Westinghouse Energy Systems

WESTINGHOUSE CLASS 3

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LOFTTR2 ANALYSIS FOR A STEAM GENERATOR TUBE RUPTURE FOR THE SOUTH TEXAS PROJECT UNITS 1 AND 2

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1. INTRODUCTION

An evaluation for a design basis steam generator tube rupture (SGTR) event has been performed for the South Texas Project (STP), Units 1 and 2, to demonstrate that the potential consequences are acceptable. This evaluation includes an analysis to demonstrate margin to steam generator overfill and an analysis to demonstrate that the calculated offsite radiation doses are less than the allowable guidelines.

The South Texas Project employs two essentially identical Westinghouse pressurized water reactor (PWR) units rated at 3800 MWt. The reactor coolant system for each unit has four reactor coolant loops with Model E2 steam generators. Since the reactors, structures, and all auxiliary equipment are substantially identical for the two units, the SGTR evaluation is applicable for both units. It is also noted that both units are currently licensed to operate with Westinghouse standard fuel with a negative moderator temperature coefficient. However, it is anticipated that the technical specifications will be changed to permit operation with a positive moderator temperature coefficient for future fuel cycles. Therefore, a more limiting positive moderator temperature coefficient was assumed for the SGTR evaluation such that the results are applicable for the current licensing basis as well as for future operation with a positive moderator temperature coefficient. The evaluation is also applicable for up to 15 percent steam generator tube plugging and for a minimum auxiliary feedwater flow rate of 500 gpm per steam generator.

The steam generator tube rupture analyses were performed for South Texas using the methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). This analysis methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group and was approved by the NRC in Safety Evaluation Reports dated December 17, 1985 and March 30, 1987. The LOFTTR2 program, an updated version of the LOFTTR1 program, was used to perform the SGTR analysis for South Texas. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluations in References 1 and 2. However, the LOFTTR1 program was subsequently modified to accommodate steam generator overfill and the revised program, designated as LOFTTR2, was used for the evaluation of the consequences of overfill in WCAP-11002 (Reference 3). The LOFTTR2 program is identical to the LOFTTR1 program, with the exception that the LOFTTR2 program has the additional capability to represent the transition from two regions (steam and water) on the secondary side to a single water region if overfill occurs, and the transition back to two regions again depending upon the calculated secondary conditions. Since the LOFTTR2 program has been validated against the LOFTTR1 program, the LOFTTR2 program is also appropriate for performing licensing basis SGTR analyses.

Plant response to the SGTR event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass in the ruptured steam generator. The analysis methodology includes the simulation of the operator actions for recovery from a steam generator tube rupture based on the South Texas Emergency Operating Procedures (EOPs), which were developed from the Westinghouse Owners Group Emergency Response Guidelines (ERGs). In subsequent references to the South Texas EOPs, the specific EOP will be listed along with the corresponding Westinghouse Owners Group ERG in parenthesis.

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. Therefore, an analysis must be performed to assure that the offsite radiation doses resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiation doses. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. An analysis was also performed to determine the offsite radiation doses, assuming the limiting single failure relative to offsite doses without steam generator overfill. The limiting single failure assumptions for these analyses are consistent with the methodology in References 1 and 2.

For the margin to overfill analysis, it was assumed that the [

]a,c The LOFTTR2 analysis to determine the margin to overfill was performed for the time period from the tube rupture until the primary and secondary pressures are equalized and the break flow is terminated. The water volume in the secondary side of the ruptured steam generator was calculated as a function of time to demonstrate that overfill does not occur. The results of this analysis demonstrate that there is margin to steam generator overfill for South Texas.

Since steam generator overfill does not occur, the results of the offsite radiation dose analysis represent the limiting consequences for South Texas. For the analysis of the offsite radiation doses, [

]a,c The primary to secondary break flow and the steam releases to the atmosphere from both the ruptured and intact steam generators were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated with the LOFTTR2 program from the initiation of the event until termination of the break flow. For the time period following break flow termination, steam releases from and feedwater flows to the ruptured and intact steam generators were determined from a mass and energy balance using the calculated RCS and steam generator conditions at the time of leakage termination. The mass release information was used to calculate the radiation doses at the exclusion area boundary and low population zone assuming that the primary coolant activity is at the maximum allowable Technical Specification limit prior to the accident. The results of this analysis show that the offsite doses for South Texas are within the allowable guidelines specified in the Standard Review Plan, NUREG-0800, Section 15.6.3, and 10CFR100.

II. ANALYSIS OF MARGIN TO STEAM GENERATOR OVERFILL

An analysis was performed to determine the margin to steam generator overfill for a design basis SGTR event for South Texas. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1. This section includes a discussion of the methods and assumptions used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

A. Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet [ ][a,c] The location of the break [ ][a,c] It was also assumed that loss of offsite power occurs at the time of reactor trip, and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The most limiting single failure with respect to steam generator overfill was determined to be [ ][a,c] However, the most limiting single failure for the four-loop South Texas plants is [ ][a,c]

The South Texas AFW system consists of four independent trains (three identical motor-driven pumps and one turbine-driven pump of equal capacity) with each feeding a dedicated steam generator. There is an AFW flow control valve for each steam generator in the flow path from the associated AFW pump. The AFW flow control valves would be normally open and are used to terminate feedwater flow to the ruptured steam generator and control inventory in the intact steam generators. [ ][a,c]

[ ][a,c] would require the operator to perform additional action to [ ][a,c] In accordance with Reference 1, it was assumed that [ ][a,c] No subsequent recovery actions are performed until the [ ][a,c] Thus, this [ ][a,c] which decreases the margin to steam generator overfill.

D. Conservative Assumptions

Sensitivity studies were performed previously to identify the initial plant conditions and analysis assumptions which are conservative relative to steam generator overfill, and the results of these studies were reported in Reference 1. The conservative conditions and assumptions which were used in Reference 1 were also used in the LOFTTR2 analysis to determine the margin to steam generator overfill for South Texas with the exception of the following differences.

1. Reactor Trip and Turbine Runback

A turbine runback can either be initiated automatically or the operator can manually reduce the turbine load following an SGTR to attempt to prevent a reactor trip. For the reference plant analysis in WCAP-10698, reactor trip was calculated to occur at approximately [ ][a,c] and turbine runback to [ ][a,c] was simulated based on a runback rate of [ ][a,c] The effect of turbine runback was conservatively simulated by [ ][a,c] However, if reactor trip occurs [ ][a,c] turbine runback to [ ][a,c] would not be possible. It is noted that earlier reactor trip will result in earlier initiation of primary to secondary break flow accumulation in the ruptured steam generator and earlier initiation of AFW flow. These effects will result in an increased secondary mass in the ruptured steam generator at the time of isolation since the isolation is assumed to occur at a fixed time after the SGTR occurs rather than at a fixed time after reactor trip. It would be overly conservative to include the simulation of turbine runback to [ ]^{a,c} in addition to the penalty in secondary mass due to earlier reactor trip. Thus, for this analysis, the time of reactor trip was determined by modeling the South Texas reactor protection system, and turbine runback was simulated [ ]^{a,c}

2. <u>Steam Generator Secondary Mass</u>

A [ ]^{a,c} initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. As noted above, turbine runback was assumed to be initiated and was simulated by [ ]^{a,c} The initial steam generator total fluid mass was conservatively assumed to be [ ]^{a,c}

3. <u>AFW System Operation</u>

For the reference plant analysis in WCAP-10698, reactor trip occurred on [ ]^{a,c} after the SGTR, and SI was initiated on low pressurizer pressure at [ ]^{a,c} after reactor trip. The reactor and turbine trip and the assumed concurrent loss of offsite power will result in the termination of main feedwater flow and actuation of the AFW system. The SI signal will also result in automatic isolation of the main feedwater system and actuation of the AFW system. The flow from the turbine-driven AFW pump will be available within approximately 10 seconds following the actuation signal, but the flow from the motor-driven AFW pumps will not be available until approximately 60 seconds due to the startup and load sequencing for the emergency diesel generators. For the reference plant analysis, it was assumed that AFW flow from both the turbine and motor-driven pumps is initiated [ ]a,c The total AFW flow from all of the AFW pumps was assumed to be distributed uniformly to each of the steam generators until operator actions are simulated to throttle AFW flow to control steam generator water level in accordance with the emergency procedures.

It is noted that if reactor trip occurs on [ ]a,c the pressure at the time of reactor trip may be significantly higher than the SI initiation setpoint. In this event, there may be a significant time delay between reactor trip and SI initiation, and it would not be conservative to model the [ ]a,c Thus, for this analysis, the time of reactor trip was determined by modeling the South Texas reactor protection system, and the actuation of the AFW system was based on the [ ]a,c It was assumed that flow from the turbine and motor-driven AFW pumps is initiated at [representative time delay for delivery of AFW flow to the steam]a,c The maximum potential AFW flow rate of 675 gpm was used in the analysis for the turbine-driven pump and also for each of the motor-driven pumps.

C. Operator Action Times

In the event of an SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in South Texas EOP POPO5-EO-EO30 (E-3), and these actions were explicitly modeled in this analysis. The operator actions modeled include identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to restore inventory, and termination of SI to stop primary to secondary leakage. These operator actions are described below.

1. Identify the ruptured steam generator.

   High secondary side activity, as indicated by the main steamline radiation monitors, steam generator blowdown radiation monitors, or condenser vacuum pump radiation monitor typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator level, or a high radiation indication on the corresponding main steamline radiation monitor or steam generator blowdown line radiation monitor, or high activity in any steam generator sample. For an SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level in that steam generator will increase more rapidly. This response, as indicated by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

2. Isolate the ruptured steam generator from the intact steam generators and isolate feedwater to the ruptured steam generator.

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator with water by 1) minimizing the accumulation of feedwater flow and 2) enabling the operator to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary to secondary leakage. In the South Texas EOP for steam generator tube rupture recovery, the operator is directed to maintain the level in the ruptured steam generator between 5% and 50% on the narrow range instrument. [ ] [a,c] It was assumed that the ruptured steam generator will be isolated when level in the steam generator reaches midway between 5% and 50% or at 10 minutes, whichever is longer. Thus, for the South Texas SGTR analysis, the ruptured steam generator was assumed to be isolated at the time when the narrow range level reaches 27.5% or at 10 minutes, whichever was longer.

3. Cool down the Reactor Coolant System (RCS) using the intact steam generators.

   After isolation of the ruptured steam generator, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure by dumping steam from only the intact steam generators. This ensures adequate subcooling in the RCS after depressurization to the ruptured steam generator pressure in subsequent actions. If offsite power is available, the normal steam dump system to the condenser can be used to perform this cooldown. However, if offsite power is lost, the RCS is cooled using the PORVs on the intact steam generators. Since offsite power is assumed to be lost at reactor trip for this analysis, the cooldown was performed by dumping steam via the PORVs on the three intact steam generators.

4. Depressurize the RCS to restore reactor coolant inventory.

   When the cooldown is completed, SI flow will increase RCS pressure until break flow matches SI flow. Consequently, SI flow must be terminated to stop primary to secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after SI flow is stopped. Since leakage from the primary side will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize, an "excess" amount of inventory is needed to ensure pressurizer level remains on span. The "excess" amount required depends on RCS pressure and reduces to zero when RCS pressure equals the pressure in the ruptured steam generator.

The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running, or auxiliary pressurizer spray or the pressurizer PORVs if the RCPs are not running. Since offsite power is assumed to be lost at the time of reactor trip, the RCPs are not running and thus normal pressurizer spray is not available. In the South Texas SGTR recovery procedure, the first alternative is to use auxiliary pressurizer spray if normal spray is not available, and the second alternative is to use a pressurizer PORV. Since the auxiliary pressurizer spray does not meet all of the requirements for safety grade equipment, credit would not normally be taken in the analysis for the use of auxiliary pressurizer spray and the analysis would be based on the use of a safety grade pressurizer PORV. However, a scoping study has indicated that the use of the auxiliary spray produces more conservative results than the use of a PORV. Therefore, for this analysis, RCS depressurization was assumed to be performed using auxiliary pressurizer spray.

The SGTR recovery procedure for South Texas instructs the operators to establish maximum charging flow after the RCS cooldown is completed but prior to the RCS depressurization. However, for RCS depressurization using the auxiliary spray system, the charging flow to the RCS must be isolated in order to utilize the auxiliary spray flow path to the pressurizer. Thus, it was assumed that the flow from two centrifugal charging pumps is supplied to the RCS, in addition to the flow from the SI pumps, for the time period from completion of the RCS cooldown until the initiation of RCS depressurization. For the

RCS depressurization, it was assumed that the normal charging flow path is isolated, and the auxiliary spray flow rate was conservatively based on the operation of only one charging pump. It was also conservatively assumed that the auxiliary spray flow rate is constant at the RCS pressure corresponding to the beginning of the depressurization, whereas the spray flow rate would actually increase as the RCS pressure decreases. After the completion of the RCS depressurization, it was assumed that the charging flow from two centrifugal charging pumps is reinitiated.

5. Terminate SI to stop primary to secondary leakage.

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, SI flow must be stopped to terminate primary to secondary leakage. Primary to secondary leakage will continue after SI flow is stopped until RCS and ruptured steam generator pressures equalize. Charging flow, letdown, and pressurizer heaters will then be controlled to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator. It was assumed that charging flow from the two centrifugal charging pumps continues for one minute following SI termination before the operators complete the action to eliminate excess charging flow.

Since these major recovery actions are modeled in the SGTR analysis, it is necessary to establish the times required to perform these actions. Although the intermediate steps between the major actions are not explicitly modeled, it is also necessary to account for the time required to perform the steps. It is noted that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to identify and isolate the ruptured steam generator, to initiate RCS cooldown, to initiate RCS depressurization, and to perform safety injection termination were developed in Reference 1 for the design basis analysis. Houston Lighting and Power Company has determined the corresponding operator action times to perform these operations for South Texas. The operator actions and the corresponding operator action times used for the South Texas analysis are listed in Table II.1.

TABLE II.1
STP SGTR ANALYSIS
OPERATOR ACTION TIMES FOR DESIGN BASIS ANALYSIS

Action	Time (min)
Identify and isolate ruptured SG	10 min or LOFTTR2 calculated time to reach 27.5% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	4
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	3
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	2
SI termination and pressure equalization	Calculated time after SI termination for equalization of RCS and ruptured SG pressures, assuming excess charging flow from two centrifugal charging pumps for one minute after SI termination.

D. Transient Description

The LOFTTR2 analysis results for the margin to overfill analysis are described below. The sequence of events for this transient is presented in Table II.2.

Following the tube rupture, reactor coolant flows from the primary into the secondary side of the ruptured steam generator since the primary pressure is greater than the steam generator pressure. In response to this loss of reactor coolant, pressurizer level decreases as shown in Figure II.1. The RCS pressure also decreases as shown in Figure II.2 as the steam bubble in the pressurizer expands. As the RCS pressure decreases due to the continued primary to secondary leakage, automatic reactor trip occurs at 19 seconds on an overtemperature delta-T trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure II.3. The main feedwater flow will be terminated and AFW flow will be automatically initiated following reactor trip and the loss of offsite power.

The RCS pressure and pressurizer level continue to decrease after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. When the RCS temperature differential begins to increase at approximately 35 seconds, the RCS pressure and pressurizer level decrease less rapidly. The decrease in RCS inventory results in a low pressurizer pressure SI signal at 376 seconds. However, before the RCS pressure decreases to the shutoff head of the high head SI pumps, the pressurizer level goes offscale low. After the RCS pressure is below the shutoff head of the high head SI pumps, the SI flow rate maintains the reactor coolant inventory and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power is assumed lost at reactor trip, the RCPs trip and a gradual transition to natural circulation flow occurs. Immediately following reactor trip the temperature differential across the core decreases as core power decays (see Figure II.4); however, the temperature differential subsequently increases at approximately 35 seconds as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperatures initially trend toward the steam generator temperature as the fluid residence time in the tube region increases. The RCS hot and cold leg temperatures then slowly decrease due to the continued addition of the auxiliary feedwater to the steam generators until operator actions are initiated to control the auxiliary feedwater flow.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   Once a tube rupture has been identified, recovery actions begin by isolating steam flow from the ruptured steam generator and isolating the auxiliary feedwater flow to the ruptured steam generator. As indicated previously, it is assumed that the ruptured steam generator will be identified and isolated when the narrow range level reaches 27.5% on the ruptured steam generator or at 10 minutes after initiation of the SGTR, whichever is longer. For the South Texas analysis, the time to reach 27.5% is less than 10 minutes, and thus it was assumed that the actions to isolate the ruptured steam generator are performed at 10 minutes. However, as noted previously, the limiting single failure was assumed to be [ ]a,c

[ ] a,c when the isolation is being performed. It was assumed that [

] a,c Thus, the isolation of AFW flow to the ruptured steam generator was assumed to be completed at 12 minutes after the SGTR. The actual time used in the analysis is 2 seconds longer because of the computer program numerical requirements for simulating the operator actions.

2. Cool Down the RCS to Establish Subcooling Margin

After isolation of the ruptured steam generator is completed at 722 seconds, a 4 minute operator action time is imposed prior to initiating the cooldown. After this time, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generators. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the PORVs on the intact steam generators. It was assumed that [ ] a,c the intact steam generator PORVs are opened at 962 seconds for the RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 35°F for subcooling uncertainty. When these conditions are satisfied at 1410 seconds, it is assumed that the operator closes the intact steam generator PORVs to terminate the cooldown. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure. The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure II.3, and the effect of the cooldown on the RCS temperature is shown in Figure II.4. As shown in Figure II.2, the RCS pressure also decreases initially during this cooldown process due to shrinkage of the reactor coolant, and then begins to increase due to the increased SI flow.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, it is assumed that normal charging flow from two centrifugal charging pumps is initiated. A 3 minute operator action time is then included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by using auxiliary pressurizer spray. The normal charging flow path is isolated in order to utilize the auxiliary spray flow path to the pressurizer. The RCS depressurization is initiated at 1600 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 8% for pressurizer level uncertainty, or pressurizer level is greater than 70%, or RCS subcooling is less than the 35°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated due to high pressurizer level. The RCS depressurization reduces the break flow as shown in Figure II.5, and increases SI flow to refill the pressurizer as shown in Figure II.1. After completion of the RCS depressurization, the charging flow from two centrifugal charging pumps was reinitiated.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 35°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is increasing, and the pressurizer level is greater than the 8% allowance for uncertainty.

After depressurization is completed, an operator action time of 2 minutes was assumed prior to SI termination. Since the above requirements are satisifed, SI termination was performed at this time. The charging flow from two centrifugal charging pumps was continued from the end of the RCS depressurization until 1 minute after SI termination, at which time it was assumed that excess charging flow is eliminated. After SI termination and the elimination of excess charging flow, the RCS pressure begins to decrease as shown in Figure II.2.

The intact steam generator PORVs also automatically open to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured steam generator pressure. The primary to secondary leakage continues after the SI flow and excess charging flow are terminated until the RCS and ruptured steam generator pressures equalize.

The primary to secondary break flow rate throughout the recovery operations is presented in Figure II.5. The water volume in the ruptured steam generator is presented as a function of time in Figure II.6. It is noted that the water volume in the ruptured steam generator when the break flow is terminated is significantly less than the total steam generator volume of 7983 ft^3. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for South Texas.

TABLE II.2

STP SGTR ANALYSIS

SEQUENCE OF EVENTS

MARGIN TO OVERFILL ANALYSIS

EVENT	Time (sec)
SG Tube Rupture	0
Reactor Trip	19
SI Actuation	376
Ruptured SG Isolated	722
RCS Cooldown Initiated	962
RCS Cooldown Terminated	1410
Two Charging Pumps Started	1416
Charging Flow to RCS Isolated	1600
RCS Depressurization Initiated	1600
RCS Depressurization Terminated	2072
Two Charging Pumps Started	2074
SI Terminated	2194
Excess Charging Flow Eliminated	2256
Steam Relief to Maintain RCS Subcooling	2674
Break Flow Terminated	3786

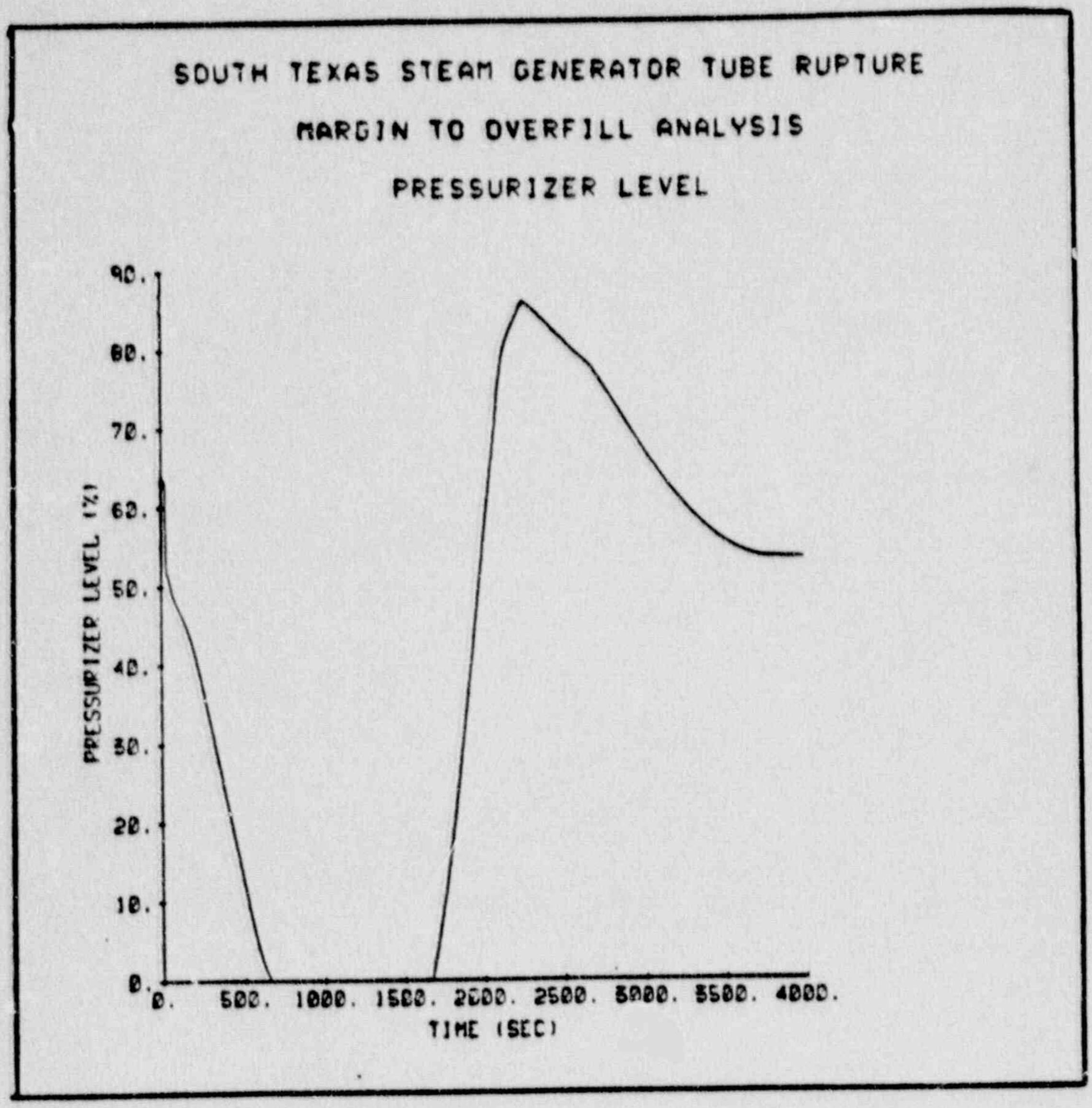

Figure II.1 Pressurizer Level - Margin to Overfill Analysis

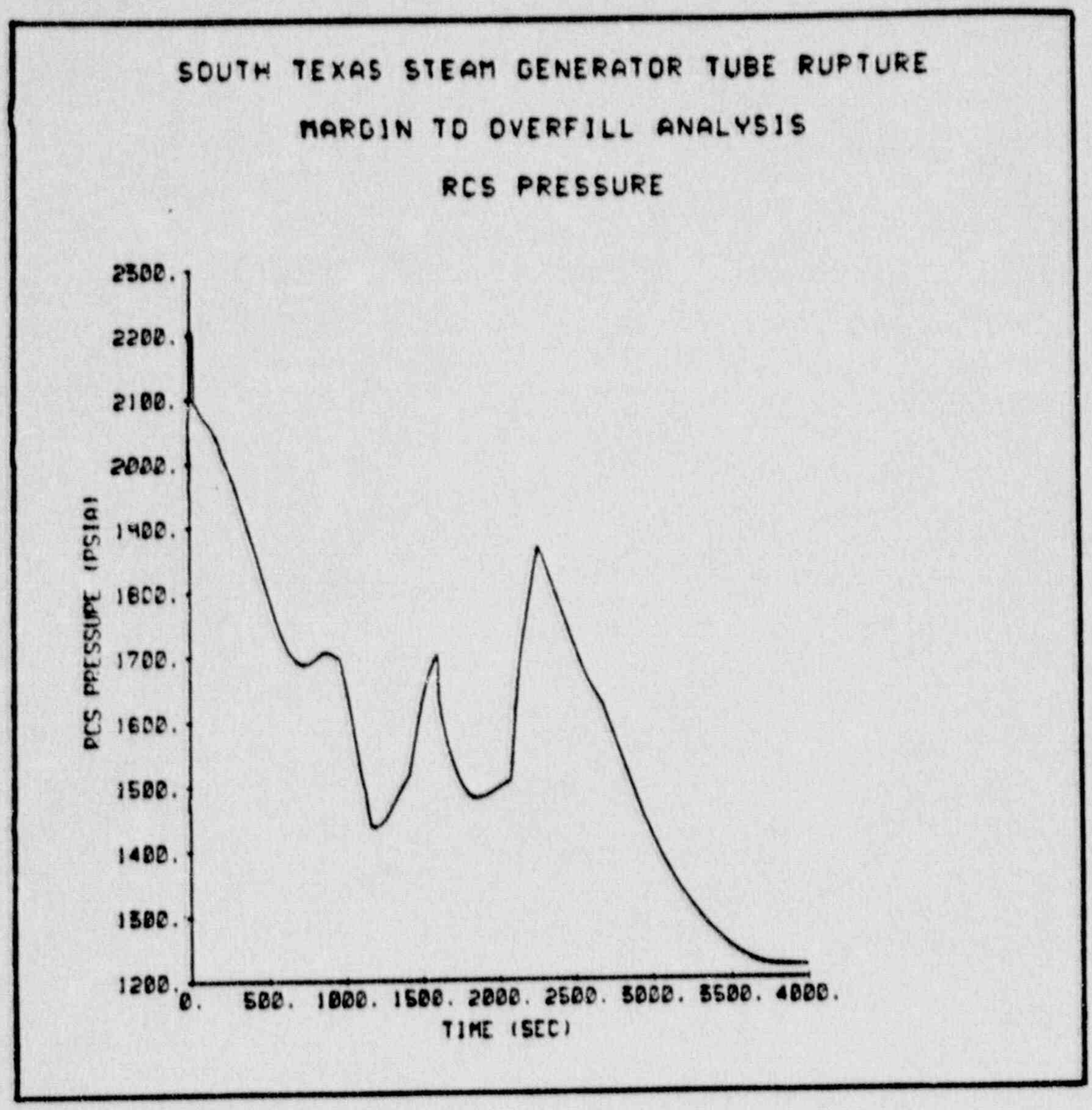

Figure II.2 RCS Pressure - Margin to Overfill Analysis

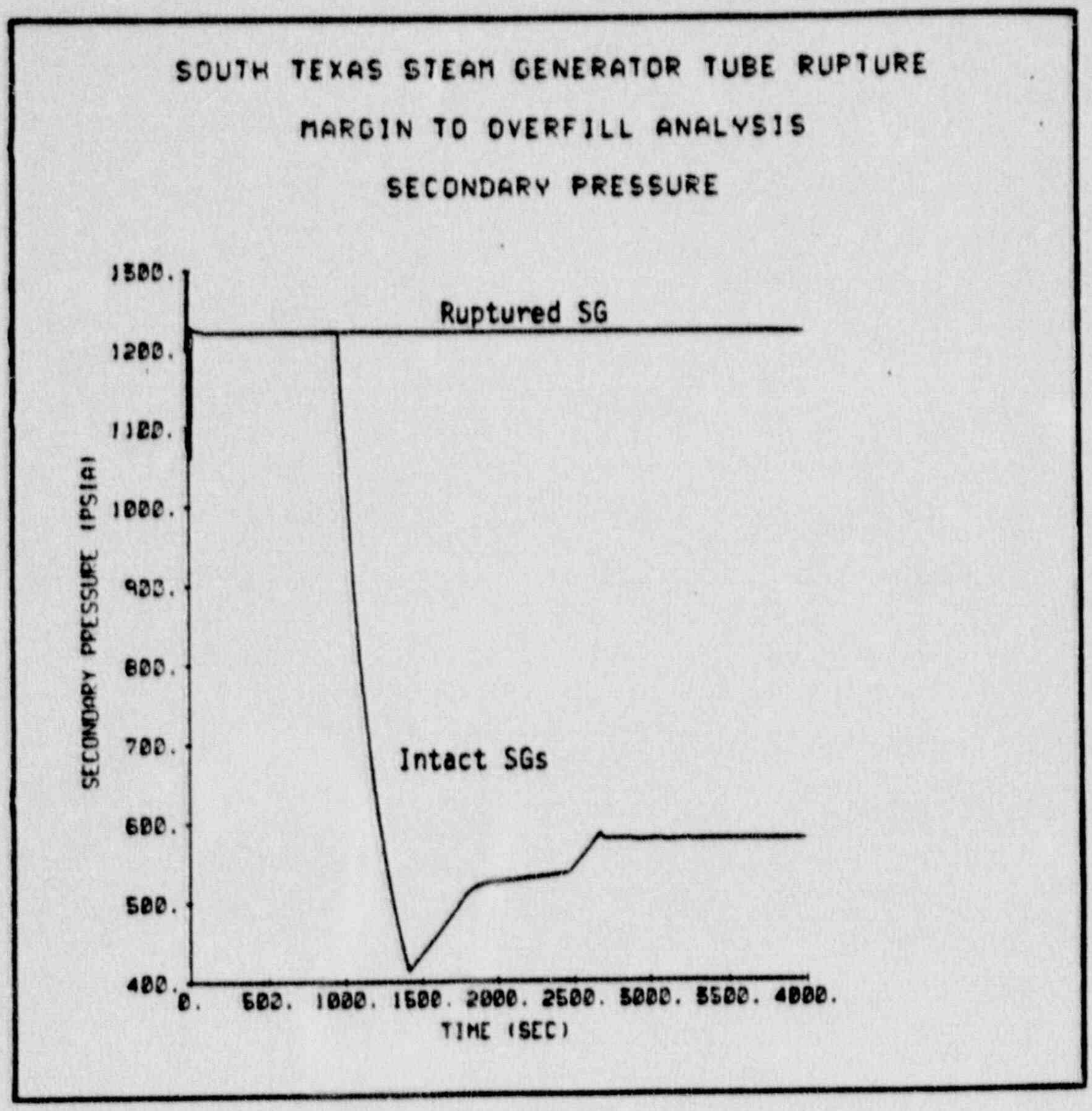

Figure II.3 Secondary Pressure - Margin to Overfill Analysis

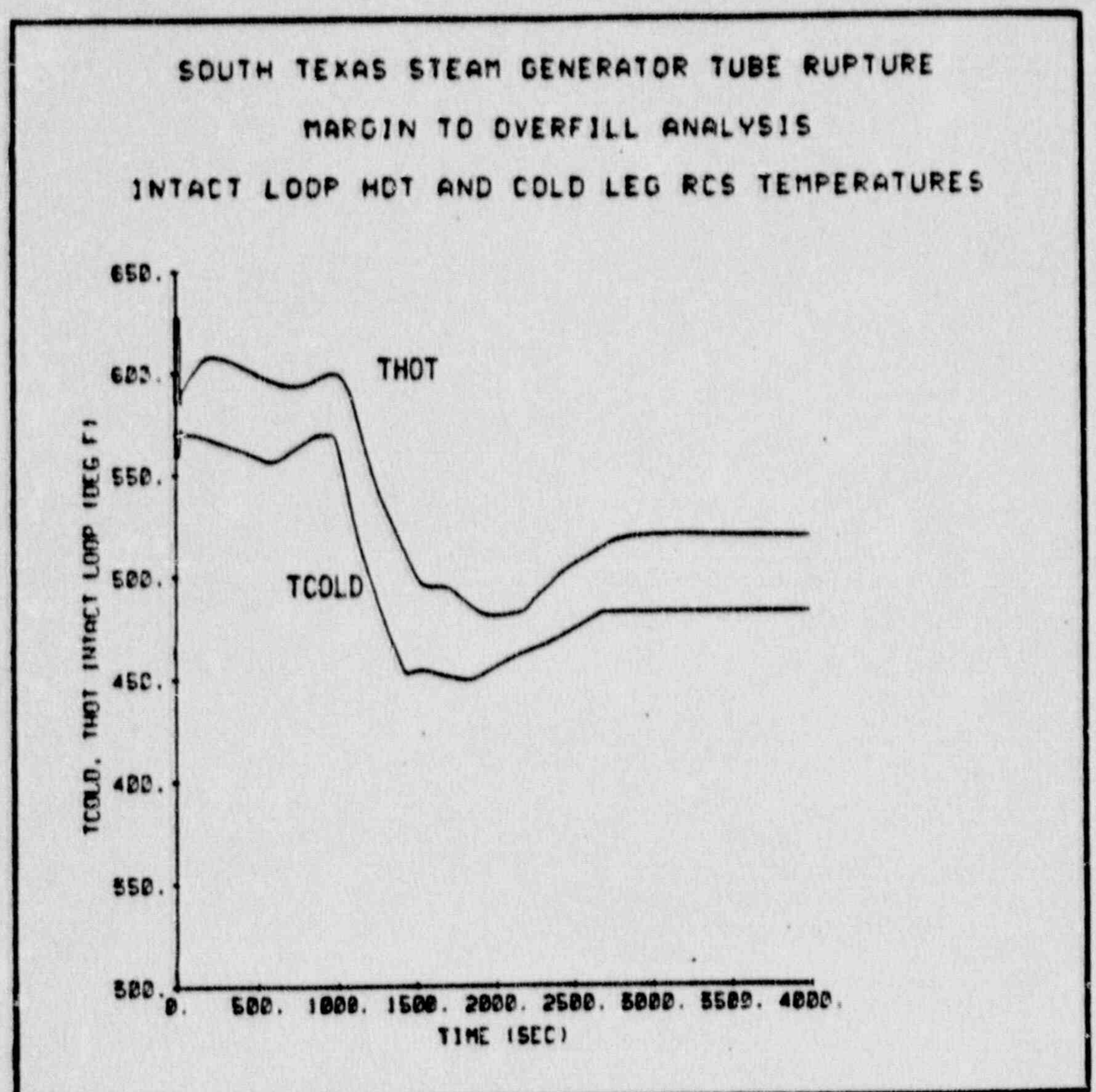

Figure II.4 Intact Loop Hot and Cold Leg RCS Temperatures - Margin to Overfill Analysis

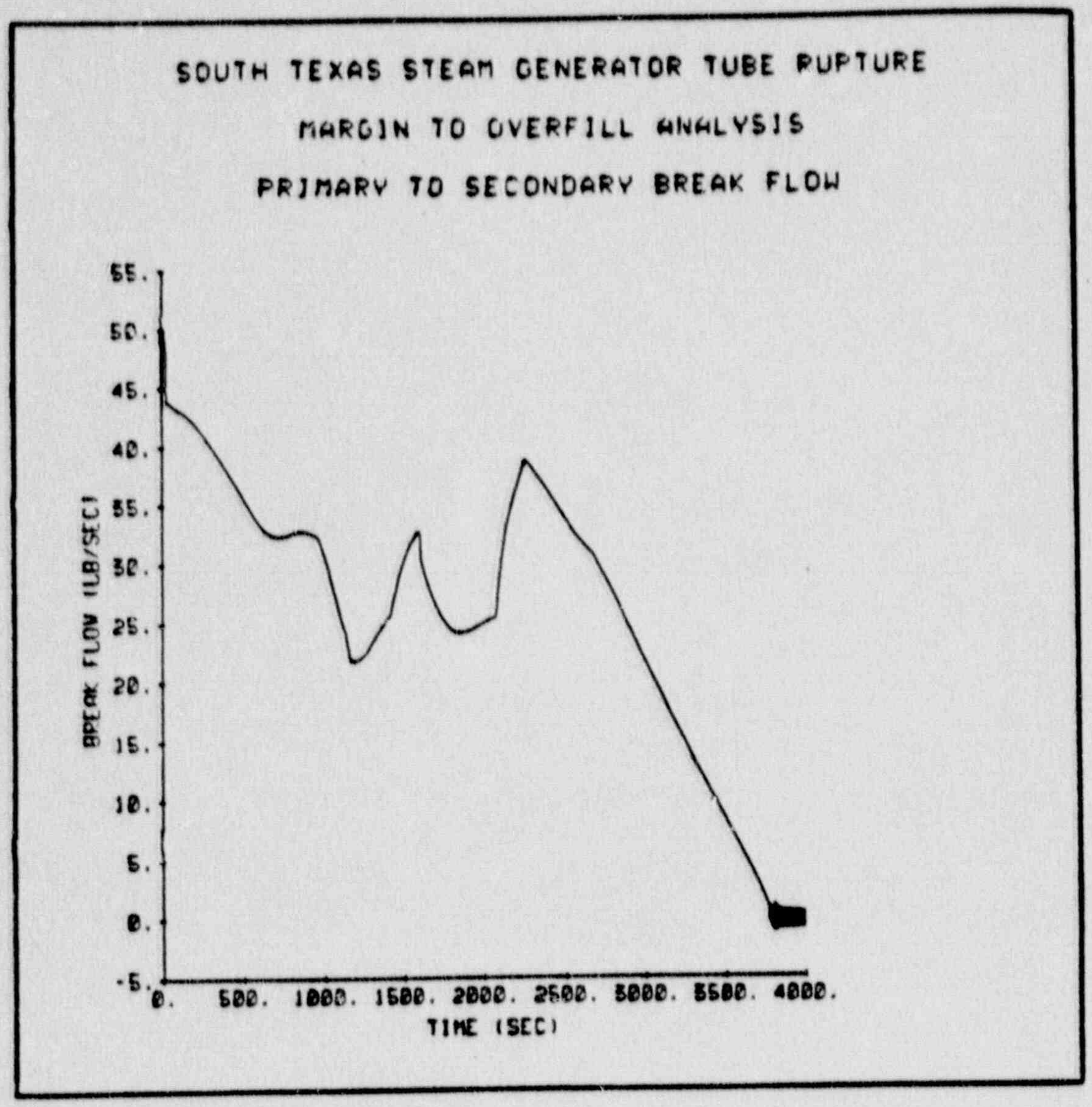

Figure II.5 Primary to Secondary Break Flow Rate - Margin to Overfill Analysis

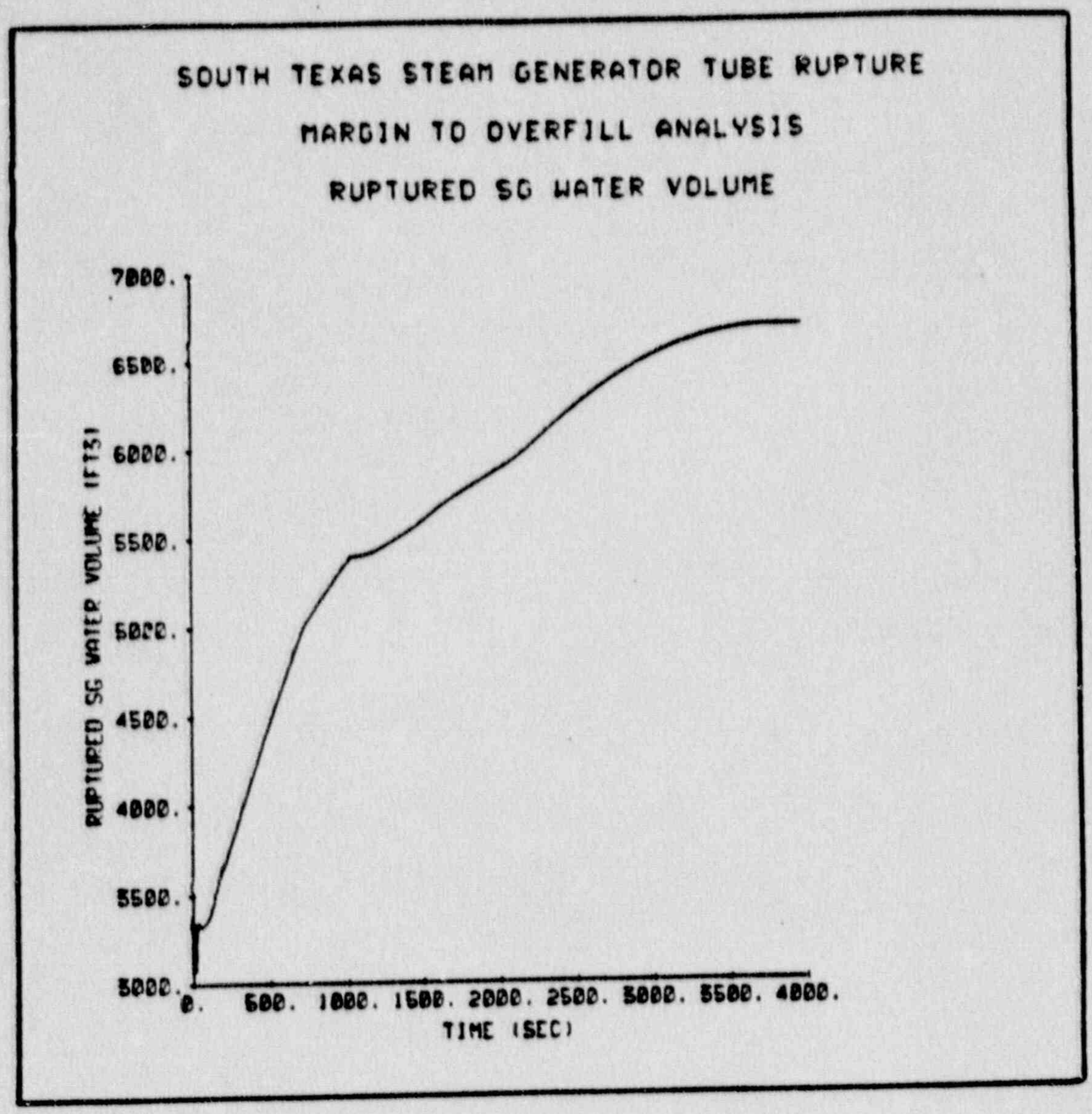

Figure II.6 Ruptured SG Water Volume - Margin to Overfill Analysis

III. ANALYSIS OF OFFSITE RADIOLOGICAL CONSEQUENCES

An analysis was also performed to determine the offsite radiological consequences for a design basis SGTR event for South Texas Units 1 and 2. The thermal and hydraulic and the offsite radiation dose analyses were performed using the methodology developed in References 1 and 2.

A. Thermal and Hydraulic Analysis

The plant response, the integrated primary to secondary break flow, and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere were calculated until break flow termination with the LOFTTR2 program for use in calculating the offsite radiation doses. This section provides a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass releases, the sequence of events during the recovery operations, and the calculated results.

1. Design Basis Accident

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet [ ]a,c The location of the break [ ]a,c However, as indicated subsequently, the break flow flashing fraction was conservatively calculated assuming that [ ]a,c In addition, the iodine scrubbing effectiveness of the steam generator water was calculated based on the conservative assumption that the rupture is located near the top of the tube bundle at the intersection of the outer tube row and the upper anti-vibration bar. The combination of these conservative assumptions regarding the break flow location results in a very conservative calculation of the offsite radiation doses. It was also assumed that loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

Based on the information in Reference 2, the most limiting single failure with respect to offsite doses is [ ]a,c Failure of [ ]a,c which will increase primary to secondary leakage and the mass release to the atmosphere. Pressure in the ruptured steam generator will remain below that in the primary system until [ ]a,c Thus, for the offsite dose analysis, it was assumed that the [ ]a,c

2. Conservative Assumptions

Most of the conservative conditions and assumptions used for the margin to overfill analysis are also conservative for the offsite dose analysis, and thus most of the same assumptions were used for both analyses. The major differences in the assumptions which were used for the LOFTTR2 analysis for offsite doses are discussed below.

a. Reactor Trip and Turbine Runback

An earlier reactor trip is conservative for the offsite dose analysis, similar to the case for the overfill analysis. Due to the assumed loss of offsite power, the condenser is not available for steam releases once the reactor is tripped. Consequently, after reactor trip, steam is released to the atmosphere through the steam generator PORVs (and safety valves if their setpoints are reached). Thus, an earlier trip time leads to more steam released to the atmosphere from the ruptured and intact steam generators. The time of the reactor trip was calculated by modeling the South Texas reactor protection system, and this time was also used for the offsite dose analysis. [

]a,c

b. Steam Generator Secondary Mass

If steam generator overfill does not occur, a [ ]a,c results in a conservative prediction of offsite doses. Thus, for the offsite dose analysis, the initial secondary mass was assumed correspond to operation [

]a,c

c. AFW System Operation

In Reference 2, it was determined that [ ]a,c results in an increase in the calculated offsite radiation doses for an SGTR, whereas it was previously concluded that [ ]a,c is conservative for the margin to overfill analysis. However, it was also demonstrated in Reference 2 that [

]a,c Since the single failure assumed for the offsite radiation dose analysis is [ ]a,c it is not necessary to assume an additional failure in the AFW system. Thus, each of the four AFW pumps were assumed to deliver flow to the associated steam generator, but a conservative minimum AFW flow of 500 gpm per pump was assumed for the offsite radiation dose analysis. In addition, the delay time assumed for initiation of AFW flow was [ ]a,c since this assumption results in a conservative calculation of the mass releases for the offsite radiation dose analysis.

d. Flashing Fraction

When calculating the amount of break flow that flashes to steam, [ ][a,c] Since the tube rupture flow actually consists of flow from the hot leg and cold leg sides of the steam generator, the temperature of the combined flow will be [ ][a,c] Thus the assumption that [ ][a,c] is conservative for the SGTR analysis.

3. Operator Action Times

The major operator actions required for the recovery from an SGTR are discussed in Section II.C and the operator action times used for the overfill analysis are presented in Table II.1. The operator action times in Table II.1 were also used for the offsite dose analysis. However, for the offsite dose analysis, the [ ][a,c] at the time the ruptured steam generator is isolated. It was assumed that the operators [ ][a,c] before proceeding with the subsequent recovery operations. The [ ][a,c] Houston Lighting and Power Company has determined that an operator can [ ][a,c] Thus, it was assumed that the [ ][a,c]

After the [ ][a,c] an additional delay time of 4 minutes (Table II.1) was assumed for the operator action time to initiate the RCS cooldown.

4. <u>Transient Description</u>

The LOFTTR2 analysis results for the offsite dose evaluation are described below. The sequence of events for the analysis of the offsite radiation doses is presented in Table III.1. The transient results for this case are similar to the transient results for the overfill analysis until the time when the ruptured steam generator is isolated. The transient behavior is different after this time since it is assumed that [ ]^{a,c} when the isolation is performed.

Following the tube rupture the RCS pressure decreases as shown in Figure III.1 due to the primary to secondary leakage. This depressurization results in reactor trip at 19 seconds on an overtemperature delta-T signal. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization continues. The steam dump system is inoperable due to the assumed loss of offsite power, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure III.2. Pressurizer level also continues to decrease following reactor trip as shown in Figure III.3. When the RCS temperature differential begins to increase at approximately 35 seconds (see Figures III.4 and III.5) as the reactor coolant pumps coast down and natural circulation flow develops, the RCS pressure and pressure level decrease less rapidly. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at 463 seconds. However, before the RCS pressure decreases to the shutoff head of the high head SI pumps, the pressurizer level goes offscale low. After the RCS pressure is below the shutoff head of the high head SI pumps, the SI flow rate maintains the reactor coolant inventory and the RCS pressure decrease is reversed.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

As indicated in Table II.1, it is assumed that the ruptured steam generator will be identified and isolated at 10 minutes after the initiation of the SGTR or when the narrow range level reaches 27.5%, whichever time is longer. Since the time to reach 27.5% narrow range level is slightly greater than 10 minutes, it was assumed that the actions to isolate the ruptured steam generator are performed at this time. The [ ][a,c] at this time. The failure causes the ruptured steam generator to rapidly depressurize, which results in an increase in primary to secondary leakage. The depressurization of the ruptured steam generator increases the break flow and energy transfer from primary to secondary which results in a decrease in the ruptured loop temperatures as shown in Figure III.4. The intact steam generator loop temperatures also decrease, as shown in Figure III.5, until the AFW flow is controlled to maintain the specified level in the intact steam generators. These effects result in a further decrease in the RCS pressure. However, when the RCS pressure decreases below the shutoff head of the high head SI pumps, the SI flow slows the rate of pressure decrease and subsequently causes the RCS pressure to increase again. It is assumed that the time required for the operator to identify that the [ ][a,c] is 15 minutes. Thus, the isolation of the ruptured steam generator is completed at 1586 seconds and the depressurization of ruptured steam generator is terminated. At this time, the ruptured steam generator pressure begins to increase to the PORV setpoint and the primary to secondary break flow begins to decrease. Because the SI flow rate exceeds the break flow rate, the rate of RCS repressurization increases.

2. Cool Down the RCS to Establish Subcooling Margin

After the [ ]^{a,c} a 4 minute operator action time is imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact steam generator PORVs. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance of 35°F for instrument uncertainty. Because of the lower pressure in the ruptured steam generator, the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. The cooldown is initiated at 1826 seconds and is completed at 2842 seconds.

The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure III.2, and the effect of the cooldown on the RCS temperature is shown in Figure III.5. The RCS pressure also decreases during this cooldown process due to shrinkage of the reactor coolant as shown in Figure III.1.

3. Depressurize to Restore Inventory

After the RCS cooldown, it is assumed that normal charging flow from two centrifugal charging pumps is initiated. A 3 minute operator action time is then included prior to the RCS depressurization. The RCS is depressurized to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS is depressurized by using auxiliary pressurizer spray. The normal charging flow path is isolated in order to utilize the auxiliary spray flow path to the pressurizer. The RCS depressurization is initiated at 3024 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 8% for pressurizer level uncertainty, or pressurizer level is greater than 70%, or RCS subcooling is less than the 35°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated due to high pressurizer level. The RCS depressurization reduces the break flow as shown in Figure III.7, and increases SI flow to refill the pressurizer as shown in Figure III.3. After completion of the RCS depressurization, the charging flow from two centrifugal charging pumps was reinitiated.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 35°F allowance for uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is increasing, and the pressurizer level is greater than the 8% allowance for uncertainty.

After depressurization is completed, an operator action time of 2 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination is performed at this time. The charging flow from two centrifugal charging pumps was continued from the end of RCS depressurization until 1 minute after SI termination, at which time it was assumed that excess charging flow is eliminated. After SI termination and the elimination of excess charging flow, the RCS pressure decreases as shown in Figure III.1. The differential pressure between the RCS and the ruptured steam generator is shown in Figure III.6. Figure III.7 shows that the primary to secondary leakage continues after the SI flow and excess charging flow are stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in Figure III.8. For this case, the water volume in the ruptured steam generator is substantially less than the total steam generator volume of 7983 ft^3 when the break flow is terminated. The mass of water in the ruptured steam generator is also shown as a function of time in Figure III.9.

TABLE III.1
STP SGTR ANALYSIS
SEQUENCE OF EVENTS
OFFSITE RADIATION DOSE ANALYSIS

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	19
SI Actuation	463
Ruptured SG Isolated	680
[ ] [a,c]	684
[ ] [a,c]	1586
RCS Cooldown Initiated	1826
RCS Cooldown Terminated	2842
Two Charging Pumps Started	2842
Charging Flow to RCS Isolated	3024
RCS Depressurization Initiated	3024
RCS Depressurization Terminated	3424
Two Charging Pumps Started	3426
SI Terminated	3546
Excess Charging Flow Eliminated	3609
Break Flow Terminated	4854

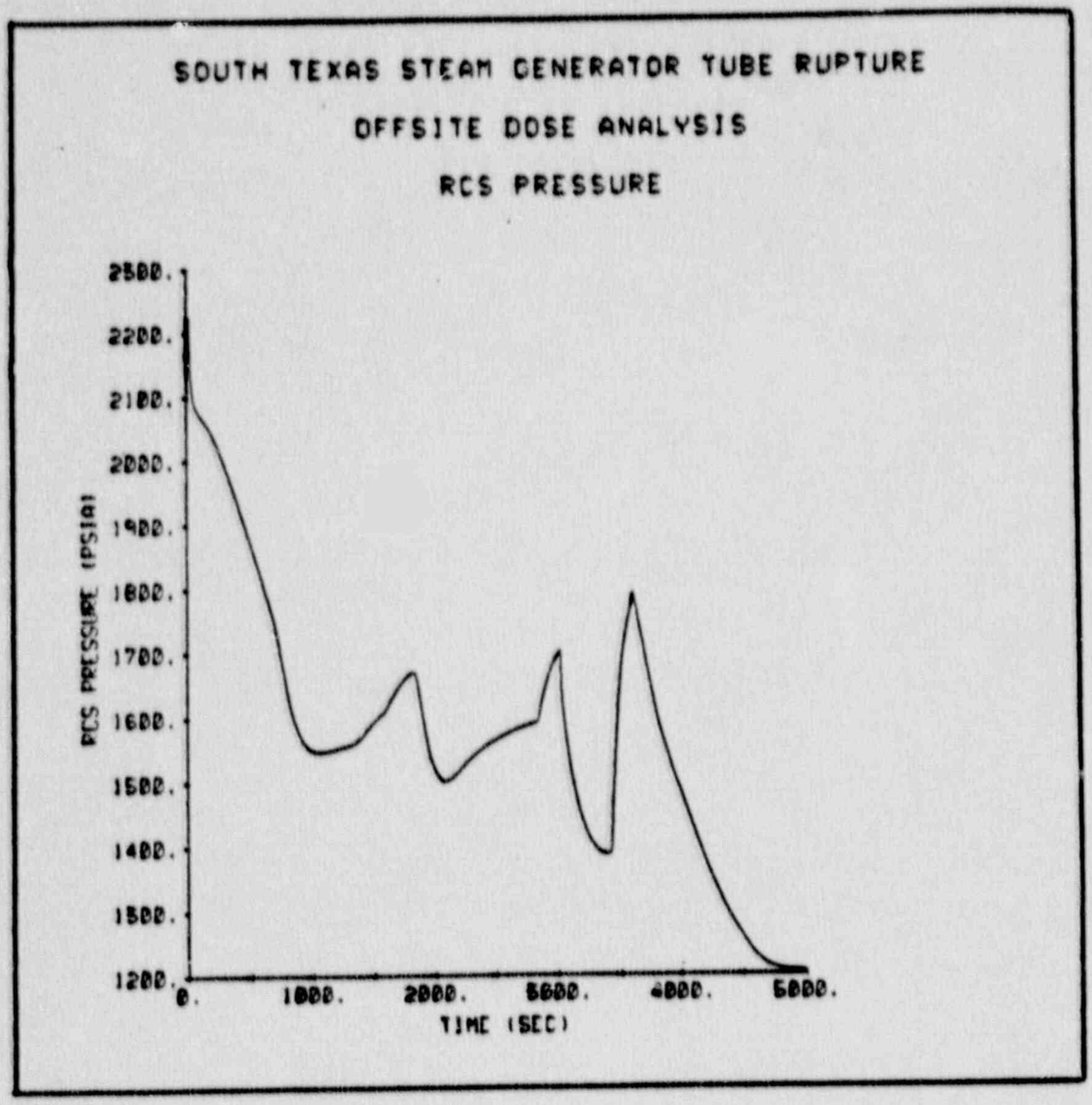

Figure III.1 RCS Pressure - Offsite Radiation Dose Analysis

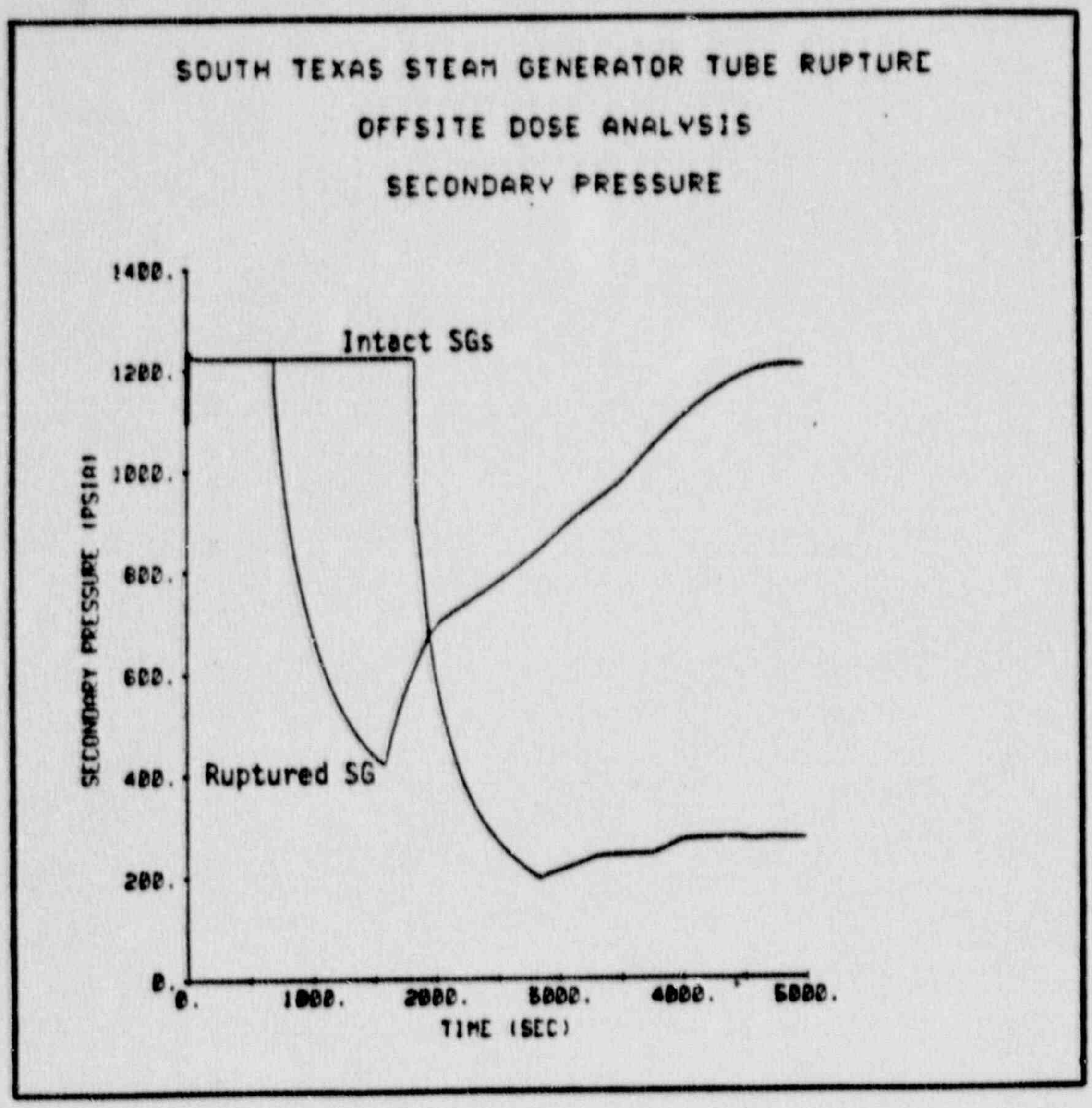

Figure III.2 Secondary Pressure - Offsite Radiation Dose Analysis

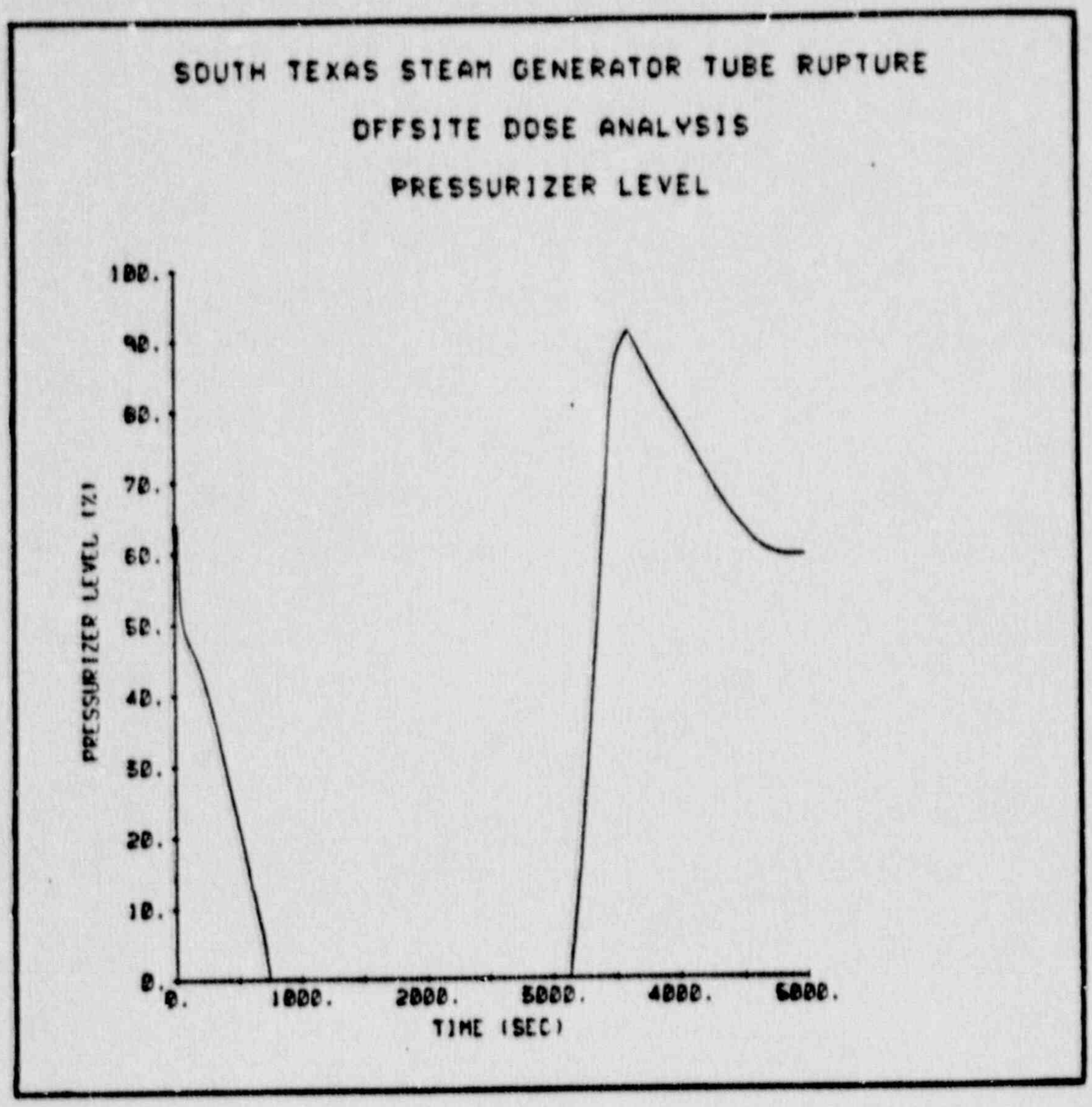

Figure III.3 Pressurizer Level - Offsite Radiation Dose Analysis

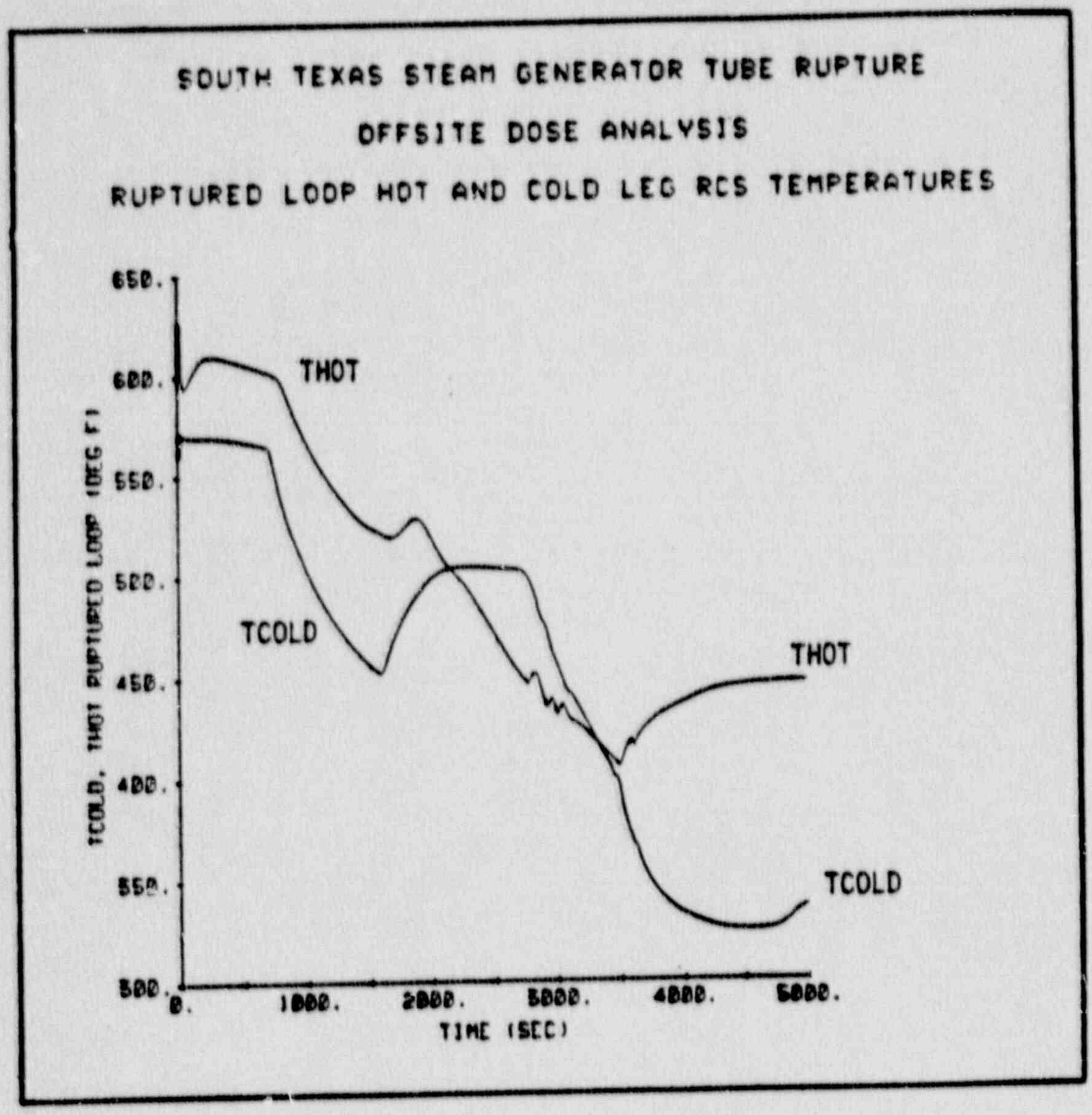

Figure III.4 Ruptured Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

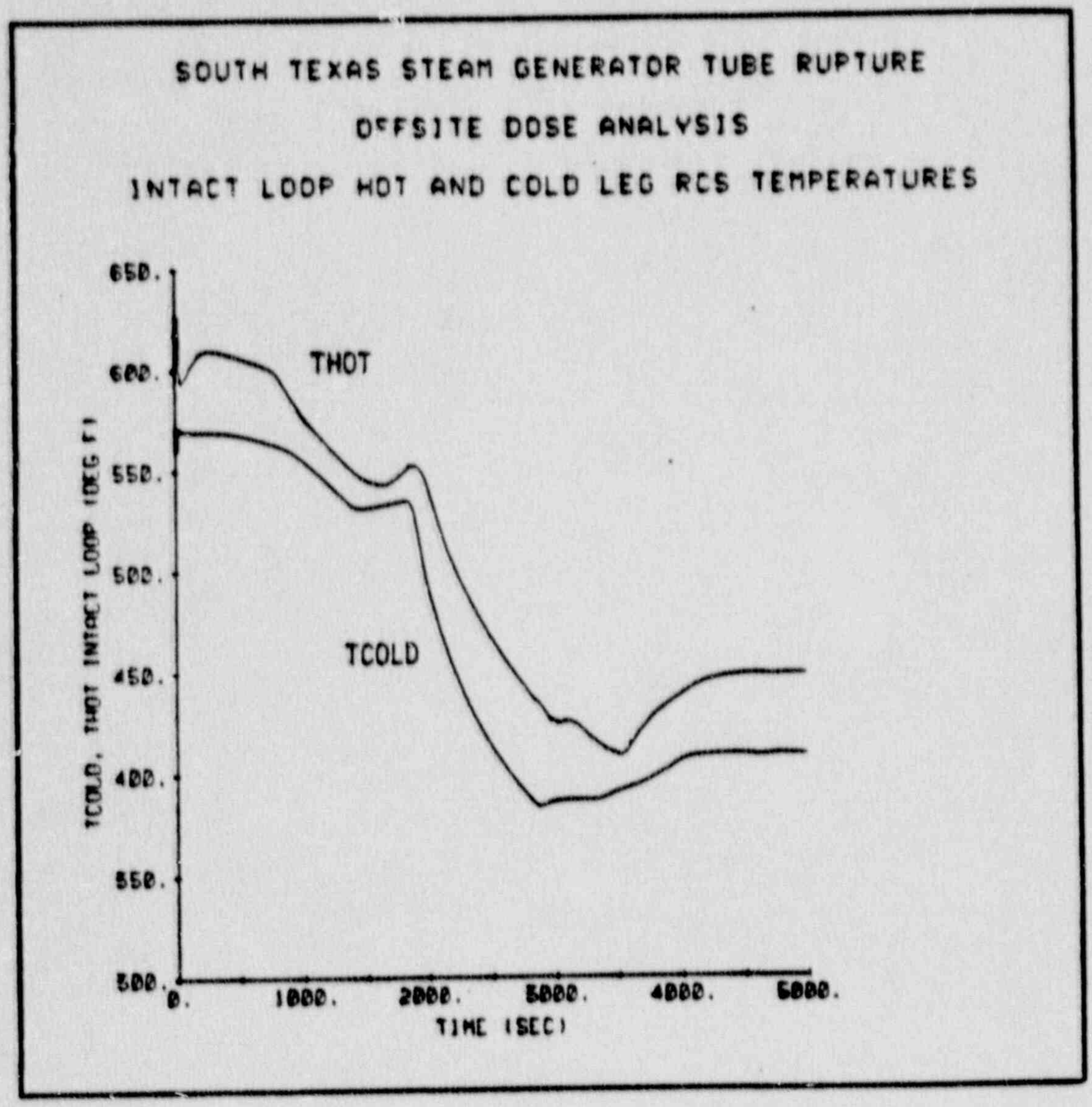

Figure III.5 Intact Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

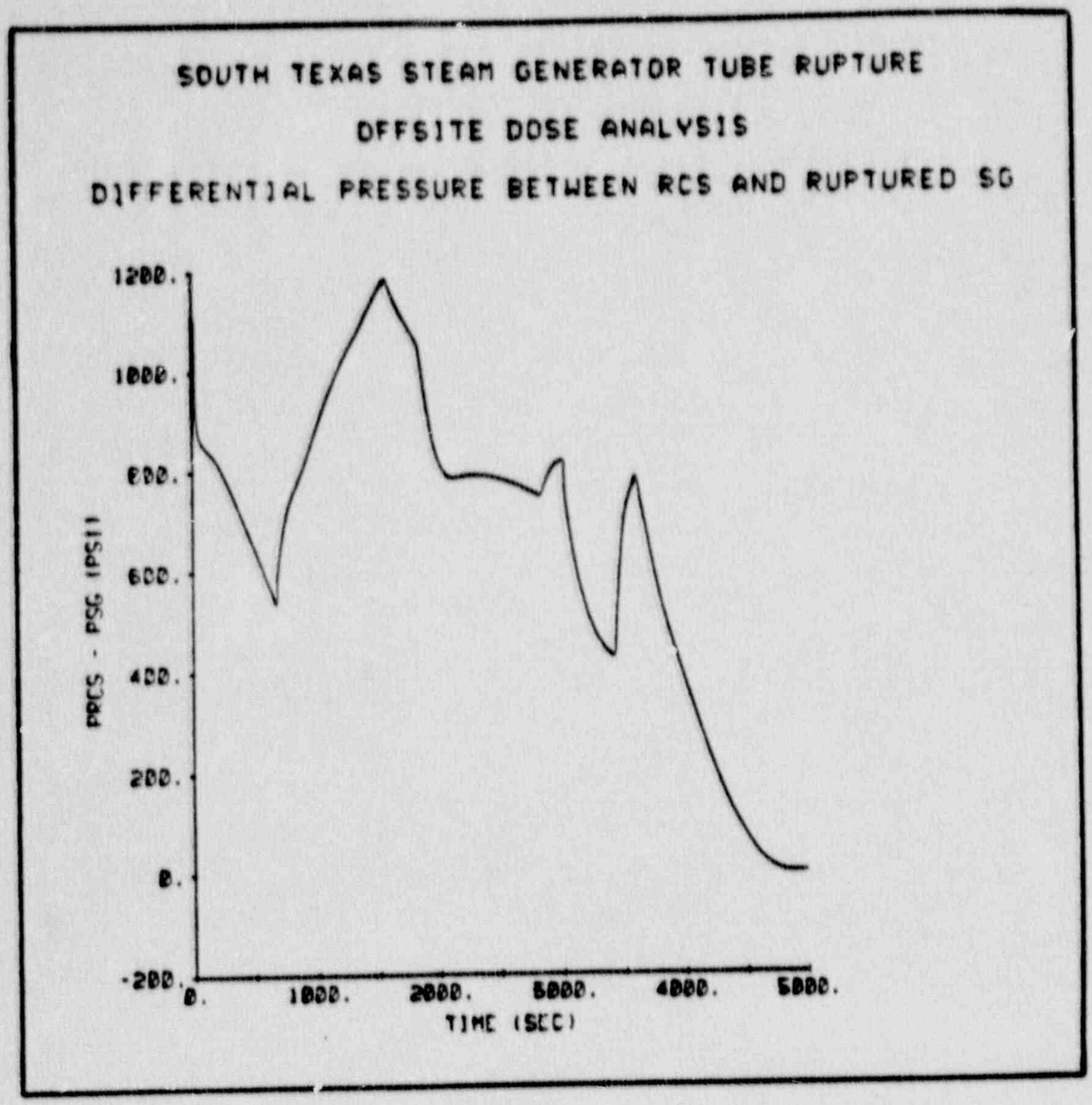

Figure III.6 Differential Pressure Between RCS and Ruptured SG - Offsite Radiation Dose Analysis

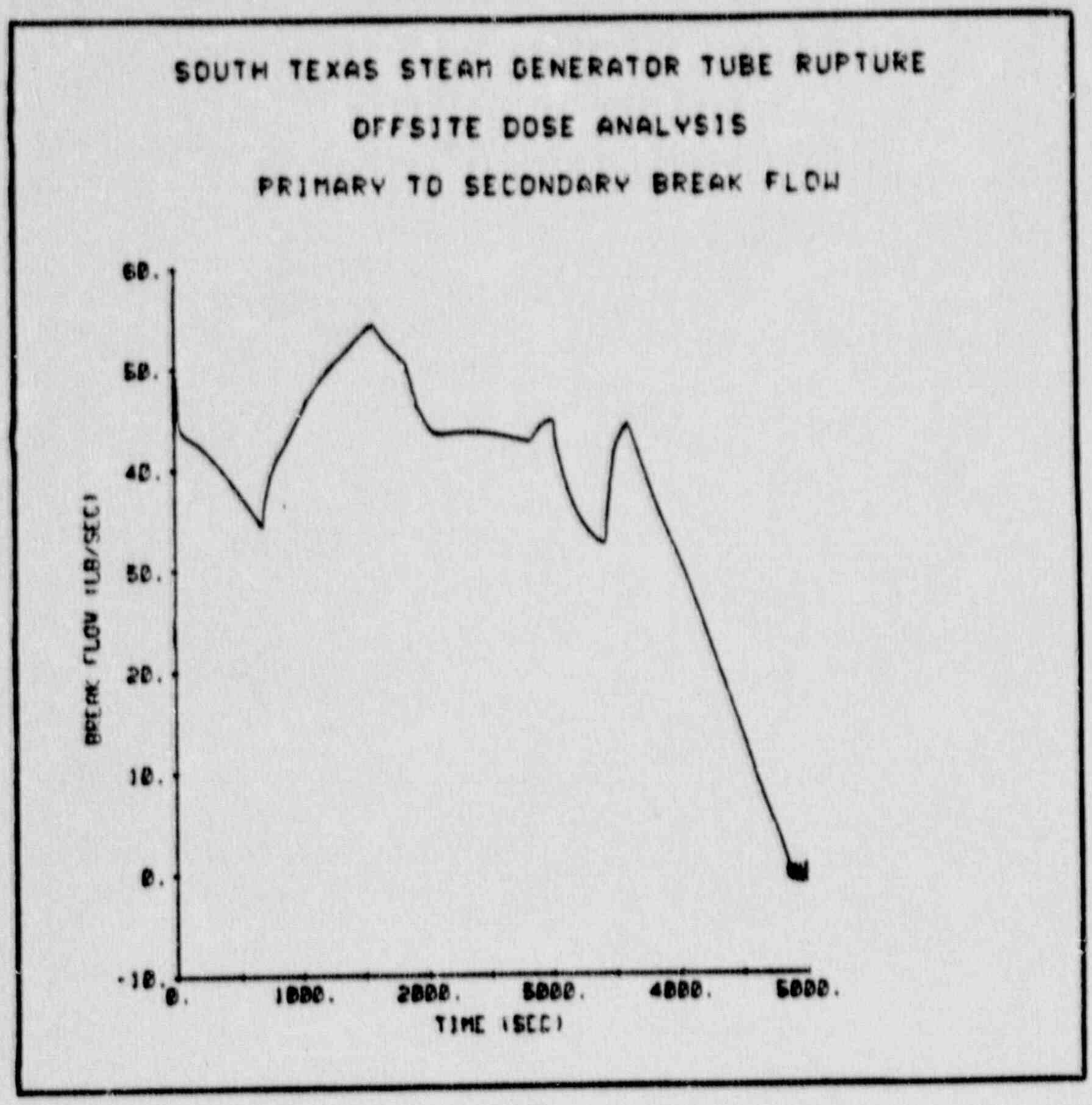

Figure III.7 Primary to Secondary Break Flow Rate - Offsite Radiation Dose Analysis

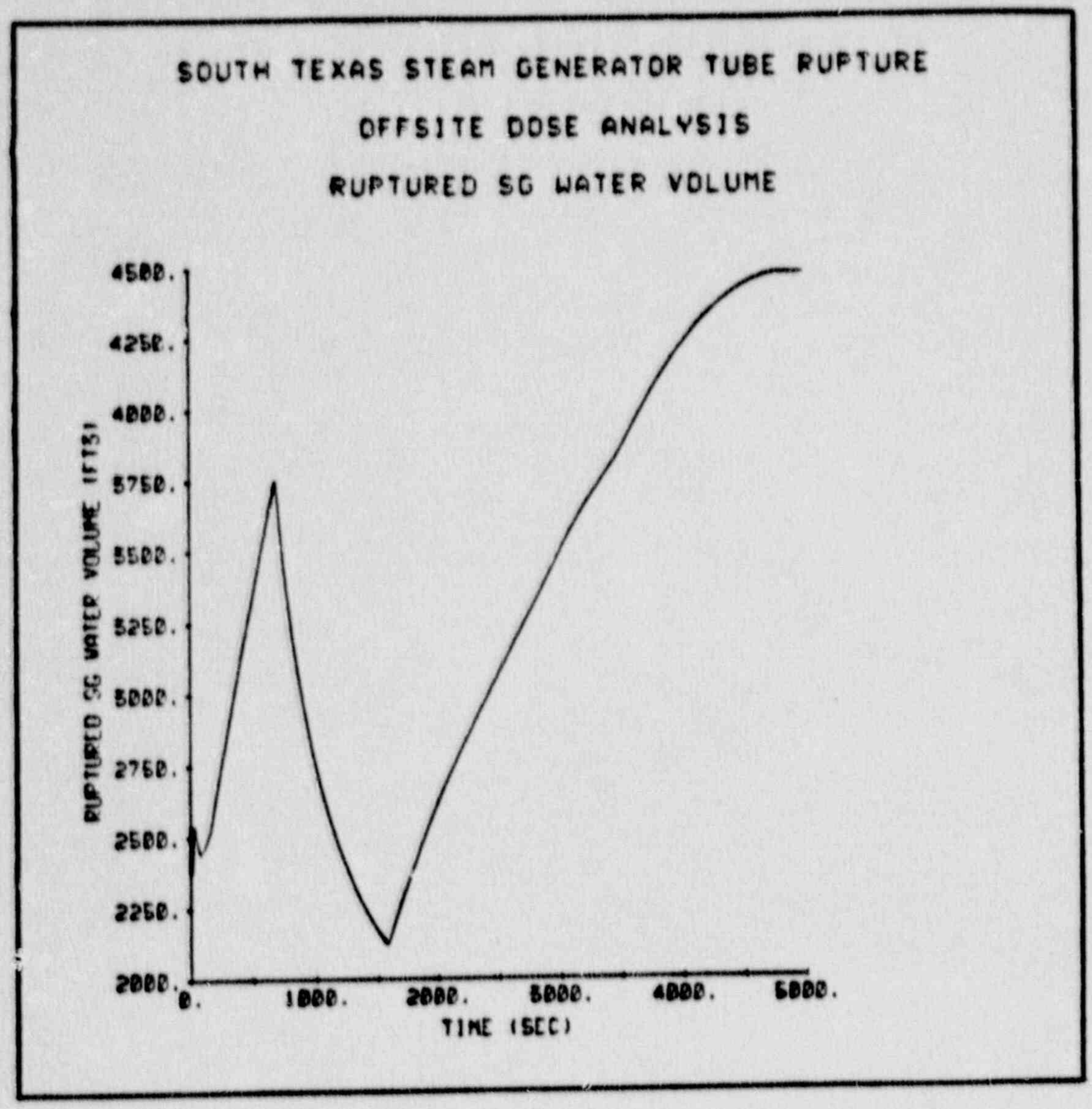

Figure III.8 Ruptured SG Water Volume - Offsite Radiation Dose Analysis

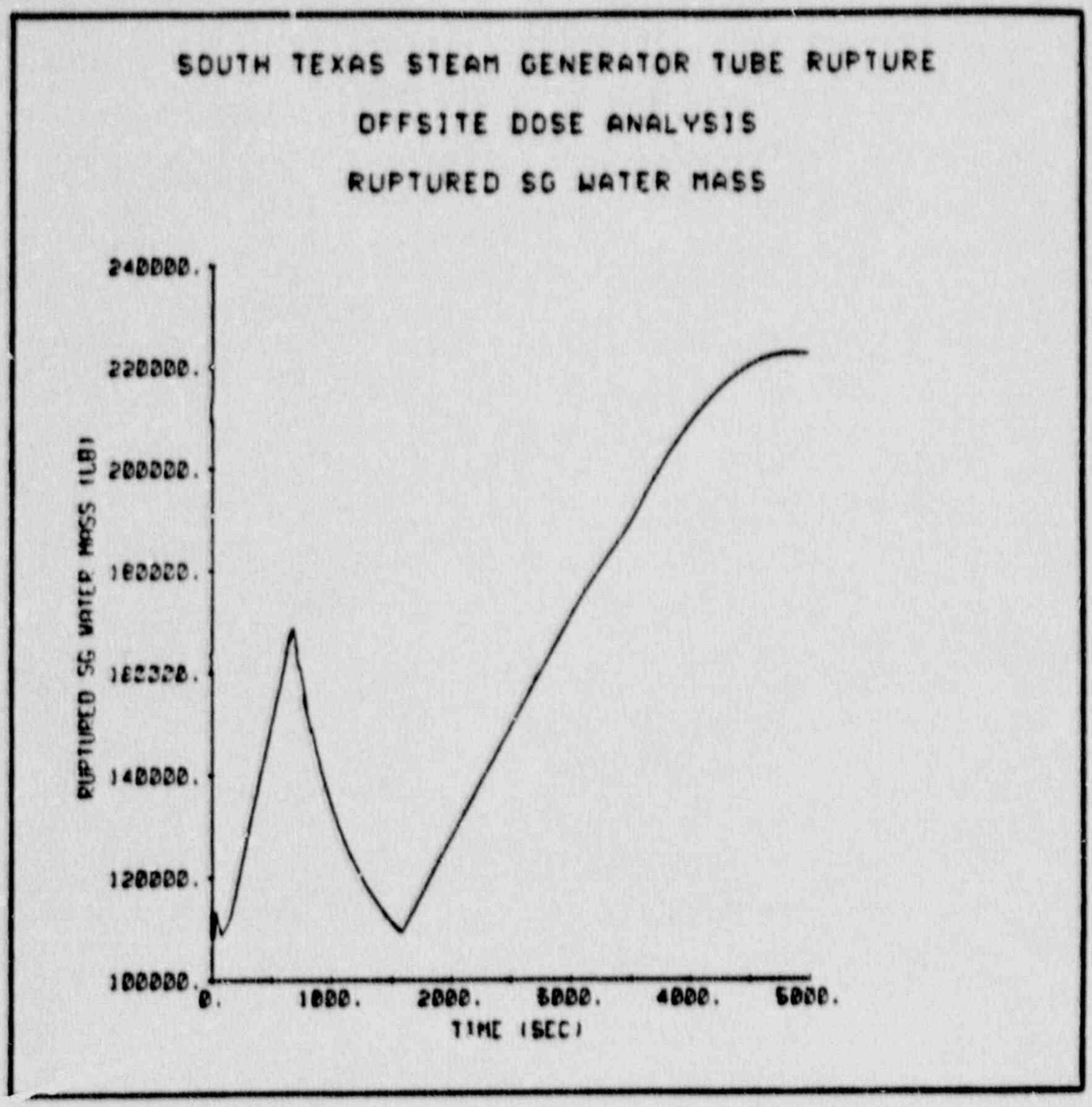

Figure III.9 Ruptured SG Water Mass - Offsite Radiation Dose Analysis

5. Mass Releases

The mass releases were determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and primary to secondary break flow into the ruptured steam generator were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident. The releases for 0-2 hours are used to calculate the radiation doses at the exclusion area boundary for a 2 hour exposure, and the releases for 0-8 hours are used to calculate the radiation doses at the low population zone for the duration of the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in South Texas EOP POPO5-EO-EO30 (E-3) were simulated until the termination of primary to secondary leakage. After the primary to secondary leakage is terminated, the operators will continue the SGTR recovery actions to prepare the plant for cooldown to cold shutdown conditions. When these recovery actions are completed, the plant should be cooled and depressurized to cold shutdown conditions. [ ]a,c it was assumed that the cooldown is performed using South Texas EOP POPO5-EO-ES33 (ES-3.3), POST-SGTR COOLDOWN USING STEAM DUMP, since this method results in a conservative evaluation of the long term mass releases for the offsite dose analysis.

The high level actions for the the post-SGTR cooldown method using steam dump in South Texas EOP POPO5-EO-ES33 (ES-3.3) are discussed below.

1. Prepare for Cooldown to Cold Shutdown

   The initial steps to prepare for cooldown to cold shutdown will be continued if they have not already been completed. A few additional steps are also performed prior to initiating cooldown.

These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and to provide for better pressure control, and assuring adequate shutdown margin in the event of potential boron dilution due to in-leakage from the ruptured steam generator.

2. Cool Down RCS to Residual Heat Removal (RHR) System Temperature

The RCS is cooled by steaming and feeding the intact steam generators similar to a normal cooldown. Since all immediate safety concerns have been resolved, the cooldown rate should be maintained less than the maximum allowable rate of 100°F/hr. The preferred means for cooling the RCS is steam dump to the condenser since this minimizes the radiological releases and conserves feedwater supply. The PORVs for the intact steam generators can also be used if steam dump to the condenser is unavailable. Since a loss of offsite power is assumed for the analysis, it was assumed that the cooldown is performed using steam dump to the atmosphere via the intact steam generator PORVs. When the RHR system operating temperature is reached, the cooldown is stopped until RCS pressure can also be decreased. This ensures that the pressure/temperature limits will not be exceeded.

3. Depressurize RCS to RHR System Pressure

When the cooldown to RHR system temperature is completed, the pressure in the ruptured steam generator is decreased by releasing steam from the ruptured steam generator. Steam release to the condenser is preferred since this minimizes radiological releases, but steam can be released to the atmosphere using the PORV on the ruptured steam generator if the condenser is not available. Consistent with the assumption of a loss of offsite power, it was assumed that the ruptured steam generator is depressurized by releasing steam via the PORV. As the ruptured steam generator pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured steam generator in order to prevent in-leakage of secondary side water or additional primary to secondary leakage. Although normal pressurizer spray is the preferred means of RCS pressure control, auxiliary spray or a pressurizer PORV can be used to control RCS pressure if pressurizer spray is not available.

4. Cool Down to Cold Shutdown

When RCS temperature and pressure have been reduced to the RHR system in-service values, RHR system cooling is initiated to complete the cooldown to cold shutdown. When cold shutdown conditions are achieved, the pressurizer can be cooled to terminate the event.

The methodology in Reference 2 was used to calculate the mass releases for the South Texas analysis. The methodology and the results of the calculations are discussed below.

a. Methodology for Calculation of Mass Releases

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage are simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact steam generators, the feedwater flows to the ruptured and intact steam generators, and the primary to secondary leakage into the ruptured steam generator were determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following the termination of leakage, it was assumed that the RCS and intact steam generator conditions are maintained stable for a [ ][a,c] until the cooldown is initiated. The PORVs for the intact steam generators were then assumed to be used to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact steam generator temperatures at 2 hours were then determined [

]a,c The steam releases and the feedwater flows for the intact steam generator for the period from leakage termination until 2 hours were determined from [

]a,c Since the ruptured steam generator is isolated, no change in the ruptured steam generator conditions is assumed to occur until subsequent depressurization.

The RCS cooldown was assumed to be continued after 2 hours until the RHR system in-service temperature of 350°F is reached. Depressurization of the ruptured steam generator was then assumed to be performed immediately following the completion of the RCS cooldown. The ruptured steam generator was assumed to be depressurized to the RHR in-service pressure of 350 psia via steam release from the ruptured steam generator PORV, since this maximizes the steam release from the ruptured steam generator to the atmosphere which is conservative for the evaluation of the offsite radiation doses. The RCS pressure is also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It is assumed that the continuation of the RCS cooldown and depressurization to RHR operating conditions are completed within 8 hours after the accident since there is ample time to complete the operations during this time period. The steam releases and feedwater flows from 2 to 8 hours were determined for the intact steam generator from [

]a,c The steam released from the ruptured steam generator from 2 to 8 hours was determined based on [

]a,c

After 8 hours, it is assumed that further plant cooldown to cold shutdown as well as long-term cooling is provided by the RHR system. Therefore, the steam releases to the atmosphere are terminated after RHR in-service conditions are assumed to be reached at 8 hours.

b. Mass Release Results

The mass release calculations were performed using the methodology discussed above. For the time period from initiation of the accident until leakage termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser is in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip will be through the condenser vacuum pump exhaust. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator PORVs. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in Figures III.10 and III.11 for the ruptured and intact steam generators, respectively, for the time period until leakage termination.

The mass releases calculated from the time of leakage termination until 2 hours and from 2-8 hours are also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the SGTR event for each of the time intervals considered are presented in Table III.2. The mass releases prior to break flow termination, from break flow termination until 2 hours, and from 2 to 8 hours are summarized in Table III.3. The results indicate that approximately 129,300 lbm of steam are released from the ruptured steam generator to the atmosphere in the first 2 hours. A total of 186,000 lbm of primary water is transferred to the secondary side of the ruptured steam generator before the break flow is terminated.

TABLE III.2

STP SGTR ANALYSIS

MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

TIME PERIOD

	O-TRIP	TRIP - TMSEP	TMSEP - TTBRK	TTBRK - T2HRS	T2HRS - TRHR
Ruptured SG					
- Condenser	23,000	0	0	0	0
- Atmosphere	0	20,400	108,900	0	41,700
- Feedwater	22,000	49,200	3,300	0	0
Intact SGs					
- Condenser	68,400	0	0	0	0
- Atmosphere	0	57,300	285,800	228,900	1,051,100
- Feedwater	68,400	149,700	483,600	243,500	1,063,400
Break Flow	900	24,800	160,300	0	0

TRIP = Time of reactor trip = 19 sec.

TMSEP = Time when water reaches the moisture separators = 631 sec.

TTBRK = Time when break flow is terminated = 4854 sec.

T2HRS = Time at 2 hours = 7200 sec.

TRHR = Time to reach RHR in-service conditions, 8 hours = 28,800 sec.

TABLE III.3

STP SGTR ANALYSIS

SUMMARIZED MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

	0 - TTBRK	TTBRK - 2HRS	2HRS - 8HRS
Ruptured SG			
- Condenser	23,000	0	0
- Atmosphere	129,300	0	41,700
- Feedwater	74,500	0	0
Intact SGs			
- Condenser	68,400	0	0
- Atmosphere	343,100	228,900	1,051,100
- Feedwater	701,700	243,500	1,063,400
Break Flow	186,000	0	0

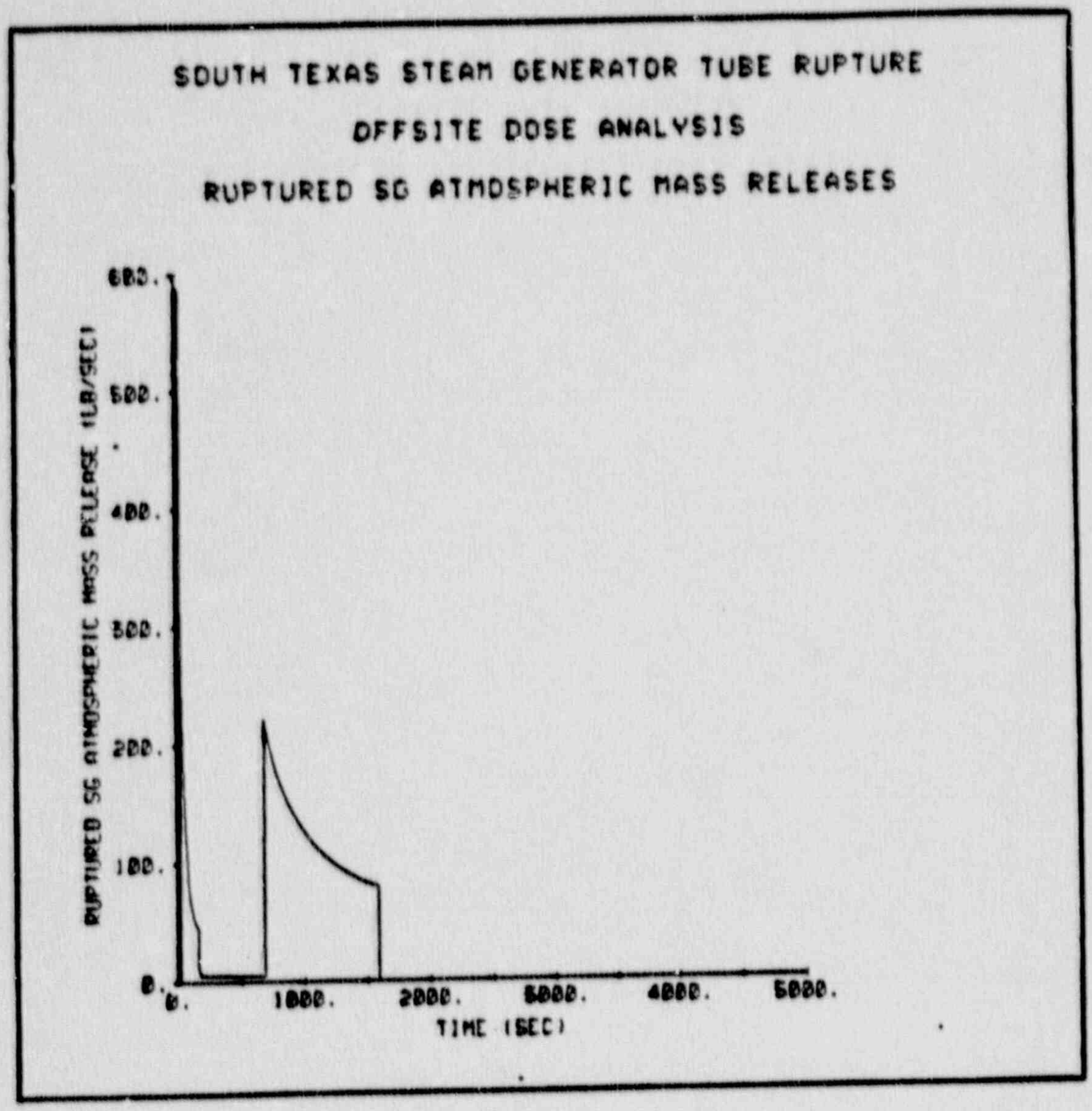

Figure III.10 Ruptured SG Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

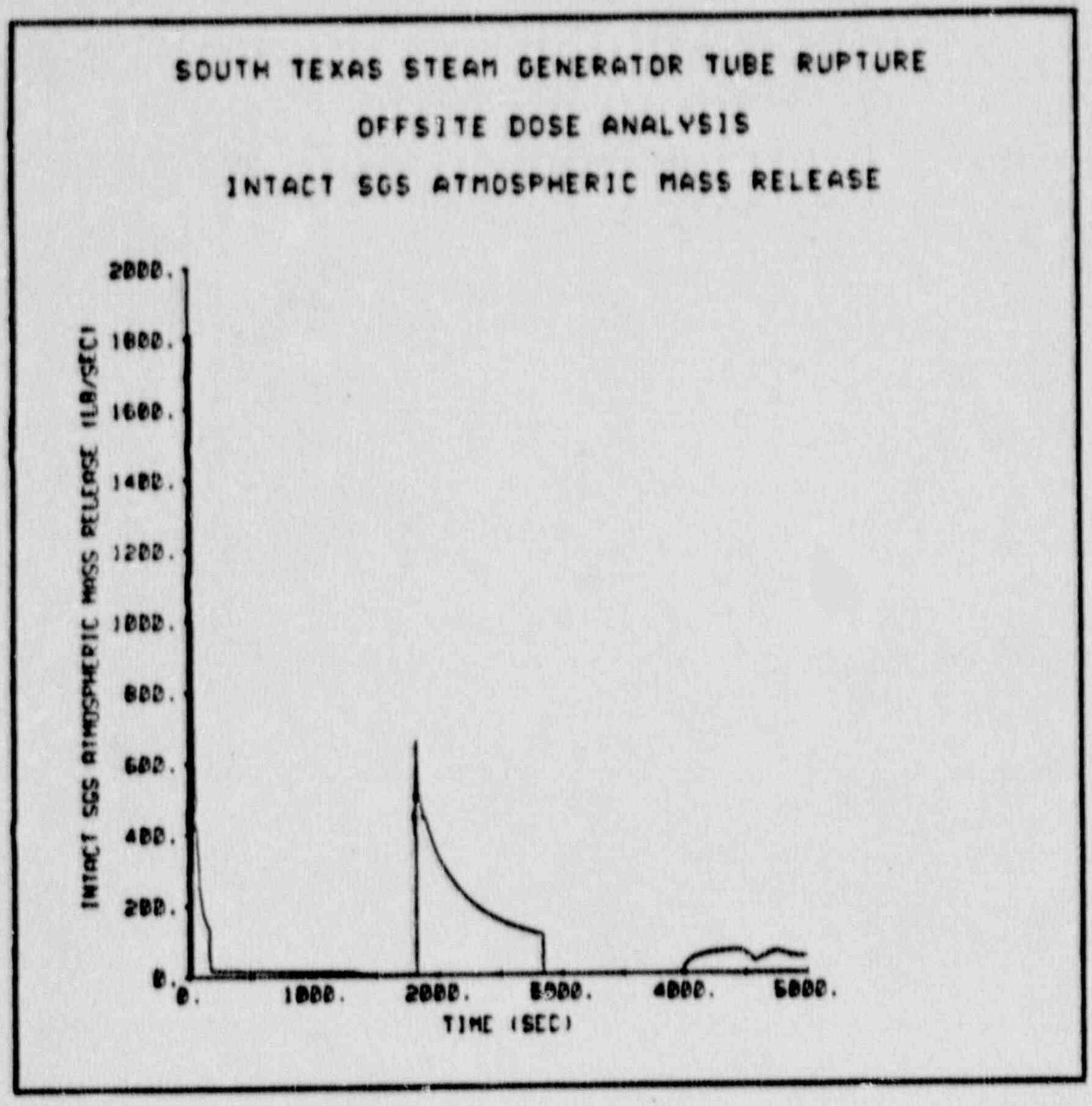

Figure III.11 Intact SGs Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

B. Offsite Radiation Dose Analysis

The evaluation of the radiological consequences of a steam generator tube rupture event assumes that the reactor has been operating at the maximum allowable Technical Specification limit for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator, via the ruptured tube, and are released to the atmosphere through the steam generator PORVs and safety valves and via the condenser vacuum pump exhaust.

The quantity of radioactivity released to the environment, due to an SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated in a manner consistent with the recommendations of Standard Review Plan Section 15.6.3.

1. Design Basis Analytical Assumptions

The major assumptions and parameters used in the analysis are itemized in Table III.4.

2. Source Term Calculations

The radionuclide concentrations in the primary and secondary system, prior to and following the SGTR are determined as follows:

a. The iodine concentrations in the reactor coolant will be based upon preaccident and accident initiated iodine spikes.

i. Accident Initiated Spike - The initial primary coolant iodine concentration is 1 μCi/gm of Dose Equivalent (D.E.) I-131. Following the primary system depressurization associated with the SGTR, an iodine spike is initiated in the primary system which increases the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to the initial primary system iodine concentration. The initial appearance rate can be written as follows:

$$P_i = A_i \lambda_i$$

where:

P_i = equilibrium appearance rate for iodine nuclide i

A_i = equilibrium RCS inventory of iodine nuclide i corresponding to 1 μCi/gm of D.E. I-131

λ_i = removal coefficient for iodine nuclide i

The duration of the spike, [ ]^{a,c} is sufficient to increase the initial RCS I-131 inventory by a factor of [ ]^{a,c}

ii. Preaccident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration from 1 to 60 μCi/gram of D.E. I-131.

b. The initial secondary coolant iodine concentration is 0.1 μCi/gram of D.E. I-131.

c. The chemical form of iodine in the primary and secondary coolant is assumed to be elemental.

d. The initial noble gas concentrations in the reactor coolant are based upon 1% fuel defects. These concentrations were taken from Table 15.A-2 of the South Texas FSAR.

3. <u>Dose Calculations</u>

The iodine transport model utilized in this analysis was proposed by Postma and Tam (Reference 4). The model considers break flow flashing, droplet size, bubble scrubbing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. Removal credit is taken for scrubbing of iodine contained in the atomized coolant droplets when the rupture site is below the secondary water level. The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100 between the steam generator liquid and steam phases. The model takes no scrubbing credit when the rupture site is above the secondary water level. Droplet removal by the dryers is assumed to be negligible. The iodine transport model is illustrated in Figure III.12.

The following assumptions and parameters were used to calculate the activity released to the atmosphere and the offsite doses following a SGTR.

a. The mass of reactor coolant discharged into the secondary system through the rupture and the mass of steam released from the ruptured and intact steam generators to the atmosphere are presented in Table III.2.

b. The time dependent fraction of rupture flow that flashes to steam and is immediately released to the environment is presented in Figure III.13.

c. In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface conservatively assumes that the rupture is located at the insersection of the outer tube row and the upper anti-vibration bar. However, in accordance with the methodology in Reference 2, the tube rupture break flow was conservatively calculated assuming that the break is at the top of the tube sheet. The collapsed water level relative to the top of the tubes in the ruptured and intact steam generators is shown in Figure III.14. The iodine scrubbing efficiency is determined by the method suggested by Postma and Tam (Ref. 4). However, since the collapsed water level in the ruptured steam generator is below the rupture site for most of the time when the rupture flow is flashing, the effect of iodine scrubbing is very small and has been conservatively neglected for this analysis.

The activity released to the environment by the flashed rupture flow can be written as follows:

$$A_r = \sum_j IA_j \, (1 - eff_j)$$

where:

A_r = total iodine released to the environment by flashed primary coolant

IA_j = (integrated activity in rupture flow during time interval j) (flashing fraction for time interval j)

eff_j = iodine scrubbing efficiency during time interval j

d. The total primary to secondary leak rate is assumed to be 1.0 gpm as allowed by the Technical Specifications. The leak rate is assumed to be 0.70 gpm for the three intact steam generators and 0.3 gpm for the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident.

e. The iodine partition coefficient between the liquid and steam of the ruptured and intact steam generators is assumed to be 100.

f. 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.

g. Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table III.8. The breathing rates were obtained from NRC Regulatory Guide 1.4, (Ref. 5).

4. Offsite Dose Calculation

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} \quad (BR)_j \quad (x/Q)_j \right) \right]$$

where

$(IAR)_{ij}$ = integrated activity of iodine nuclide i released during the time interval j in Ci*

$(BR)_j$ = breathing rate during time interval j in meter^3/second (Table III.8)

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/meter^3 (Table III.8)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for iodine nuclide i in rem/Ci (Table III.9)

D_{Th} = thyroid dose via inhalation in rem

Offsite whole-body gamma doses are calculated using the equation:

$$D_\gamma = 0.25 \sum_i \left[\bar{E}_{\gamma i} \left(\sum_j (IAR)_{ij} \; (x/Q)_j \right) \right]$$

* No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

where:

$(IAR)_{ij}$ = integrated activity of noble gas or iodine nuclide i released during time interval j in Ci *

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/m^3

$\bar{E}_{\gamma i}$ = average gamma energy for noble gas or iodine nuclide i in Mev/dis (Table III.10)

D_γ = whole body gamma dose due to immersion in rem

Offsite beta-skin doses are calculated using the equation:

$$D_\beta = 0.23\sum_i \left[\bar{E}_{\beta i}\left(\sum_j (IAR)_{ij}\ (x/Q)_j\right)\right]$$

where:

$(IAR)_{ij}$ = integrated activity of noble gas or iodine nuclide i released during time interval j in Ci *

$(x/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/m^3

$\bar{E}_{\beta i}$ = average beta energy for noble gas or iodine nuclide i in Mev/dis (Table III.10)

D_β = beta-skin dose due to immersion in rem

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* No credit is taken for cloud depletion by ground deposition or by radioactive decay during transport to the exclusion area boundary or to the outer boundary of the low-population zone.

5. Results

Thyroid, whole-body gamma, and beta-skin doses at the Exclusion Area Boundary and the outer boundary of the Low Population Zone are presented in Table III.11. All doses are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100.

TABLE III.4

STP SGTR ANALYSIS

PARAMETERS USED IN EVALUATING

RADIOLOGICAL CONSEQUENCES

I. Source Data	
A. Core power level, MWt	4100
B. Total steam generator tube leakage, prior to accident, gpm	1.0
C. Reactor coolant activity:	
1. Accident Initiated Spike	The initial RC iodine activities based on 1 μCi/gram of D.E. I-131 are presented in Table III.5. The iodine appearance rates assumed for the accident initiated spike are presented in Table III.6.
2. Preaccident Spike	Primary coolant iodine activities based on 60 μCi/gram of D.E. I-131 are presented in Table III.5.
3. Noble Gas Activity	The initial RC noble gas activities based on 1% fuel defects are presented in Table III.7.

TABLE III.4 (Sheet 2)

D.	Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table III.5.
E.	Reactor coolant mass, grams	2.6×10^8
F.	Initial Steam generator water mass (each), grams	4.9×10^7
G.	Offsite power	Lost at time of reactor trip
H.	Primary-to-secondary leakage duration for intact SG, hrs.	8
I.	Species of iodine	100 percent elemental

II. Activity Release Data

A. Ruptured steam generator

1.	Rupture flow	See Table III.2
2.	Rupture flow flashing fraction	See Figure III.13
3.	Iodine scrubbing efficiency	Negligible
4.	Total steam release, lbs	See Table III.2
5.	Iodine partition coefficient	100

TABLE III.4 (Sheet 3)

6. Location of tube rupture	Intersection of outer tube row and upper anti-vibration bar
B. Intact steam generators	
1. Total primary-to-secondary leakage, gpm	0.7
2. Total steam release, lbs	See Table III.2
3. Iodine partition coefficient	100
C. Condenser	
1. Iodine partition coefficient	100
D. Atmospheric Dispersion Factors	See Table III.8

TABLE III.5

STP SGTR ANALYSIS

IODINE SPECIFIC ACTIVITIES

IN THE PRIMARY AND SECONDARY COOLANT

BASED ON 1, 60 AND 0.1 μCi/gram OF D.E. I-131*

	Specific Activity (μCi/gm)		
	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.75	45.0	0.075
I-132	0.88	52.8	0.088
I-133	1.19	71.4	0.120
I-134	0.18	10.8	0.018
I-135	0.66	39.6	0.066

*Consistent with the STP Technical Specifications.

TABLE III.6

STP SGTR ANALYSIS

IODINE SPIKE APPEARANCE RATES

(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
2.2	12.1	4.8	5.7	4.4

TABLE III.7

STP SGTR ANALYSIS
NOBLE GAS SPECIFIC ACTIVITIES IN THE REACTOR COOLANT BASED ON 1% FUEL DEFECTS

Nuclide	Specific Activity (μCi/gm)
Xe-131m	2.0
Xe-133m	16.0
Xe-133	250.0
Xe-135m	0.46
Xe-135	6.8
Xe-138	0.64
Kr-85m	2.0
Kr-85	7.3
Kr-87	1.2
Kr-88	3.6

TABLE III.8
STP SGTR ANALYSIS
ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

Time (hours)	Exclusion Area Boundary x/Q (Sec/m^3)	Low Population Zone x/Q (Sec/m^3)	Breathing Rate (m^3/Sec) [5]
0-2	1.3×10^{-4}	3.8×10^{-5}	3.47×10^{-4}
2-8	-	1.6×10^{-5}	3.47×10^{-4}

TABLE III.9

STP SGTR ANALYSIS

THYROID DOSE CONVERSION FACTORS

(Rem/Curie) [Ref. 6]

Nuclide	
I-131	1.49×10^6
I-132	1.43×10^4
I-133	2.69×10^5
I-134	3.73×10^3
I-135	5.60×10^4

TABLE III.10

STP SGTR ANALYSIS

AVERAGE GAMMA AND BETA ENERGY FOR NOBLE GASES AND IODINES

(Mev/dis) [Ref. 7]

Nuclide	$\bar{E}_\gamma$	$\bar{E}_\beta$
Xe-131m	0.0029	0.16
Xe-133m	0.02	0.212
Xe-133	0.03	0.153
Xe-135m	0.43	0.099
Xe-135	0.246	0.325
Xe-138	1.2	0.66
Kr-85m	0.156	0.253
Kr-85	0.0023	0.251
Kr-87	0.793	1.33
Kr-88	2.21	0.248
I-131	0.38	0.19
I-132	2.2	0.52
I-133	0.6	0.42
I-134	2.6	0.69
I-135	1.4	0.43

TABLE III.11

STIP SGTR ANALYSIS

OFFSITE RADIATION DOSES

	Doses (Rem) Calculated Value	Allowable Guideline Value [Ref. 8]
1. Accident Initiated Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid Dose	4.0	30
Whole - Body Gamma Dose	0.067	2.5*
Beta - Skin Dose	0.110	2.5*
Low Population Zone (0-8 hr.)		
Thyroid Dose	1.2	30
Whole - Body Gamma Dose	0.020	2.5*
Beta - Skin Dose	0.033	2.5*
2. Pre-Accident Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid Dose	15.6	300
Whole - Body Gamma Dose	0.069	25*
Beta - Skin Dose	0.110	25*
Low Population Zone (0-8 hr.)		
Thyroid Dose	4.6	300
Whole - Body Gamma Dose	0.020	25*
Beta - Skin Dose	0.034	25*

*Assumed to apply to the sum of the whole-body gamma and beta-skin doses.

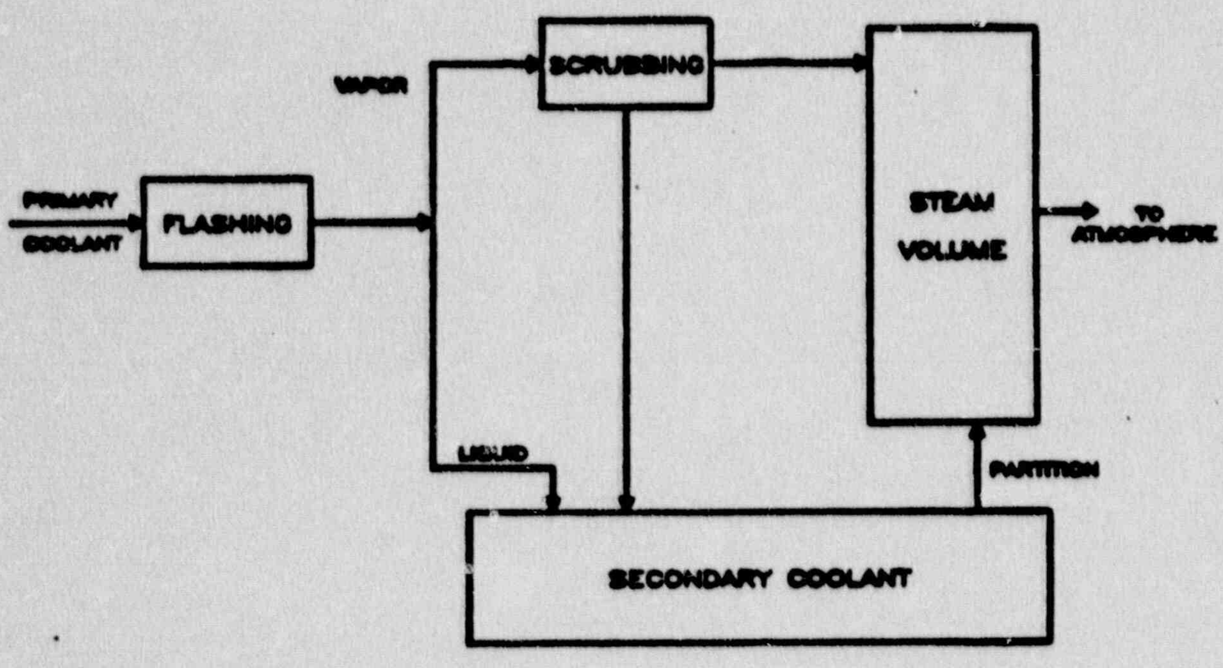

Figure III.12 Iodine Transport Model - Offsite Radiation Dose Analysis

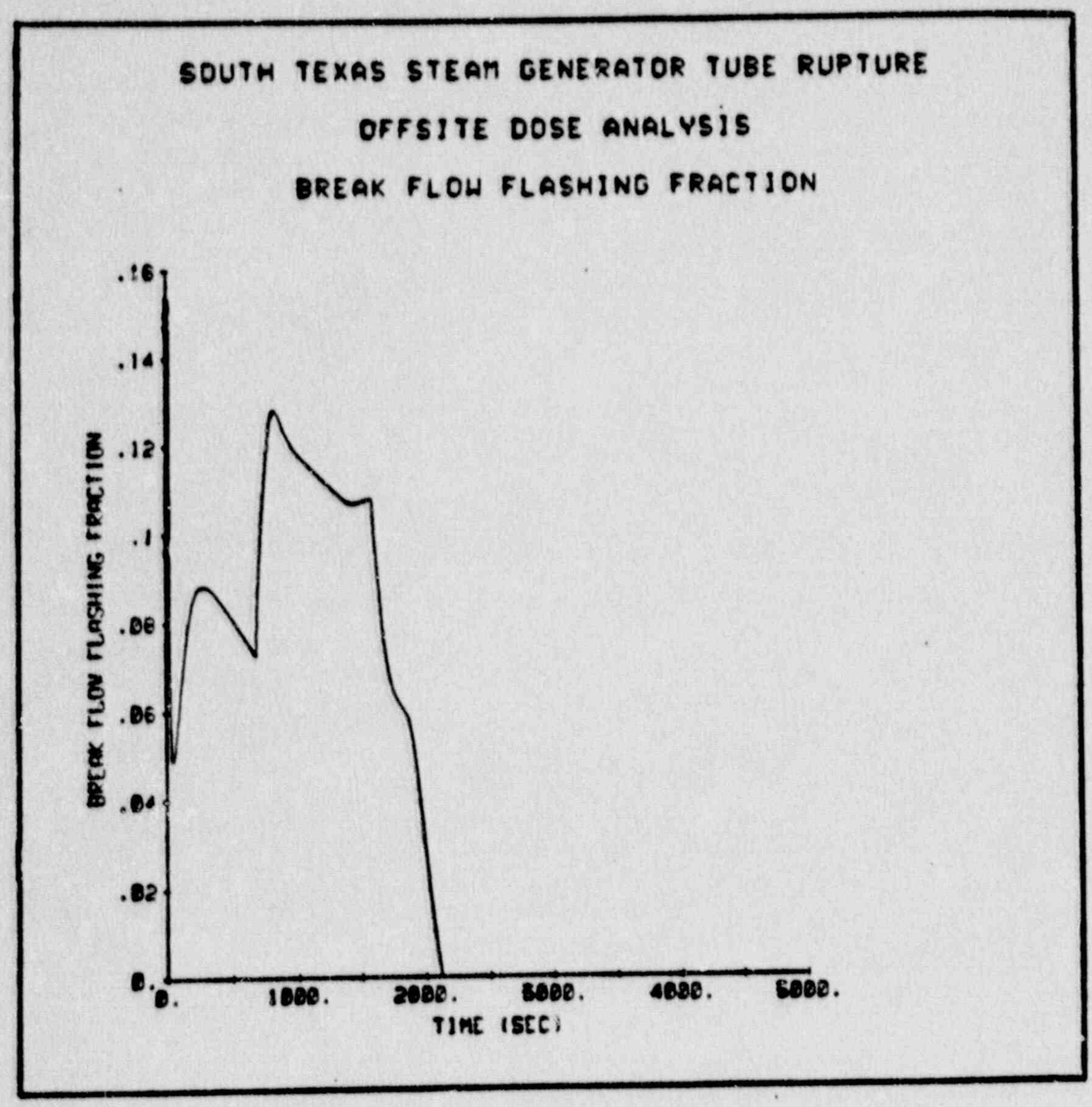

Figure III.13 Break Flow Flashing Fraction - Offsite Radiation Dose Analysis

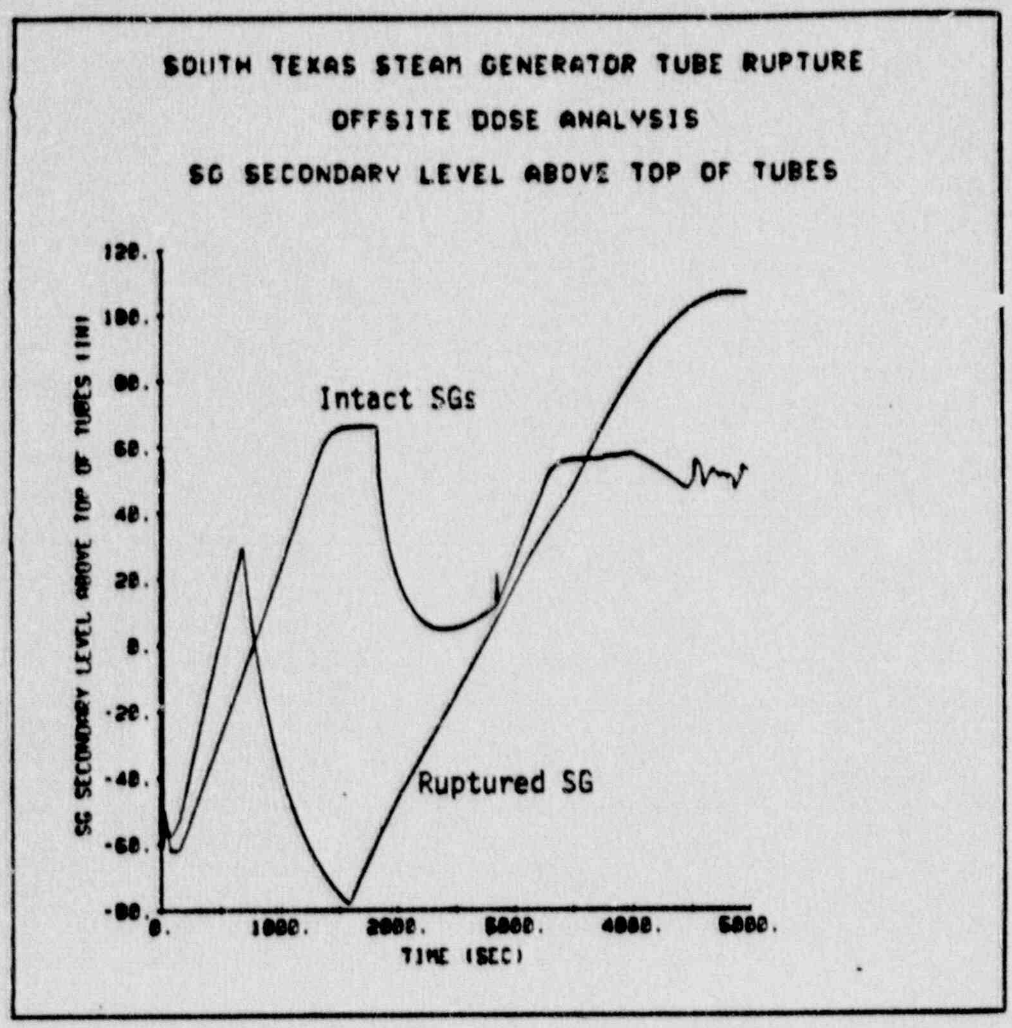

Figure III.14 SG Water Level Above Top of Tubes - Offsite Radiation Dose Analysis

IV. CONCLUSION

An evaluation has been performed for a design basis SGTR event for the South Texas Units 1 and 2 to demonstrate that the potential consequences are acceptable. An analysis was performed to demonstrate margin to steam generator overfill assuming the limiting single failure relative to overfill. The limiting single failure is the failure of [

] a,c The results of this analysis indicate that the recovery actions can be completed to terminate the primary to secondary break flow before overfill of the ruptured steam generator would occur.

Since it is concluded that steam generator overfill will not occur for a design basis SGTR, an analysis was also performed to determine the offsite radiation doses assuming the limiting single failure for offsite doses. For this analysis, it was assumed that [

] a,c The primary to secondary break flow and the mass releases to the atmosphere were determined for this case, and the offsite radiation doses were calculated using this information. The resulting doses at the exclusion area boundary and low population zone are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100. Thus, it is concluded that the consequences of a design basis steam generator tube rupture at South Texas would be acceptable.

V. REFERENCES

1. Lewis, Huang, Behnke, Fittante, Gelman, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A [PROPRIETARY]/WCAP-10750-A [NON-PROPRIETARY], August 1987.

2. Lewis, Huang, Rubin, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10698-P-A [PROPRIETARY]/Supplement 1 to WCAP-10750-A [NON-PROPRIETARY], March 1986.

3. Lewis, Huang, Rubin, Murray, Roidt, Hopkins, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident," WCAP-11002 [PROPRIETARY]/WCAP-11003 [NON-PROPRIETARY], February 1986.

4. Postma, A. K., Tam, P. S., "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture," NUREG-0409.

5. NRC Regulatory Guide 1.4, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors," June 1974.

6. NRC Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," October 1977.

7. Bell, M. J., "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, 1973.

8. Standard Review Plan, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," NUREG-0800, July 1981.

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1. Lewis, Huang, Behnke, Fittante, Gelman, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10750-A, August 1987.

2. Lewis, Huang, Rubin, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Supplement 1 to WCAP-10750-A, March 1986.

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7. Bell, M. S. "ORIGEN - The ORNL Isotope Generation and Depletion Code", ORNL-8628, 1973.

8. Standard Review Plan, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure", NUREG-0800, July 1981.