and letdown. During the time interval required to accomplish these tasks, the RCS repressurizes causing renewed or continued leakage from the RCS into the ruptured SG.

Once SI flow has been stopped, RCS pressure decreases gradually due to continued flow through the broken tube until pressure equilibrium with the ruptured SG is obtained and leakage from the RCS is terminated. If conditions are such that the specified PORV closure criteria are no longer met, the operators may reopen a pressurizer PORV during this time interval in order to terminate leakage from the RCS.

4.0 RETRAN MODEL OF SEABROOK

4.1 Qualification of RETRAN-02 for Use in SGTR Analysis

The Seabrook plant-specific tube rupture analyses have been performed using RETRAN-02. A model of the Seabrook plant was developed for these analyses, using an approach similar to that used by INPO in their RETRAN analysis of the Ginna Steam Generator Tube Rupture (SGTR) Reference (9). This approach was proven to be adequate by the excellent agreement INPO achieved to an actual tube rupture event.

The RETRAN-02 computer code Reference (7) was developed by Energy, Inc., under the sponsorship of the Electric Power Research Institute (EPRI). RETRAN-02 is a one-dimensional, best-estimate, transient thermal-hydraulic computer code capable of simulating most pressurized water reactor transients. It has been reviewed and approved by the Nuclear Regulatory Commission (NRC) for use in the analysis of non-LOCA transients, including steam generator tube ruptures (Reference (8)). Yankee Atomic Electric Company has extensive experience applying the RETRAN code to both the Yankee (Reference (22)) and Maine Yankee (Reference (23)) plants.

The capability of RETRAN-02 to adequately simulate SGTR transients has been demonstrated by post-event analyses of actual tube rupture transients at the Prairie Island and Ginna nuclear power plants (References (9) and (10)). These analyses, performed by the Institute of Nuclear Power Operations (INPO) and the EPRI-sponsored Nuclear Safety Analysis Center (NSAC), demonstrate that the response of each plant to the SGTR predicted using RETRAN-02, compares favorably with recorded plant responses.

4.2 Description of Seabrook SGTR RETRAN Model

Seabrook Station Unit 1 is a four-loop plant with a rated thermal power of 3411 MWt. Its Nuclear Steam Supply System (NSSS) was designed by Westinghouse and is similar to other Westinghouse four-loop NSSSs of the same vintage.

A two-loop RETRAN model was used to perform the analyses presented in this report. One loop in the model represents the single coolant loop containing the ruptured SG and the other model loop represents the three lumped coolant loops with intact SGs.

The nodalization used in the Seabrook SGTR RETRAN-02 model (see Figures 4.1 and 4.2) is very similar to that used by INPO to model the Ginna plant in Reference (9). The primary differences between the Seabrook and INPO Ginna models are that the Seabrook model does not include the detailed modeling of the core bypass and upper head regions used in the Ginna model and does not model the surge line explicitly.

The core bypass and upper head flow paths were not modeled in detail since the limiting SGTR event scenarios for minimum margin to overfill and maximum off-site mass release both assume loss of off-site power at the time of reactor trip. Since the flow through these regions is negligible under natural circulation conditions, the use of a "stagnant" upper head volume and lumping of the core bypass region with the rest of the core region are conservative modeling techniques that minimize computer run time.

The surge line is modeled as a junction. The junction form loss coefficients for forward and reverse flow appropriately account for the hot leg tee junction and pressurizer inlet flow diffuser losses, and the volume of the liquid region of the pressurizer has been artificially increased by the volume of the surge line. The purpose of using this technique is also to minimize computer run time, since the surge line volume would otherwise be the smallest volume in the model, and explicit modeling of this volume would result in the need to use smaller time steps during times when the pressurizer empties and refills than would otherwise be the case. The accuracy of transient results is not affected because of the relatively small volume of the surge line.

In the model descriptions which follow, numbers enclosed in curly braces, e.g., {1}, designate volume numbers used in the model and correspond to the numbers shown in Figures 4.1 and 4.2. Numbers enclosed in

vertical bars, e.g., |2|, designate heat conductors in the model. Numbers enclosed in angled braces, e.g., <3>, designate junction numbers in the model. Heat conductors are not shown in Figures 4.1 and 4.2.

4.2.1 Description of Reactor Vessel Model

The Reactor Vessel (RV) model consists of six volumes, five flow junctions, and seven heat conductors. The six volumes represent the upper {42} and lower {11} portions of the reactor vessel inlet and downcomer annulus, the lower plenum {12}, core and lumped core bypass {1}, outlet plenum {2} and upper head {3} regions. The upper head volume is modeled using the nonequilibrium pressurizer option, since it is expected that liquid in this region (which is stagnant after the loss of forced recirculation flow accompanying the loop) will flash during the event recovery, causing this volume to act as a second pressurizer. The values for bubble rise velocity and void distribution parameters used in this volume are those recommended by Moore and Rettig in Reference (11).

The seven heat conductors represent the fuel |1|, RV lower head metal |3|, RV downcomer outer wall |2,24|, lower and upper core barrel |4,25|, and the portion of the upper guide structure |5| exposed to liquid in the outlet plenum volume.

The RV upper closure head and other metal structures exposed to liquid in the upper head region are not explicitly modeled due to the fact that the upper head volume uses the RETRAN nonequilibrium pressurizer option, which does not allow heat conductors to be specified as connected to this volume. Since there is negligible flow through this volume under natural circulation conditions, this metal would remain hot along with the liquid in the region.

Seabrook is a T_{COLD} upper head plant. This means that the design of the RV internals is such that a sufficient fraction of the cold leg fluid entering the RV is diverted into the upper head to maintain the upper head fluid and metal temperatures at T_{COLD} during normal operation with the RCPs running. Under natural circulation conditions, however, flow direction in this flow path reverses and fluid at T_{HOT} enters the upper head region.

Realistic modeling of these two flow configurations would require a more detailed upper head model than was deemed desirable for this analysis due to anticipated increases in computer runtime. Instead, this behavior is conservatively simulated by assuming a stagnant upper head volume initialized at the core exit fluid temperature. This is conservative because hotter fluid in the upper head tends to keep RCS pressure elevated and thus, maximizes the leak flow rate.

The RETRAN point-kinetics and decay heat models are used to determine the power generation in the core heat conductor.

4.2.2 Description of Intact Coolant Loop Model

A single lumped loop is used in the model to represent the three coolant loops containing intact SGs. This model loop consists of twelve volumes, fifteen junctions, and fifteen heat conductors. The twelve volumes represent the hot leg piping {5}, SG inlet plenums {35}, primary side of the SG U-tubes {7,24,25,26,27,28}, SG outlet plenums {37}, crossover piping and reactor coolant pumps {39,9} and discharge leg piping {41}.

The fifteen heat conductors are used to model the U-tubes [12,13,14,15,16,17], SG inlet and outlet plenum walls and tubesheets [33,35,37,39], and RC piping, pump casing and impeller metal [8,9,27,29].

The RCPs are modeled, in <61>, using the RETRAN centrifugal pump model and built-in quadrant curves for Westinghouse RCPs.

The charging and letdown flow in these loops is modeled via a fill junction <29>.

The SI flow to these loops is modeled via fill junction <50>.

4.2.3 Description of the Ruptured Loop and Pressurizer Models

The model of the single coolant loop containing the ruptured SG consists of thirteen volumes, seventeen junctions, fifteen heat conductors, and three nonconducting heat exchangers.

The thirteen volumes represent the hot leg piping {6}, combined pressurizer and surge line {4}, SG inlet plenum {36}, primary side of the SG U-tubes {8,29,30,31,32,33}, SG outlet plenum {34}, crossover piping and reactor coolant pump {38,10} and discharge leg piping {40}. The RETRAN nonequilibrium pressurizer option is used in volume {4}.

The fifteen heat conductors are used to model the U-tubes [18,19,20,21,22,23], SG inlet and outlet plenum walls and tubesheet [32,34,36,38], and RC piping, pump casing and impeller metal [7,10,11,28,31].

The three nonconducting heat exchangers are used to represent the normal and back-up pressurizer heaters.

The RCP is modeled, in <62>, using the RETRAN centrifugal pump model and built-in quadrant curves for Westinghouse RCPs.

The SI flow to this loop is modeled via fill junction <49>.

As noted earlier, the pressurizer and surge line are represented by a single volume, {4}. This volume is connected to the loop containing the ruptured SG, rather than the model loop representing the lumped intact coolant loops. The effect of this placement was judged to be negligible.

The hydraulic losses associated with the pressurizer surge line are modeled via junction <4>. Pressurizer spray flow is modeled using fill junction <30> and the pressurizer PORV is modeled using junction <31>. The

area of junction <31> is set to a value equivalent to a single PORV and the discharge side of this junction is connected to a semi-infinite steam-filled volume at atmospheric pressure.

4.2.4 Description of SG Secondary and Main Steam Line Models

The secondary side of the lumped intact SGs and the single ruptured SG are each modeled using six volumes, eight junctions, and 24 heat conductors.

The six volumes represent the SG downcomer {15,16}, region inside the tube bundle shroud below the top of the tube bundle {43,44,45,46,47,48}, lumped steam riser and separator region {49,50}, and the combined recirculating water plenum and steam dome regions {19,20}. The action of the steam separators and dryers is simulated by using the RETRAN bubble rise model in volumes {19,20}, with a semi-infinite bubble rise velocity to provide maximum steam-water separation.

The flow of main feedwater to the SGs is modeled using two fill junctions <27,28>, control valves, and appropriate trip and control system logic to determine valve position and pump status.

The emergency feedwater flow to each SG is also modeled using two fill junctions <36,37>, control valves, and appropriate trip and control logic to determine valve position and pump status.

In addition to the 12 heat conductors used to model the SG U-tubes |12-23|, 12 additional heat conductors are used to model the SG vessel walls and internals |40-51|.

The main steam system is modeled using three volumes, nine junctions, and two heat conductors. The three volumes represent the lumped main steam line piping between the intact SGs and the Main Steam Isolation Valves (MSIVs) {21}, the piping between the ruptured SG and its associated MSIV {22}, and the piping between the MSIVs and turbine steam admission valves {23}. The turbine and condenser are simulated by a control valve

junction and semi-infinite volume at atmospheric conditions. The junction loss coefficient is determined by RETRAN during the initialization process such that the resultant flow through the junction to the atmospheric volume matches the rated steam flow rate for the initial condition.

The turbine stop valves and MSIVs are modeled using RETRAN control valves in the appropriate junctions <26,24 and 25>, controlled by trip logic.

The SG MSSVs <53,54> and ASDVs <32,33> are also modeled using RETRAN control valves in flow junctions between the main steam line volumes and the semi-infinite atmospheric pressure volume {100}. The areas of these junctions are computed by control block and trip logic to simulate the valve position control system in the case of the ASDVs, and the staggered opening setpoints of the individual MSSVs, with 3% accumulation.

Heat conductors [52,53] are used to model the metal walls of the main steam piping between the SGs and MSIVs.

Several negative fill junctions (<73>, <74>, <75>, and <76>) were added to the model to represent the EFW pump turbine steam supply line vents, and the EFW pump turbine exhaust for the analysis of the limiting radiological release scenario discussed in Section 9.0. These flow paths were conservatively neglected for the analysis of the limiting margin to overfill scenario discussed in Section 7.0.

4.2.5 Description of the Tube Rupture Model

The location of the tube rupture is assumed to be just above the SG outlet plenum tubesheet. This yields a conservatively high leak flow rate.

The broken tube is simulated by two junctions <51,52>. Junction <51> models the longer portion of the ruptured tube, extending from the SG inlet plenum tubesheet to the rupture location just above the outlet plenum tubesheet. Junction <52> simulates the "stub" portion of the broken tube, extending from the SG outlet plenum upwards to just above the tubesheet.

The flow area of each junction is set to the cross-sectional flow area of a single Model F SG U-tube, and controlled by RETRAN control valves. The valves in these junctions are initially closed. The opening of the break is simulated by fully opening these valves in one-tenth of a second.

The flow loss coefficients determined for each junction account for the tubesheet entrance, tube wall friction, and rupture exit losses for the associated segment of the broken tube, assuming the rupture to be located just above the tubesheet on the "cold leg" side of the tube bundle as described above.

4.2.6 Description of SG Narrow-Range (NR) Level Model

Identification of the ruptured SG is assumed to occur as a result of operator observation of an uncontrolled increase in that SG's NR level. The entire span of the NR level instrumentation is located within model volumes {19} and {20}. These volumes model the combined volumes of the recirculating water plenum and steam dome. The NR level in the model is calculated as the collapsed liquid level in volume {19} or {20} corrected for the difference in elevation between the lower NR level tap and the bottom of the volumes.

The areas of volumes {19,20} are set equal to the estimated average flow area in the region of the SG covered by the SG NR level instrumentation span. The volume inside the steam separator riser tubes has been lumped with the volume of the region inside the tube bundle shroud between the top of the U-tubes and the lower separator deck plate into volumes {49, 50}, since the void fraction in these volumes is the same at the start of the transient.

The lumping of the volume inside the riser tubes into the volume below the separator entrance deck results in a conservative delay in the return of the SG water level onto the NR span after a reactor trip. This is true because the elevations in the model are such that after a simulated reactor trip, volumes {49,50} must fill prior to level returning in volumes {19,20}. As a result, the volume of liquid in the ruptured SG is higher in the model than it would be in the real SG once level returns to the NR span. This results in a conservative delay in the assumed time of

identification and isolation of the faulted SG and a decrease in the calculated margin to overfill. This conservatism has been retained in the analysis results.

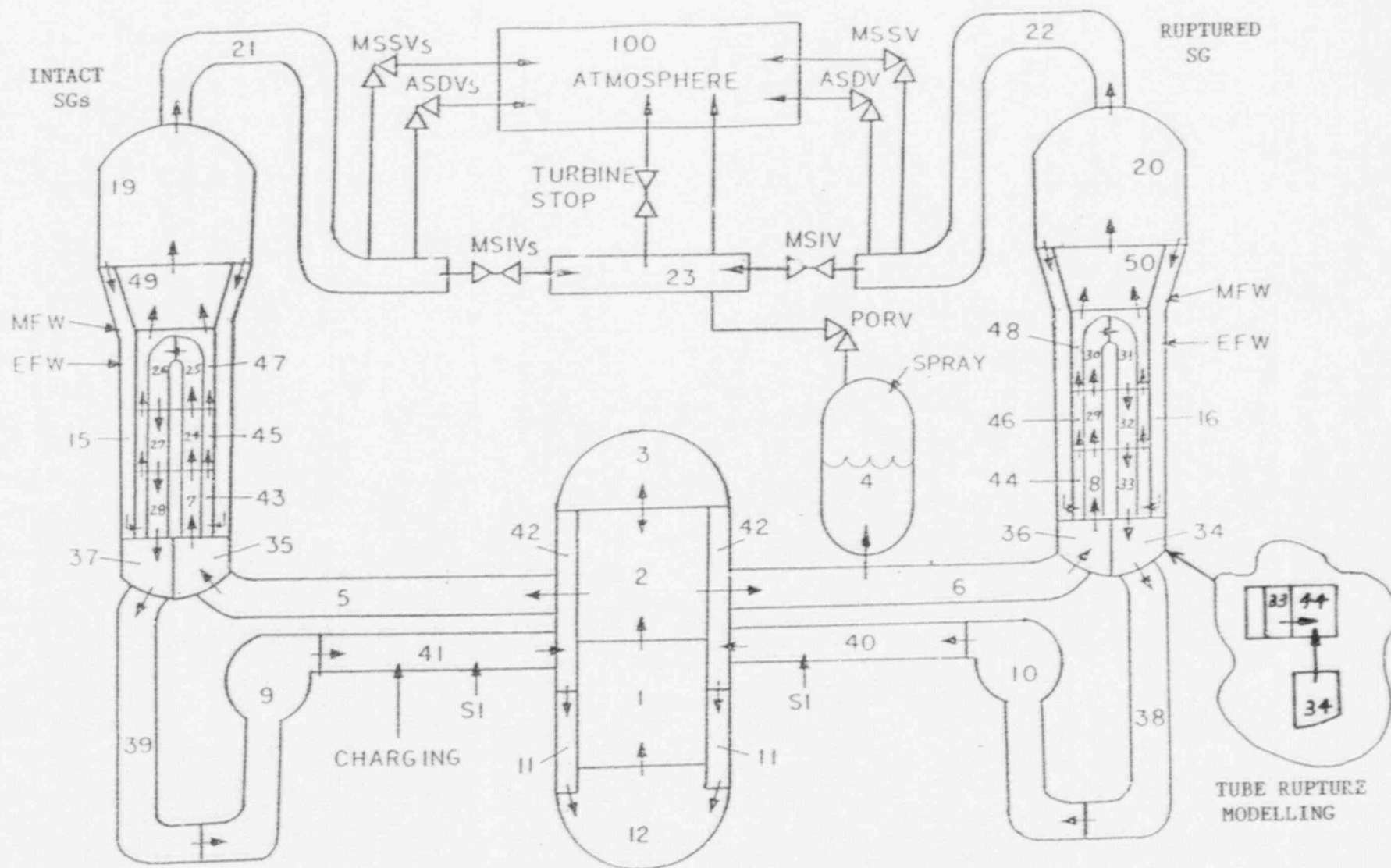


FIGURE 4.1

SEABROOK RETRAN MODEL - VOLUMES

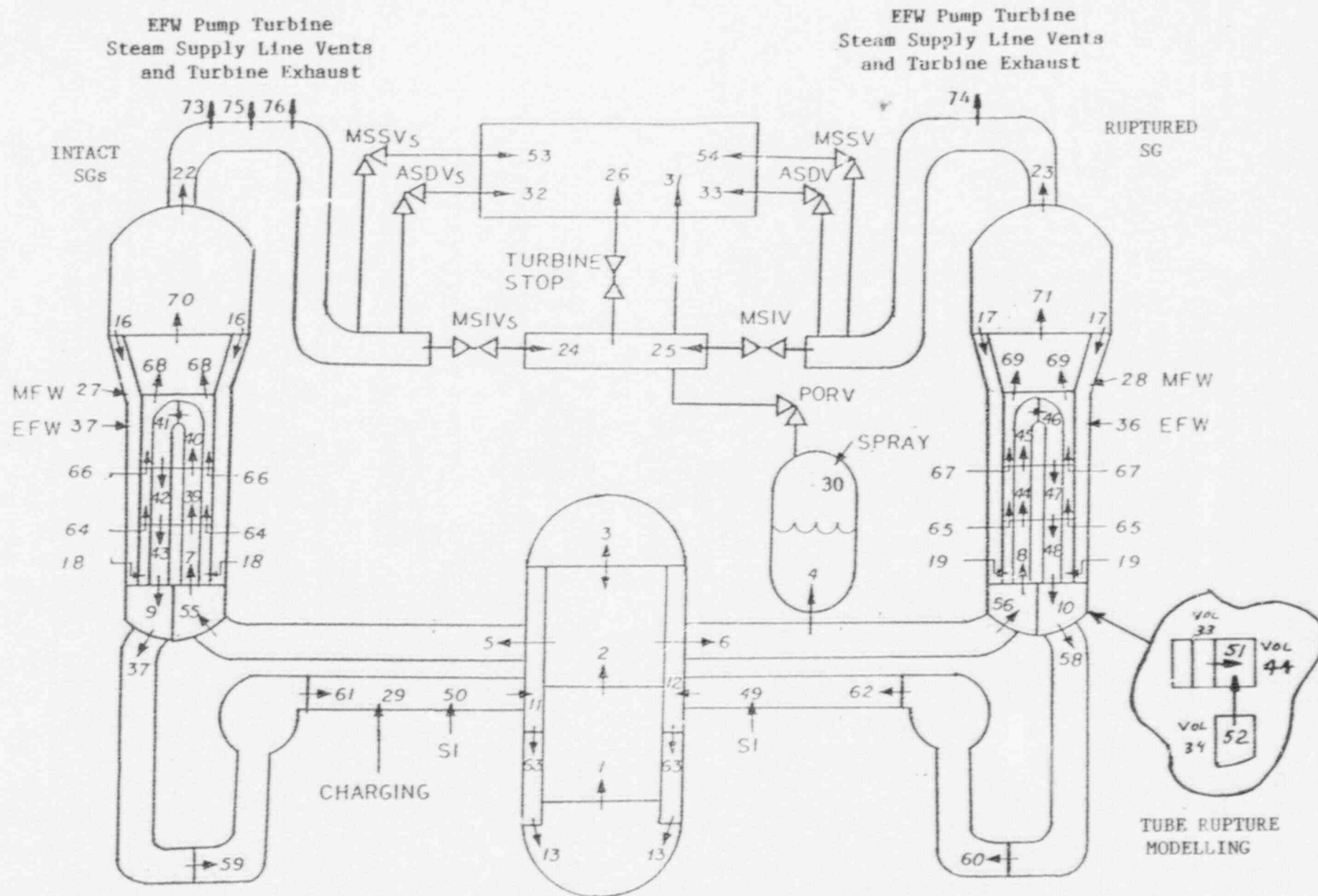


FIGURE 4.2

SEABROOK RETRAN MODEL - JUNCTIONS

5.0 OPERATOR RESPONSE TIME EVALUATION

Mitigation of a SGTR requires manual intervention by the operators during the course of plant recovery from the event. Table 5.1 lists the significant operator actions required to isolate the ruptured SG and terminate break flow. The times assumed for these operator actions in the Seabrook plant-specific analysis are also shown in Table 5.1. The action times assumed were derived from times observed during SGTR simulation runs on the Seabrook plant-specific training simulator and by plant walkdown. Simulator results are shown in Table 5.2. These simulations were performed using design basis SGTR scenario assumptions, regular Seabrook operating crews, and Seabrook plant-specific emergency response procedures.

The scenario for the simulations was a single tube rupture during normal operation at 100% RTP. No credit was taken for possible high radiation readings from the main steam line or condenser air evacuation system radiation detectors. During the simulations, these systems were declared inoperative and could not be used by the operators in the diagnosis of the event. Loss of off-site power was simulated coincident with the reactor trip.

The assumed single failure for the six SGTR scenarios was the failure of one intact SG ASDV to open. This single failure was determined to be the limiting failure with respect to minimizing margin to overfill for the Seabrook plant (see Section 7.0).

The early SGTR simulations assumed a break flow rate (75 lb/sec) equal to the conservatively high flow rate assumed in the Seabrook FSAR. The break flow area assumed in the Seabrook FSAR corresponds to a tube diameter larger than the actual diameter of the tubes used in Seabrook's Model F SGs. The correct initial break flow rate for Seabrook's Model F SGs was determined to be 47 lb/sec. Hence, simulations conducted after Run No. 2 assume a break size set to obtain an initial flow of 47 lb/sec. instead of 75 lb/sec.

Unlike most other Westinghouse plants, Seabrook has both a qualified wide range (WR) and narrow range (NR) SG level channel on each SG. Therefore, WR level indications were available during the simulations for the operators to track SG level. However, the operators were instructed to adhere closely

to existing plant procedures which specifically direct operator attention to NR level in SGTR diagnostic steps. Hence, diagnosis of the event as a tube rupture was confirmed only after ruptured SG level returned onto the NR scale and operators observed the uncontrolled increase in SG level.

The first operator actions of interest are identification and isolation of the ruptured SG. Once the ruptured SG has been identified, the emergency response procedures direct the operators to isolate the SG by terminating EFW flow to that SG (once SG level has returned to 5% NR) and closing the associated MSIV. Table 5.2 provides the ruptured SG NR level at the time of completion of closure of the ruptured SG MSIV during the simulations. SG NR level is used as the measurement criterion rather than the absolute time interval between initiation of the break and completion of isolation in order to provide a common yardstick which is unaffected by differences in assumed initial rupture flow rates among the simulations.

The next operator action of interest is the initiation of the RCS cooldown by opening the intact SG ASDVs. Table 5.2 presents the observed time interval between completion of ruptured SG MSIV closure and opening of the intact SG ASDVs.

During the simulations, the ruptured SG was identified and closure of the associated MSIV initiated shortly after its level returned onto the NR scale. As noted in Table 5.1, the criterion used in the analysis is that the ruptured SG will be identified and isolation completed (closure of the associated MSIV and termination of EFW flow to that SG) by the time its level reaches 33% NR. The observed SG level at the completion of MSIV closure during two of the simulations exceeds 33% NR. Thus, 33% is not a bounding value for this action time, but rather can be considered a median value. As will be explained further below, however, the use of this median value, along with a five minute duration for the time interval between the completion of MSIV closure and initiation of the RCS cooldown and other conservatisms in the model used in the analysis, constitute a conservative combination of assumptions relative to the observed combinations of these two parameters.

The analysis assumptions for SG level at completion of ruptured SG isolation and time interval between MSIV closure and initiation of the RCS

cooldown were 33% NR and five minutes, respectively. These analysis assumptions are not individually bounding but are bounding when considered together in determining the time of initiation of RCS cooldown.

For example, in simulation Number 1, the observed time interval between MSIV closure and initiation of the cooldown exceeded the analysis assumption of five minutes by 22 seconds. The isolation of the ruptured SG, however, was completed at a time when the SG level was 16% below the analysis assumption of 33%. The rate of level increase prior to isolation during this simulation was approximately 9% per minute. Isolation of the ruptured SG was, therefore, completed one minute and 47 seconds sooner than would have been the case if level reached the criterion used in the analysis, 33% NR. Thus, the analysis assumption that closure of the MSIV is not complete until level reaches 33% NR, combined with the assumptions of a five-minute interval between MSIV closure and the start of the cooldown results in roughly a 1-1/2 minute delay in the start of the RCS cooldown relative to the time it would start if the observed combination of action times for this simulation were to have been assumed.

Similarly, although ruptured SG level at the time of isolation observed in simulations two and six exceeds the 33% NR value assumed in the analysis, the observed time intervals between MSIV closure and initiation of the cooldown are much shorter than the five minutes assumed in the analysis. The combination of values assumed in the analysis results in later initiation of RCS cooldown than the times that would result using the observed combinations in both these cases as well. Thus, the combined assumptions of 33% NR as the isolation criterion and the use of a five-minute time interval between completion of isolation and the start of the RCS cooldown results in conservatively long times for initiation of the cooldown (and thereby conservatively delay termination of the rupture).

This combination of assumptions also results in conservative values for the ruptured SG liquid inventory at the time of initiation of the RCS cooldown, even though the observed ruptured SG levels in simulations Numbers 2 and 6 exceed the level criterion used for determining the completion of ruptured SG in the analysis. This is due to two factors.

First, the ruptured SG liquid inventory continues to increase during the time interval between completion of MSIV closure and the start of the cooldown due to continuing leakage from the RCS into the SG. As noted earlier, the value for this interval assumed in the analysis is much longer than the observed interval during these two simulations, resulting in a larger level increase during this interval.

Second, due to the method of modelling the ruptured SG secondary side, as discussed earlier in Section 4.2.6, the SG liquid volume at the time the ruptured SG level in the model returns to 33% NR, is larger than that which would be present in the real SG at that indicated level. The ruptured SG liquid volume at 33% NR in the model corresponds to the liquid volume which would be present at a real SG at a level of 44% NR. This conservatism in the model, in combination with the level increase attributable to the use of a longer than observed time interval between isolation and start of the cooldown, results in conservatively high ruptured SG liquid inventories at the start of the cooldown.

The next operator action of interest is the opening of a pressurizer PORV to reduce RCS pressure. The observed time intervals between the end of the cooldown and the start of the RCS depressurization phase of the recovery process were also bounded by the time interval of two minutes assumed in the plant-specific analysis.

The Seabrook plant-specific analysis assumes a conservative five minute time interval between the end of the first RCS depressurization (defined as closure of the pressurizer PORV) and the termination of SI pump operation. The observed times for this interval during the SGTR simulations performed at the Seabrook simulator were well under two minutes in five of the six runs. In the sixth run (Run No. 4 in Table 5.2), the duration of this interval was approximately 1 minutes and 25 seconds.

During the time interval between PORV closure and termination of SI pump operation, the RCS repressurizes due to the fact that the ECCS flow rate significantly exceeds the rate of leakage from the primary to the ruptured SG secondary side. Longer values for this time interval result in more pronounced increases in RCS pressure, which are more likely to draw the attention of the operators to the fact that the RCS has repressurized and result in their taking action to reduce RCS pressure and terminate the renewed break flow (typically by re-opening the PORV).

This is exactly the situation observed in the simulations at the Seabrook simulator. During Run No. 4, the only run where the time interval for the termination of SI flow exceeded two minutes, the operators noticed the large increase in RCS pressure during this interval and reopened the pressurizer PORV shortly after terminating SI pump flow (1 minute and 20 seconds after terminating SI flow, which corresponds to a point in time roughly 4 minutes and 45 seconds following the initial closure of the PORV at the end of the first depressurization). This action successfully terminated the break flow. In the remaining five simulation runs, where the RCS repressurization was much less pronounced due to the shorter time interval for termination of the SI flow, it took the operators much longer to realize that the RCS pressure had increased above the ruptured SG pressure and open the PORV a second time to reduce RCS pressure and terminate the break flow (the longest observed time was approximately fifteen minutes during Run No. 2).

The analysis includes bounding values for both these two time intervals, i.e., the time interval between the end of the first RCS depressurization and termination of SI flow and the time interval between the first and second openings of the pressurizer PORV. The action times assumed were five minutes and fifteen minutes, respectively. The assumed time intervals bound the individual observed times for all runs.

Besides verifying the operator action time intervals for normal operator actions, the additional time (30 minutes) required by the operators to perform corrective actions in response to single active failures of equipment required to mitigate the SGTR were verified by a walk-through at the Seabrook site (Reference (21)).

The additional time required to terminate SI pump flow by opening the pump breaker in the Electrical Switchgear Room, if an SI pump could not be stopped from the control board, was estimated to be four minutes.

Failure of remotely initiated isolation of the steam supply line from a ruptured SG to the EFW pump turbine is a potential single active failure. Operator response is local isolation within thirty minutes. This is not the limiting single active failure.

Justification for operator entry into the potentially adverse thermal and radiological environments near isolation valves which may have to be closed manually is provided in References (20) and (21).

The simulator runs also indicate that it is a distinct possibility that the operators will manually initiate SI prior to an automatic reactor trip in an attempt to restore pressurizer level. In the simulation run on March 14, 1989, the operators manually initiated SI three minutes and 35 seconds after the tube rupture. Since SI actuation causes a reactor trip, this resulted in a reactor trip much earlier than the automatic RPS trips would have tripped the plant. In order to account for this potential operator action, the plant-specific analysis included a sensitivity study on time of reactor trip to determine the most conservative assumption to use in the analysis.

Varying the assumed time of reactor trip in the Seabrook plant-specific analysis resulted in differing times to SG isolation, but equivalent ruptured SG liquid volumes at the time of ruptured SG isolation, since in each case the ruptured SG was assumed to be isolated when its level returned to 33% NR, which was typically more than ten minutes after the opening of the break.

Hence, the determination of whether to assume an automatic or earlier manual reactor trip was made on a case-specific basis. The bases for the trip times chosen are provided in the discussion for the individual cases presented in Sections 7.0 and 9.0.

TABLE 5.1

Operator Actions/Time Intervals for SGTR Recovery Actions

<u>Description of Action/Interval</u>	<u>Analysis Assumption</u>
Identification and isolation of ruptured SG.	33% NR Level
Time interval between isolation and initiation of cooldown.	5 Minutes
Time interval between end of cooldown and start of depressurization.	2 Minutes
Time interval between end of first depressurization and initiation of SI termination.	5 Minutes
Time interval between end of first depressurization and second attempt.	15 Minutes
Time to identify and isolate an ASDV that fails to close.	30 Minutes
Time to close MSIVs in lines from intact SGs if MSIV in line from ruptured SG fails to close.	2 Minutes
Time to open another pressurizer PORV if first PORV fails to open on demand.	2 Minutes
Time to open SI pump breaker if an SI pump cannot be stopped from the control board.	4 Minutes
Time to locally close EFW pump turbine steam supply isolation valve in the event of failure of remotely initiated isolation.	30 Minutes

TABLE 5.2
Operator Action Times Observed During Design-Basis
Tube Rupture Simulations at Seabrook Simulator

SIMULATION NUMBER (DATE)	1 (7/12/88)	2 (7/23/88)	3 (2/28/89)	4 (3/07/89)	5 (3/14/89)	6 (3/21/89)
RUPTURED SG NARROW RANGE LEVEL @ TIME OF ISOLATION	~ 17%	~ 42%	~ 25%	~ 30%	~ 32%	~ 52%
INTERVAL BETWEEN MSIV CLOSURE AND START OF COOLDOWN	5:22	3:00	2:52	2:54	2:46	2:25
INTERVAL BETWEEN END OF COOLDOWN AND START OF DEPRESSURIZATION	1:03	0:30	0:15	0:17	1:35	0:20
INTERVAL BETWEEN PORV CLOSURE AND TERMINATION OF SI	1:10	1:30	1:30	3:25	1:06	1:15
INTERVAL BETWEEN 1ST PORV CLOSURE AND 2ND OPENING	NA	15:05	1:55	4:45	NA	9:15

6.0 REQUIRED EQUIPMENT AND ITS SAFETY CLASS

Recovery from a SGTR requires the use of a variety of plant systems and equipment. A comprehensive list of the equipment required to function during the recovery process was compiled by reviewing the pertinent Seabrook plant-specific Emergency Response Procedures (ERPs). The procedures reviewed were the principal procedures used by the operators to recover from a design basis SGTR, Procedure E-0, "REACTOR TRIP OR SAFETY INJECTION" and Procedure E-3, "STEAM GENERATOR TUBE RUPTURE." Procedure ECA-3.1, "SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED," was not included since the RCS subcooling and pressure predicted during the recovery process remain above the values which would result in a transition to this procedure.

Table 6.1 provides a summary of the systems (or portions of systems), components, and instrumentation that are credited for mitigation of the SGTR event. All items are nuclear safety related unless specifically noted and justified. The analysis considers single active failures in equipment required to mitigate the SGTR.

TABLE 6.1

Summary of Systems, Components, and Instrumentation Credited for
Mitigation of the SGTR Event

A. SYSTEMS

- o Solid-State Protection
- o Rod Control
- o Reactor Coolant
- o Main Steam, Note 1
- o Main Steam Drains, Note 1
- o Emergency Electrical Distribution
- o Diesel Generator
- o Service Water
- o Primary Component Cooling Water
- o Safety Injection, including the Refueling Water Storage Tank
- o Charging
- o Feedwater, Note 1
- o Condensate (Condensate Storage Tank)
- o Turbine Generator (Turbine Stop Valves), Note 1

Note:

1. Includes non-nuclear safety-related components for backup of safety-related components required for steam generator isolation.

B. COMPONENTS (Notes 1 and 2)

- o Emergency Feedwater (EFW) Flow Control Valves (AC Motor), Note 3
- o EFW Pumps
- o Atmospheric Steam Dump Valves (ASDVs)(Pneumatic), Note 4
- o ASDV Block Valves (Manual)
- o Main Steam Isolation Valves (MSIVs)(Electrohydraulic/Pneumatic), Note 5
- o MSIV Bypass Valves (AC Motor), Note 6
- o Turbine Generator Stop and Control Valves (Electrohydraulic), Notes 7 and 12
- o Main Feedwater Pump Turbine Stop Valves (Electrohydraulic), Note 12
- o Main Steam Drain Valves (MSDVs), Notes 11 and 12
- o Auxiliary Steam System Supply Valve (AC Motor), Note 12
- o Moisture Separator Reheater Steam Supply Valves (AC Motor), Note 12
- o Turbine-Driven EFW Pump Steam Supply Isolation Valves (Pneumatic), Note 8
- o Turbine-Driven EFW Pump Steam Supply Control Valve (Pneumatic)
- o SG Blowdown Isolation Valves (Pneumatic)
- o Main Steam Safety Valves
- o Feedwater Control Valves (Pneumatic), Note 9
- o Feedwater Isolation Valves (Electrohydraulic), Note 9
- o Pressurizer Power-Operated Relief Valves (PORV)(DC Solenoid-Operated Pilot)
- o PORV Block Valves (AC Motor)
- o Safety Injection Reset, Note 10
- o Emergency Diesel Generators
- o Safety Injection Pumps
- o Charging Pumps

TABLE 6.1

Summary of Systems, Components, and Instrumentation Credited for
Mitigation of the SGTR Event
(Continued)

Notes:

1. Motive power for valve is in parenthesis.
2. Only components specifically discussed in the emergency response procedures or Reference (2) are listed. Components that are directly associated with the listed component or are part of required supporting systems listed in Section A above are not listed separately.
3. The EFW flow control valves are in series and fail as-is. No single failure will prevent the isolation of EFW to a SG.
4. There is only one ASDV per SG. A failed open ASDV on the ruptured SG will require isolation with the block valve. No single failure will prevent the opening of more than one ASDV on the intact SGs.
5. The MSIVs have redundant closing controls. There is only one MSIV and actuator for each steam line. Backup for isolating the ruptured SG is provided by the intact SG MSIVs and nonqualified valves in the Turbine Building and has been accepted in Enclosure 3 to the NRC's SER on WCAP-10698, Reference (6).
6. There is only one bypass valve per steam line, and they are all powered from Train A. The MSIV bypass valves are normally closed when the MSIVs are open. Backup is provided by nonqualified valves in the Turbine Building.
7. The turbine generator stop valves are not qualified but are supplied with highly reliable closing controls that have been reviewed by the NRC and found to be adequate for the important function of rapidly stopping flow to the turbine after a reactor trip.
8. There is only one fully nuclear safety-related isolation valve, including pneumatic supply, in each steam supply line. Redundant isolation is available by manual closure of separate nuclear safety-related valves.
9. The single nuclear safety-related isolation valve in the feedwater line to each SG is backed up by a nonqualified feedwater control valve. Redundant closing controls are provided.
10. Safety injection reset is not required but is listed to be consistent with Reference (2).
11. Only the drain valves for the MSIV body (manual) and the steam header upstream of the MSIV (ac motor) are nuclear safety related. The upstream drain valve for each steam header is provided with a manual isolation valve.

TABLE 6.1

Summary of Systems, Components, and Instrumentation Credited for
Mitigation of the SGTR Event
(Continued)

12. Nonnuclear safety-related component used for backup isolation of ruptured SG.

C. INSTRUMENTATION (Note 1)

- o Steam Generator Level, Note 2
- o EFW Flow, Note 3
- o SG Pressure
- o Core Exit Temperature, Note 4
- o Wide-Range Reactor Coolant System (RCS) Pressure
- o Pressurizer Level
- o RCS Subcooling Margin
- o Containment Pressure

Notes:

1. Status indication is provided for all required components and has the same qualifications as the component. This indication is not listed separately.
2. One, fully qualified narrow-range loop is provided for each SG. Backup is provided by the safety-related wide-range loop that is redundant to the safety-related narrow-range loop.
3. There is one qualified EFW flow indication loop per SG. Nonqualified backup indication is provided by the station computer.
4. The redundant average core exit temperature indication is displayed on the nonqualified plasma display at the Shift Superintendent's Station. The value can also be obtained from the station computer or the maintenance panel on MM-CP-486B.

7.0 TRANSIENT ANALYSIS OF THE CASE WITH MINIMUM MARGIN TO OVERFILL

The plant equipment relied upon to mitigate the effects of a design basis tube rupture was identified in Section 6.0 of this report. The designs of these systems were reviewed to identify potential single failures of this equipment. The effects of these single failures were then evaluated in order to determine the limiting single failure with respect to minimizing the margin to overfill of the ruptured SG. The single failure resulting in minimum margin to overfill was found to be the failure of one of the intact SG ASDVs to open. A summary of the analysis of this single failure follows.

7.1 Description of Case Scenario

The initiating event for this scenario is a design basis tube rupture, e.g., the instantaneous guillotine severance of a single SG tube in a single SG. The operators are assumed to manually trip the reactor and initiate SI approximately 2-1/2 minutes after the occurrence of the rupture (see discussion in Section 7.2.2). A loss of off-site power is assumed to occur coincident with the reactor trip.

No further operator actions are assumed or modeled until the level in the ruptured SG returns to an indicated level of 33% NR, by which time it is assumed that the operators will have identified the ruptured SG (based on an uncontrolled level increase in the ruptured SG), closed its associated MSIV, and terminated EFW flow to it. These actions are simulated in the RETRAN run by closing the MSIV and terminating EFW flow to the ruptured SG precisely when the predicted ruptured SG level returns to 33% NR.

From this point in time, the operators are assumed to implement the actions specified in Seabrook ERP E-3. These actions are simulated in the RETRAN analysis and are assumed to occur in accordance with the time intervals determined to be bounding from the Seabrook SGTR simulations (see Table 5.1). Five minutes after isolating the ruptured SG, the operators are assumed to open the intact SG ASDVs to cool the plant down to the target average core exit temperature specified in the E-3 procedure.

EFW flow to the intact SGs is assumed to be throttled to a conservatively low value prior to the start of the cooldown, since level has returned to the level restoration band specified in E-3 (5-50% NR level).

The assumed single failure for this case is the failure of one of the ASDVs in a line from an intact SG to open. This is simulated in the RETRAN analysis by reducing the area of the junction representing the three intact SG ASDVs by one-third. The reduction in steam relief flow slows the cooldown, allowing leakage into the ruptured SG to occur for a longer period of time.

Once the target average core exit temperature has been reached, the operators are assumed to control the intact SG ASDVs to maintain the target average core exit temperature, and after two minutes open a single pressurizer PORV to reduce RCS pressure. The PORV is closed when one of the three conditions specified in the E-3 procedure for closing the valve is met.

Five minutes later, the operator is assumed to terminate SI flow per procedure. If the PORV is not reopened, RCS pressure will gradually decrease due to continued break flow until it equalizes with the ruptured SG pressure. This action terminates the break flow and the analysis of the event.

7.2 Initial Conditions and Conservative Assumptions

The event was initiated from conditions which were determined to be conservative with respect to minimizing the resultant margin to overfill. Table 7.1 provides a summary of these conditions.

Westinghouse performed a series of sensitivity studies in Reference (2) to determine the most conservative direction for changes to the nominal initial operating conditions with respect to reducing margin to overfill. In order to determine the most conservative direction and parameter values for the Seabrook plant-specific analysis, the logic behind Westinghouse's choices in Reference (2) was reviewed to determine applicability to Seabrook in light of any differences in plant systems or differences in assumed operator action times.

7.2.1 Initial Power Level

The analysis assumes the plant to be initially operating at 100% RTP. The Seabrook FSAR gives a power level uncertainty of 2% RTP. For the overfill analysis, the most conservative direction for a change in power level with respect to maximizing SG initial inventory is to reduce the nominal initial power by the 2% RTP uncertainty. The initial SG liquid mass increases with decreasing power level due to the lower void content in the tube bundle region. However, this would also result in less decay heat generation during the cooldown phase of the SGTR recovery, and thus, would be nonconservative from the viewpoint of prolonging the time to break flow termination. Hence, rather than reducing the plant power level, the analysis conservatively assumes an initial power level of 100% RTP, but uses an initial SG inventory in the RETRAN model corresponding to 98% RTP. Additional modifications in initial SG inventory were made to bound operation at lower power levels and to account for the effect of automatic turbine run back. These modifications are discussed in Sections 7.2.4 and 7.2.17 below.

7.2.2 Initial RCS Pressure and Time of Reactor Trip

The analysis assumes a conservatively high initial operating pressure. The reason for assuming a high initial RCS pressure follows.

Variations in initial RCS pressure can influence the plant response to a tube rupture in two ways. First, the time of an automatic reactor trip signal on low pressurizer pressure is affected. A higher initial pressure delays the low pressurizer pressure signal. Second, the break flow rate is affected. A higher RCS pressure yields a greater differential pressure across the rupture site. This, in turn, yields a higher leak flow rate.

The impact of a delayed low pressure trip is discussed first. The time of ruptured SG isolation for the Seabrook plant-specific analysis is assumed to be the time of return of the ruptured SG level to 33% NR level. Since the Steam Generator Water Level Control System (SGWLCS) compensates for the break flow into the ruptured SG up until the time of main feedwater isolation and the ruptured SG is assumed to be isolated at a fixed SG level, the time of reactor trip has negligible influence on margin to overfill for the Seabrook

analysis. The same liquid inventory condition will exist at the time of ruptured SG isolation regardless of reactor trip time. This was verified via a RETRAN sensitivity study in which the reactor was first assumed to be tripped manually at 140 seconds after the tube rupture versus a case where no manual trip was assumed and only the automatic low pressure RPS trip was credited. The difference in margin to overfill was less than 12 seconds where margin to overfill is determined as the remaining ruptured SG void volume divided by the "equilibrium" break volumetric flow rate as in Reference (2). Determination of a conservative initial pressure for the Seabrook analysis was therefore based on criteria other than its potential effect on time of generation of an automatic RPS reactor trip.

The effect of initial pressure on break flow rate is discussed now. If the analysis were to assume an automatic low pressure RPS trip, any effect of higher or lower initial RCS pressure on the rupture flow rate would be moot, since the flow into the ruptured SG is automatically compensated for by the SGWLCS prior to the trip and the RCS pressure (and, therefore, rupture flow) at the time of the trip would be the same regardless of initial pressure. However, the assumption of an early manual reactor trip and SI actuation results in a higher RCS pressure and pressurizer level at the time of the trip, which results in a higher RCS pressure and rupture flow rate after the trip. The RCS pressure remains higher until the start of the depressurization phase of the event recovery, resulting in a larger primary to secondary ΔP , and therefore, a higher rupture flow rate during the event.

The initial condition for RCS pressure assumed in the Seabrook plant-specific overfill analysis therefore is a conservatively high RCS pressure, combined with an early manual reactor trip and manual SI actuation.

7.2.3 Pressurizer Water Level

A higher initial pressurizer level is conservative since it delays the point in time when the pressurizer empties and, thereby, results in higher RCS pressures, primary to secondary ΔP and, thereby, a higher rupture flow rate

for a longer period of time. This reduces the margin to SG overfill. Therefore, the Seabrook plant-specific analysis assumed a conservatively high initial pressurizer level.

7.2.4 Initial Steam Generator Secondary Mass

Intuitively, one would expect a higher initial SG liquid mass reduces the margin to overfill. Thus, as previously discussed, the analysis assumed an initial SG mass consistent with operation at the SG normal water level for an initial power level of 98% RTP. The initial SG level was then increased to the high level deviation alarm setpoint plus an allowance for uncertainty.

For the analysis of the case with minimum margin to overfill, the initial SG inventory was further increased by an additional 11,500 lb., per SG, to bound the effects of operation at lower power levels (down to 70% rated thermal power) or automatic turbine runback.

However, since the criterion used in the analysis for determining the time of ruptured SG isolation is the time ruptured SG level returns to 33% NR after the reactor trip, the SG inventory at the time of isolation is not affected by this additional initial mass, only the time of isolation is affected. Small variations in the initial SG inventory have negligible influence on margin to overfill.

7.2.5 Break Location

The Seabrook plant-specific analysis assumes the location of the rupture to be on the cold leg side of the SG just above the SG outlet plenum tubesheet. This yields a conservatively high leak flow rate.

7.2.6 Off-Site Power Availability

The analysis assumes a loss of off-site power to occur coincident with the reactor trip. The loss of off-site power at reactor trip results in less margin to SG overfill than continued off-site power availability.

The physical reasons for this are the higher average core exit temperature which exists after the trip at the lower natural circulation RCS flow rates relative to that which would exist if forced circulation with the RCPs was maintained. This results in longer cooldown times due to the larger difference between initial and final cooldown temperatures and the reduced heat transfer efficiency on natural circulation.

The rate of cooldown is also substantially slowed for a loss of off-site power by the resulting loss of condenser vacuum, which results in the inability to use the steam dump system. The SG ASDVs are used for the cooldown instead, and since they have a lower capacity than the steam dumps, the rate of cooldown is much slower. A slower cooldown results in a larger integrated break flow into the ruptured SG, decreasing the margin to overfill.

7.2.7 Reactor Trip Delay

Since the analysis assumes an early manual reactor trip, no delay in the trip signal was assumed for plant-specific analysis.

7.2.8 Turbine Trip Delay

Following a reactor trip, the turbine is automatically tripped. Since some steam would be extracted from the ruptured SG during this interval, one would intuitively expect that it is nonconservative for the overfill analysis to account for this delay. However, since the assumed time of ruptured SG isolation in the overfill analysis is based on return of ruptured SG NR level to 33% NR, any loss of inventory due to the delay in turbine trip would be made up before ruptured SG level returned to 33% NR. Thus, the length of the turbine trip delay has no influence on the margin to overfill, and no delay was assumed.

7.2.9 Steam Generator Relief Valve Pressure Setpoint

The use of the SG relief valve (MSSV or ASDV) with the lowest setpoint was determined to be conservative.

A lower relief valve setpoint results in smaller margins to overfill since it results in lower ruptured SG pressures after isolation. A lower ruptured SG pressure results in the need to extend the RCS cooldown to lower temperatures than would otherwise be the case in order to achieve the desired degree of RCS subcooling relative to the ruptured SG pressure which underlies the cooldown target temperature table in the E-3 procedure. Since the intact SG pressures must be reduced further in order to cool the RCS to the lower temperatures, the steam flow rate through the intact SG ASDVs is reduced, which reduces the rate of temperature decrease. Thus, it takes longer to cool the RCS down to a lower temperature (even if the overall cooldown temperature differential is the same), resulting in larger integrated break flows during the cooldown process and lower margin to overfill. Therefore, the Seabrook plant-specific analysis assumed the ASDVs to function since they are the relief valves with the lowest setpoint.

7.2.10 Low Pressurizer Pressure Reactor Trip Setpoint

The analysis of the case with minimum margin to overfill conservatively assumed a manual reactor trip for reasons noted earlier.

7.2.11 Pressurizer Pressure for SI Initiation

The analysis of this case assumed an early manual SI initiation and reactor trip, thus the pressure setpoint for SI actuation is moot. Early SI actuation is conservative since it helps maintain a higher RCS pressure after the trip and thereby maximizes break flow.

7.2.12 Safety Injection Flow Rate

In order to maximize the RCS pressure and break flow rate, maximum SI flow rates were assumed.

7.2.13 Emergency Feedwater Flow

Due to the ruptured SG isolation criteria used in the plant-specific overfill analysis, minor differences in the amount of EFW flow to the ruptured

SG have no impact on margin to overfill, since EFW is assumed to be isolated to the ruptured SG once its level reaches 33% NR, and the ruptured SG inventory will be the same at the time of EFW isolation. Therefore, the best-estimate EFW flow rate was used for the portion of the transient prior to ruptured SG isolation.

Differences in EFW flow to the intact SGs after ruptured SG isolation have an influence on the rate of RCS cooldown, since the EFW water is cooler than the RCS water. Thus, lower EFW flow rates will result in slightly longer cooldown times, increasing the amount of time RCS water flows into the ruptured SG, and thereby, reducing margin to overfill. The EFW flow rate to the intact SGs after ruptured SG isolation was reduced to a total of 500 gpm in the analysis of the case with minimum margin to overfill in order to conservatively account for this effect.

7.2.14 Emergency Feedwater System Delay

No delay in EFW delivery to the SGs, after EFW initiation, was assumed for the overfill analysis. This maximizes the EFW flow to the ruptured SG; however, the ruptured SG inventory at the time of isolation is unaffected. The additional EFW flow to the intact SGs, prior to ruptured SG isolation and the start of the RCS cooldown, has no effect on the cooldown either, since this fluid will be heated to saturation temperature in the intact SGs in the process of decay heat removal.

7.2.15 Emergency Feedwater Temperature

The temperature of the EFW affects the time required to cooldown the RCS using the intact SGs. A higher EFW temperature increases the time required to cooldown and is, therefore, conservative with respect to margin to overfill. The overfill analysis therefore assumed the maximum EFW temperature, 100°F.

7.2.16 Chemical and Volume Control System and Pressurizer Heater Operation

For the plant-specific overfill analysis, normal operation of the CVCS and pressurizer heaters prior to the reactor trip and loss of off-site power was assumed, since this maximizes the RCS pressure at the time of the trip and subsequent primary to secondary delta-P and break flow rate.

7.2.17 Turbine Runback

The potentially adverse effects on margin to overfill of an automatic turbine runback were discussed in Reference (2). Briefly, the pressure reduction in the RCS caused by the tube rupture causes a reduction in the overtemperature Δ -T trip setpoint. This results in periodic automatic reductions in turbine load and, by action of the T_{avg} control system (using automatic control rod motion), a reduced core power level. The SG level control system responds to maintain SG level stable, and since the amount of steam voiding in the tube bundle is reduced at lower power levels, the SG liquid inventory at the time of trip and ruptured SG isolation is thereby increased.

For the Seabrook plant-specific analysis, however, since ruptured SG isolation is assumed to occur at a fixed SG NR level, the effects of a turbine runback are negligible. The runback increases the SG inventory at the time of reactor trip, but this only results in a slightly shorter time interval for the time between the trip and recovery of the ruptured SG level to 33% NR, whereupon ruptured SG isolation is assumed.

This was confirmed as part of the trip time sensitivity analysis discussed earlier, wherein the difference in overfill margin between a case assuming an early manual trip and one crediting only the automatic low pressure trip, in which a turbine runback was simulated, was found to be negligible. Hence, turbine runback was not explicitly modeled in the analysis of the case with minimum margin to overfill (although as mentioned earlier, the initial SG inventory was increased by 11,500 lbs. to bound the effects of either turbine runback or operation at power levels down to 70% RTP).

7.2.18 RCP Operation

As discussed earlier under the section on the availability of off-site power, loss of power to the pumps is conservative with respect to minimizing margin to overfill.

7.2.19 Decay Heat

Maximizing decay heat lengthens the time required to cool down the RCS. As noted earlier in the discussion on availability of off-site power, longer cooldown times result in reduced margin to overfill. Therefore, the analysis assumes maximum decay heat. RETRAN has built-in the 1971 ANS decay heat standard, including the effect of actinides. An uncertainty of 20% was applied.

7.2.20 Average Temperature

As noted earlier in Section 4.0, the RETRAN model used in the plant-specific analysis conservatively assumes the RV upper head to be a stagnant volume, initialized with a fluid temperature equal to T_{HOT} . A higher initial RCS average temperature results in a higher initial core exit temperature, and therefore, a higher RV upper head fluid temperature. Since the RV upper head region is relatively stagnant after the loss of off-site power, fluid in it remains hot and flashes during the depressurization accompanying either the cooldown of the RCS or the opening of the PORV. The upper head then acts as a second pressurizer, helping to maintain RCS pressure. Thus, a hotter initial upper head fluid temperature is conservative for the overfill case since it will result in higher RCS pressures during the cooldown and/or depressurization, resulting in larger break flow rates than would otherwise be the case.

Therefore, the analysis assumed a conservatively high initial average temperature.

A higher initial RCS average temperature will increase the steam release required to cool down the RCS to the post-trip temperature, thereby increasing the integrated steam release and off-site dose. This effect reduces the ruptured SG inventory immediately after the reactor trip, but has no effect on the ultimate margin to overfill since any additional mass release must be made up before the ruptured SG is assumed to be isolated; i.e. before ruptured SG level returns to 33% NR.

Since the SG relief valve setpoint determines both the post-reactor trip RCS temperature and the lower bound of the RCS cooldown via the target temperature table in the E-3 procedure, the post-trip cooldown span is independent of the assumed initial core average temperature.

7.3 Transient Description

Table 7.2 provides the sequence of events for the case with minimum margin to overfill. Figures 7-1 through 7-18 provide plots of the parameters of major interest.

In the following discussion, letters enclosed within square brackets, [A], indicate labeled points of interest on the plots.

7.3.1 RCS Response

7.3.1.1 Break Initiation to Ruptured SG Identification

The rupture is assumed to occur at ten seconds into the RETRAN run. After the tube ruptures, the flow of coolant out of the primary system results in a gradual reduction in primary system pressure and pressurizer level, causing the charging system to increase charging flow as the deviation between the actual and the CVCS "target" level setpoint slowly grows. Since the leakage flow rate exceeds the maximum make-up flow capacity of the CVCS, pressurizer level and pressure continue to decrease uncontrollably.

The operators are assumed to initiate a manual reactor trip at 150 seconds [A] and to manually initiate SI ten seconds later at 160 seconds. Coincident loss of off-site power is assumed to occur at the reactor trip.

Core power level and primary system temperatures are nearly constant until the reactor is tripped by the operator. The rapid decrease in core heat production causes the core average and exit temperatures to drop rapidly. This drop in coolant temperature results in a shrinkage of the coolant volume and an accompanying drop in RCS pressure from approximately 2240 psia at the time of the trip to 2020 psia shortly thereafter [A].

The reactor trip results in a turbine trip and the closing of the turbine stop valves, bottling up the SG secondary side. As a result, primary to secondary heat transfer is reduced causing the cold leg temperatures to begin to rise. Pressure in the main steam lines rises to the ASDV and MSSV setpoints, causing these valves to open. Primary to secondary heat transfer then increases, which results in a reduction of cold leg temperatures to close to the saturation temperature of the SG secondary side.

The loss of off-site power results in the loss of motive power for the RCPs which coast down, resulting in decreasing RCS flow rates until natural circulation is established.

The loss of power also causes the Emergency Feedwater (EFW) pumps to be actuated. No delay between start of the EFW pumps and actual feedwater delivery to the SGs was assumed for this scenario.

The effects of the loss of forced reactor coolant flow on RCS temperatures can be observed in Figure 7.2, which shows the resulting increase in hot leg temperature between about 180 seconds and 300 seconds, during the development of sufficient driving head for natural circulation to occur.

As noted earlier, the operators were assumed to manually actuate SI ten seconds after tripping the reactor trip. This action causes the SI pumps to start and the CCP pumps discharge to be redirected to the cold legs. It

also results in the tripping of the steam-driven MFWPs and closure of the main feed line isolation and control valves.

The increase in primary system mass inventory, caused by the SI flow, results in the RCS pressure increasing to around 2200 psia by 1100 seconds and pressurizer level increasing very gradually by a few percent. RCS temperatures drop slightly due to the combined effect of the drop-off of decay heat and the relatively cold temperature of the injected water.

7.3.1.2 Ruptured SG Isolation and RCS Cooldown

Since no credit is taken for the expected high radiation indications on the main steam line and or condenser air evacuation system detectors, which are part of the Radiation Data Management System, the operator is assumed to identify the ruptured SG by observation of the uncontrolled increase in level. The ruptured SG is assumed to be isolated by closure of the associated MSIV and termination of EFW flow to it by the time its level reaches an indicated 33% NR, at 768 seconds [B]. This assumption is supported by observations of the action of Seabrook Station operating crews during several simulated SGTR scenarios at the plant simulator.

Five minutes following isolation, the operators are assumed to open the operable intact SG ASDVs [C], per procedure, to begin cooling the RCS to the target core exit temperature specified in Step 14.a of the E-3 procedure. The purpose of the cooldown is to establish adequate subcooling to allow RCS pressure to subsequently be reduced to the ruptured SG pressure without causing voids to form in the reactor core region.

EFW flow to the intact SGs is assumed to be throttled to a conservatively low value prior to the start of the cooldown, since level has returned to the level restoration band specified in E-3 (5-50% NR level).

The cooldown proceeds until the core exit temperature specified in the E-3 procedure is reached, at which time the operator closes the intact SG ASDVs [D] and sets them to maintain this temperature.

The cooldown results in significant shrinkage of the volume of the RCS coolant inventory causing a reduction in RCS pressure to about 1770 psia and the pressurizer level dropping to around 15%.

7.3.1.3 RCS Depressurization and Termination of Break Flow

Two minutes after terminating the cooldown, the operator is assumed to open one pressurizer PORV, per procedure, in an attempt to further reduce RCS pressure below the ruptured SG pressure [E]. As a result, since there is very little coolant flow into the RV upper head region under natural circulation cooling conditions, and the water and metal in this region have remained hot during the cooldown, a steam bubble forms in the upper head when the RCS pressure drops below the saturation pressure of the fluid in this region.

The steam bubble in the RV upper head volume increases in size as pressure is decreased and more of the liquid in that region flashes to steam. The formation of the steam bubble causes a displacement of coolant from the upper head into the pressurizer during the depressurization. This phenomenon is the cause of the "bump" in the plot of intact loop hot leg temperature during this time interval, which shows the effect of the outsurge of hot fluid from the upper head into the RV outlet plenum.

The venting of the pressurizer steam space through the PORV results in RCS pressure decreasing rapidly to 1225 psia (the ruptured SG pressure is still around 1140 psia), at which time the operator is assumed to close the PORV [F] since the pressurizer level has reached 75% - one of the PORV closure criteria in the E-3 procedure.

After another five minutes have elapsed, the operator is assumed to terminate SI flow [G]. During this time interval, the continuing safety injection flow repressurizes the RCS to around 1725 psia, causing renewed leakage from the RCS into the ruptured SG (the ruptured SG pressure is still 1140 psia).

The RCS pressure decreases gradually during the next 30 minutes due to the continuing break flow. Break flow stops when the RCS pressure drops to the ruptured SG pressure [H] at 4690 seconds.

7.3.2 SG and Secondary System Response

7.3.2.1 Break Initiation to Ruptured SG Identification

Steam Generator pressures remain steady until the reactor trip [A]. The reactor trip results in a turbine trip and the closing of the turbine stop valves, bottling up the SG secondary side. As a result, pressure in the main steam lines rises to the ASDV and MSSV setpoints, causing these valves to open. Shortly after the trip, the reduced heat input from the primary (due to the drop in core power level and exit temperature) results in SG pressures falling to just below the ASDV set pressure.

The main steam flow is constant until the turbine trip at 150 seconds [A], whereupon flow stops as the turbine stop valves close.

The ruptured SG level shows a small increase shortly after the break opens, until the action of the level controller can reduce the MFW flow rate to compensate for the leak flow. The level drops out of the NR span in all SGs immediately after the reactor trip [A].

Level returns onto the narrow-range scale in the ruptured SG first, reaching 33% at 768 seconds [B], at which time the intact SG levels have yet to return on-scale.

7.3.2.2 Ruptured SG Isolation and RCS Cooldown

Once the operators identify the ruptured SG, the MSIV in the steam line from the ruptured SG is closed, and EFW to the ruptured SG is terminated. These actions are assumed to occur when the ruptured SG level reaches 33% NR at 768 seconds [B]. These actions result in a pressure rise in the ruptured SG until its ASDV reopens. The action of the ASDV maintains ruptured SG pressure near 1140 psia throughout the remainder of the transient.

The ruptured SG narrow-range level continues to increase during this time interval until it goes off-scale high at around 2000 seconds.

The intact SG levels return onto the narrow range scale around 900 seconds, and gradually increase until the start of the RCS cooldown [C]. At this time there is an initial swell of intact SG level as bubbles form in the tube bundle region from the renewed steaming, causing water to be displaced into the SG recirculating water plenum. Since the steaming rate exceeds the throttled EFW flow rate, level then falls during the cooldown.

The operators are assumed to open the operable intact SG ASDVs to cooldown the RCS five minutes after the ruptured SG level returned to 33% NR [C]. This results in a gradual drop in intact SG pressures until the target core exit temperature is reached and the valves are closed [D]. Pressure in the intact SGs then remains relatively stable through the end of the transient.

7.3.2.3 RCS Depressurization and Break Flow Termination

The liquid volume in the ruptured SG continues to increase until the RCS pressure is finally reduced below the ruptured SG pressure around 4690 seconds [H].

The liquid volume in the ruptured SG at this time is approximately 5675 cubic feet, which corresponds to a margin to overfill of 227 cubic feet. This corresponds to a margin to overfill of 3.6 minutes at the equilibrium break flow rate for Seabrook.

TABLE 7.1

Summary of Conservative Assumptions and Initial Conditions
for Case With Minimum Margin to Overfill

<u>Parameter</u>	<u>Assumption</u>
Off-Site Power	Lost on Reactor Trip
Initial Power	100% RTP
Initial RCS Pressure	2,300 psia
Initial Pressurizer Water Level	71% Span
Initial Core Average Temperature	594.5°F
RCS Flow Rate	Thermal Design
Initial Steam Generator Pressure	1,000 psia
Additional Steam Generator Initial Mass	+11,500 lbm
Reactor Trip Delay	None
Turbine Trip Delay	None
Steam Generator Atmospheric Steam Dump Valve Setpoint	1,125 psig
Reactor Trip	Manual at 140 seconds
ECCS Actuation	Manual at 150 seconds
Main Feedwater Isolation	On Safety Injection Actuation
Emergency Feedwater Flow Initiation Delay	None
Safety Injection Initiation Delay	None
ECCS Flow Rates	Maximum
Emergency Feedwater Flow Rates	Best-Estimate Prior to Initiation of RCS Cooldown; Minimum 500 gpm Thereafter.
Pressurizer Pressure and Level Control	Prior to Reactor Trip
Steam Generator Water Level Control	Prior to Reactor Trip
Decay Heat	120% ANSI/ANS-5.1-1973
Limiting Single Active Failure	ASDV on an Intact SG Fails Closed

TABLE 7.2

Sequence of Events for Case With
Minimum Margin to Overfill

<u>RETRAN Time (Seconds)</u>	<u>Event/Condition</u>	<u>Transient Time* (minutes/seconds)</u>
0	Steady-state operation at 100% RTP	
10	Complete severance of one U-tube	0:00
150	Manual Reactor trip with coincident loss of off-site power (RCPs trip, EFW actuated)	2:20
160	Manual SI Actuation (CVCS isolated and MFWPs trip)	2:30
161	SG ASDVs open	2:31
162	SG MSSVs open	2:32
180	SG MSSVs close	2:50
662	Ruptured SG level returns to 33% NR (EFW to ruptured SG isolated, and MSIV in associated steam line closed)	10:52
962	Operator opens intact ASDVs to start RCS cooldown	15:52
962	Intact SG level >5% NR, operators throttle EFW to intact SG	15:52
2581	E-3 target average core exit temperature reached, operator throttles intact SG ASDVs	42:51
2701	Operator opens one pressurizer PORV to depressurize RCS below ruptured SG pressure	44:51
2822	Pressurizer level $\geq 75\%$, operator closes pressurizer PORV	46:52

*Relative to break initiation.

TABLE 7.2
(Continued)

Sequence of Events for Case With
Minimum Margin to Overfill

<u>RETRAN Time</u> <u>(Seconds)</u>	<u>Event/Condition</u>	<u>Transient Time*</u> <u>(minutes/seconds)</u>
3122	Operator terminates SI flow	51:52
4690	RCS and ruptured SG pressures equalized	78:00
	(Maximum ruptured SG liquid volume = 5675 ft ³)	

*Relative to break initiation.

Figure 7.1

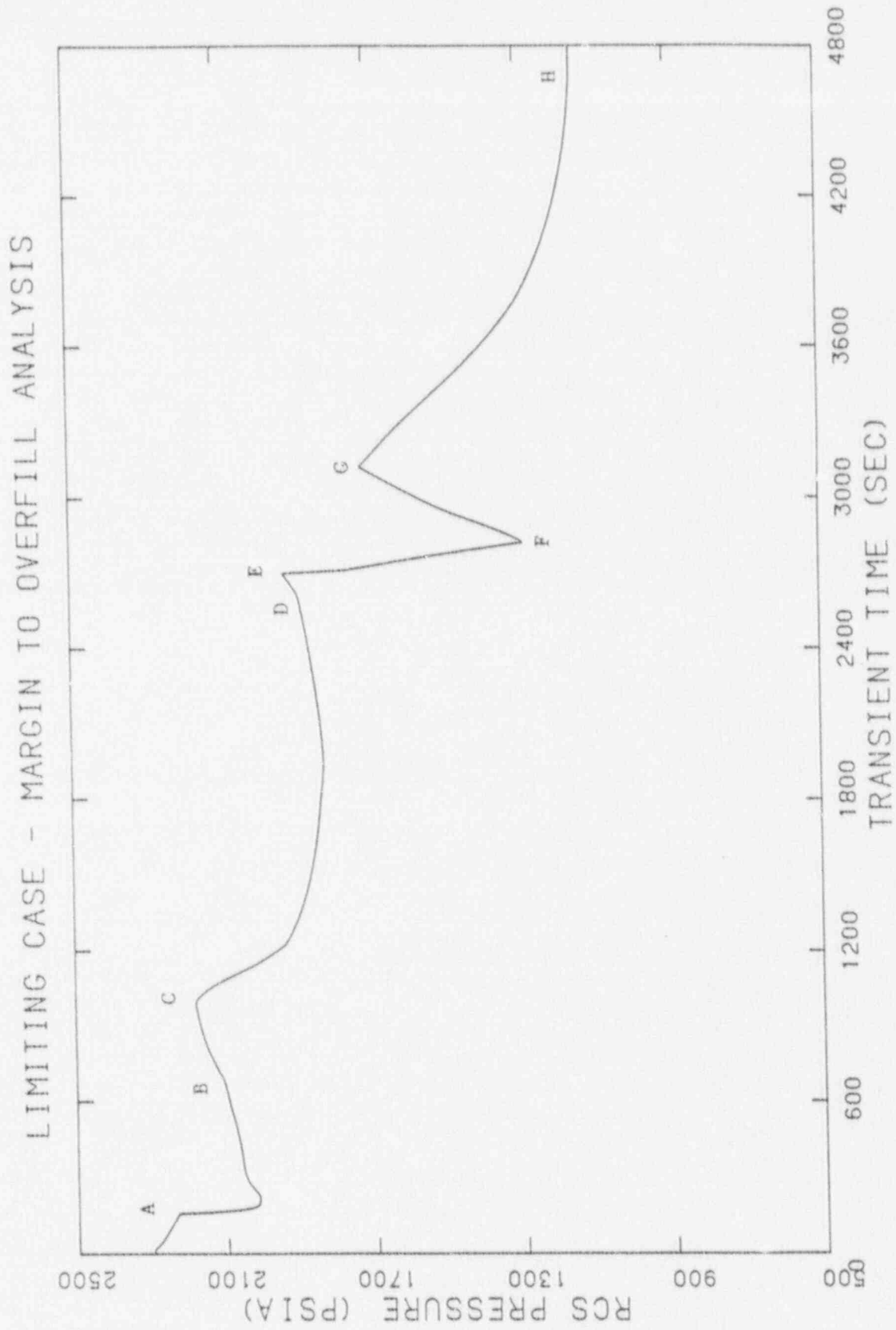
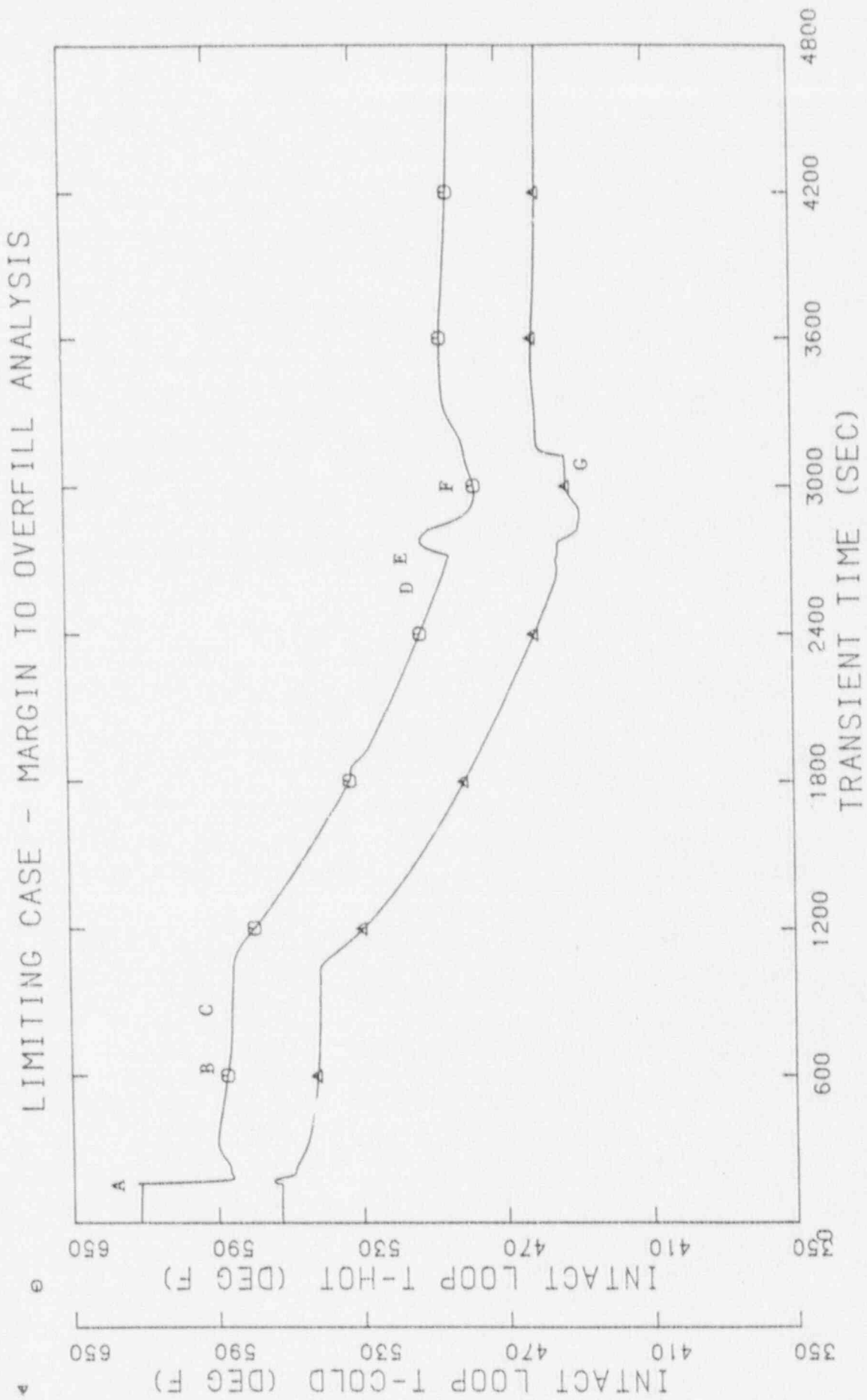


Figure 7.2



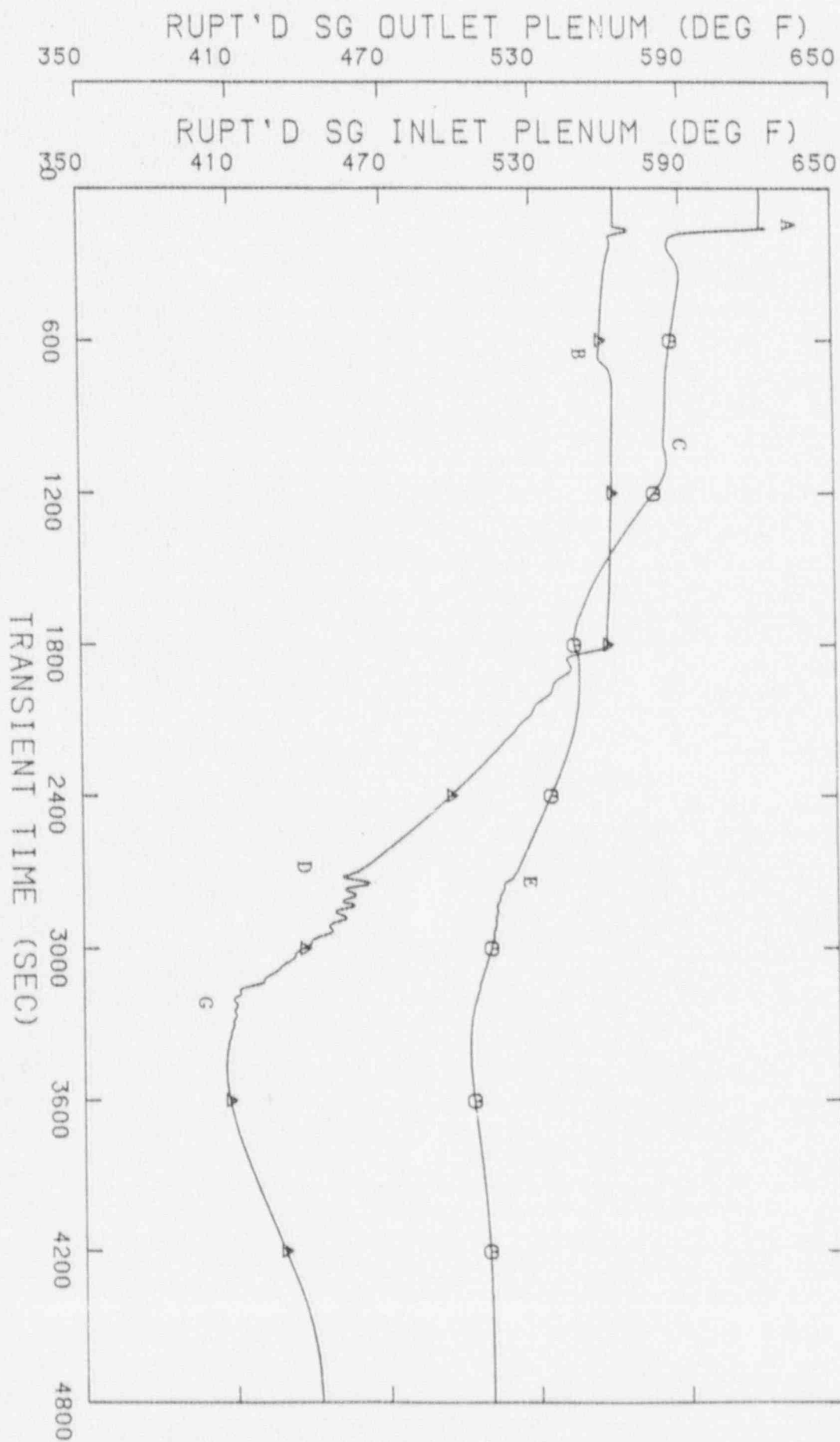


Figure 7.3

LIMITING CASE - MARGIN TO OVERFILL ANALYSIS

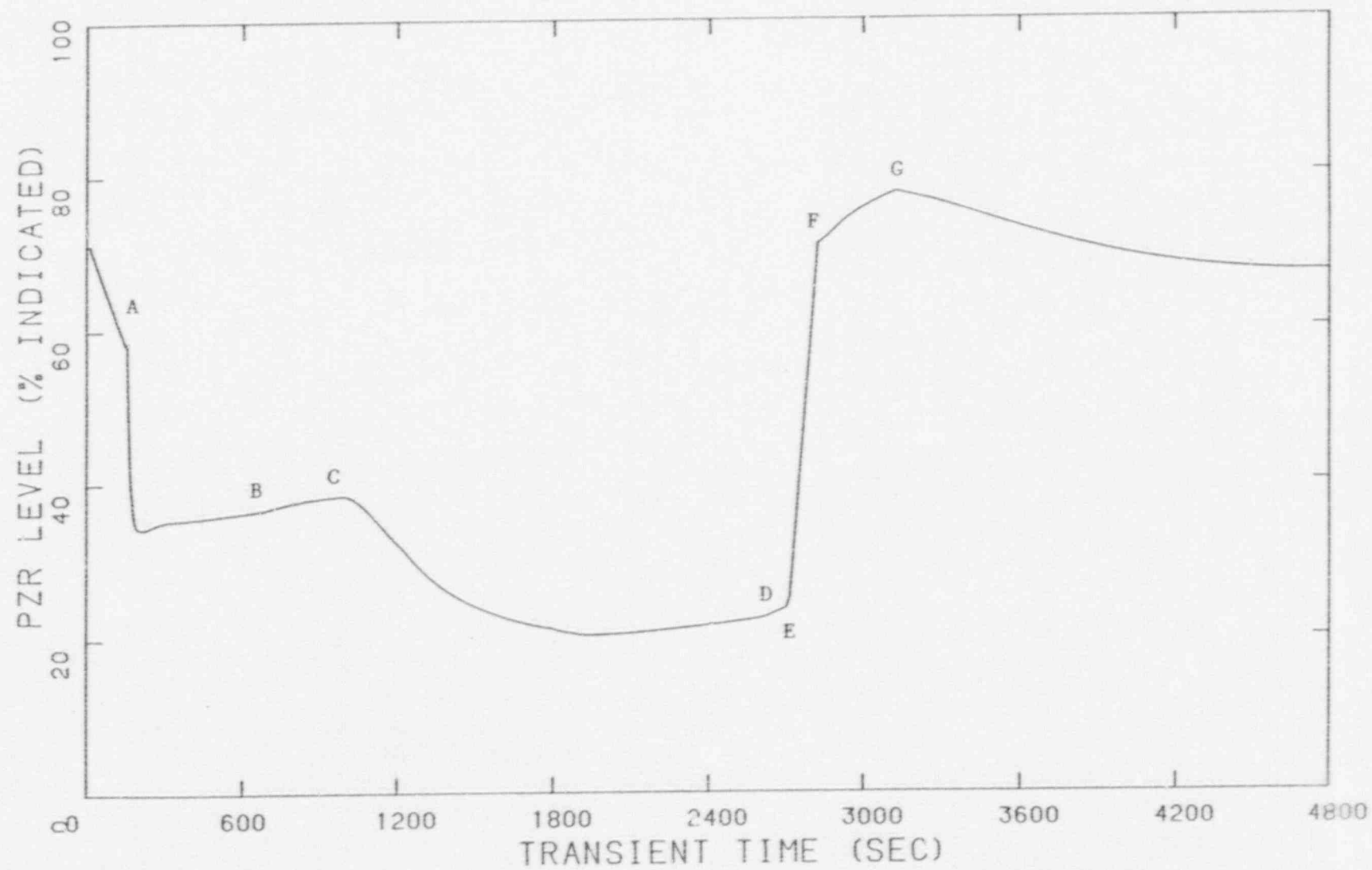


Figure 7.4

Figure 7.5

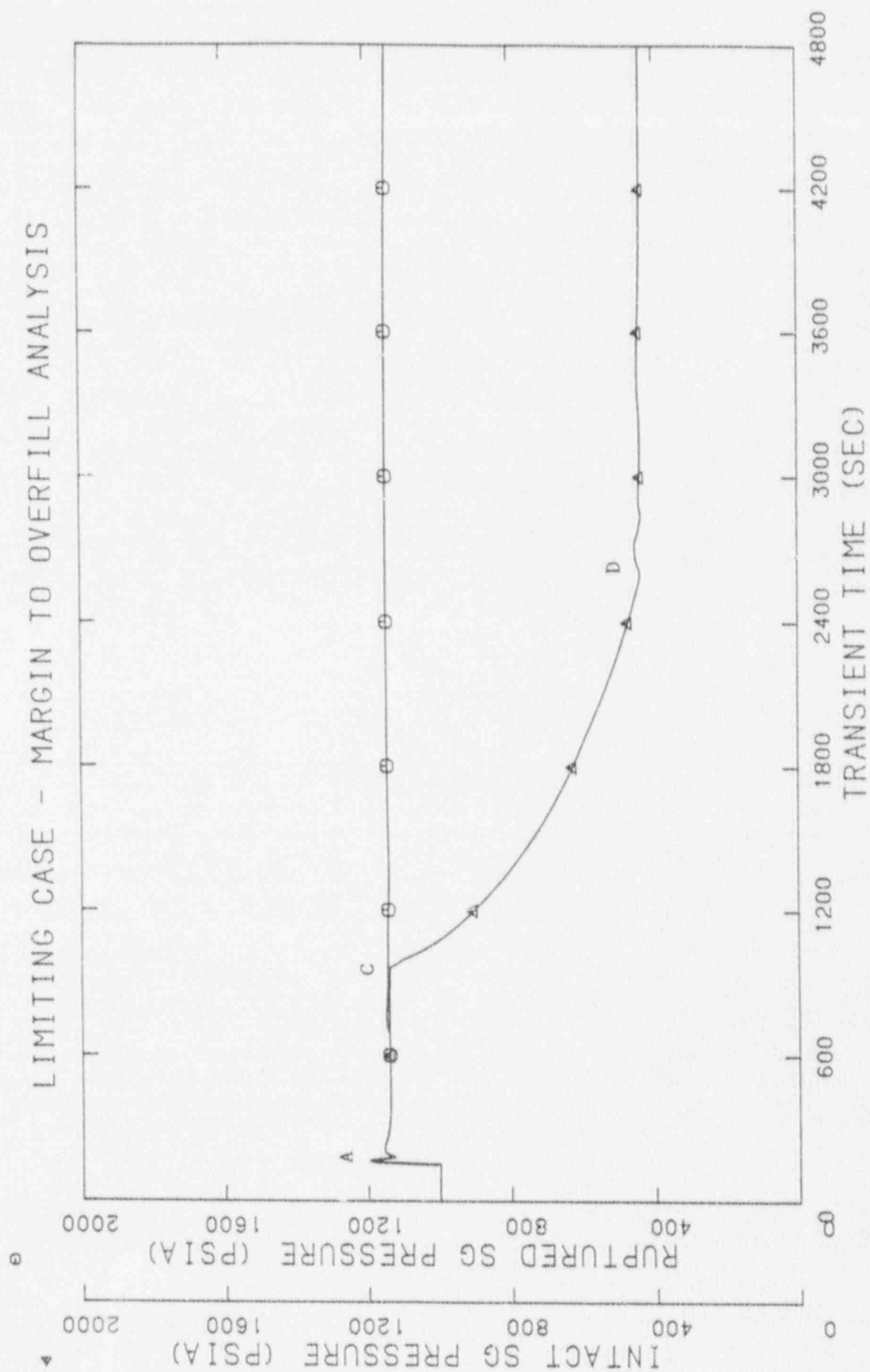
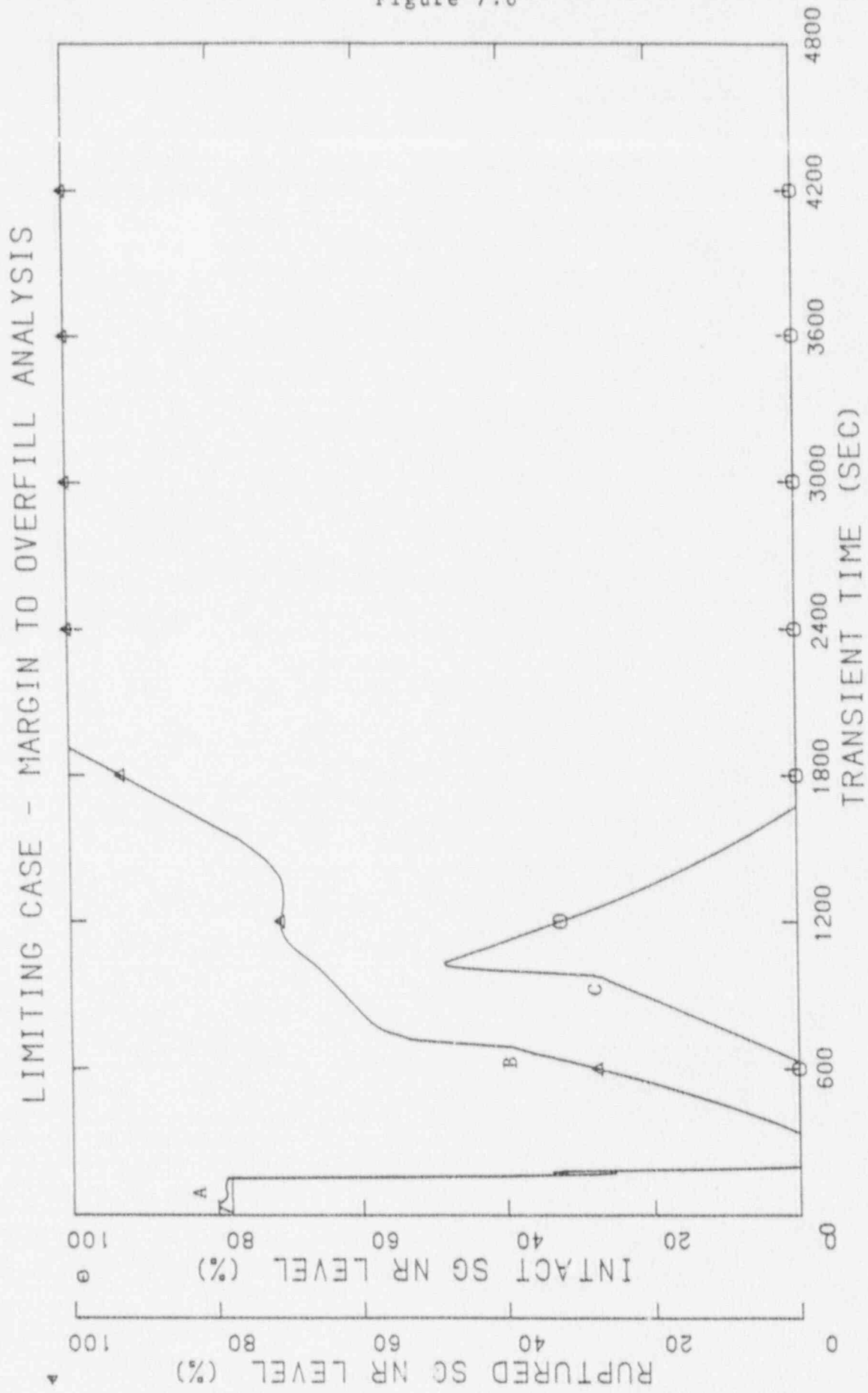


Figure 7.6



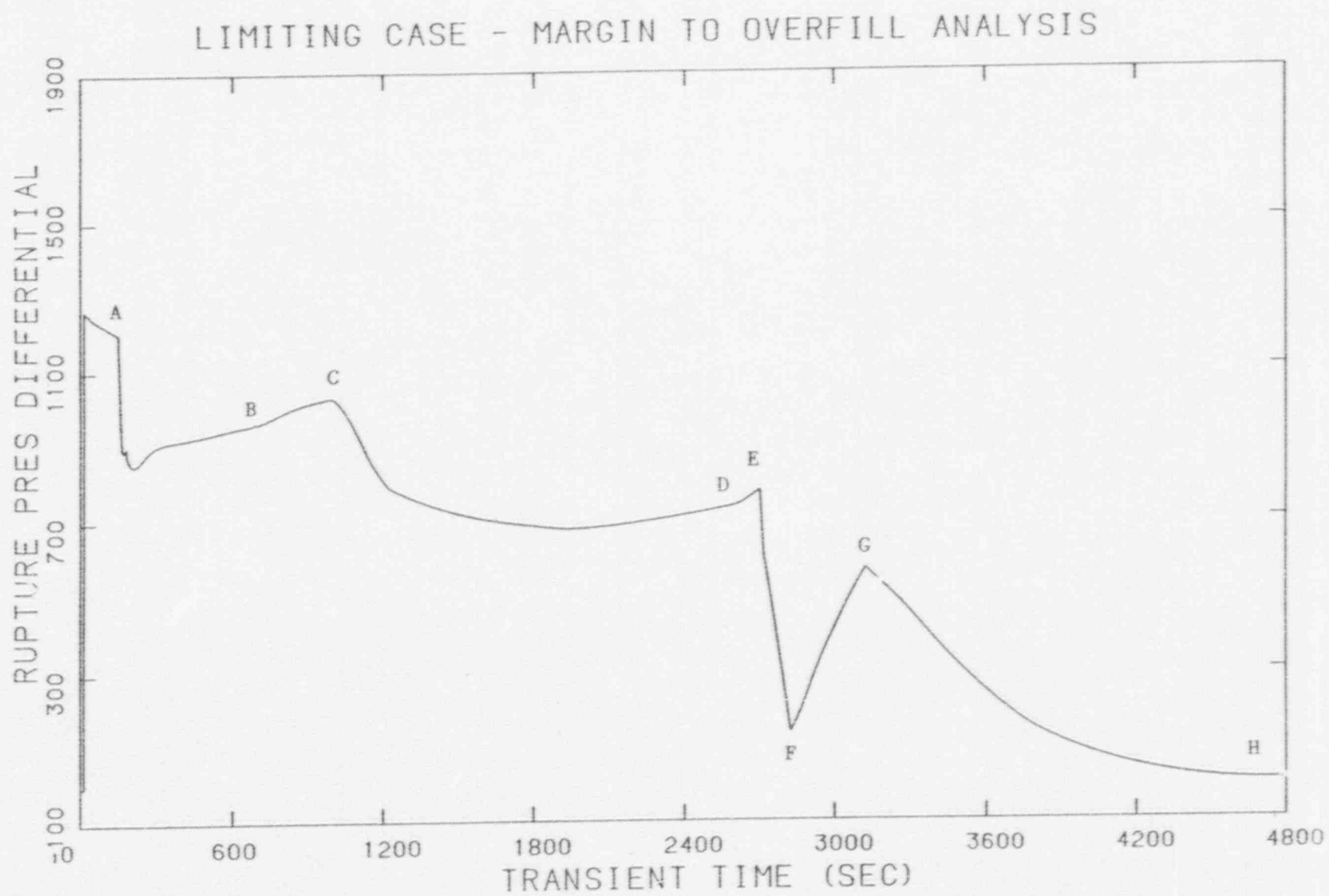


Figure 7.7

LIMITING CASE - MARGIN TO OVERFILL ANALYSIS

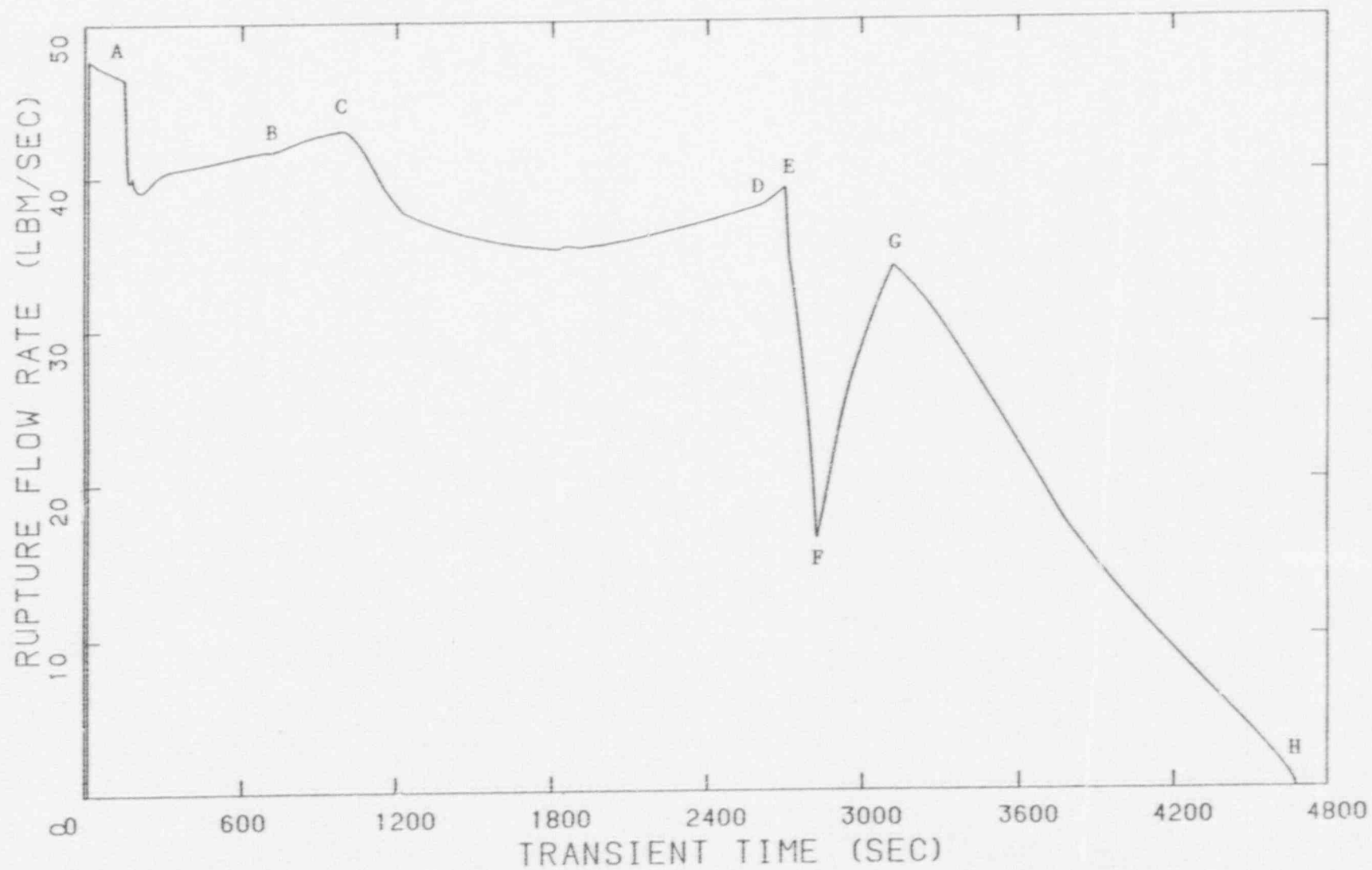
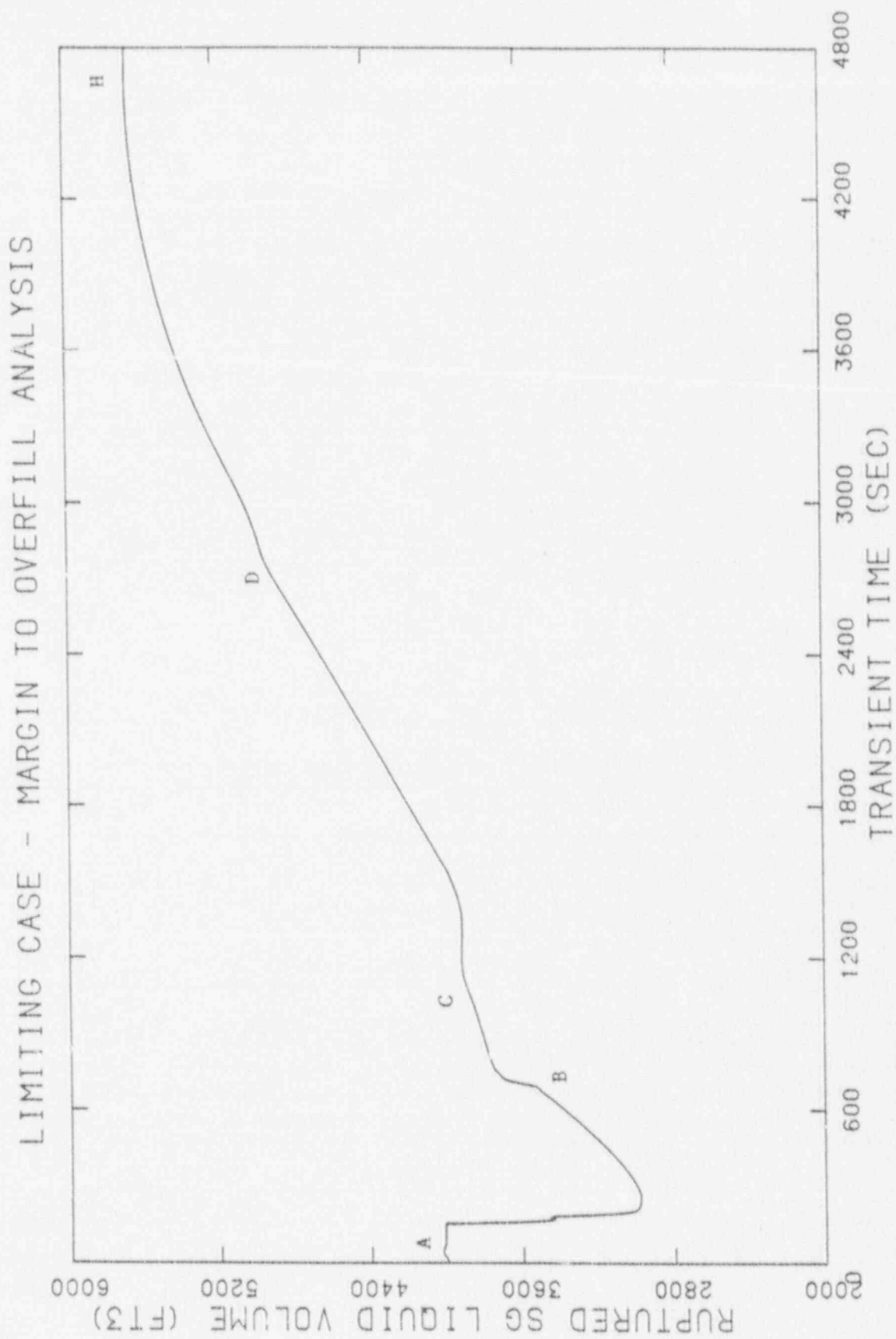


Figure 7.8

Figure 7.9



8.0 STATIC LOAD ANALYSIS OF FLOODED MAIN STEAM LINES

The analysis shown in Section 7.0 concludes that steam generator overfill will not occur due to a design basis SGTR accident. However, a static analysis of the main steam lines in the flooded condition was performed in order to satisfy the requirements of Reference (6). The results of the analysis showed that the main steam lines are structurally adequate for this condition.

The analysis assumed flooding of the main steam lines from the steam generators to the main steam isolation valves. The emergency feedwater turbine supply lines were flooded up to the first normally closed safety class power-operated valves. The atmospheric relief and safety valve discharge lines were assumed completely flooded.

The static analysis considered deadweight, pressure, and thermal expansion loads. Pipe stresses, nozzle loads, containment penetration loads, and support stresses were all shown to be less than emergency condition allowables. Therefore, the main steam lines are structurally adequate for the flooded condition.

9.0 TRANSIENT ANALYSIS OF THE CASE WITH THE MOST SEVERE RADIOLOGICAL CONSEQUENCES

The plant equipment relied upon to mitigate the effects of a design basis tube rupture was identified in Section 6.0 of this report.

The design of these systems were reviewed to identify potential single active failures of equipment required to mitigate the SGTR. The consequential effects of these single failures were then considered in determining the limiting single failure with respect to maximizing off-site doses for a SGTR at Seabrook.

The limiting single failure with respect to off-site doses for Seabrook was determined to be the failure of the ASDV on the ruptured SG to close after isolation of the ruptured SG. The analysis of this event is presented in the remainder of this section, and the evaluation of the resultant off-site doses is discussed in Section 10.0

9.1 Description of Case Scenario

The initiating event for this scenario is a design basis tube rupture; e.g., the instantaneous guillotine severance of a single SG U-tube in a single SG. The operators are conservatively assumed to take no manual actions prior to the occurrence of an automatic reactor trip on low RCS pressure. Initiation of SI after the reactor trip is also assumed to occur as a result of automatic actuation on low RCS pressure. A loss of off-site power is assumed to occur coincident with the reactor trip.

As part of the initial response to the reactor trip, it is conservatively assumed that the operator throttles EFW flow to a total of 500 gpm. This is the minimum flow value specified in Procedure E-0 until level in two steam generators recovers to greater than 65% Wide Range (WR) or one steam generator recovers to greater than 5% Narrow Range (NR). This flow is assumed to be uniformly distributed to the four steam generators until level in the ruptured steam generator returns to 25% NR. This is the lower end of the post-trip steam generator level restoration interval (25% NR - 50% NR) specified in Procedure E-0. At this time, EFW flow to the ruptured

steam generator is conservatively assumed to be stopped. These assumptions prolong the time to achieve diagnosis and isolation of the ruptured steam generator.

No further operator actions are modeled until the level in the ruptured SG returns to an indicated level of 33% NR, by which time it is assumed that the operators will have identified the ruptured SG (based on observation of an uncontrolled increase in ruptured SG level), and closed its associated MSIV. This action is simulated in the RETRAN run by closing the MSIV when the predicted ruptured SG level returns to 33% NR after the trip.

At this point in the transient evolution, it is assumed that the ASDV associated with the ruptured SG fails to a fully open position. This is the worst single failure for off-site dose consequences.

During the next thirty minutes, it is assumed that the operators realize that the ASDV has failed to close when ruptured SG pressure falls below 1125 psig, and that after failing to obtain valve closure using the controller or control switch on the control board, proceed to close the upstream ASDV block valve. The ASDV block valve is assumed to be fully closed at the end of this thirty-minute interval (References (20) and (21)).

Once the ASDV block valve has been closed, operator actions to implement the remaining steps of E-3 are assumed to resume. These actions are assumed to occur in accordance with the time intervals determined to be bounding from the Seabrook SGTR simulations (see Table 5.1). Hence, five minutes after isolating the failed ASDV, the operators are assumed to open the intact SG ASDVs to cool the plant down to the target average core exit temperature specified in the E-3 procedure.

Once the target average core exit temperature has been reached, the operators are assumed to set the intact SG ASDVs to maintain current intact SG pressures and after two minutes, open a single pressurizer PORV to reduce RCS pressure. The PORV is closed when one of the three conditions specified in Step 18.b for closing the valve is met.

The operators were conservatively assumed not to terminate SI flow until five minutes later. This assumption bounds the longest time interval observed during the SGTR simulations performed to quantify operator action time intervals, discussed earlier in Section 5.0 of this report.

Ten minutes later (fifteen minutes after closing the PORV), the operators are assumed to reopen the PORV (if the pressurizer and RCS conditions are not such that they meet one of the three PORV closure criteria) and depressurize the RCS below the ruptured SG pressure if the RCS pressure is still higher than the ruptured SG pressure. This action or alternatively, if the PORV is not re-opened, the decay of RCS pressure from continued break flow after SI termination, terminates the break flow and the portion of the transient analyzed using RETRAN.

The assumed transient evolution for the remainder of the eight-hour period, over which radiological consequences were determined, is discussed in Section 9.4.

9.2 Initial Conditions and Conservative Assumptions

The initial conditions assumed for the case with the most severe radiological consequences are given in Table 9.1.

Differences in assumptions from those used in the analysis of the limiting overfill case are noted in the discussions which follow.

9.2.1 Initial RCS Pressure and Time of Reactor Trip

A maximum time interval between break initiation and time of reactor trip was used. This assumption maximizes the level of radiological contamination in the ruptured SG at the time of the trip and loss of off-site power, when direct steam release from the ruptured SG through the ASDV and MSSV begins. This also maximizes the integrated break flow into the ruptured SG during the event and thus, maximizes the radioactivity transferred through the rupture. Early manual reactor trip or SI actuation would yield lower total mass releases (and lower radiological consequences) due to the shorter time interval from rupture to termination of break flow. Therefore, no manual

reactor trip or SI actuation was assumed, instead, these functions are assumed to occur automatically on low RCS pressure.

The conservative initial condition for RCS pressure for this case is a high RCS pressure in order to increase the initial pressure differential to the trip setpoint, thereby delaying the time of trip.

Thus, as in the limiting overfill case, the Seabrook limiting radiological consequence case assumes a conservatively high initial RCS pressure, but unlike the overfill case, reactor trip and SI actuation are delayed until automatically initiated.

9.2.2 Initial Steam Generator Secondary Mass

In order to reduce the amount of radiologically "clean" water in the ruptured SG, and delay the time of ruptured SG isolation (return of ruptured SG level to 33% NR), the additional mass, which was added to the initial SG inventory to bound low power operation and the potential effects of turbine runback in the limiting overfill analysis, was not added for this case. Reducing the amount of initially clean water in the SG reduces the dilution of radioactive contaminants in the ruptured SG, which is conservative with respect to radiological consequences. It also extends the time interval between reactor trip and SG isolation by reducing the ruptured SG liquid inventory at the time of the trip, thereby requiring more mass (contaminated RCS water and EFW flow) to be added before level returns to 33% NR, the criterion assumed for determining the time of SG isolation.

The initial SG inventory was further reduced to the low SG level alarm setpoint minus uncertainty.

9.2.3 Reactor Trip Delay

Since it was determined in Section 9.2.1 that delaying the reactor trip as long as possible is conservative for this case, the maximum trip delay time was assumed.

9.2.4 Steam Generator Relief Valve Pressure Setpoint

As noted in Section 7.2.9, operation of the ASDVs maximizes the integrated break flow during the event. Therefore, the ASDVs are assumed to function in this analysis. The ASDV on the faulted SG is assumed to operate normally until it fails open as the single active failure.

9.2.5 Low Pressurizer Pressure Reactor Trip Setpoint

In order to delay the time of the reactor trip as long as possible, the minimum low pressure trip setpoint was used.

9.2.6 Pressurizer Pressure for SI Initiation

In order to delay the addition of radiologically clean water to the RCS, the minimum low pressure SI actuation setpoint was used. Since RCS pressure drops rapidly after reactor trip, the time of SI actuation is not sensitive to the SI actuation setpoint.

9.2.7 Emergency Feedwater System Delay and Flow Rate

Minimum EFW flow is conservative for this analysis since it results in larger integrated steam releases from the ruptured and intact SGs to the atmosphere. Minimum EFW flow is also conservative in that it delays the point in time at which ruptured SG isolation is assumed to occur by delaying the return of ruptured SG level to 33% NR. This delay results in more leakage of radiologically contaminated RCS fluid into the ruptured SG and, thus, higher levels of contamination in the ruptured SG fluid inventory.

Therefore, in order to minimize EFW flow to the ruptured SG, the maximum delay (60 seconds from loss of off-site power) in EFW delivery to the SGs was assumed.

EFW flow is minimized further by assuming the operator throttles total EFW flow to 500 gpm immediately upon initiation as explained earlier in Section 9.1.

9.2.8 Turbine Runback

In order to minimize the accumulation of clean water in the ruptured SG, no automatic turbine runback or additional initial SG inventory intended to bound the effects of a runback or operation at lower power levels was assumed.

9.3 Transient Description

Table 9.2 provides the sequence of events for this case. Figures 9.1 through 9.18 provide plots of the parameters of major interest. In the discussion which follows, letters enclosed within square brackets, [A], indicate labeled points of interest on the plots.

9.3.1 RCS Response

9.3.1.1 Break Initiation to Ruptured SG Identification

The rupture is assumed to occur at ten seconds into the RETRAN run. After the tube ruptures, the flow of coolant out of the primary system results in a gradual reduction in primary system pressure and pressurizer level, causing the charging system to increase charging flow as the deviation between the actual and the programmed pressurizer level slowly grows. Since the leakage flow rate exceeds the maximum make-up flow capacity of the CVCS, pressurizer level and pressure continue to decrease.

An automatic reactor trip on low RCS pressure occurs at 904 seconds [A]. Coincident loss of off-site power is assumed to occur at the reactor trip.

Core power level and primary system temperatures are nearly constant until the reactor is tripped. After the trip the rapid decrease in core heat production causes the core average and exit temperatures to drop rapidly. This drop in coolant temperature results in a shrinkage of the coolant volume, emptying of the pressurizer, and an accompanying drop in RCS pressure from 1940 psia at the time of the trip to 1750 psia shortly thereafter. The drop

in pressure results in flashing of the hot fluid in the RV upper head and a small steam bubble is formed.

The reactor trip results in a turbine trip and the closing of the turbine stop valves, bottling up the SG secondary side. As a result, primary to secondary heat transfer is reduced causing the cold leg temperatures to begin to rise. Pressure in the main steam lines rises to the ASDV and MSSV setpoints, causing these valves to open. Primary to secondary heat transfer then increases, which results in a reduction of core inlet (and cold leg) temperatures to approximately the saturation temperature corresponding to the SG secondary side pressure shortly after the trip.

The loss of off-site power results in the loss of motive power for the RCPs which coast down, resulting in decreasing RCS flow rates until natural circulation is established. The effects of the loss of forced reactor coolant flow on RCS temperatures can be observed in the intact loop temperature plot, which shows the resulting increase in hot leg temperature between 900 and 1000 seconds, during the development of sufficient driving head for natural circulation to occur.

The loss of power also causes the EFW pumps to be actuated. Steam release to the atmosphere via the steam supply to the EFW pump turbine also begins. EFW delivery to the SGs was assumed to begin after a 60-second delay for this scenario. Total EFW flow is assumed to be throttled to 500 gpm by immediate operator action.

SI is automatically actuated on low RCS pressure during the drop in pressure following the reactor trip. This action causes the SI pumps to start and the Centrifugal Charging Pump (CCP) discharge to be redirected to the cold legs. It also results in the tripping of the steam-driven MFWPs and closure of the main feed line isolation valves. The increase in primary system mass inventory caused by the SI flow results in the RCS pressure increasing to around 1900 psia by 1300 seconds and restoration of level in the pressurizer. RCS temperatures drop slightly over this time period due to the combined effect of the drop-off of decay heat and the relatively cold temperature of the injected SI water.

EFW flow to the ruptured SG is assumed to be terminated when its level returns to 25% NR at 2320 seconds.

9.3.1.2 Ruptured SG Isolation and RCS Cooldown

No credit is taken for the expected high radiation indications on the main steam line or condenser air evacuation system detectors. The operator is assumed to identify the ruptured SG by observation of the uncontrolled increase in narrow-range level indication. The ruptured SG is assumed to be isolated by closure of the associated MSIV and EFW turbine steam line valve when level reaches an indicated 33% NR at 2378 seconds in this case [B].

The ASDV associated with the ruptured SG is assumed to fail to a fully open position at this time. Further operator actions are assumed to be delayed for thirty minutes until the associated ASDV block valve is manually closed.

Five minutes after closure of the ASDV block valve, the operators are assumed to open the intact SG ASDVs [C], per procedure, to begin cooling the RCS to the target core exit temperature specified in the E-3 procedure. The operators are assumed to track the changes in this target temperature as ruptured SG level changes during the cooldown. The cooldown proceeds until the average core exit temperature specified for the current ruptured SG pressure in the E-3 procedure is reached at 5490 seconds, at which time the operator is assumed to set the intact SG ASDVs [D] to maintain the existing intact SG pressures to maintain the target RCS temperature.

The cooldown results in significant shrinkage of the volume of the RCS coolant inventory causing a reduction in RCS pressure to about 1690 psia and the pressurizer to once again empty.

9.3.1.3 RCS Depressurization and Termination of Break Flow

Two minutes after terminating the cooldown, the operator is assumed to open one pressurizer PORV, per procedure, in an attempt to further reduce RCS pressure below the ruptured SG pressure [E]. As a result, the steam bubble in the RV upper head volume increases in size as pressure is decreased and more

of the liquid in that region flashes to steam. The increase in the size of this steam bubble causes a rapid displacement of coolant from the upper head into the pressurizer during the depressurization. This phenomenon is the cause of the "bump" in the plot of intact loop hot leg temperature during this time interval, which shows the effect of the outsurge of hot fluid from the upper head into the RV outlet plenum.

The venting of the pressurizer steam space results in RCS pressure decreasing rapidly to 635 psia at 5831 seconds, whereupon the operator is assumed to close the PORV [F] since RCS pressure is less than the ruptured SG pressure and the pressurizer level has reached 75%, two of the PORV closure criteria in the E-3 procedure. The ruptured SG pressure is 750 psia at this time.

Five minutes are then assumed to elapse before the operator completes the termination of SI flow at 6131 seconds [G]. During this time interval, the continuing safety injection flow repressurizes the RCS to around 1300 psia, causing renewed leakage from the RCS into the ruptured SG.

The operators are assumed to not re-open the PORV since pressurizer level remains above the 75% closure criteria. RCS pressure subsequently decreases gradually due to the continuing break flow. At 7930 seconds, RCS pressure equilibrizes with the ruptured SG pressure, terminating the leak.

9.3.2 SG and Secondary System Response

9.3.2.1 Break Initiation to Ruptured SG Identification

SG pressures remain steady until the reactor trip [A]. The reactor trip results in a turbine trip causing the pressure in the main steam lines to rise to the ASDV and MSSV setpoints, causing these valves to open. Shortly after the trip, the reduced heat input from the primary (due to the drop in core power level and RCS temperatures) results in SG pressures falling to just below the ASDV set pressure.

The reactor trip and SI actuation result in termination of MFW flow to all SGs, and actuation of EFW. Steam flow to the atmosphere via the supply to the EFW pump turbine begins immediately upon EFW actuation. Delivery of EFW to the SGs is assumed to start 60 seconds after the loss of off-site power.

The main steam flow is nearly constant until the reactor/turbine trip [A], whereupon flow stops as the turbine stop valves close.

The ruptured SG NR level shows a small increase shortly after the break occurs, until the action of the level controller can reduce the MFW flow rate to compensate for the leak. The level drops out of the NR span in all SGs immediately after the reactor trip [A].

Level returns onto the narrow-range scale in the ruptured SG first, reaching 25% NR at 2320 seconds. At this time, EFW flow to the ruptured SG is assumed to be terminated.

9.3.2.2 Ruptured SG Isolation and RCS Cooldown

Once the operators identify the ruptured SG, the MSIV and EFW turbine supply valve in the steam line from the ruptured SG are closed. This action is assumed to occur at 2378 seconds [B] when the NR level reaches 33%. EFW flow to the ruptured SG was previously terminated when NR level reached 25%.

At this point in time, the ASDV associated with the ruptured SG is assumed to fail to a fully open position. The ruptured SG narrow-range level swells in response to liquid being displaced from the tube bundle region by the formation of steam bubbles, then drops gradually over the next thirty minutes until the ASDV block valve is closed at 4178 seconds [K]. The level in the ruptured SG then drops as the bubbles in the tube bundle region collapse.

The operators are assumed to open the intact SG ASDVs to cooldown the RCS five minutes after the ruptured SG ASDV is isolated [C]. This results in a gradual drop in intact SG pressures until the target core exit temperature is reached at 5490 seconds and the valves are set to maintain SG

pressure [D]. Pressure in the intact SGs then remains relatively stable through the end of the transient.

9.3.2.3 RCS Depressurization and Break Flow Termination

The intact SG ASDVs are automatically modulated during this time interval to remove decay heat and maintain the target average core exit temperature by maintaining intact SG pressure.

The liquid volume in the ruptured SG continues to increase until the RCS pressure is finally reduced below the ruptured SG pressure at 7930 seconds. The liquid volume in the ruptured SG at this time is approximately 4330 cubic feet, which corresponds to a margin to overfill of 1572 cubic feet. This corresponds to a temporal margin to overfill of 25 minutes at the equilibrium break flow rate for Seabrook (where equilibrium flow rate was determined as in Reference (2)).

9.4 Determination of Mass Releases

Mass releases were determined for an eight-hour period, beginning at the time of the rupture, for use in evaluating the radiological consequences of the event. The mass releases from the intact and ruptured SGs were taken from the RETRAN results for the period of time from break initiation through termination of break flow at 7930 seconds. For the remainder of the 0- to 2-hour interval, and the subsequent 2- to 8-hour period after the rupture, mass releases required to remove core decay heat, LCP heat, and to cool down the RCS and SG fluid and metal to RHR operating conditions were determined via conservative engineering calculations.

As in the analysis of the limiting overfill case, a 20% multiplier was applied to the decay heat computed by RETRAN using the built-in ANS 1971 standard decay heat calculation for the period from rupture through break flow termination. The decay heat during the period from break flow termination through eight hours was determined using the formulation prescribed in Branch Technical Position (BTP) ASB 9-2 Reference (12), except that a 20% multiplier was used for all time periods. This is more conservative than the BTP which

allows a reduction of the uncertainty to 10% for decay times longer than 1000 seconds.

The assumed sequence of events following termination of the break flow is as follows.

The operators are assumed to maintain the RCS and SGs in relatively stable states for the first 20 minutes following break flow termination while preparing to cooldown to conditions suitable for implementing cooling using the RHR System. A single RCP is assumed to be started during this time interval, which homogenizes the RCS temperature and adds additional heat which must be removed during the cooldown, conservatively increasing the estimated steam releases. At the end of this 20-minute interval a maximum rate cooldown (100°F/hr) to the RHR System operating temperature of 350°F is assumed to begin.

In order to maximize off-site releases during the cooldown to RHR entry conditions, it is assumed that the operators are unable to restore steam dump to the condenser and, therefore, perform the cooldown by dumping steam from the intact SGs to atmosphere using the associated ASDVs. Steam is also released via the supply line to the EFW pump turbine.

Once the RCS temperature has been reduced to the RHR operating temperature, the ruptured SG pressure is assumed to be reduced to that required for RHR System operation by reopening the ruptured SG ASDV block valve and dumping steam from the ruptured SG to the atmosphere. The RCS pressure is assumed to be reduced concurrently to the RHR operating pressure.

This entire evolution is assumed to be completed within the eight-hour time period. At the end of the eight-hour period, the transition to cooling using the RHR System is assumed to occur, and the subsequent plant cooldown to cold shutdown conditions is accomplished using the RHR System, without causing any further mass releases.

Table 9.3 provides a summary of the mass releases during the eight-hour period. Since the condenser is assumed to be in service until the loss of off-site power following the reactor trip, steam releases from the intact and

ruptured SGs prior to the trip are to the condenser. After the trip, all steam releases are assumed to be through either the SG ASDVs or MSSVs directly to the atmosphere, until the transition to RHR cooling occurs at the end of the eight-hour period.

TABLE 9.1

Summary of Conservative Assumptions and Initial Conditions
for Case With Most Severe Radiological Consequences

<u>Parameter</u>	<u>Assumption</u>
Off-Site Power	Lost on Reactor Trip
Initial Power	100% RTP
Initial RCS Pressure	2300 psia
Initial Pressurizer Water Level	71% Span
Initial Core Average Temperature	594.5°F
RCS Flow Rate	Thermal Design
Initial Steam Generator Pressure	1000 psia
Initial Steam Generator Level	42% NR
Reactor Trip Delay	2 Seconds
Turbine Trip Delay	None
Steam Generator Atmospheric Steam Dump Setpoint	1125 psig
Low Pressurizer Pressure for Reactor Trip	1920 psig
Low Pressurizer Pressure for ECCS Actuation	1840 psig
Main Feedwater Isolation	On Reactor Trip/Loss of Off-site Power
Emergency Feedwater Flow Initiation Delay	60 Seconds
Safety Injection Initiation Delay	None
ECCS Flow Rates	Maximum
Emergency Feedwater Flow Rate	Minimum 500 gpm Total
Pressurizer Pressure and Level Control	Prior to Reactor Trip
Steam Generator Water Level Control	Prior to Reactor Trip
Decay Heat	120% ANSI/ANS-5.1-1973
Limiting Single Active Failure	ASDV on Faulted SG Fails Open

TABLE 9.2

Sequence of Events for Case With
Maximum Radiological Consequences

<u>RETRAN Time (Seconds)</u>	<u>Event/Condition</u>	<u>Transient Time (minutes/seconds)</u>
0	Steady-state operation at 100% RTP	-0:10
10	Complete severance of one U-tube	0:00
904	Reactor trip on low pressurizer pressure with coincident loss of off-site power (RCPs trip, EFW flow starts after 60-second delay, turbine trips)	14:54
908	SG ASDVs open	14:58
917	SG MSSVs open	15:07
918	SI actuation on low pressurizer pressure (CVCS and MFW isolated)	15:08
935	SG MSSVs close	15:20
950	Pressurizer empties	15:40
964	EFW flow commences	15:54
2320	Ruptured SG level returns to 25% NR EFW to ruptured SG isolated	38:30
2378	Ruptured SG level returns to 33% NR MSIV in associated steam line closed	39:28
2378	Ruptured SG ASDV fails to full open position	39:28
4178	Operator closes ruptured SG ASDV block valve	69:28
4478	Operator opens intact ASDVs to start RCS cooldown	74:28
5490	E-3 target core exit temperature reached, operator throttles intact SG ASDVs	91:20
5610	Operator opens one pressurizer PORV to depressurize RCS	93:20

TABLE 9.2
(Continued)

Sequence of Events for Case With
Maximum Radiological Consequences

<u>RETRAN Time</u> <u>(Seconds)</u>	<u>Event/Condition</u>	<u>Transient Time</u> <u>(minutes/seconds)</u>
5831	Operator closes pressurizer PORV (indicated pressurizer level = 75%)	97:01
6131	Operator terminates SI flow	102:01
7930	Break flow terminated	132:00

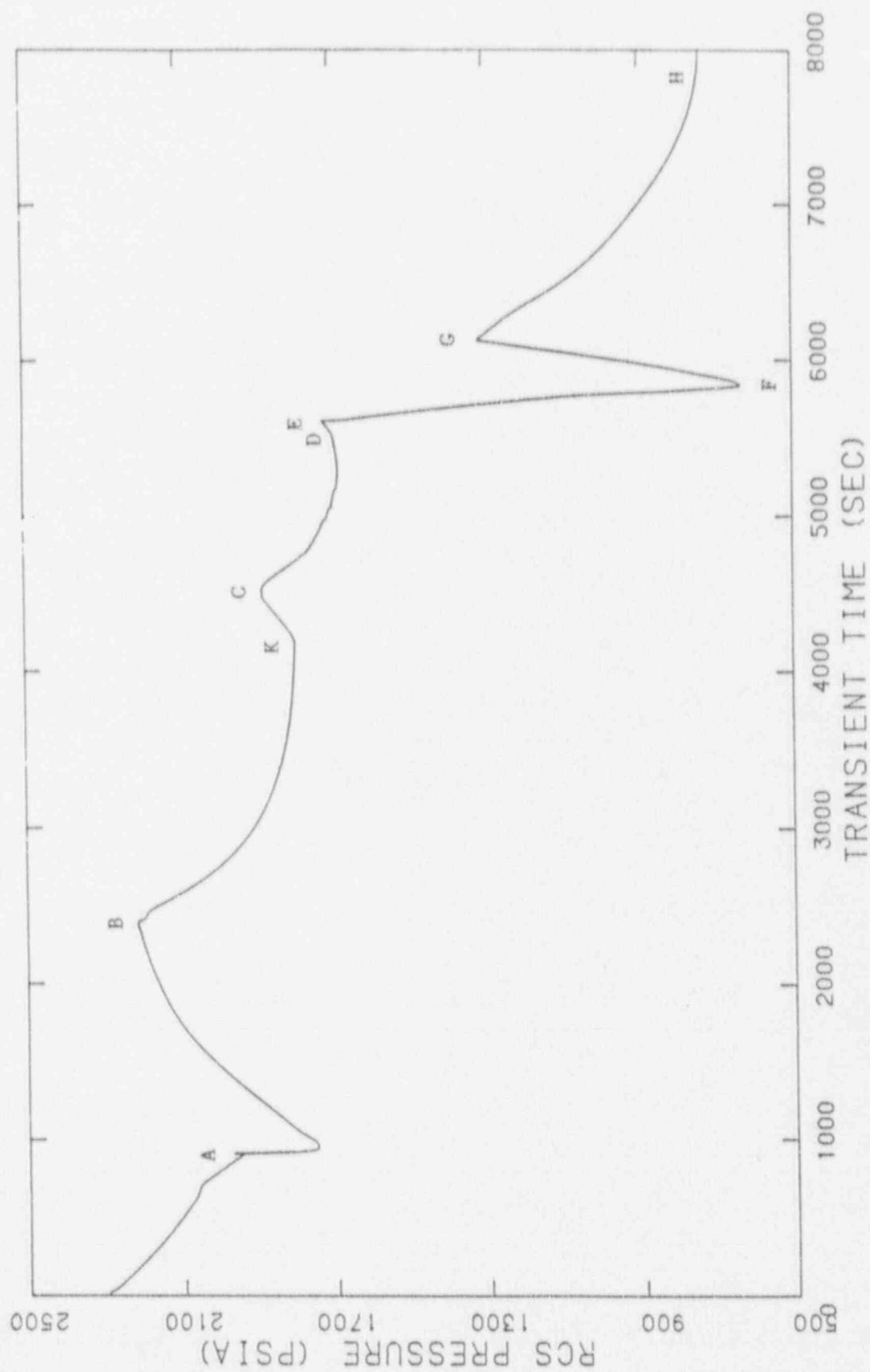
TABLE 9.3

Mass Flows for a Design Basis SCTR Assuming
Failure of the Ruptured SG ASDV Fully Open
at Time of SG Isolation

	<u>Integrated Mass Flows (lbm) for Each Time Period</u>			
	<u>0 - TTRIP</u>	<u>TTRIP-T2HRS</u>	<u>T2HRS-TTBRK</u>	<u>TTBRK-TRHR</u>
Ruptured SG				
To condenser	940,289	0	0	0
To atmosphere	0	166,601	0	24,130
From feedwater	900,233	24,452	0	0
Intact SG				
To condenser	2,790,770	0	0	0
To atmosphere	0	422,730	29,826	904,659
From feedwater	2,791,018	408,208	49,732	897,983
Break Flow	39,405	244,690	5,297	0

TTRIP = Time of reactor trip = 894 seconds
 T2HRS = Two hours after rupture = 7,200 seconds
 TTBRK = Time of break flow termination = 7,920 seconds
 TRHR = Time to reach RHR in-service conditions = 28,800 seconds
 (eight hours after rupture)

Figure 9.1



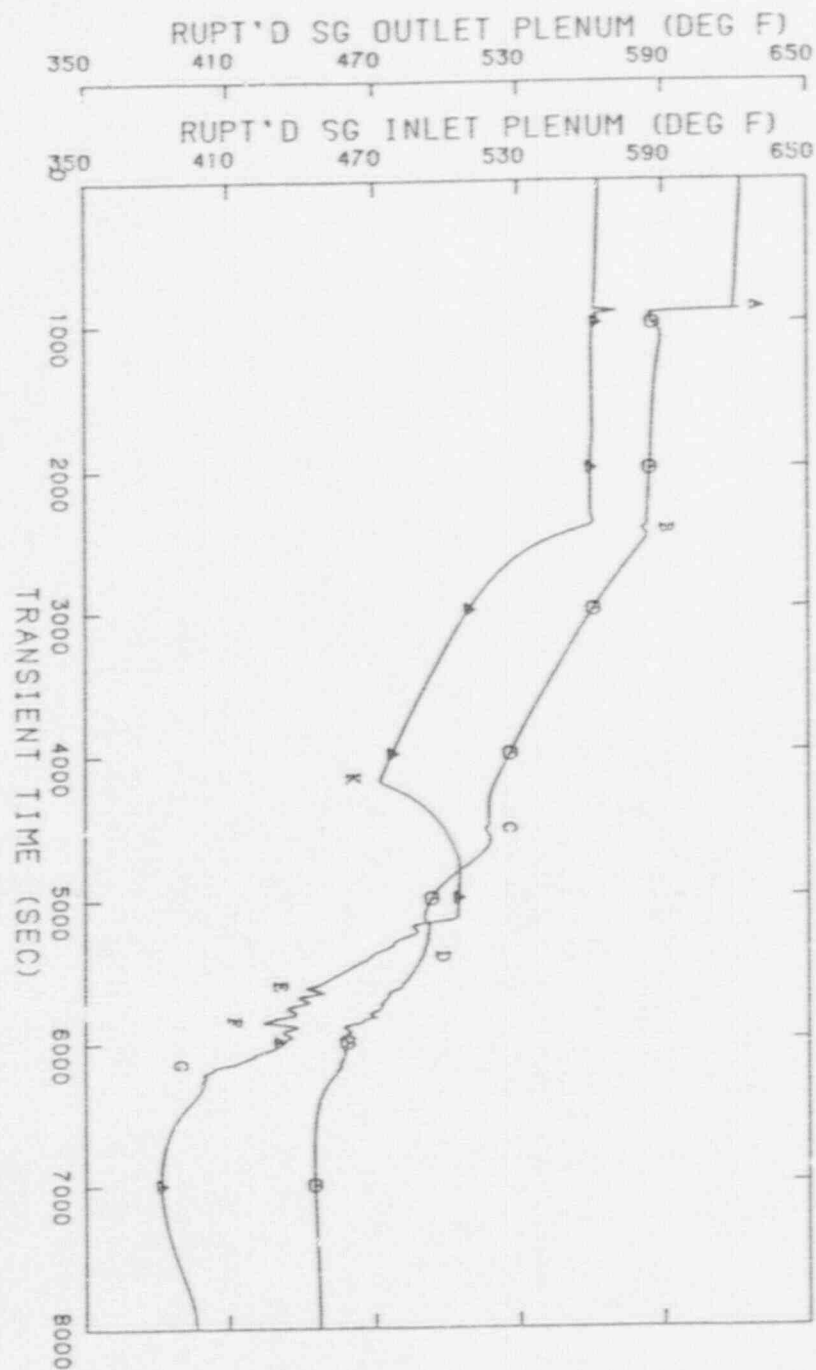


Figure 9.3

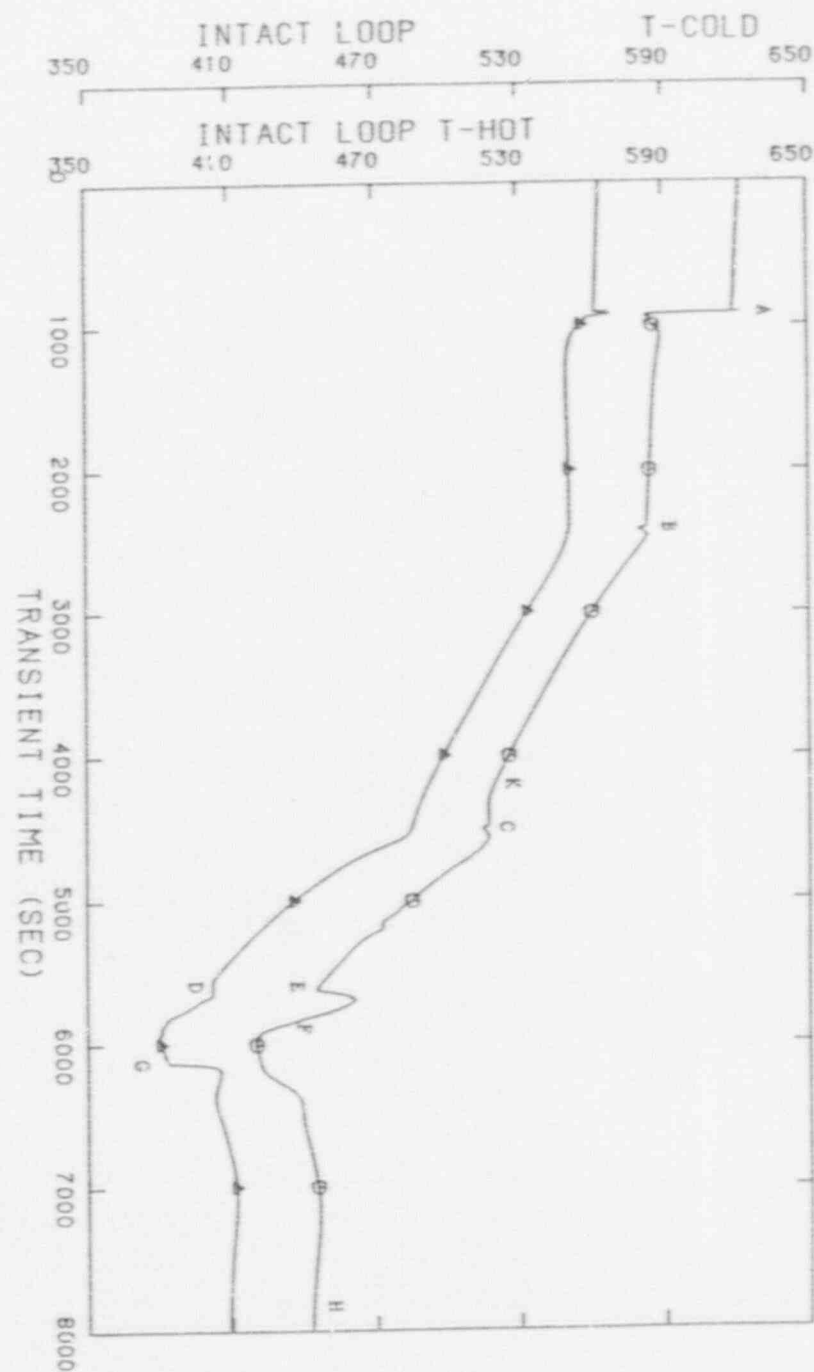
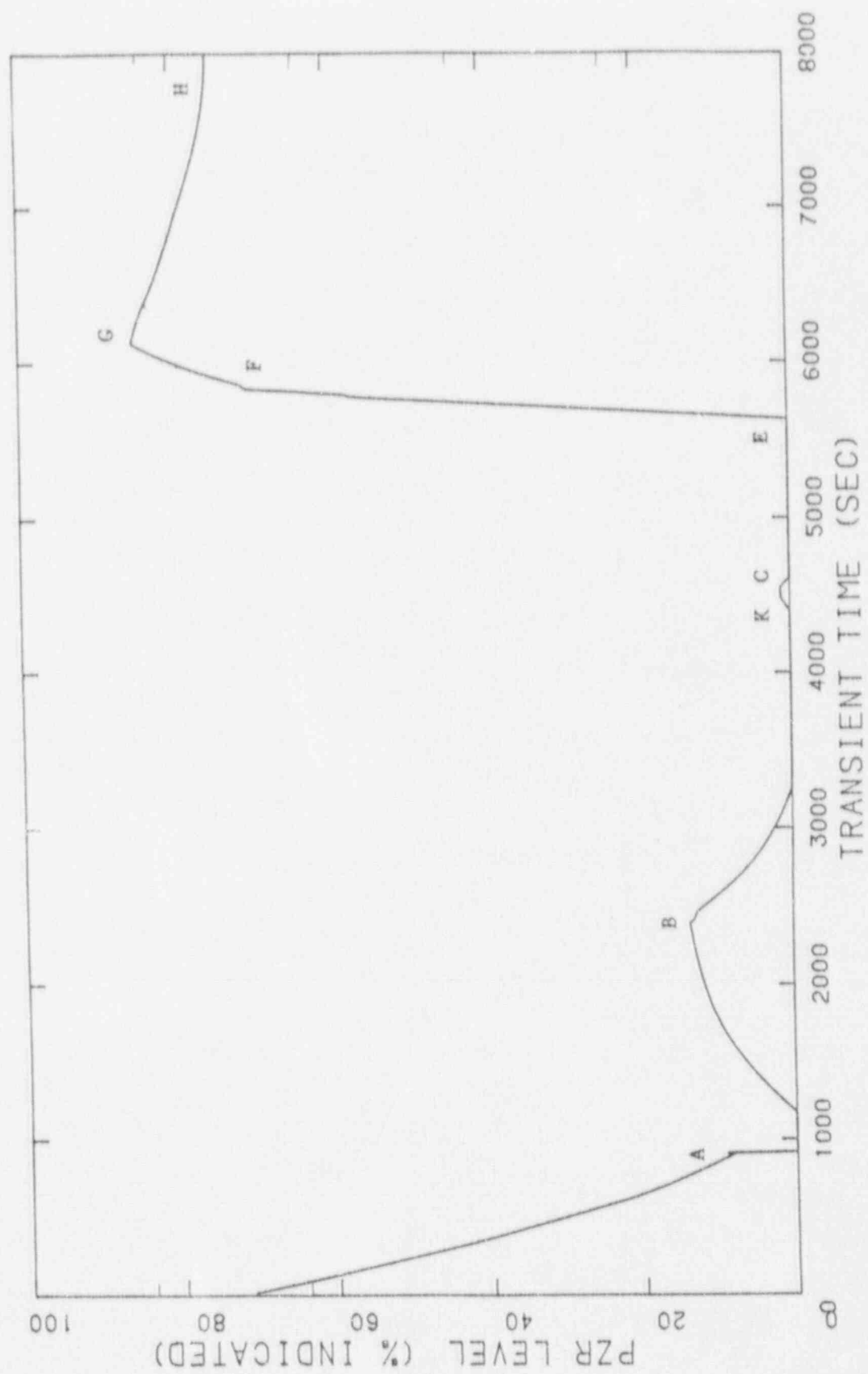


Figure 9.2

Figure 9.4



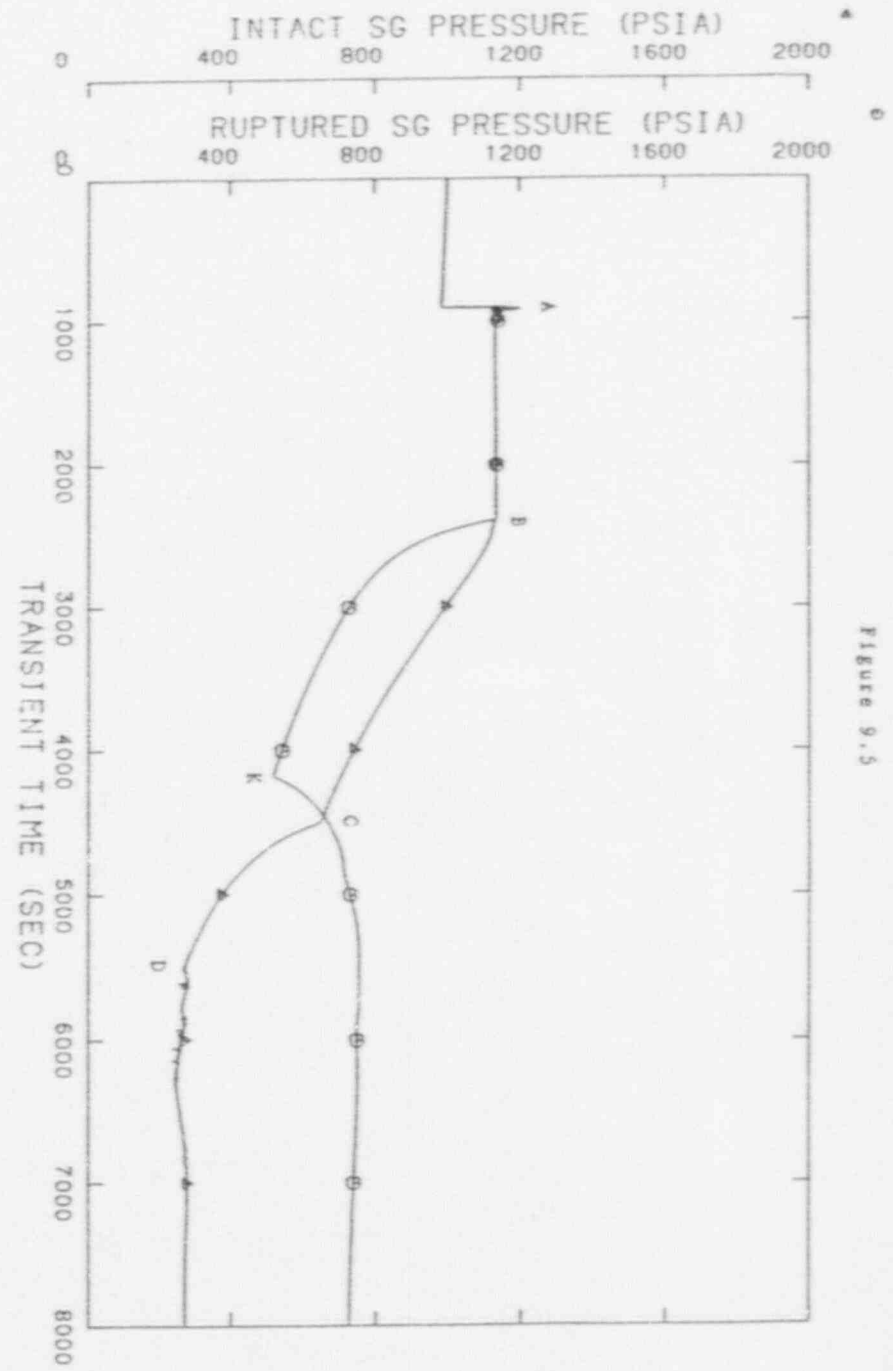


Figure 9.5

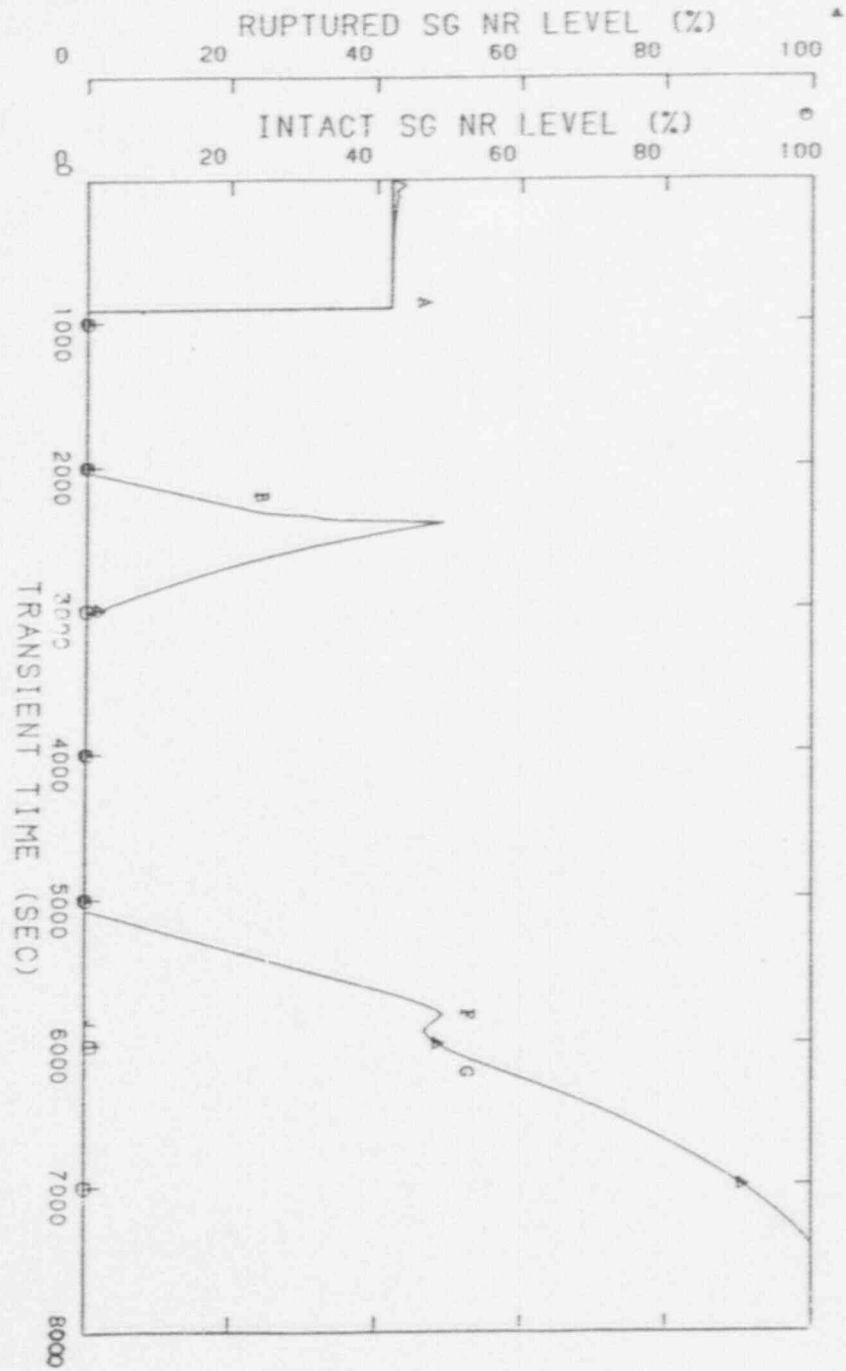


Figure 9.6

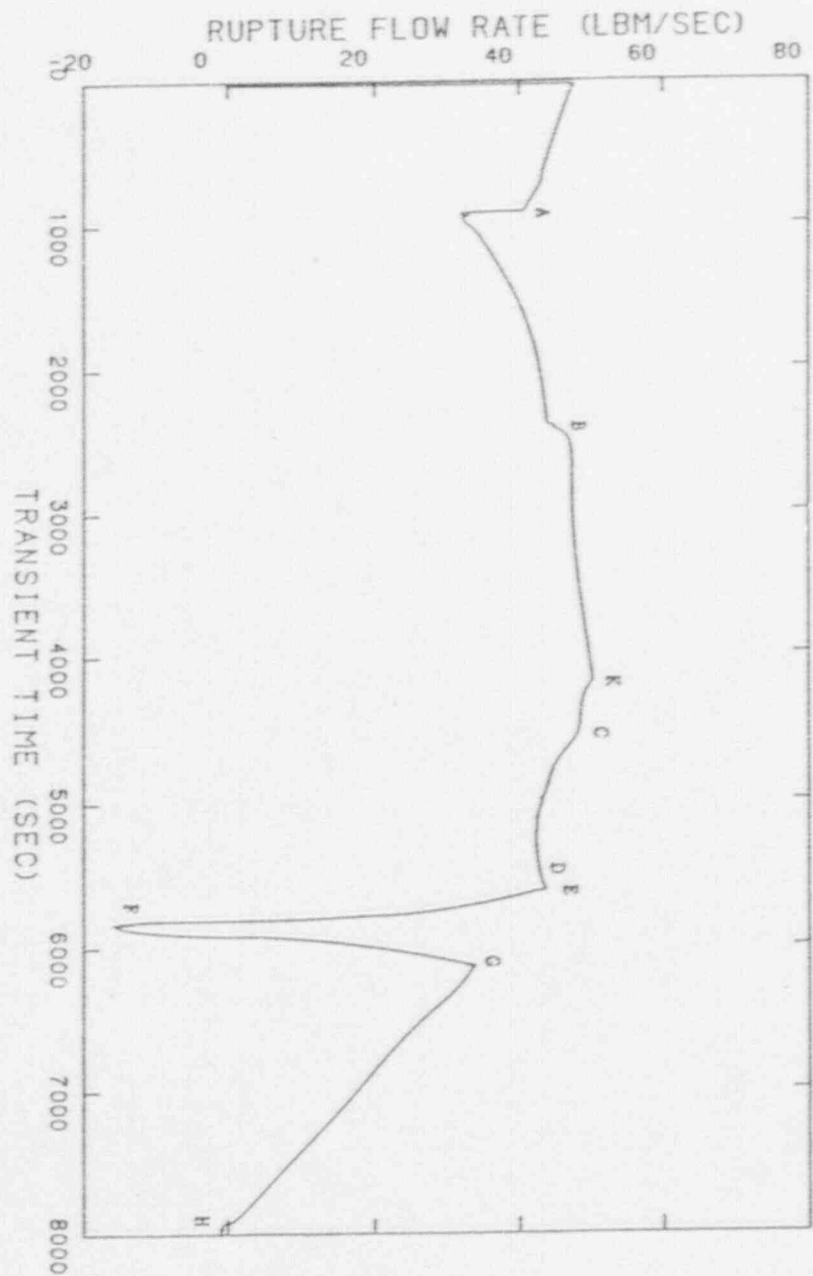


Figure 9.8

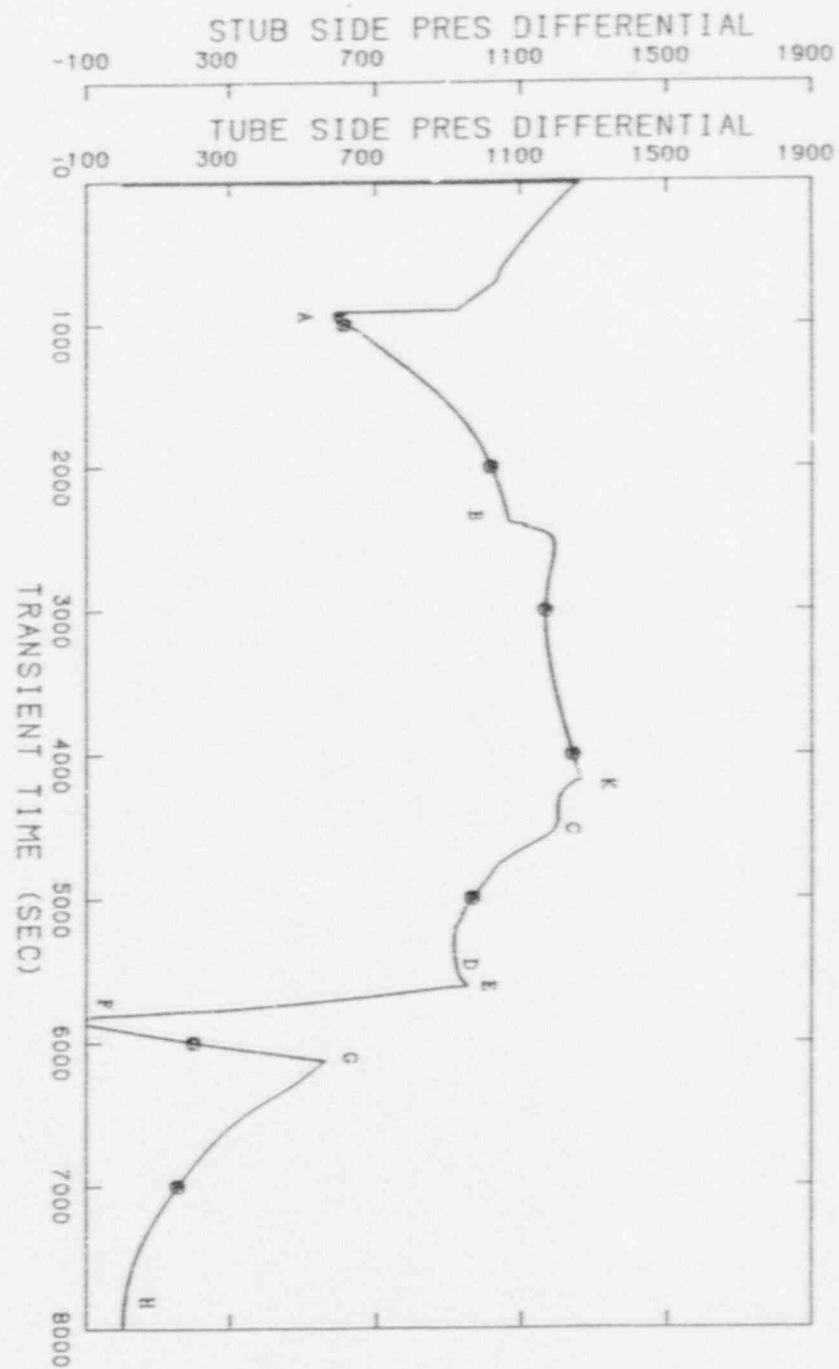


Figure 9.7

Figure 9.9

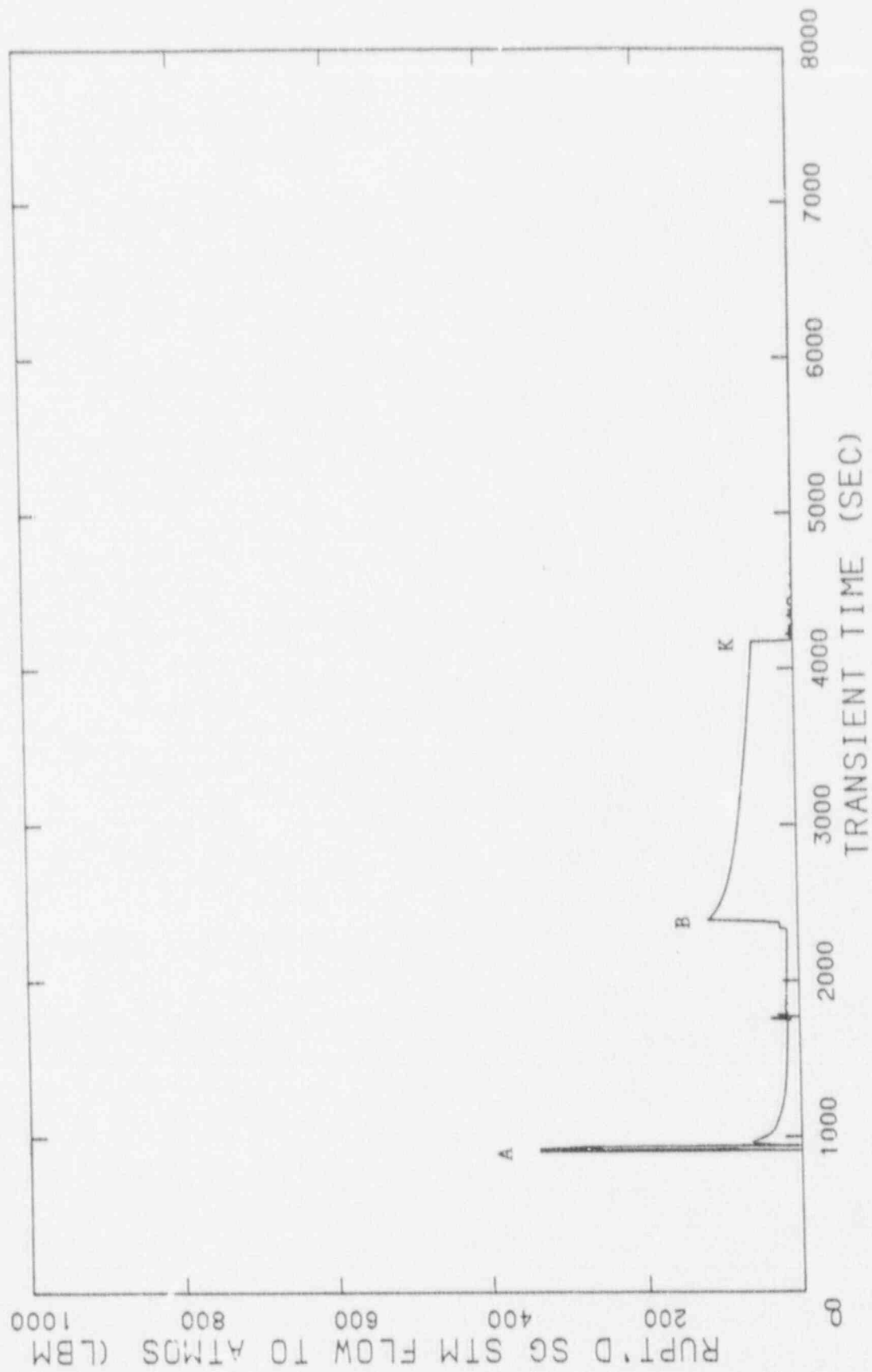


Figure 9.10

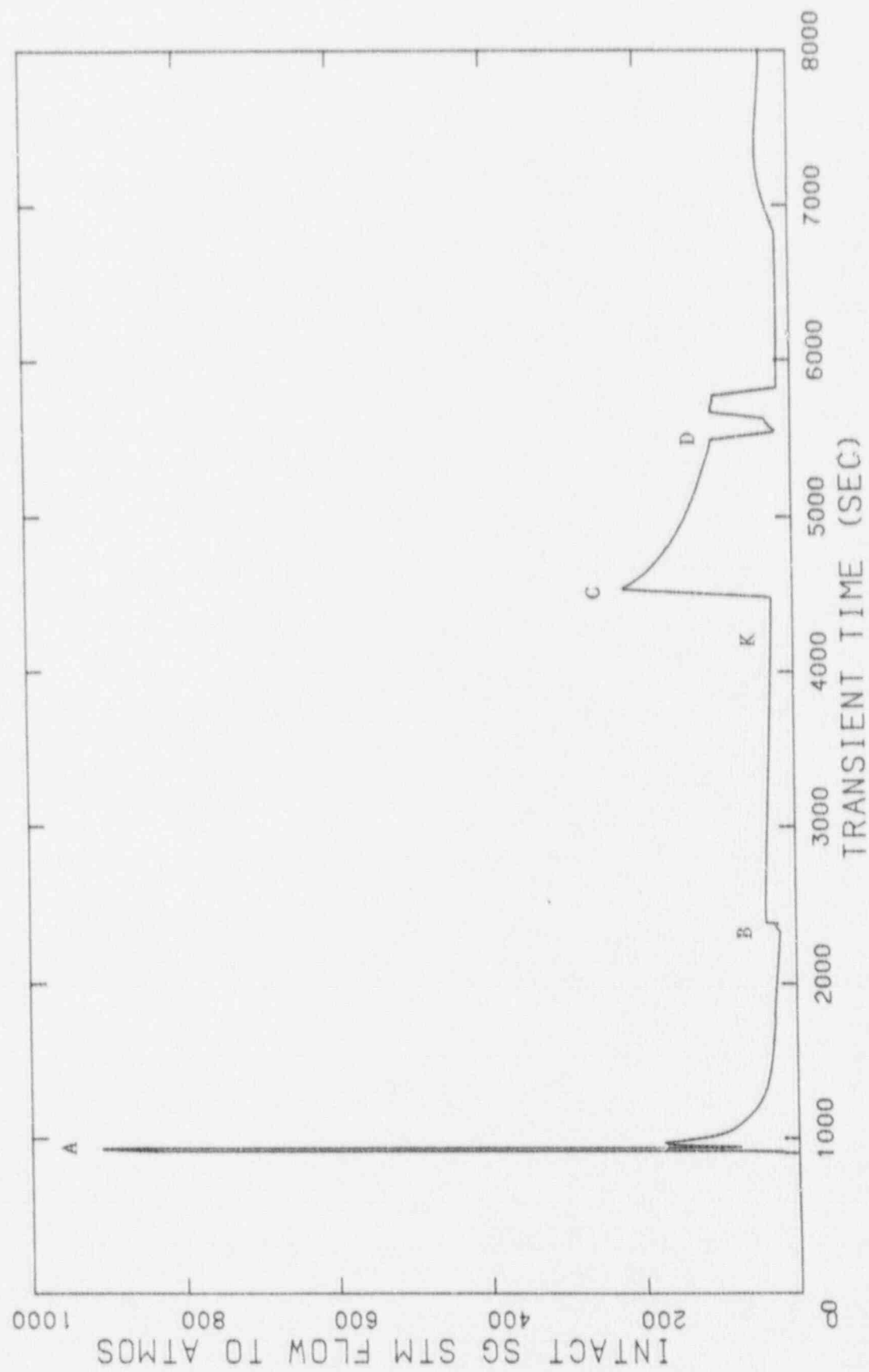


Figure 9.11

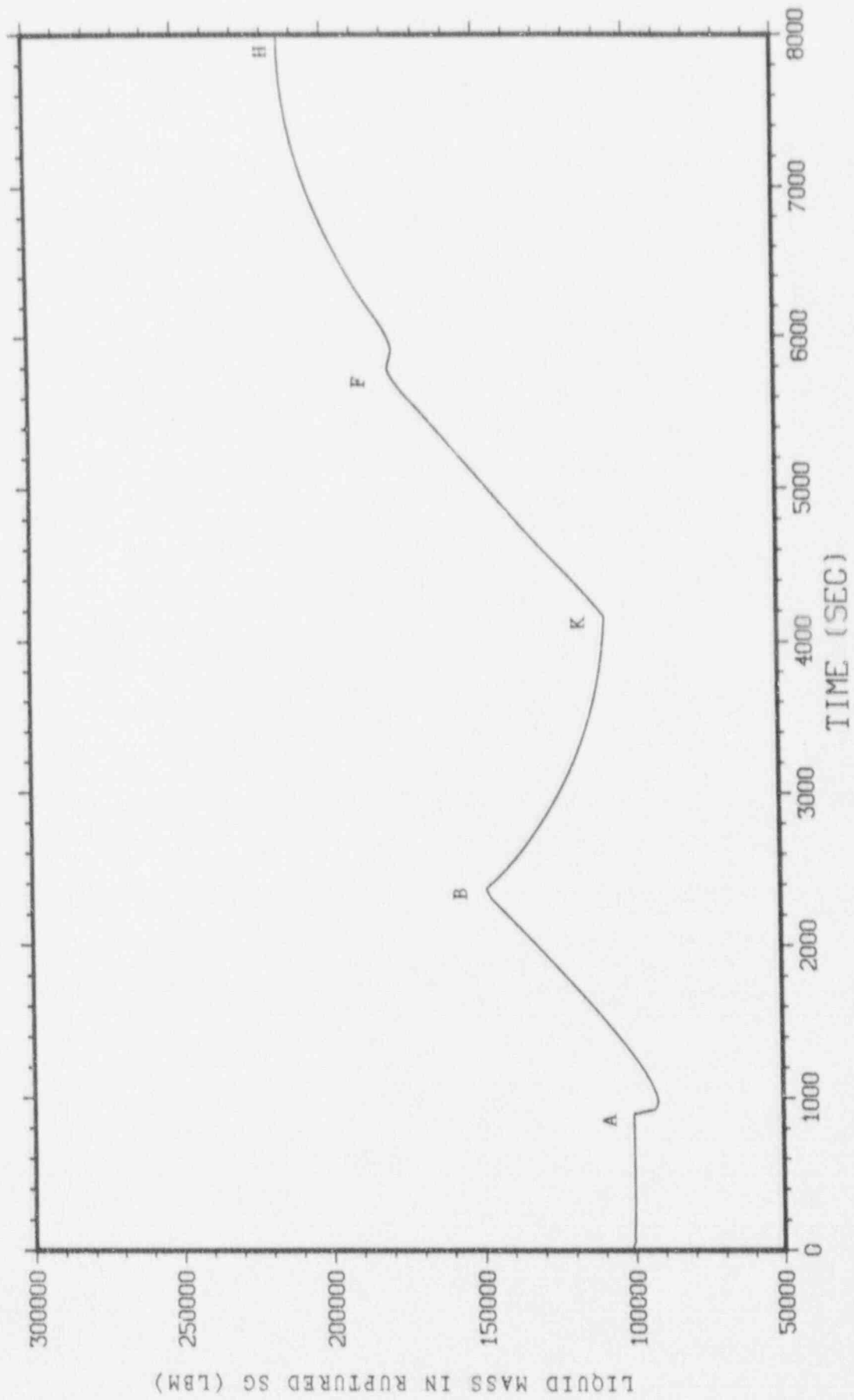
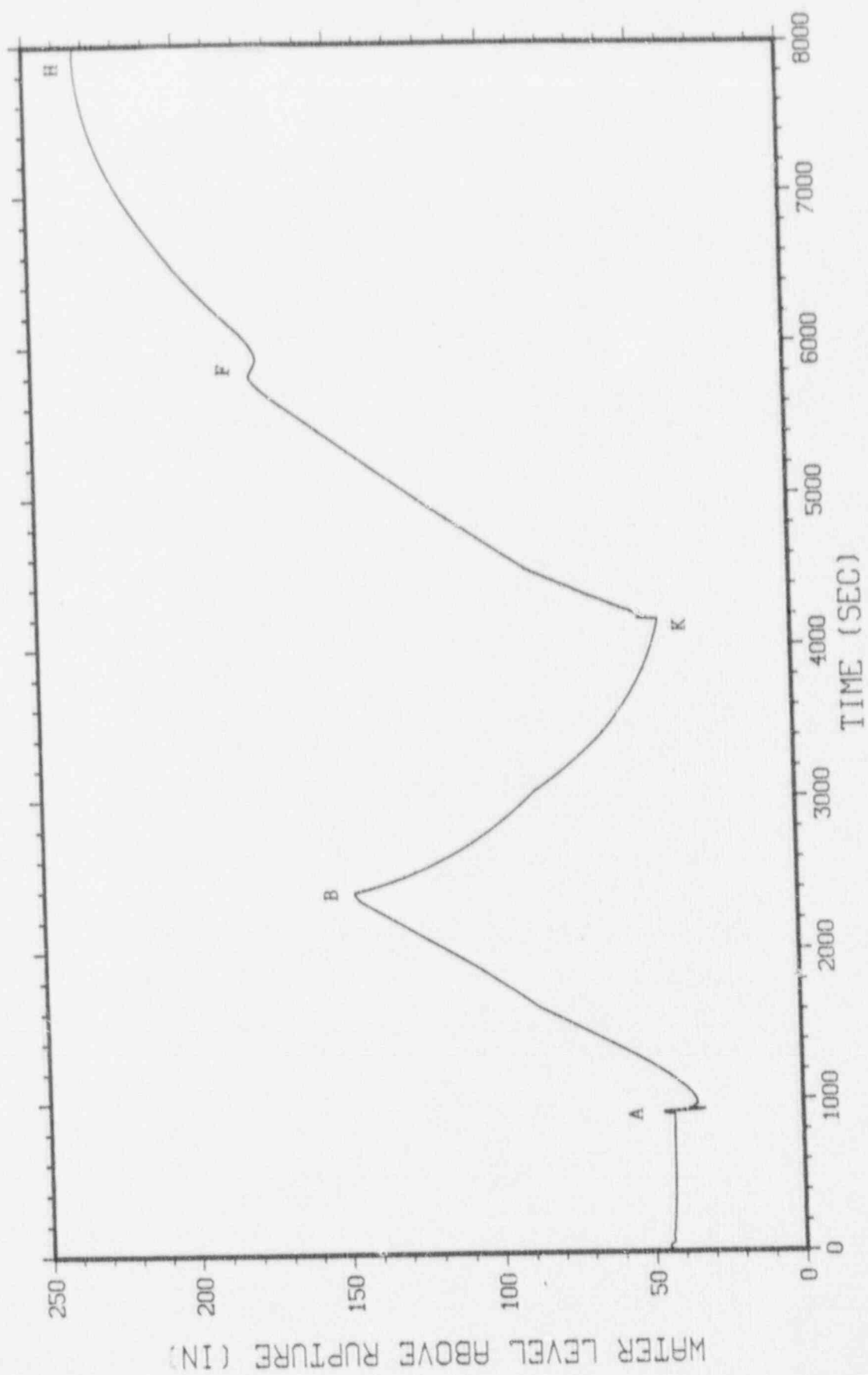


Figure 9.12



10.0 RADIOLOGICAL DOSE CALCULATIONS

Methodology and consequences for the radiological analysis of the SGTR case with the most severe results (Section 9.0) are presented here. The worst single failure for off-site radiological consequences in the event of a Steam Generator Tube Rupture (SGTR), is a failed open Atmospheric Steam Dump Valve (ASDV). This failure provides an open pathway to the environment for the release of radioactive materials. The SGTR event has been radiologically evaluated assuming a failed open ASDV in conjunction with a loss of off-site power and assuming elevated RCS iodine levels due to a preaccident iodine spike and an accident initiated iodine spike. The most severe radiological results occur for the preaccident iodine spike.

The minimum margin to Standard Review Plan (SRP) acceptance criteria occurs with an accident initiated iodine spike due to the more restrictive acceptance criteria.

For the calculation of the radiological consequences of a SGTR, it is assumed that the reactor has been operating with a small percent of defective fuel for a sufficient time to establish equilibrium concentrations of radionuclides in the primary and secondary coolants. Coolant-specific activity concentrations are conservatively assumed to be at the limits of Technical Specifications 3/4.4.8 and 3.7.1.4. (Reference (14)). Radionuclides from the primary coolant enter the steam generator via the ruptured tube and are released to the atmosphere via the turbine condenser air ejector exhaust, the steam generator ASDVs, or the safety relief valves. Steam releases via the steam-driven auxiliary feedwater pump exhaust and vents are also considered.

The radioactivity released to the atmosphere from a SGTR depends upon the primary and secondary coolant activity, iodine spiking effects, primary to secondary break and leak flow, time-dependent break flow flashing fractions, time-dependent scrubbing of flashed activity, partitioning of activity between the steam generator liquid and steam, and the mass of fluid discharged to the environment. For simplification and clarification, the term iodine partition factor (PF) used in this report will be defined as a mass based partition factor (i.e., PF_m), therefore:

$$PF_m = \frac{[I]_w}{[I]_s}$$

where: $[I]_w$ = iodine concentration in liquid, $\mu\text{Ci/gram}$
 $[I]_s$ = iodine concentration in steam, $\mu\text{Ci/gram}$

All of these parameters were conservatively evaluated and the radiological consequences calculated for each case considered in determining the one with the most severe consequences. Doses due to fission product releases were calculated using the guidance and recommendations of USNRC Standard Review Plan (SRP) 15.6.3 (Reference (13)). The calculational methods and the results of the off-site dose calculations are discussed in the following sections.

10.1 Analysis, Methodology, and Assumptions

The guidance and recommendations given in Reference (15), NUREG-0409, have been followed to the extent possible.

The major assumptions and parameters used in the calculation of the off-site radiation doses are summarized below.

10.1.1 Source Term Calculations

The concentrations of radionuclides in the primary and secondary system, prior to and following the accident, were determined as follows:

1. The equilibrium iodine activity in the reactor coolant is based on 1 $\mu\text{Ci/gram}$ dose equivalent I-131 as shown in Table 10.1. The iodine concentrations in the reactor coolant were then determined for both preaccident and accident-initiated iodine spikes as prescribed in NRC Standard Review Plan 15.6.3, Reference (13).
 - a. Preaccident Spike - It is assumed that a reactor transient has occurred prior to the SGTR which raises the primary coolant iodine concentration from 1 $\mu\text{Ci/gram}$ to 60 $\mu\text{Ci/gram}$ of dose equivalent I-131.

- b. Accident Initiated Spike - It is assumed that the primary system depressurization associated with the SGTR creates an iodine spike in the primary system which increases the iodine release rate from the fuel to the primary coolant to a value 500 times greater than the release rate corresponding to the maximum equilibrium primary system iodine concentration of 1 $\mu\text{Ci/gram}$ of dose equivalent I-131. The duration of the spike is assumed to be 4.0 hours. The iodine appearance rates in the reactor coolant for this case are presented in Table 10.2.
2. The noble gas activity in the reactor coolant is based on a Technical Specification limit of $100/\bar{E}$ as shown in Table 10.1. The assumption of $100/\bar{E}$ as applying only to the noble gas activity is conservative.
 3. The initial secondary coolant activity is based on the Technical Specification limit of a dose equivalent of 0.1 $\mu\text{Ci/gram}$ of I-131 (Reference (14)).

10.1.2 Radioactivity Transport and Release

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the resulting off-site doses following a SGTR:

1. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the intact and ruptured steam generators to the atmosphere are presented in Section 9.0, Table 9.3.
2. The time-dependent fraction of rupture flow that flashes to steam and is immediately available for release to the environment was determined based on thermodynamic considerations. Results are shown in Figure 10.1.

3. The iodine removal efficiency for scrubbing of steam bubbles as they rise from the leak site (assumed to be at the top of the tube bundle) to the water surface was also determined based on NUREG-0409 (Reference (15)) recommendations. The iodine removal efficiency is a function of the bubble rise time, partition factor for iodine, and the water level above the top of the tube bundle. The iodine partition factor used was $PF_m = 100$. A bubble rise time of 30 cm/sec (rise time for largest stable bubble of 3.6 cm) was used (Reference (15)). A conservative scrubbing efficiency of 0.0 is used from $t=0$ to $t=900$ seconds while two-phase water exists above the break location. The iodine removal efficiency used once the collapsed steam generator water level is at a minimum of approximately 3 feet above the break location (at approximately 900 seconds) is 0.40.
4. The maximum allowable preaccident primary to secondary leak rate of 1 gpm is assumed to be evenly divided among the intact steam generators.
5. All noble gas activity in the reactor coolant which is transported to the secondary system via the tube rupture is assumed to be immediately available for release to the atmosphere.
6. The iodine partition factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 100 on a mass basis (Reference (13)). The iodine partitioning factor for flashing primary coolant is conservatively assumed to be equal to 1.0 (Reference (15)). This means that the fraction of primary coolant iodine volatilized at any point in time equals the flashing fraction.
7. No credit was taken for radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the site boundary or outer boundary of the low population zone.

8. A series of tests were conducted at the MB-2 test facility (Reference (18)) to determine the primary coolant bypassing and secondary coolant carryover fractions during a SGTR event. The results of these tests were published by the Electric Power Research Institute (Reference (18)). This report recommends that 0.001% of the (primary coolant) break flow be assumed to be released directly from an open valve. This 0.001% would result in an insignificant increase in this iodine release compared to the iodine release from primary coolant flashing as discussed in Paragraph No. 3 above. The EPRI report also recommends that secondary moisture carryover be taken as 0.005% of total mass flow from the ruptured steam generator. This value would also not significantly increase the iodine release during the accident. For the analyses discussed here, it was conservatively assumed that 190 lb/hr of secondary liquid carryover was released through the ruptured steam generator ASDV. This is equivalent to approximately 0.2% of the total release and would more than account for coolant bypass and carryover.
9. One hundred percent of the iodine activity in the secondary liquid carryover and coolant bypass is assumed to be released. This conservatively accounts for flashing and atomization.
10. Loss of off-site power at 894 seconds is assumed to cause loss of main condenser availability and its assigned iodine decontamination fraction of 99%.

10.1.3 Dose Calculations

1. Short-term atmospheric dispersion factors (X/Qs) for accident analysis and breathing rates are provided in Table 10.3. The breathing rate was obtained from Reference (16). The X/Qs were obtained from the Seabrook Final Safety Analysis Report (Reference (1)).

2. Decay constants and dose conversion factors for the whole-body and thyroid doses are presented in Table 10.4. The dose conversion factors were obtained from Reference (17).
3. The off-site thyroid doses were calculated using the equation:

$$D_{Th} = \sum_i DCF_i \sum_j \tilde{R}_{ij} (Br)_j (X/Q)_j$$

where:

\tilde{R}_{ij} = integrated activity of isotope i released during the time interval j in Ci.

$(Br)_j$ = breathing rate during time interval j in meter³/second.

$(X/Q)_j$ = off-site atmospheric dispersion factor during time interval j in second/meter³.

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope i in rem/Ci inhaled.

D_{Th} = thyroid dose via inhalation in rems.

4. The off-site whole-body doses were calculated using the following equation, for an infinite cloud of gamma emitters:

$$D_{TB} = \sum_i DCF_{Yi} \sum_j \tilde{R}_{ij} (X/Q)_j$$

where:

\tilde{R}_{ij} = integrated activity of isotope i released during the jth time interval in Ci.

$(X/Q)_j$ = off-site atmospheric dispersion factor during time interval j in second/meter³.

DCF_{Yi} = gamma whole-body dose conversion factor for
isotope i in rem-m³/Ci-second

D_{TB} = gamma whole-body dose in rems.

10.2 Off-Site Radiation Doses

The off-site radiation doses were calculated using the methodology and assumptions described above. The thyroid and whole-body doses were determined at the site boundary for a two-hour exposure period and at the outer boundary of the low population zone for an eight-hour period. The results are presented in Table 10.5. The doses were determined for both a preaccident and an accident-initiated iodine spike. Standard Review Plan 15.6.3 indicates that the calculated doses for a preaccident initiated iodine spike should not exceed the guideline values of 10CFR100; while for an accident initiated iodine spike, the calculated doses should not exceed a small fraction of the 10CFR100 guidelines, i.e., 10% of the 10CFR100 limits. The acceptance criteria for the two iodine spike conditions are also presented in Table 10.5 for comparison with the calculated doses.

It can be seen from Table 10.5 that all of the calculated doses are well within the guidelines, and that the thyroid doses are controlling. In addition, it is noted that the thyroid doses for the accident initiated iodine spike are more limiting than for the preaccident iodine spike relative to these criteria, although the magnitude of the thyroid dose for the preaccident iodine spike is greater. Based on this information, it is concluded that the thyroid dose for an accident initiated spike represents the most limiting condition for the evaluation of the off-site radiation exposure. All off-site doses were calculated using the computer code GENRUP (see Reference (19) and Appendix A). Thermal-hydraulic parameters were obtained from Section 9.0. The Control Room doses are bounded by the LOCA results in FSAR Section 15.6.5.4.

TABLE 10.1

Reactor Coolant Iodine and Noble Gas Activity

<u>Nuclide</u>	<u>Iodine Activity Based on 1 $\mu\text{Ci}/\text{gram}$ of Dose Equivalent I-131 ($\mu\text{Ci}/\text{gram}$)</u>
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I-131	5.43×10^{-1}
I-132	2.53×10^0
I-133	1.69×10^0
I-134	4.10×10^0
I-135	3.14×10^0

<u>Nuclide</u>	<u>Noble Gas Activity Based on 100/E Coolant Technical Specification Limit (assumed to be all due to noble gas) ($\mu\text{Ci}/\text{gram}$)</u>
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Xe-131m	2.48×10^1
Xe-133m	2.38×10^0
Xe-133	8.83×10^1
Xe-135m	4.42×10^0
Xe-135	2.89×10^1
Xe-137	1.16×10^0
Xe-138	4.08×10^0
Kr-85m	5.44×10^0
Kr-85	1.46×10^1
Kr-87	5.10×10^0
Kr-88	9.51×10^0

TABLE 10.1a

Secondary Coolant Iodine Activity

<u>Nuclide</u>	<u>Iodine Activity Based on 0.1 $\mu\text{Ci}/\text{gram}$ of Dose Equivalent I-131 ($\mu\text{Ci}/\text{gram}$)</u>
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I-131	6.10×10^{-2}
I-132	1.05×10^{-1}
I-133	1.63×10^{-1}
I-134	8.14×10^{-2}
I-135	2.24×10^{-1}

TABLE 10.2

Iodine Appearance Rates in the
Reactor Coolant for a Design Basis SGTR
(Curies/Hour)

	<u>I-131</u>	<u>I-132</u>	<u>I-133</u>	<u>I-134</u>	<u>I-135</u>
Equilibrium Appearance Rates Due to Technical Specification Fuel Defects	1.03×10^1	2.16×10^2	4.31×10^1	7.96×10^2	1.30×10^2
Appearance Rates Due to an Iodine Spike-500X Equilibrium Rates	5.14×10^3	1.08×10^5	2.16×10^4	3.98×10^5	6.50×10^4

TABLE 10.3

Short-Term Atmospheric Dispersion Factors and Breathing Rates
for Accident Analysis

Time (hours)	Site Boundary x/Q (sec/m ³)	Low Population Zone x/Q (sec/m ³)	Breathing Rate (m ³ /sec)
0-1	2.67×10^{-4}	1.31×10^{-4}	3.47×10^{-4}
1-2	1.88×10^{-4}	9.17×10^{-5}	3.47×10^{-4}
2-8	--	4.82×10^{-5}	3.47×10^{-4}

TABLE 10.4

Isotopic Data

Isotope	Decay Constant (1/Hour)	Dose Conversion Factors	
		Whole-Body (rem-m ³ /Ci-hr)	Thyroid (rem/Ci)
I-131	3.59×10^{-3}	3.17×10^2	1.49×10^6
I-132	3.01×10^{-1}	1.93×10^3	1.43×10^4
I-133	3.33×10^{-2}	4.95×10^2	2.69×10^5
I-134	7.91×10^{-1}	2.01×10^3	3.73×10^3
I-135	1.05×10^{-1}	1.48×10^3	5.60×10^4
Xe-131m	2.43×10^{-3}	1.04×10^1	--
Xe-133m	1.32×10^{-2}	2.83×10^1	--
Xe-133	5.51×10^{-3}	3.36×10^1	--
Xe-135m	2.72×10^0	3.60×10^2	--
Xe-135	7.63×10^{-2}	2.08×10^2	--
Xe-138	2.94×10^0	1.02×10^3	--
Kr-85m	1.55×10^{-1}	1.35×10^2	--
Kr-85	7.38×10^{-6}	1.85×10^0	--
Kr-87	5.45×10^{-1}	6.82×10^2	--
Kr-88	2.44×10^{-1}	1.69×10^3	--

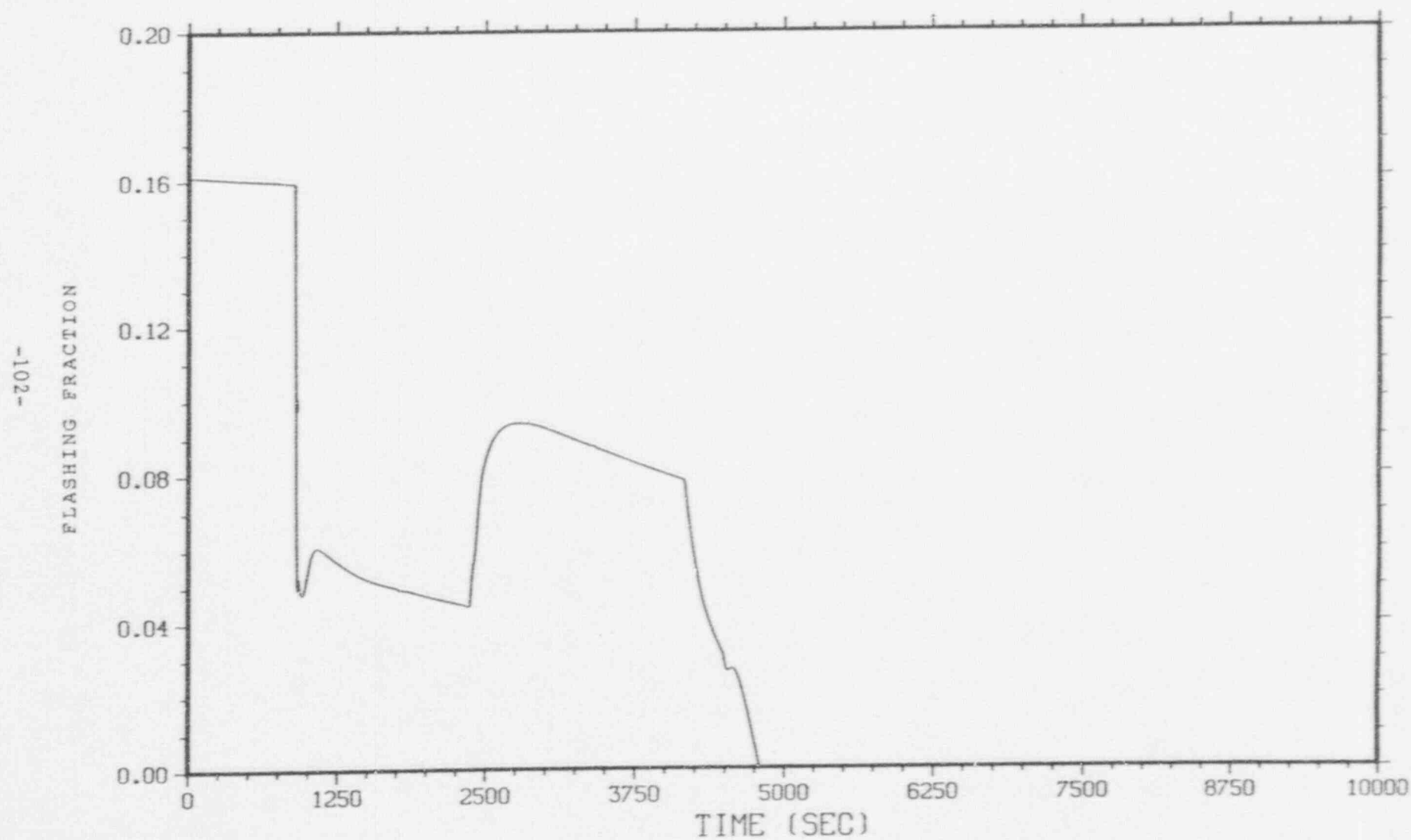
TABLE 10.5

Calculated Off-Site Radiation Doses for a Design Basis SGTR
Assuming a Failed Open ASDV and 30-Minute Isolation Time

		Doses (Rem)			
		Site Boundary (0-2 hr)		Low Pop. Zone (0-8 hr)	
		Thyroid	Total Body	Thyroid	Total Body
1.	Preaccident Iodine Spike				
	Calculated Dose	2.3×10^1	5.6×10^{-1}	1.1×10^1	4.2×10^{-1}
	Standard Review Plan Guidelines	300	25	300	25
2.	Accident Initiated Iodine Spike				
	Calculated Dose	1.8×10^1	1.1×10^0	9.0×10^0	7.7×10^{-1}
	Standard Review Plan Guidelines	30	2.5	30	2.5

FIGURE 10.1

PRIMARY COOLANT FLASHING IN RUPTURED STEAM GENERATOR



11.0 CONCLUSIONS

This report documents an analysis of a postulated design basis SGTR at the Seabrook Nuclear Power Station. The analysis complies with recent NRC SERs in References (5) and (6). The RETRAN computer code was used to predict the thermal-hydraulic plant response. This code has been reviewed and approved by the NRC for use in the analysis of non-LOCA transients, including SGTRs.

Systems, components, and instrumentation credited for mitigation of the design basis SGTR are nuclear safety related and meet the single failure criteria for the required function.

The most limiting single active failure for SG overfill is a failed closed ASDV on an intact SG. Analysis results show that overfill of the faulted SG will not occur. Nevertheless, a static load analysis of flooded main steam lines has been performed as required by the NRC. Results show the main steam lines will remain intact in this highly unlikely consequence of a SGTR.

The most limiting single active failure for radiological consequences was determined to be a failed open ASDV on the faulted steam generator. Analysis results show that the off-site radiological doses received at the exclusion area boundary and the low population zone boundary would be within the limits of 10CFR100 and the SRP guideline values. This is true provided the failed open ASDV is isolated within 30 minutes after failure. Corrective action is manual closure of the associated block valve in accordance with Emergency Response Procedure E-3, "Steam Generator Tube Rupture." Plant tests verified that auxiliary operators can perform this corrective action within 30 minutes after failure.

12.0 REFERENCES

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- (12) NUREG-75/087, Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."
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- (16) NRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors," Revision 2, June 1974.
- (17) NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10CFR, Part 50, Appendix I," Revision 1, October 1977.
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APPENDIX A

"GENRUP" Description

GENRUP is a computer code developed for the radiological assessment of SGTR accidents. In such accidents halogens and noble gases from the primary coolant enter the secondary side of the failed steam generator and are released to the environment either through the air ejectors in the main condenser, or through the pressure relief valves.

It is clear that any theoretical model for the prediction of radiological impacts must contain a fairly extensive set of both thermal-hydraulic and radiological considerations. In GENRUP, the thermal-hydraulic and radiological effects were decoupled, and all thermal-hydraulic data must be provided as input to the code, as a function of post-tube rupture time.

With respect to the radiological effects, the radionuclide pathways employed in GENRUP include four compartments where the accumulation of radioactivity depends on both leakage and decay (namely, the primary coolant, the intact steam generators, the ruptured steam generator, and the Control Room), and six components where halogens may accumulate during the accident (namely, the intact and ruptured steam generator internal surfaces, the main condenser, the water pool resulting from safety relief valve leakage, and the Control Room intake and recirculation filters). The list of radionuclides includes 11 halogens and 13 noble gases. Initial concentrations in the primary coolant and the steam generator waters can be provided either individually for each isotope, or in lump sum equivalent units, and halogen production in the primary coolant as a result of coincident spiking effects is available. The transfer of halogens from the steam generator waters to the steam spaces is based on partitioning, and the effects of flashing and scrubbing in the ruptured steam generator are accounted for. The code predicts the isotopic release rates and cumulative releases to the atmosphere, as well as the radiation exposure rates and total exposures (thyroid, whole body, and skin) at either an off-site location or in a nearby building or room (such as the Control Room).