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LOFTTR2 ANALYSIS FOR A STEAM GENERATOR TUBE RUPTURE FOR WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

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I. INTRODUCTION

An evaluation for a design basis steam generator tube rupture (SGTR) event has been performed for the Watts Bar Nuclear Plant 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 a thermal and hydraulic analysis to determine the input for use in calculating the offsite radiation doses.

The Watts Bar Nuclear Plant employs two essentially identical Westinghouse pressurized water reactor (PWR) units each rated at 3411 MWt. The reactor coolant system has four reactor coolant loops with Model D3-2 steam generators. Since the reactors, structures, and all auxiliary equipment are essentially identical for the two units, the SGTR evaluation is applicable for both units. The SGTR evaluation is based on the current Watts Bar Units 1 and 2 plant design. The evaluation is applicable for operation with either Westinghouse Standard or Vantage-5 Hybrid (V5H) fuel installed. The evaluation is also applicable for up to 10 percent steam generator tube plugging to provide an allowance for future tube plugging up to this level.*

The SGTR analyses were performed for Watts Bar Units 1 and 2 using the analysis methodology developed in WCAP-10698 (Reference 1) and Supplement 1 to WCAP-10698 (Reference 2). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the NRC in Safety Evaluation Reports (SERs) 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 Watts Bar Units 1 and 2. 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

*Assumes 10% of steam generator tubes in each generator are plugged and corresponds to the worst plugging level of any steam generator.

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 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 Watts Bar Units 1 and 2 Emergency Operating Procedures (EOPs), which were developed from the Westinghouse Owners Group Emergency Response Guidelines (ERGs). The operator action times used for the analysis are based on the results of simulator studies of the SGTR recovery operations which were performed by the TVA operations personnel using the plant training simulator.

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. A thermal and hydraulic analysis was also performed to determine the input for use in calculating 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, the single failure was assumed to be the failure of [

]^{a,c}

[ ]^{a,c} The LOFTTR2 analysis to determine the margin to overfill was performed for the time period from the steam generator tube rupture until the primary and secondary pressures are equalized (break flow termination). 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 Watts Bar Units 1 and 2.

Since steam generator overfill does not occur, the thermal and hydraulic analysis results for the offsite radiation dose analysis represent the limiting case for the analysis of the radiological consequences for Watts Bar Units 1 and 2. For the thermal and hydraulic analysis for the offsite radiation dose analysis, the ruptured steam generator power-operated relief valve (PORV) was assumed to fail open at the time the isolation of the ruptured steam generator is performed. 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 will be used to calculate the radiation doses at the site boundary and low population zone.

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 Watts Bar Units 1 and 2. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1, and using the plant specific parameters for Watts Bar Units 1 and 2. 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 on the outlet (cold leg) side of the steam generator. [

.]^{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.

Based on the sensitivity studies in Reference 1 and previous sensitivity studies for four-loop plants, the limiting single failure may be [

]^{a,c}

The Watts Bar AFW system consists of two motor-driven AFW (MDAFW) pumps, and one turbine-driven AFW (TDAFW) pump with a capacity approximately equal to the combined capacity of the two MDAFW pumps. Each MDAFW pump normally feeds two steam generators and the TDAFW pump feeds all four steam generators. There is a level control valve (LCV) in the flow path from the MDAFW pump to each steam generator and a LCV in the flow path from the TDAFW pump to each steam generator. The LCVs

for the MDAFW and TDAFW pumps are automatically controlled to throttle the feed flow as required to maintain the level in the associated steam generator between 33% and 43% of the narrow range span (NRS). The LCVs in the MDAFW and TDAFW flow paths can also be used to manually control the AFW flow to the steam generators and to isolate the flow to the ruptured steam generator.

The failure of [

]^{a,c} thereby reducing the margin to steam generator overfill. Since the MDAFW pump flow rate to the ruptured steam generator is slightly greater than the TDAFW pump flow rate, [

]^{a,c}

The failure of [

]^{a,c} would increase the time required to perform the RCS cooldown, which results in additional primary to secondary leakage and decreases the margin to steam generator overfill.

A sensitivity study for these two single failures, has shown that the limiting single failure for Watts Bar is [

]^{a,c}

B. 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 applied with the Watts Bar Units 1 and 2 parameters in the LOFTTR2 analysis to determine the margin to steam generator overfill for Watts Bar with the exception of the following differences.

1. Reactor Trip and Turbine Runback

A turbine runback can be initiated automatically or the operator can manually reduce the turbine load following an SGTR to attempt to prevent a reactor trip. The simulation of turbine runback results in [

]a,c For the reference plant analysis in Reference 1, 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, the Watts Bar steam generator level program increases linearly from 38% NRS at 0% rated thermal power to 66.5% NRS at 100% rated thermal power. With this level program, the initial secondary mass for power levels between [ ]a,c is less than the 100% power initial mass. [

]a,c

2. Steam Generator Secondary Mass

A [ ]a,c initial secondary water mass in the ruptured steam generator was determined by Reference 1 to be conservative for overfill. The initial steam generator total fluid mass was conservatively assumed to be [

]a,c

3. AFW System Operation

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 TDAFW pump will be available within approximately 10 seconds following the actuation signal, but the flow from the MDAFW 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 TDAFW and MDAFW pumps is initiated [ ]a,c

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 the Watts Bar analysis, the time of reactor trip was determined by modeling the Watts Bar Units 1 and 2 reactor protection system, and the actuation of the AFW system was based on the [ ]a,c It was assumed that the flow from both the MDAFW and TDAFW pumps is initiated at [ ]a,c

For the margin to overfill analysis, the maximum total AFW flow rate with both MDAFW pumps and the TDAFW pump operating was determined to be 2467 gpm, with the flow divided to the four steam generator as follows: 759 gpm to the ruptured steam generator, and 739 gpm, 498 gpm, and 471 gpm to each of the intact steam generators. The operation of the LCVs to throttle the AFW flow to the intact steam generators was simulated in the analysis to maintain the level at approximately 43%. With the LCVs for the MDAFW and TDAFW pumps at the maximum possible open positions allowed by the control system (narrow range water level below 33% NRS), the AFW flow rate to the ruptured steam generator is 759 gpm (394 gpm from the MDAFW pump and 365 gpm from the TDAFW pump). For the narrow range water level in the ruptured steam generator between 33% and 43% NRS, the AFW flow rate will be decreased linearly from 759 gpm to 0 gpm. [

]^{a,c} the AFW flow rate to the ruptured steam generator would decrease linearly from 759 gpm to 394 gpm between 33% and 43% NRS, and continue at 394 gpm until 15.00 minutes when the operator isolates the ruptured steam generator.

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 the Watts Bar Units 1 and 2 EOP E-3 which is based on the Westinghouse Owners Group ERG E-3, and these actions were explicitly modeled in this analysis. The operator actions modeled include (1) identification and isolation of the ruptured steam generator, (2) cooldown of the RCS to ensure subcooling, (3) depressurization of the RCS to restore inventory, and (4) 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 condenser vacuum exhaust radiation monitor, steam generator blowdown radiation monitor, or steam generator discharge 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 narrow range level, high radiation from a steam generator blowdown radiation monitor, high radiation from a steam generator discharge radiation monitor, a RADCON survey, or a chemistry laboratory sample. For an SGTR that results in 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 be distributed to each of the steam generator and will begin to refill the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly than expected in that steam generator. This response, as displayed 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 the steam generator with a tube rupture has been identified, recovery actions begin by isolating steam flow from and stopping auxiliary feedwater flow to the ruptured steam generator. In addition to minimizing radiological releases, this also reduces the possibility of overfilling the ruptured steam generator 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. For the reference plant analysis in Reference 1, it was assumed that the ruptured steam generator will be

isolated when [

]^{a,c} For Watts Bar Units 1 and 2, the steam generator narrow range level corresponding to being just on span is 10% and the comparable operator action time is 15.00 minutes. Thus, applying the Reference 1 methodology for the Watts Bar Units 1 and 2 analysis, the ruptured steam generator was assumed to be isolated at 30% narrow range level or at 15.00 minutes, whichever was longer.

3. Cool down the 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 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 to release steam from 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 the 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. However, 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 this event, RCS depressurization can be performed using the pressurizer PORVs or auxiliary pressurizer spray. Because the pressurizer PORVs are the preferred alternative, it was assumed that a pressurizer PORV is used for the RCS depressurization for this analysis.

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 re-initiation of leakage into the ruptured steam generator.

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 SI termination were developed for the design basis analysis in Reference 1. Tennessee Valley Authority has determined the corresponding operator action times to perform these operations for Watts Bar Units 1 and 2. The operator actions and the corresponding operator action times used for the Watts Bar Units 1 and 2 analysis are listed in Table II.1.

TABLE II.1

WATTS BAR UNITS 1 AND 2 SGTR ANALYSIS

OPERATOR ACTION TIMES FOR DESIGN BASIS ANALYSIS

Action	Time Intervals
Identify and isolate ruptured SG	15.00 min or LOFTTR2 calculated time from event initiation to reach 30% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	7.15 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	2.45 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	4.07 min
SI termination and pressure equalization	Calculated by LOFTTR2

D. Transient Description

The LOFTTR2 analysis results for the Watts Bar Units 1 and 2 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 on an overtemperature delta-T trip signal at approximately 92 seconds.

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 was assumed to be terminated and AFW flow was assumed to 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. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 307 seconds. After SI actuation, the SI flow rate initially exceeds the tube rupture break flow rate, and the pressurizer level and

RCS pressure begin to increase and approach the equilibrium values 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 as the reactor coolant pumps coast down and natural circulation flow develops. The cold leg temperature trends toward the steam generator temperature as the fluid residence time in the tube region increases. The hot leg temperature reaches a peak and then slowly decreases as steady state conditions are reached, until operator actions are initiated to cooldown the RCS.

The loss of offsite power at reactor trip results in the termination of main feedwater and actuation of the AFW system. It was assumed that AFW flow to the steam generators is [ ]^{a,c} The water level in the steam generators decreases significantly following reactor trip and then begins to increase due to the AFW addition. The water level in the ruptured steam generator increases more rapidly due to the primary to secondary break flow into that steam generator, and the water level reaches 33% at approximately 132 seconds. The operation of the LCV in the TDAFW pump flow path was simulated to throttle the flow from the TDAFW pump as the level increases to 43%, [ ]^{a,c} Thus, the [ ]^{a,c} until the operator action is taken to isolate the ruptured steam generator at 15.00 minutes. The AFW flow to the intact steam generators was assumed to be throttled to maintain the level at approximately 43%.

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, the ruptured steam generator is assumed to be identified and isolated when the narrow range level reaches 30% on the ruptured steam generator or at 15.00 minutes after initiation of the SGTR, whichever is longer. For the Watts Bar Units 1 and 2 analysis, the time to reach 30% is less than 15.00 minutes, and thus the ruptured steam generator is assumed to be isolated at 15.00 minutes or 900 seconds. However, because of the computer program limitations for simulating the operator actions, the ruptured steam generator was isolated 902 seconds.

Because the water level in the ruptured steam generator is greater than 43%, the AFW flow from the TDAFW pump will have been throttled to zero prior to this time. However, because of the [ ]a,c full flow from the MDAFW pump was assumed to continue and must be isolated at this time. The [ ]a,c For either case, it is expected that the operators would recognize that the flow was not being throttled when the level is above 33% since indications of the level and AFW flow rates are available in the control room. If [ ]a,c the operator can still manually throttle the MDAFW flow as required to maintain level and to isolate the MDAFW flow to the ruptured steam generator. However, if [ ]a,c it may not be possible to use the valve to throttle or isolate the MDAFW flow. It is expected that [ ]a,c would be detected if the valve fails to respond to manual throttling action, and the operators would be expected to turn off the

associated MDAFW pump or to locally close the isolation valve in the MDAFW flow path. Thus, it is assumed that the flow from the MDAFW pump to the ruptured steam generator is terminated at this time by turning off the associated MDAFW pump or by manually closing the isolation valve in the flow path. It is noted that stopping the MDAFW pump feeding the ruptured steam generator would also stop the MDAFW flow to the associated intact steam generator. However, adequate AFW flow would still be available from the TDAFW pump to maintain inventory in that intact steam generator.

2. Cool down the RCS to Establish Subcooling Margin

After isolation of the ruptured steam generator, there is a 7.15 minute operator action time imposed prior to initiating the cooldown. The actual delay time used in the analysis is 2 seconds longer because of the computer program limitations for simulating the operator actions. 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 three intact steam generators. The intact steam generator PORVs are assumed to be opened at 1333 seconds for RCS cooldown. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for subcooling uncertainty. When these conditions are satisfied at 1816 seconds, it is assumed that the operator closes the PORVs on the intact steam generators 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. The pressurizer level and RCS pressure also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures II.1 and II.2.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a 2.45 minute operator action time is 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 opening a pressurizer PORV. The RCS depressurization is initiated at 1963 seconds and continued until any of the following conditions are satisfied: pressurizer level is greater than 65%, or RCS subcooling is less than the allowance for subcooling uncertainty, or RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 10%. For this case, the RCS depressurization is terminated at 2048 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10%. 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.

4. Terminate SI to Stop Primary to Secondary Leakage

The previous actions establish 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, 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 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 stable or increasing, and the pressurizer level is greater than 10%. For the Watts Bar Units 1 and 2 analysis, SI was not terminated until the RCS pressure increased to 50 psi above the ruptured steam generator pressure to assure that RCS pressure is increasing.

After depressurization is completed, an operator action time of 4.07 minutes was assumed prior to initiation of SI termination. Since the above requirements are satisfied, SI termination actions were performed at 2292 seconds by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure II.2. The intact steam generator PORVs are also opened 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 is 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 and 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 approximately 5828 ft^3 which is 119 ft^3 less than the total steam generator volume of 5947 ft^3. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for Watts Bar Units 1 and 2.

TABLE II.2

WATTS BAR UNITS 1 AND 2 SGTR ANALYSIS

SEQUENCE OF EVENTS

MARGIN TO OVERFILL ANALYSIS

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	92
Safety Injection	307
Ruptured SG Isolated	902
RCS Cooldown Initiated	1333
RCS Cooldown Terminated	1816
RCS Depressurization Initiated	1963
RCS Depressurization Terminated	2048
SI Terminated	2292
Break Flow Terminated	3884

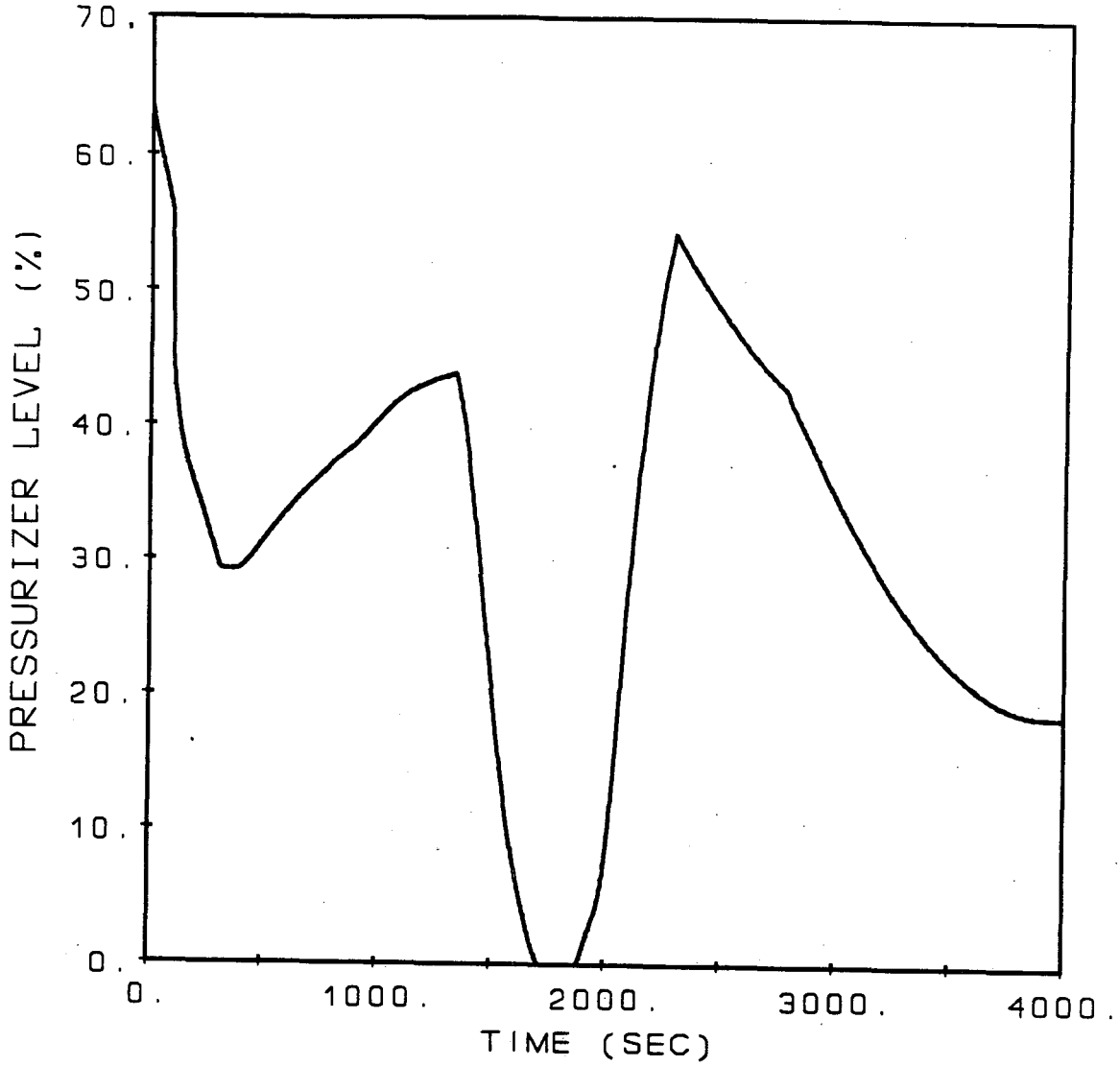

Figure II.1 Pressurizer Level - Margin to Overfill Analysis

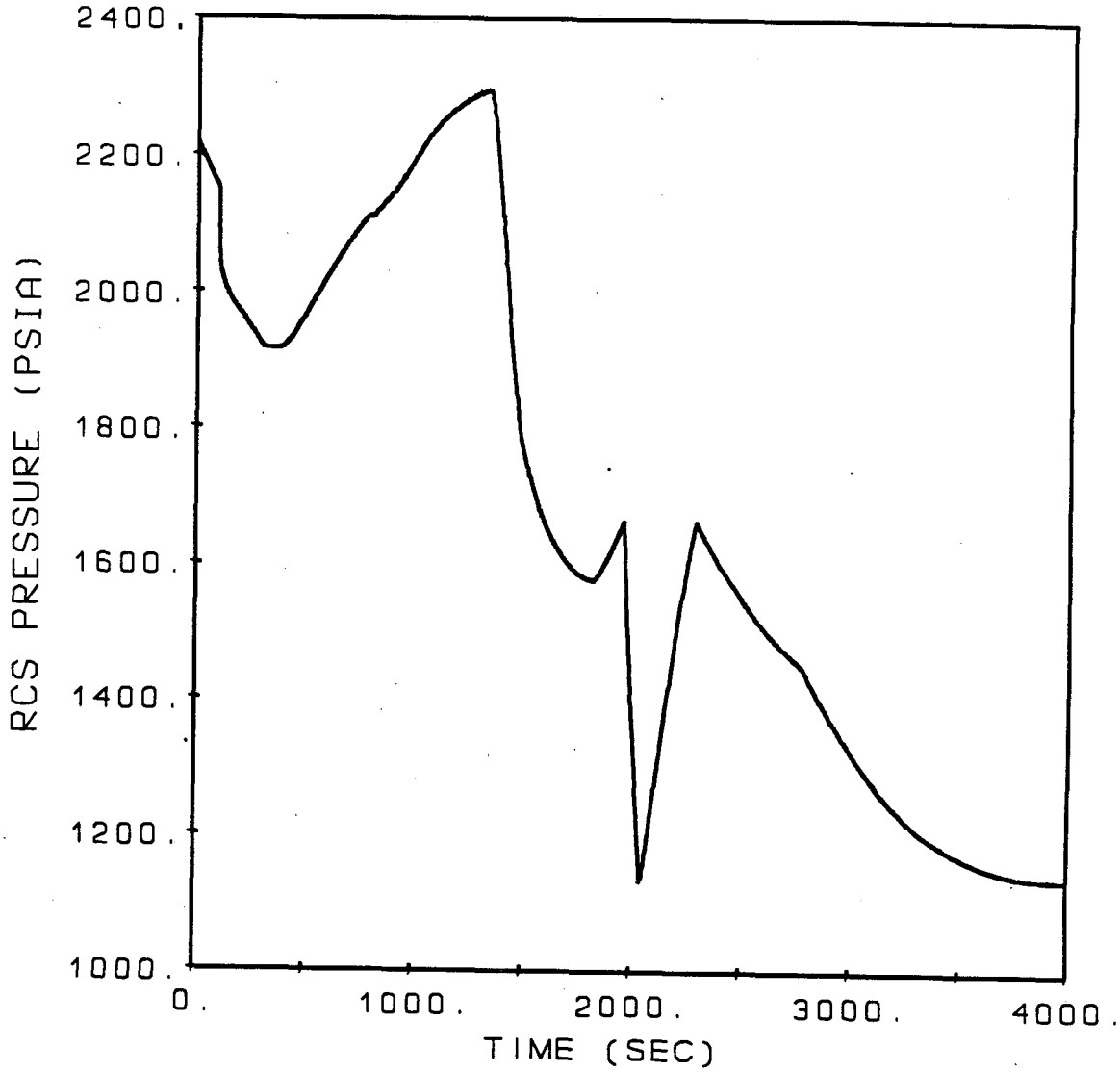

Figure II.2 RCS Pressure - Margin to Overfill Analysis

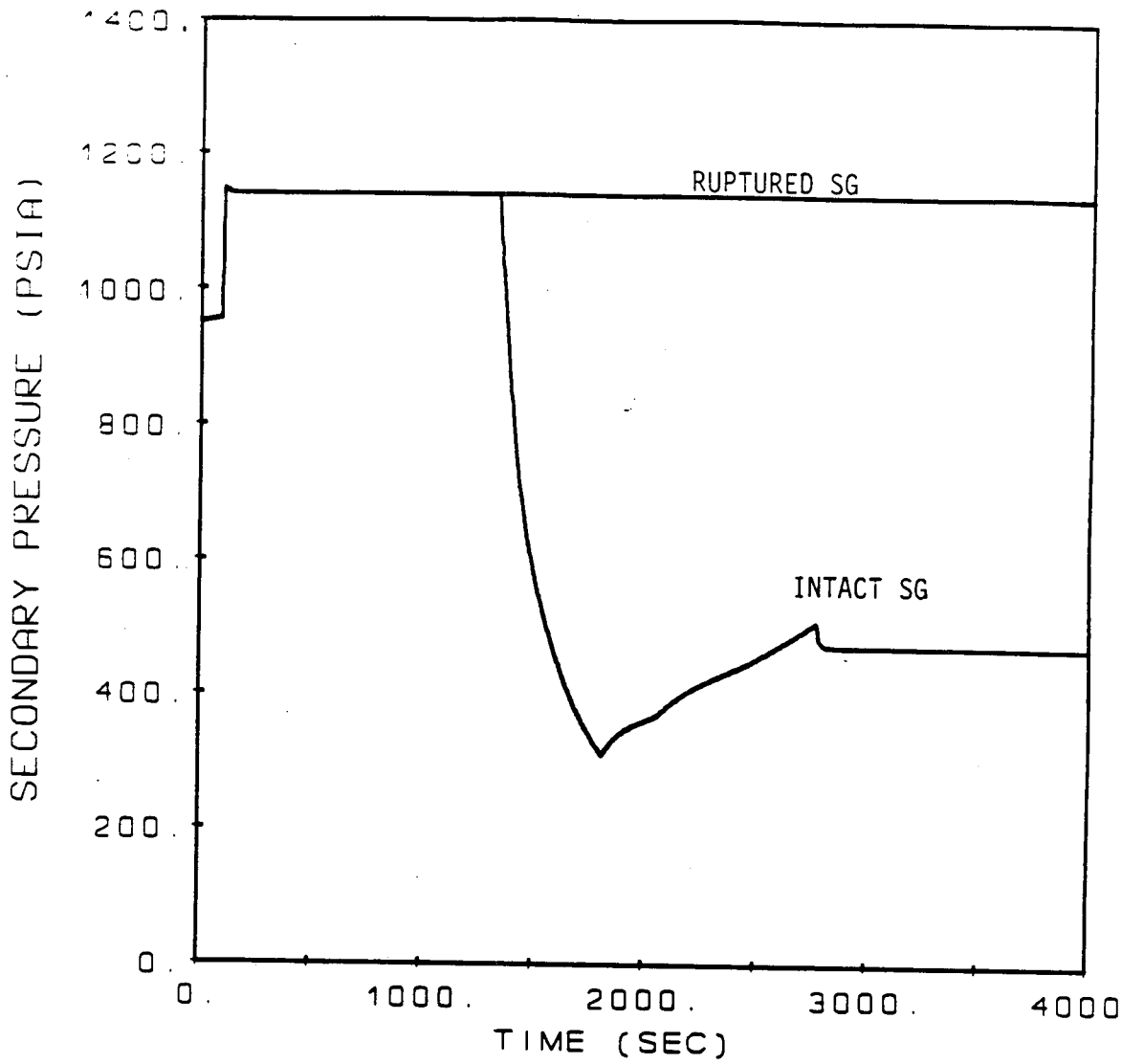

Figure II.3 Secondary Pressure - Margin to Overfill Analysis

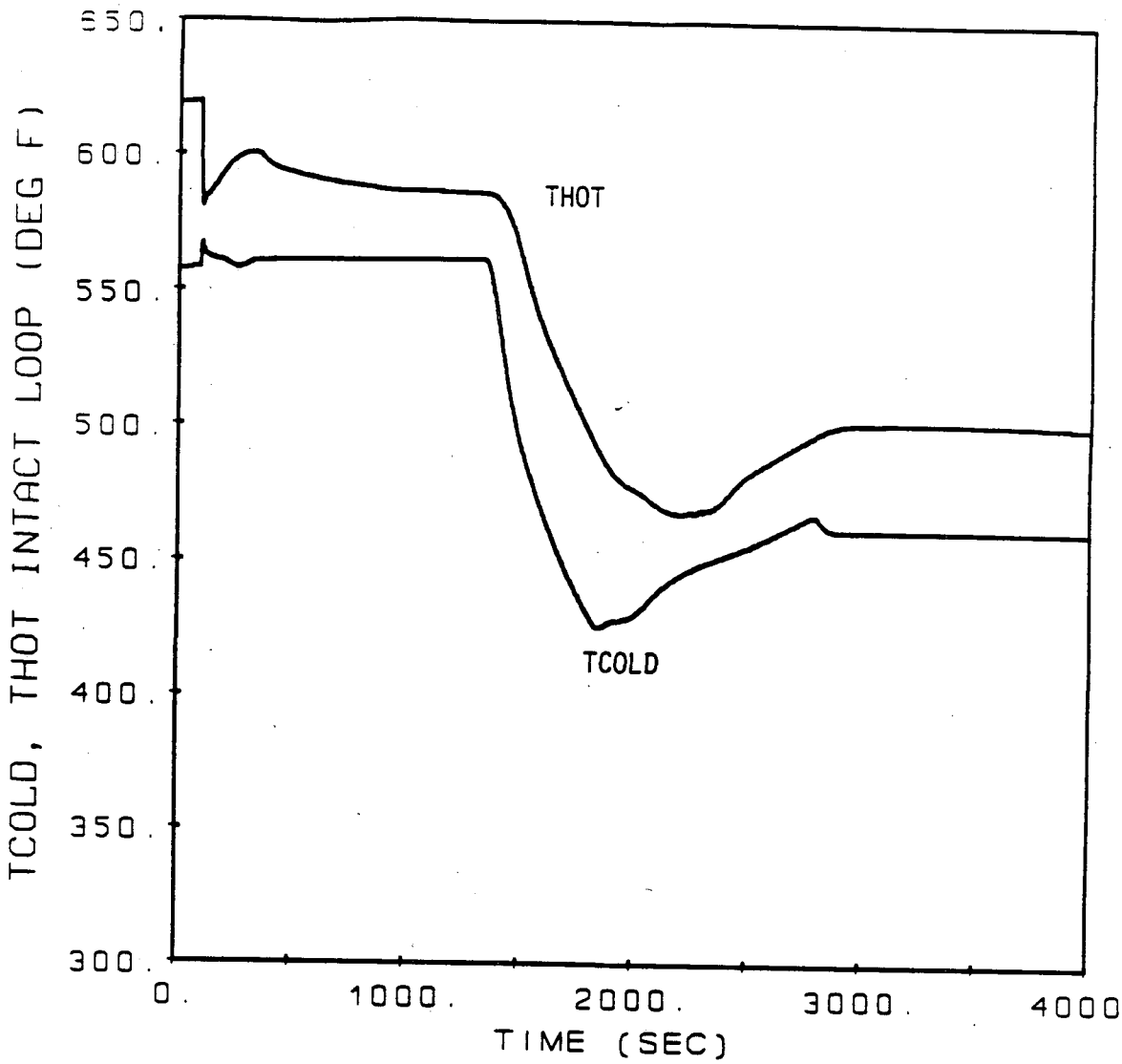

Figure II.4 Intact Loop Hot and Cold Leg RCS Temperatures - Margin to Overfill Analysis

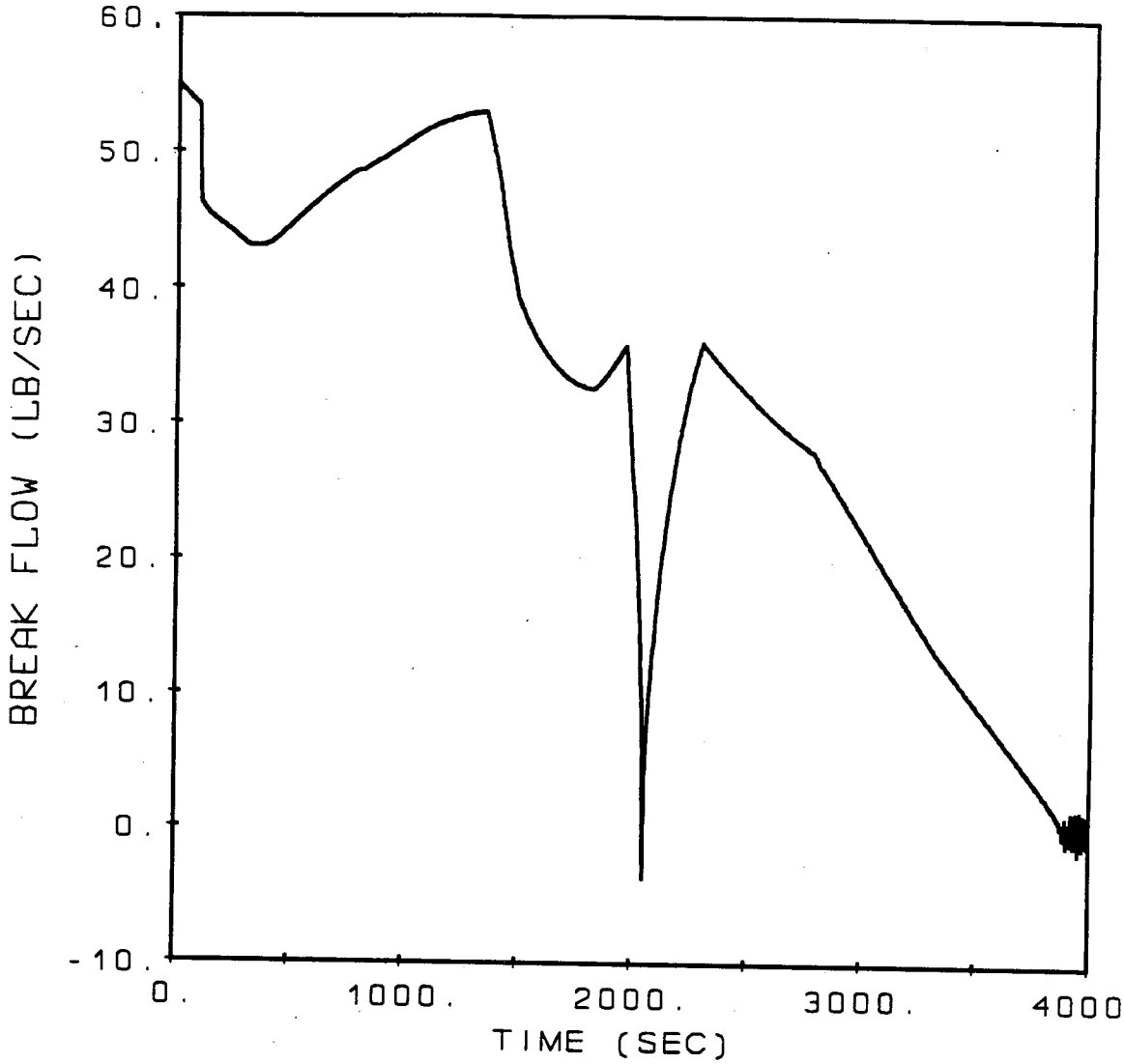

Figure II.5 Primary to Secondary Break Flow Rate - Margin to Overfill Analysis

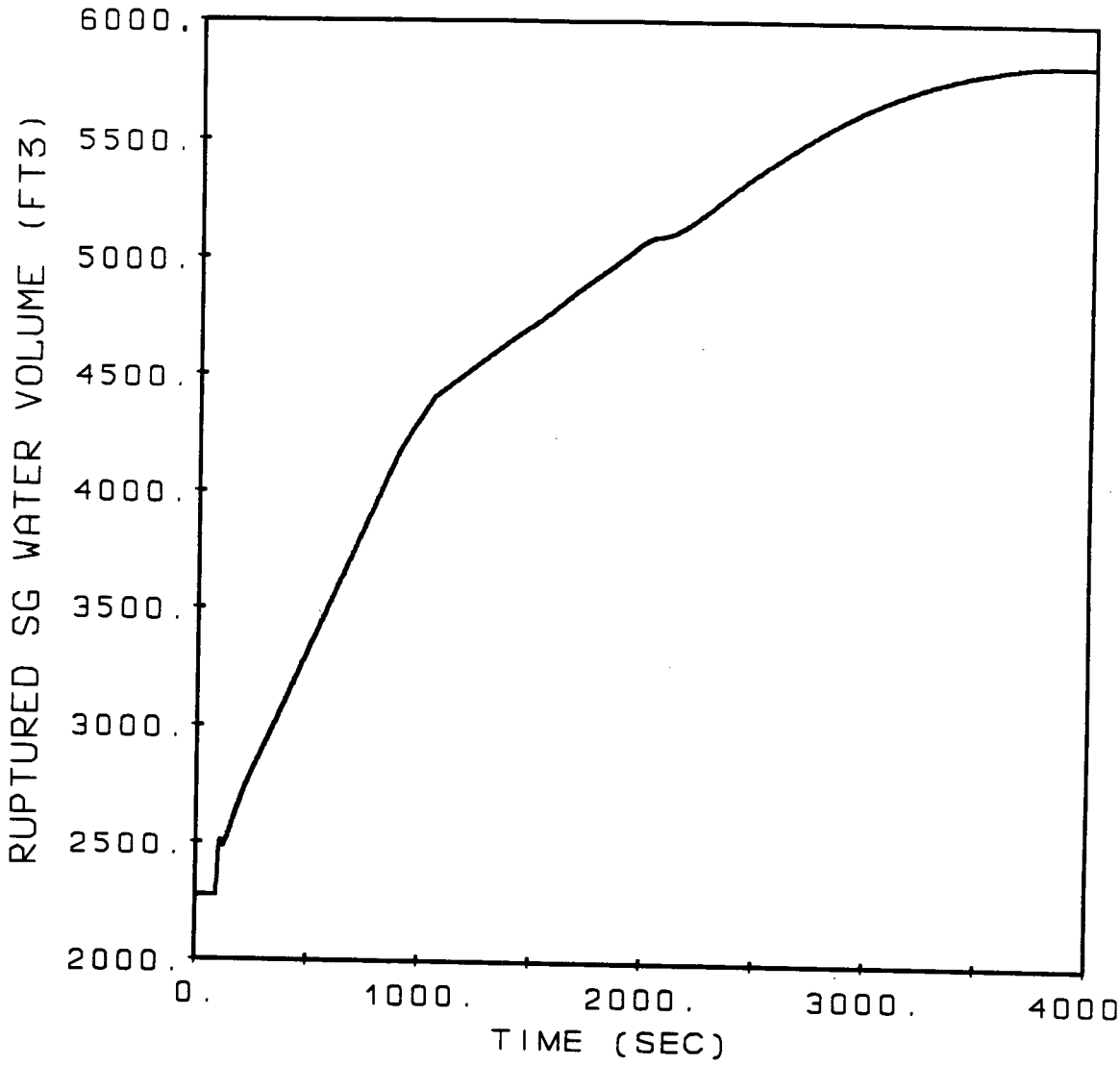

Figure II.6 Ruptured SG Water Volume - Margin to Overfill Analysis

III. THERMAL AND HYDRAULIC ANALYSIS FOR OFFSITE RADIOLOGICAL CONSEQUENCES

A thermal and hydraulic analysis was performed to determine the input for the offsite radiological consequences analysis for a design basis SGTR for Watts Bar Units 1 and 2. The thermal and hydraulic analysis was performed using the methodology developed in References 1 and 2 and the plant specific parameters for Watts Bar Units 1 and 2.

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 until break flow termination were calculated 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.

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 on the outlet (cold leg) side of the steam generator. [

]^{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.

Based on the information in Reference 2, the most limiting single failure with respect to offsite doses for Watts Bar Units 1 and 2 is a failed open PORV on the ruptured steam generator. Failure of this PORV will cause an uncontrolled depressurization of the ruptured steam generator 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 the failed PORV is isolated by locally closing the associated block valve, and the recovery actions are completed. Thus, for the offsite dose analysis, it was assumed that the ruptured steam generator PORV fails open and must be locally isolated.

B. 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.

1. 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 Watts Bar Units 1 and 2 reactor protection system, and this time was used for the offsite dose analysis. In addition, turbine runback was simulated for the analysis since the reduced initial water mass for lower power levels is conservative for the offsite dose analysis. Turbine runback was simulated until the time of reactor trip at [ ]^{a,c} by conservatively determining the secondary mass for the runback power level, and reducing the initial secondary mass by the difference in the nominal mass for 100% power and the runback power level.

2. Steam Generator Secondary Mass

If steam generator overfill does not occur, [ ]a,c results in a conservative prediction of offsite doses. Thus, for the offsite dose analysis, the initial secondary mass was assumed to correspond to operation at nominal steam generator mass minus a [ ]a,c allowance for uncertainties. As noted above, the effect of turbine runback was simulated by further reducing the secondary water mass to account for the decrease in the nominal mass at the runback power level.

3. AFW System Operation

In Reference 2, it was determined that a [ ]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 [ ]a,c Since the single failure assumed for the offsite radiation dose analysis is a failed open PORV on the ruptured steam generator, it is not necessary to assume an additional failure in the AFW system. Thus, the TDAFW pump and both MDAFW pumps were assumed to deliver flow to all four steam generators. A minimum AFW flow of 310 gpm per steam generator was assumed for the offsite radiation dose analysis. The delay time assumed for delivery of the AFW flow was conservatively increased to 60 seconds after SI actuation since this represents the maximum allowable delay time for the Watts Bar Units 1 and 2 AFW pumps. The AFW flow to the ruptured steam generator was assumed to be reduced linearly from 310 gpm to zero between 33% and 43% NRS to simulate the throttling by the LCVs. The operation of the LCVs to throttle the AFW flow to the intact steam generators was also simulated in the analysis to maintain the level at approximately 43% NRS.

4. Flashing Fraction

When calculating the amount of break flow that flashes to steam, 100 percent of the break flow is assumed to come from the hot leg side of the break [ ]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 less than the hot leg temperature and the flashing fraction will be correspondingly lower. Thus the assumption that 100 percent of the break flow comes from the hot leg is conservative for the SGTR analysis.

C. 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 assumed for the overfill analysis were also used for the offsite dose analysis. However, for the offsite doses analysis, the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator is isolated. Before proceeding with the recovery operations, the failed open PORV on the ruptured steam generator is assumed to be isolated by locally closing the associated block valve. Tennessee Valley Authority has determined that an operator can locally close the block valve for the PORV on the ruptured steam generator within 11 minutes after the failure. Thus, it was assumed that the ruptured steam generator PORV is isolated at 11 minutes after the valve is assumed to fail open. After the ruptured steam generator PORV is isolated, the additional delay time of 7.15 minutes (Table II.1) was assumed for the operator action time to initiate the RCS cooldown.

D. Transient Description

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. It is noted that reactor trip occurs at a slightly different time for this case compared to the overfill analysis due to the use of different input parameters to provide conservative results for the offsite dose analysis. The transient results for this case are similar to the transient results for the overfill analysis until the ruptured steam generator is isolated. The transient behavior is different after this time since it is assumed that the ruptured steam generator PORV fails open at that time.

Following the tube rupture the RCS pressure decreases as shown in Figure III.1 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on low pressurizer pressure at approximately 94 seconds. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. 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. The RCS pressure and pressurizer level also decrease more rapidly following reactor trip as shown in Figures III.1 and III.3. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 310 seconds.

Major Operator Actions

a. Identify and Isolate the Ruptured Steam Generator

The ruptured steam generator is assumed to be identified and isolated at 15.00 minutes after the initiation of the SGTR or when the narrow range level reaches 30%, whichever time is greater. Since the time to reach 30% narrow range level is less than 15.00 minutes, it was assumed that the ruptured steam generator is isolated at 15.00 minutes or 900 seconds. However, the actual isolation time used in

the analysis is 902 seconds because of the computer program limitations for simulating the operator actions. The ruptured steam generator PORV is also assumed to fail open at this time, and the failure is simulated at 904 seconds because of the computer program limitations. 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. It is assumed that the time required for the operator to identify that the ruptured steam generator PORV is open and to locally close the associated block valve is 11 minutes. However, the actual time used in the analysis is 2 seconds longer because of the computer program limitations. Thus, at 1566 seconds the depressurization of ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown on Figure III.2.

b. Cool Down the RCS to establish Subcooling Margin

After the block valve for the ruptured steam generator PORV is closed, there is a 7.15 minute operator action time imposed prior to initiation of cooldown. Thus, the RCS cooldown was initiated at 1995 seconds. The depressurization of the ruptured steam generator due to the failed open PORV affects the RCS cooldown target temperature since the temperature is determined based on the ruptured steam generator pressure at that time. 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 for instrument uncertainty. Because of the lower pressure in the ruptured steam generator when the cooldown is initiated, the associated temperature the RCS must be cooled to is also lower, which has the net

effect of extending the time required for cooldown. The cooldown is initiated at 1995 seconds and is completed at 2740 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 and pressurizer level also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures III.1 and III.3.

c. Depressurize RCS to Restore Inventory

After the RCS cooldown, a 2.45 minute operator action time is 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 opening a pressurizer PORV. The RCS depressurization is initiated at 2887 seconds and continued until any of the following conditions are satisfied: pressurizer level is greater than 65%, or RCS subcooling is less than the allowance for subcooling uncertainty, or RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than 10%. For this case, the RCS depressurization is terminated at 2976 seconds because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 10%. 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.

d. Terminate SI to Stop Primary to Secondary Leakage

The previous actions establish 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, 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 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 stable or increasing, and the pressurizer level is greater than 10%. For the Watts Bar Units 1 and 2 analysis, SI was not terminated until the RCS pressure increased to 50 psi above the ruptured steam generator pressure to assure that RCS pressure is increasing.

After depressurization is completed, an operator action time of 4.07 minutes was assumed prior to initiation of SI termination. Since the above requirements are satisfied, SI termination actions were performed at 3220 seconds by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure III.1. The intact steam generator PORVs are also opened 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 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 is 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 when the break flow is terminated is less than the volume for the margin to overfill case and significantly less than the total steam generator volume of 5947 ft^3. The mass of water in the ruptured steam generator is also shown as a function of time in Figure III.9.

TABLE III.1

WATTS BAR UNITS 1 AND 2 SGTR ANALYSIS

SEQUENCE OF EVENTS

OFFSITE RADIATION DOSE ANALYSIS

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	94
Safety Injection	310
Ruptured SG Isolated	902
Ruptured SG Atmospheric Steam Dump Valve Fails Open	904
Ruptured SG Atmospheric Steam Dump Valve Block Valve Closed	1566
RCS Cooldown Initiated	1995
RCS Cooldown Terminated	2740
RCS Depressurization Initiated	2887
RCS Depressurization Terminated	2976
SI Terminated	3220
Break Flow Terminated	4548

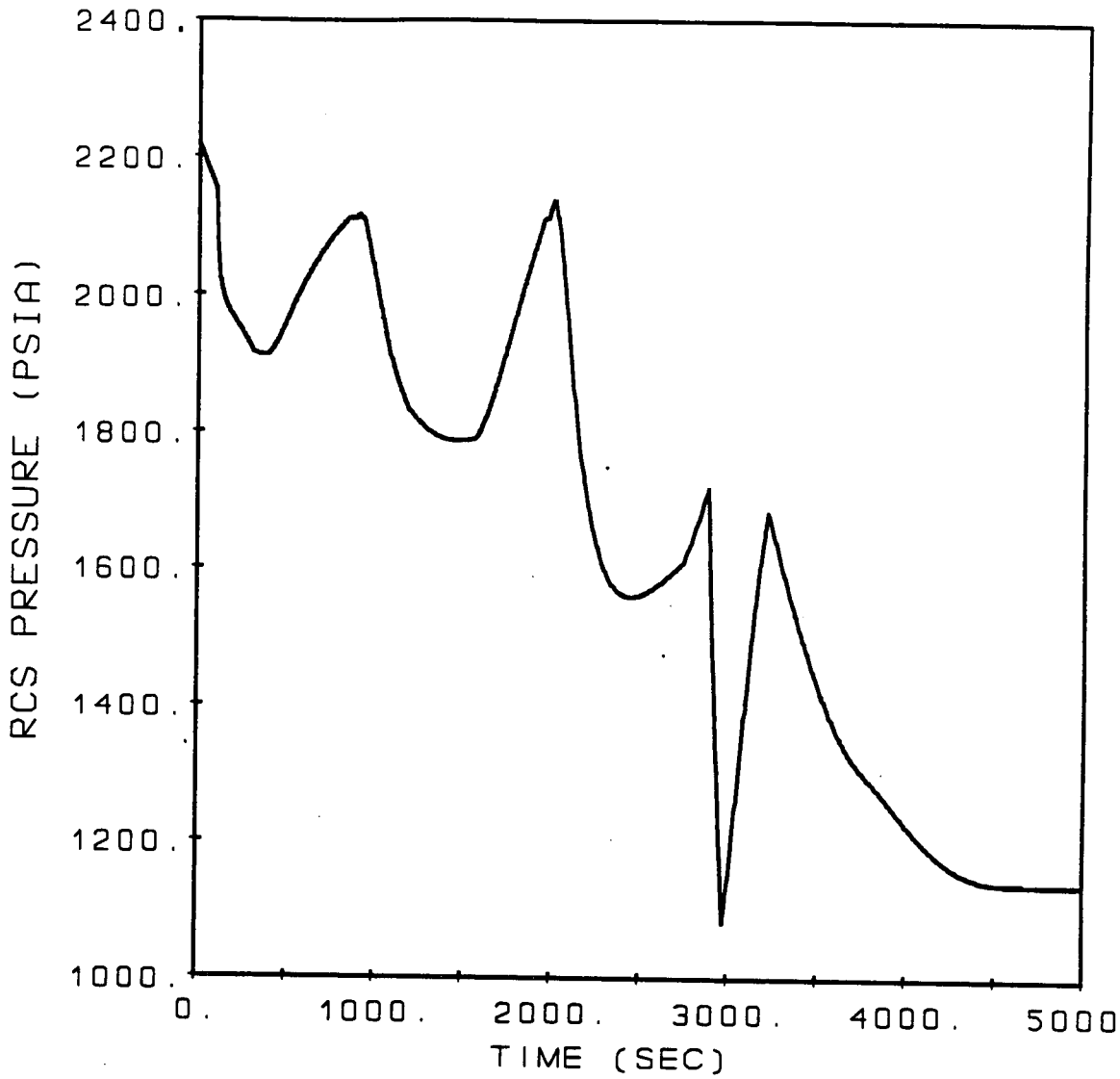

Figure III.1 RCS Pressure - Offsite Radiation Dose Analysis

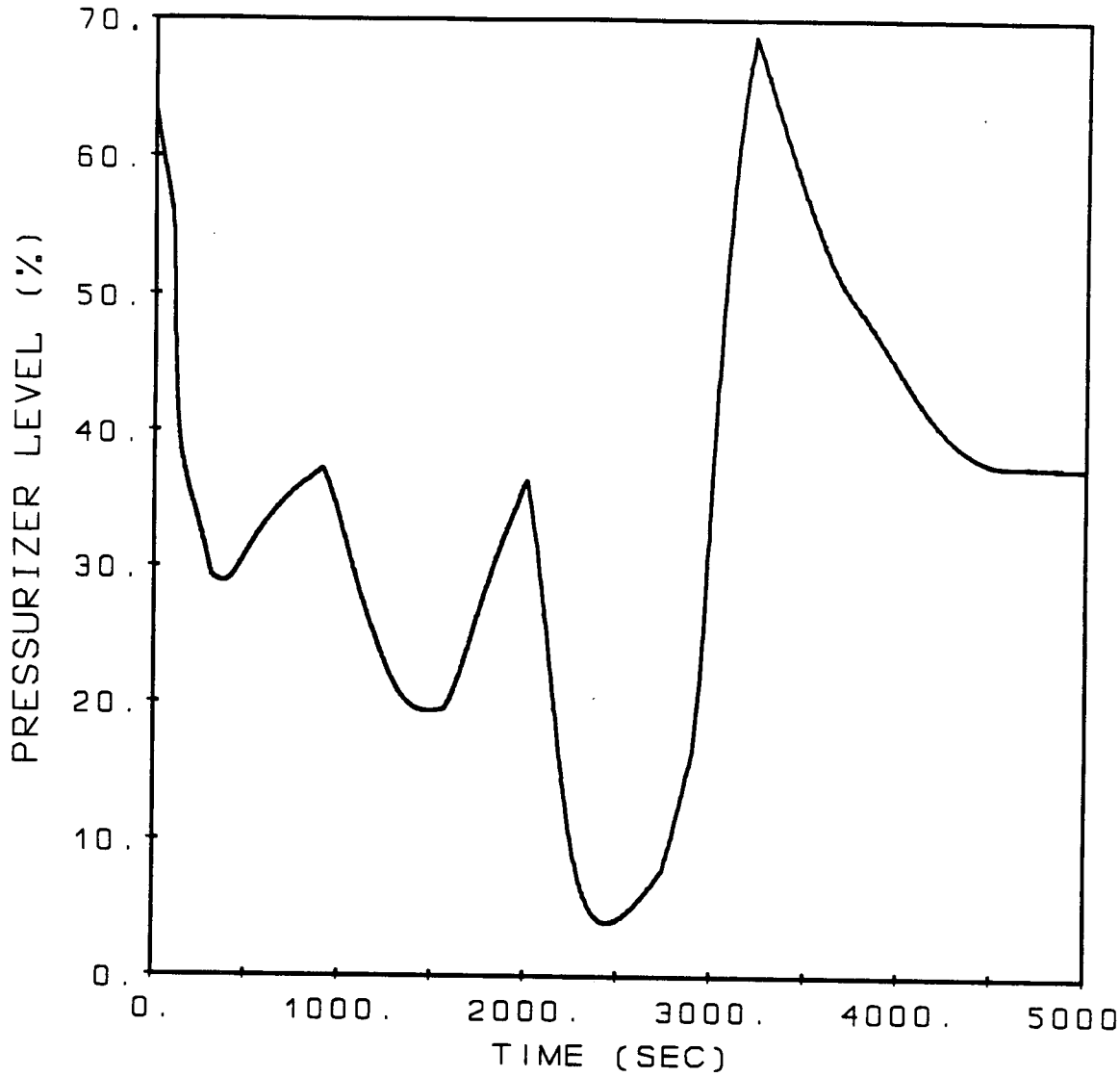

Figure III.3 Pressurizer Level - Offsite Radiation Dose Analysis

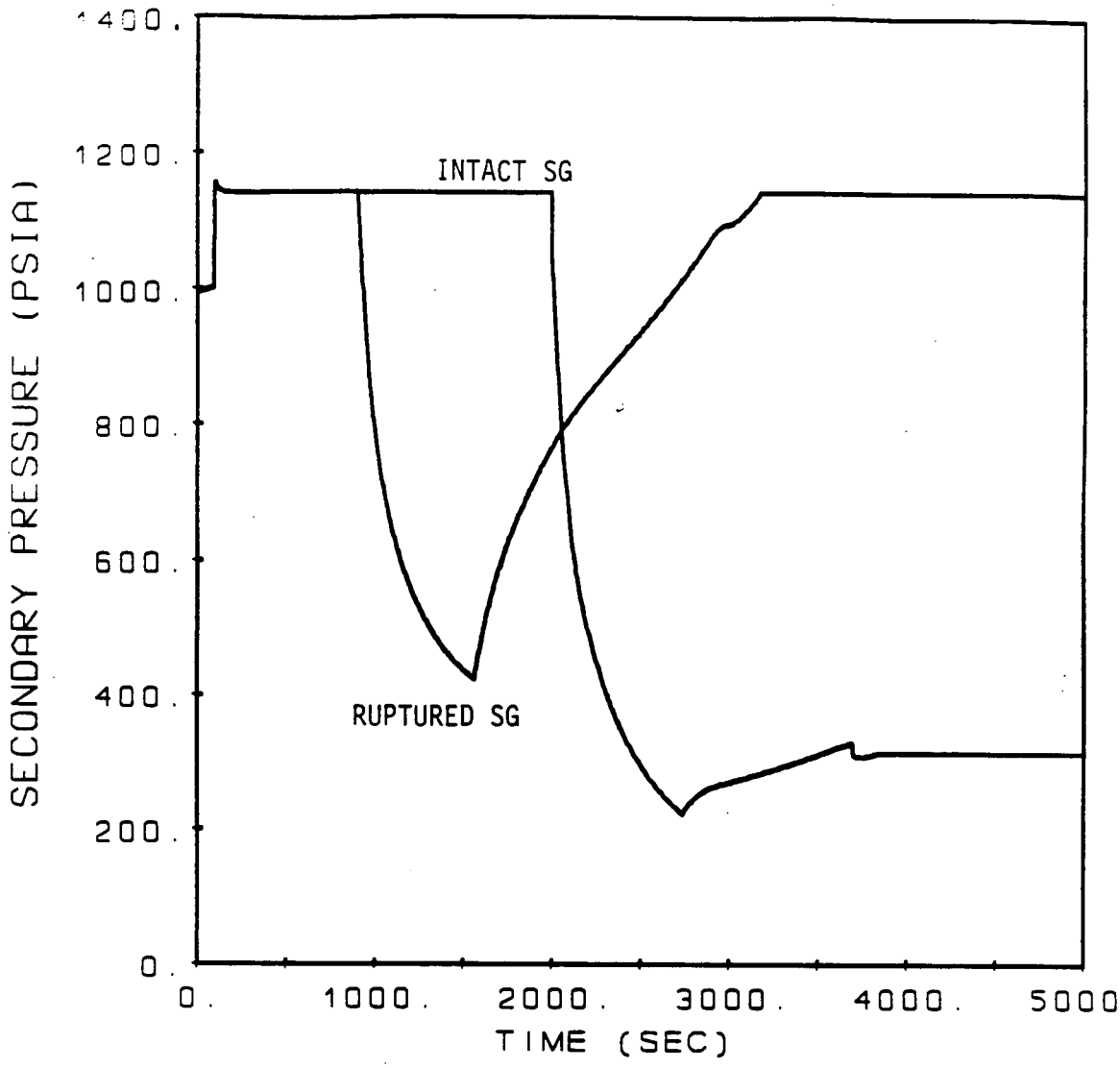

Figure III.2 Secondary Pressure - Offsite Radiation Dose Analysis

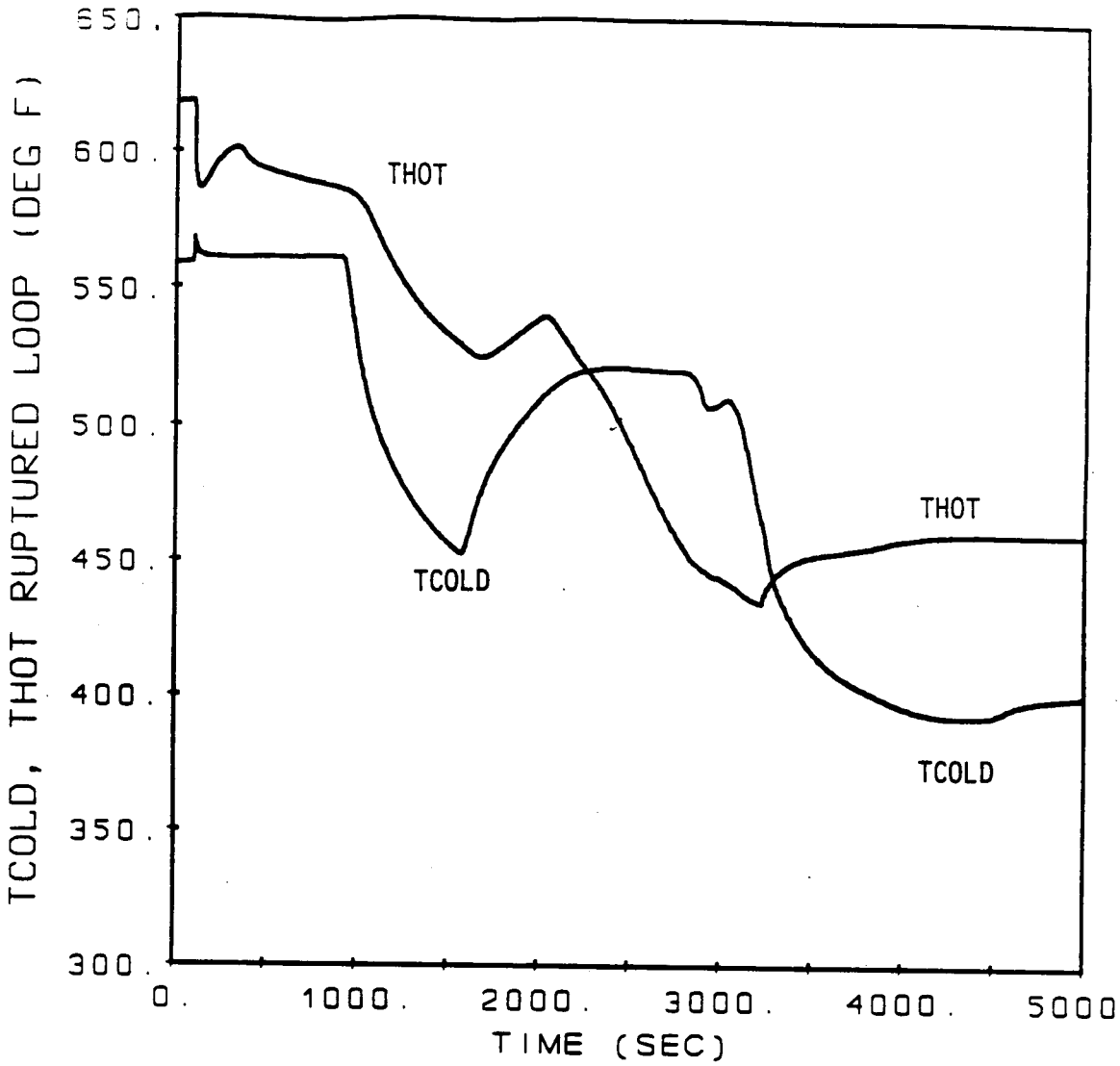

Figure III.4 Ruptured Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

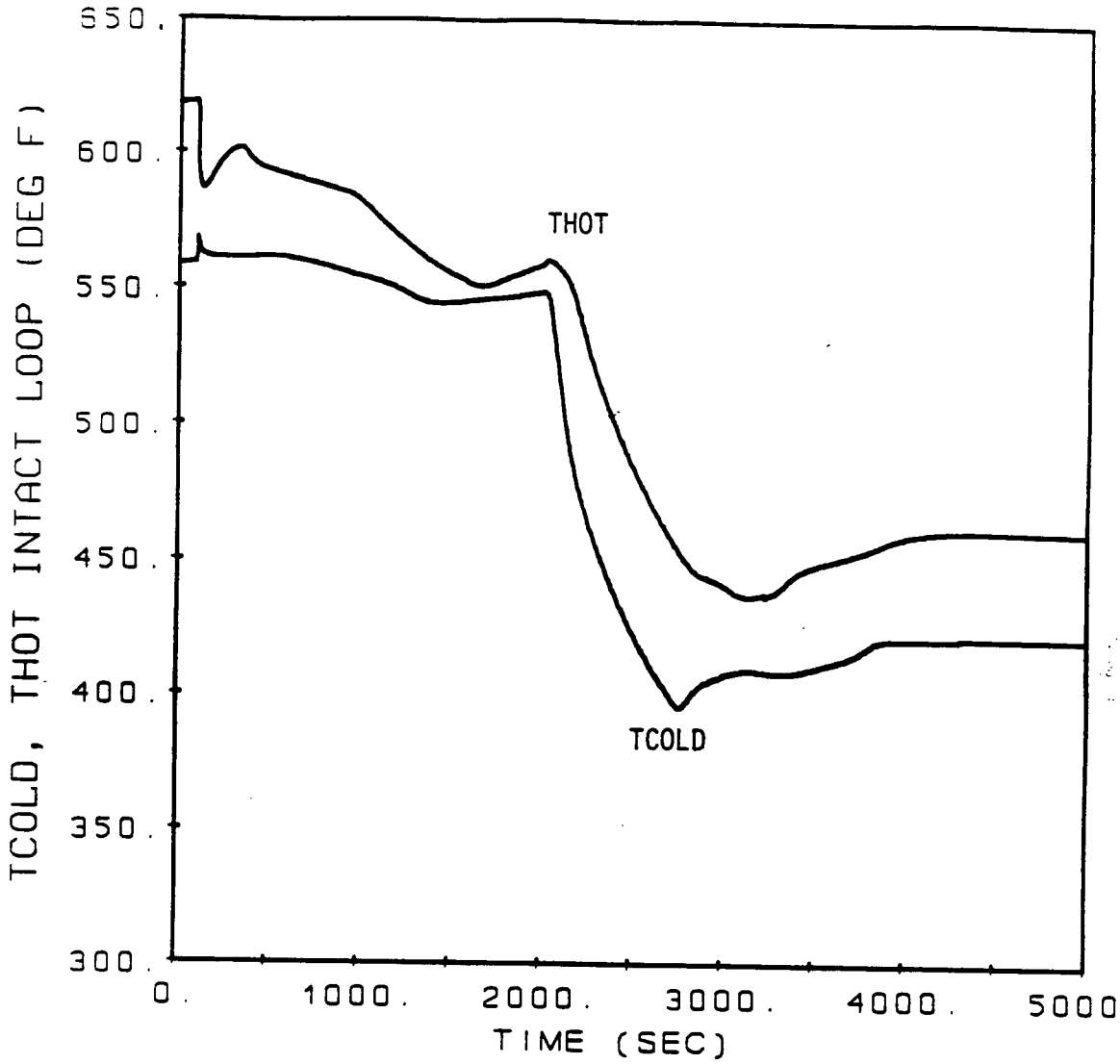

Figure III.5 Intact Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

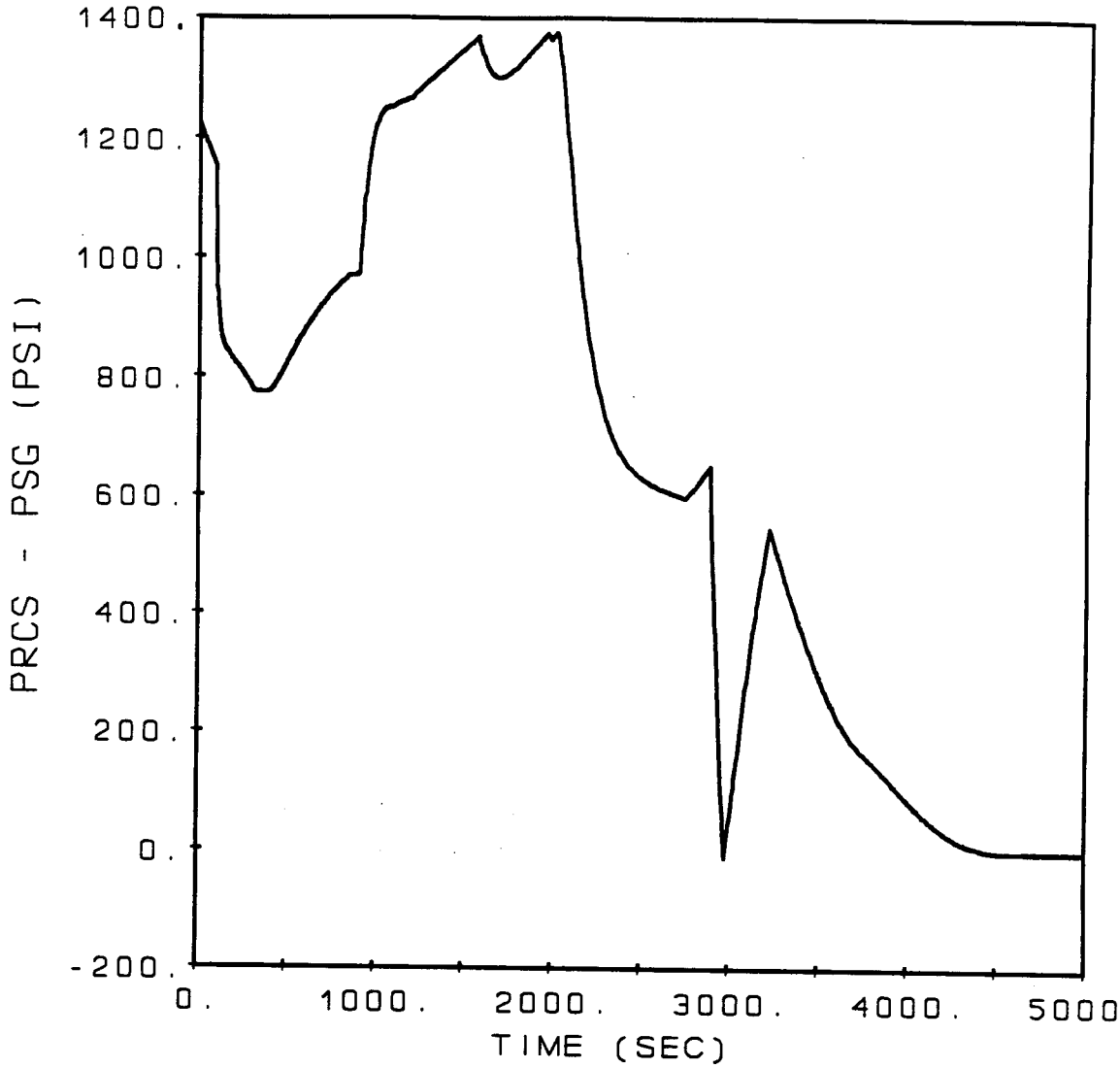

Figure III.6 Differential Pressure Between RCS and Ruptured SG - Offsite Radiation Dose Analysis

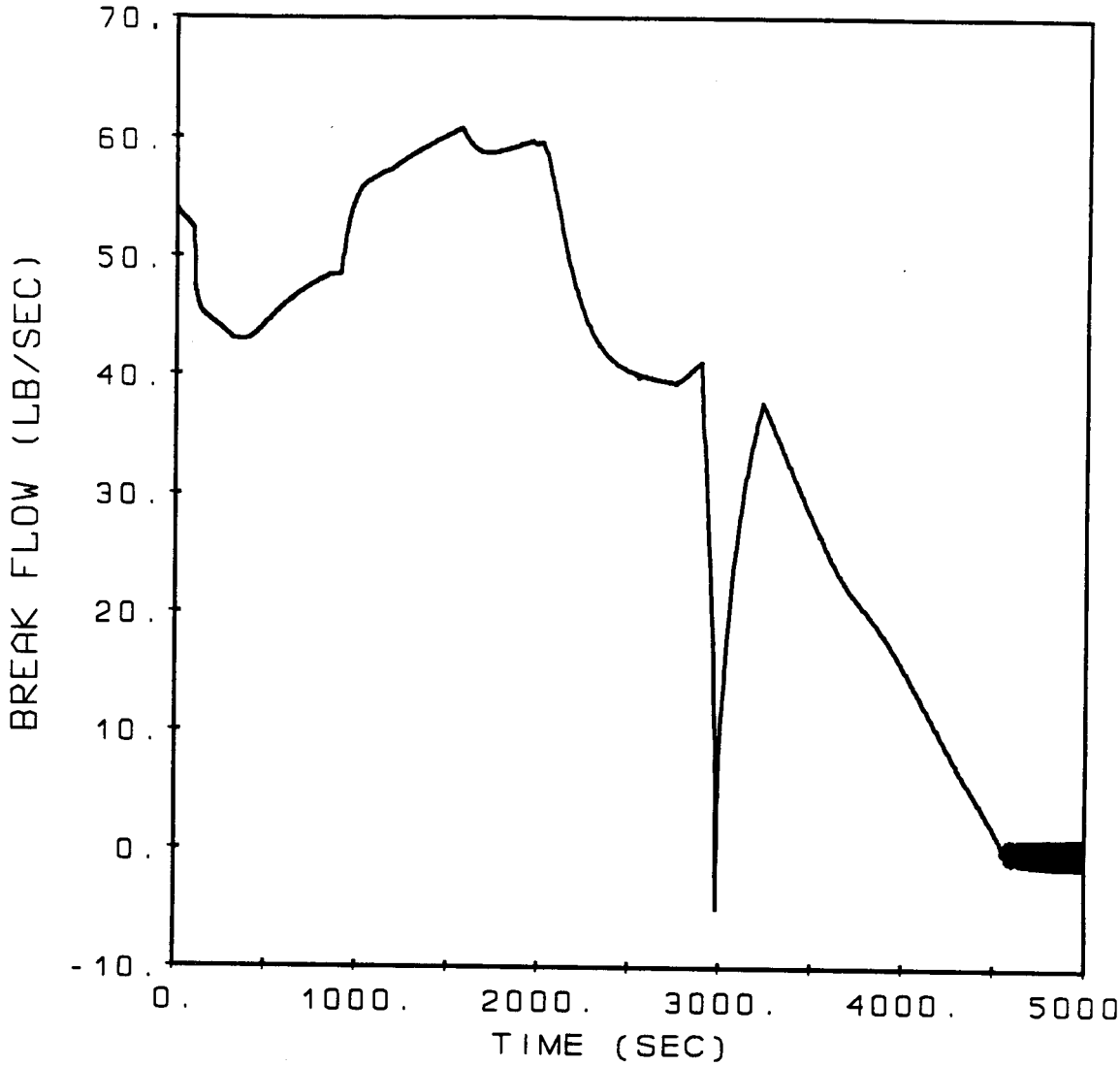

Figure III.7 Primary to Secondary Break Flow Rate - Offsite Radiation Dose Analysis

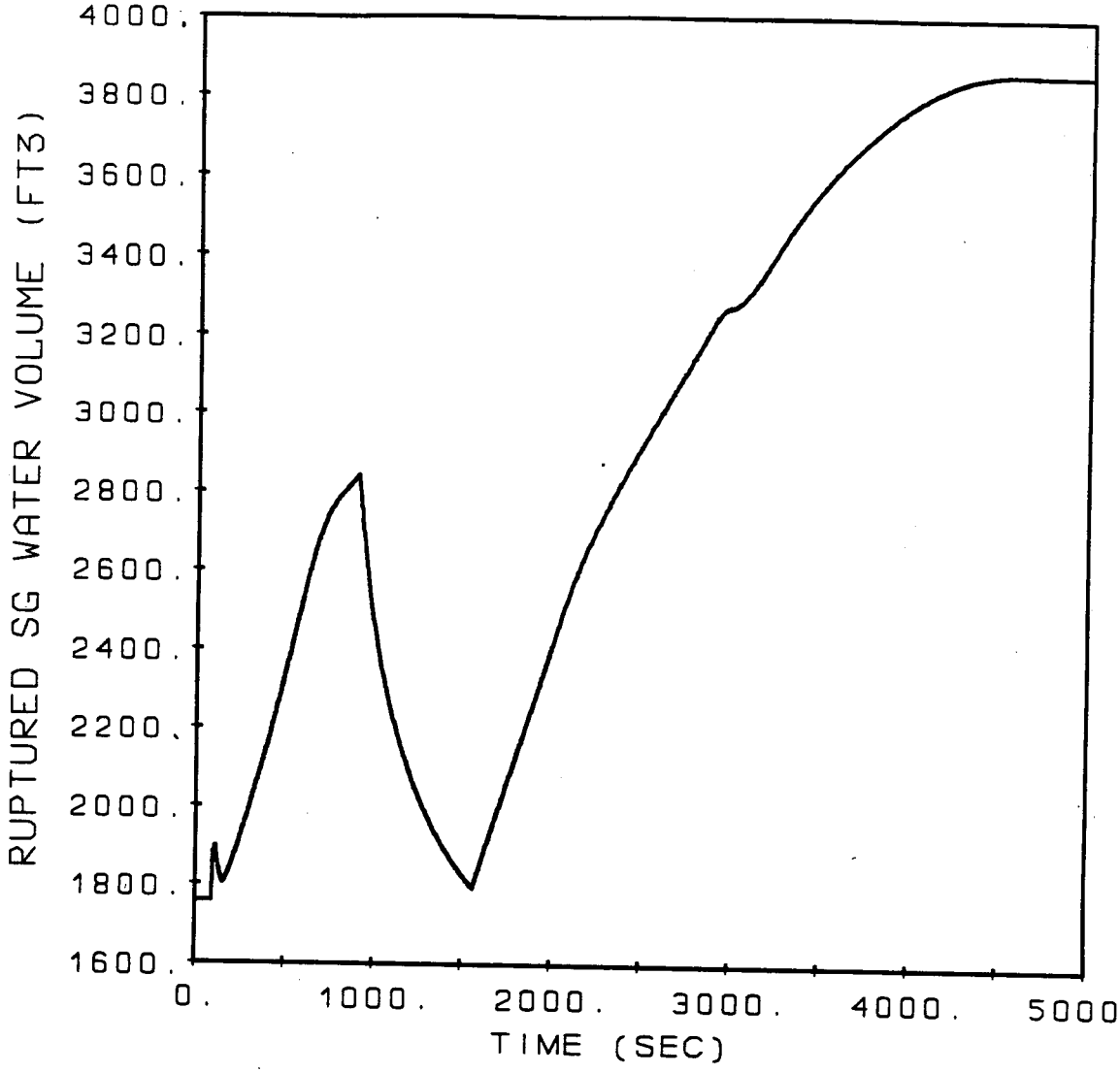

Figure III.8 Ruptured SG Water Volume - Offsite Radiation Dose Analysis

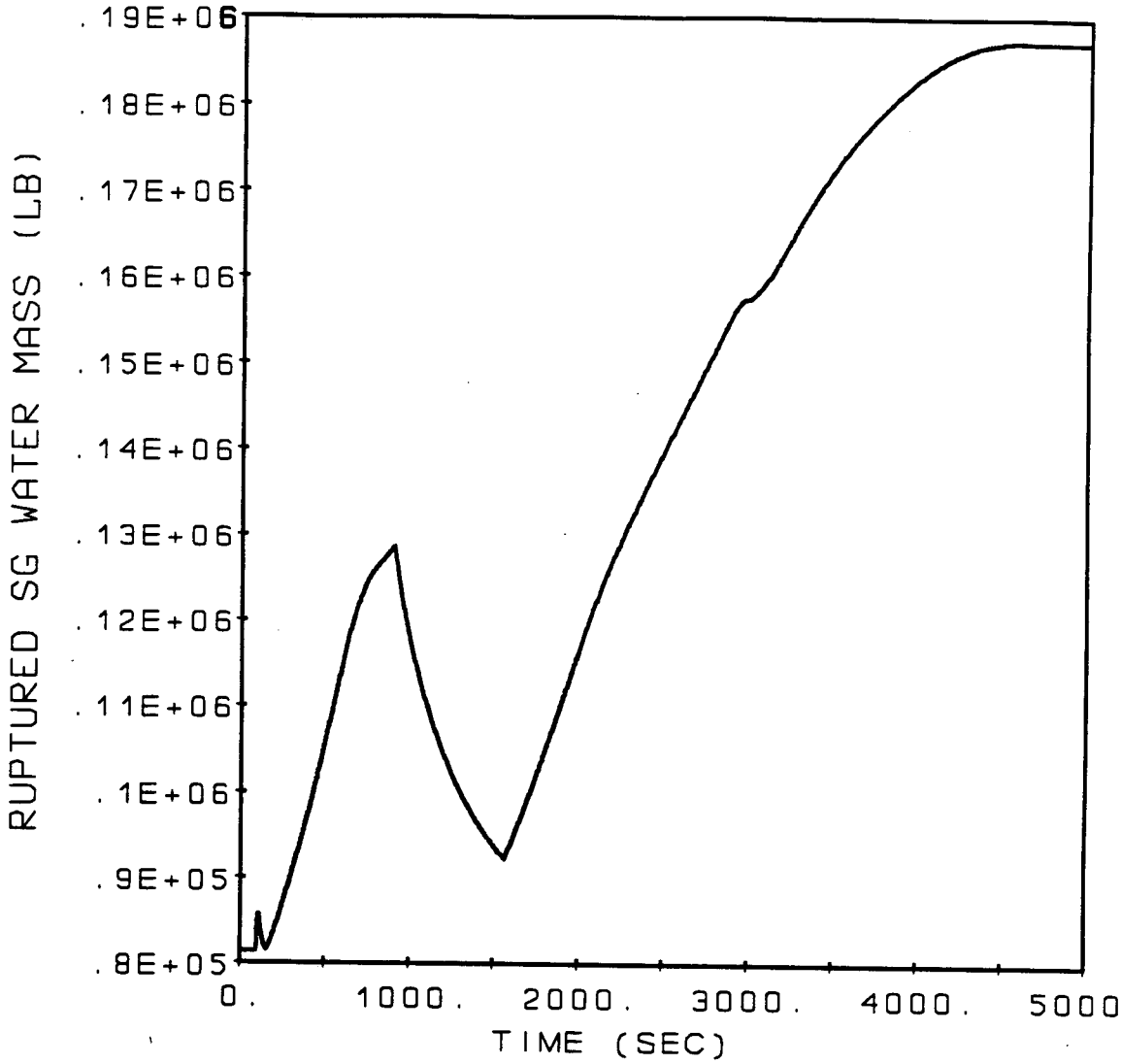

Figure III.9 Ruptured SG Water Mass - Offsite Radiation Dose Analysis

E. Mass Releases

The mass releases were determined for use in evaluating the site 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 site 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 analysis, the SGTR recovery actions in Watts Bar Units 1 and 2 EOP E-3 (ERG 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 Watts Bar Units 1 and 2 EOP ES-3.2 (ERG ES-3.3), POST-SGTR COOLDOWN BY RUPTURED STEAM GENERATOR DEPRESSURIZATION, 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 post-SGTR cooldown method using steam dump in Watts Bar Units 1 and 2 EOP ES-3.2 (ERG 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. Cooldown 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 coòling 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 Watts Bar Units 1 and 2 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 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. Cooldown 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 Watts Bar Units 1 and 2 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 [ ]^{a,c} until the cooldown to cold shutdown 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 395 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 generators 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 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 from 0 to 2 hours and from 2 to 8 hours are summarized in Table III.3. The results indicate that approximately 104,900 lbm of steam are released from the ruptured steam generator to the atmosphere in the first 2 hours. A total of 176,700 lbm of primary water is transferred to the secondary side of the ruptured steam generator before the break flow is terminated.

In addition to the mass releases, the following information was also developed for use in performing the offsite radiation dose analysis. The time-dependent fraction of

the tube rupture break flow that flashes to steam is presented in Figure III.12. As noted previously, the break flow flashing fraction was conservatively calculated by assuming that 100% of the break flow comes from the hot leg side of the break. The collapsed water level in both the ruptured and intact steam generators throughout the transient is shown in Figure III.13 for use in calculating the time-dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface.

TABLE III.2

WATTS BAR UNITS 1 AND 2 SGTR ANALYSIS

MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

	TIME PERIOD			
	0 - TRIP	TRIP - TTBRK	TTBRK - T2HRS	T2HRS - TRHR
Ruptured SG				
- Condenser	102,100	0	0	0
- Atmosphere	0	104,900	0	31,600
- Feedwater	97,000	34,200	0	0
Intact SG				
- Condenser	303,500	0	0	0
- Atmosphere	0	291,000	225,400	935,300
- Feedwater	303,500	440,700	237,000	943,800
Break Flow	5,000	171,700	0	0

TRIP = Time of reactor trip = 94 sec.

TTBRK = Time when break flow is terminated = 4548 sec.

T2HRS = Time at 2 hours = 7200 sec.

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

TABLE III.3

WATTS BAR UNITS 1 AND 2 SGTR ANALYSIS

SUMMARIZED MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

	TIME PERIOD	
	0 - 2HRS	2HRS - 8HRS
Ruptured SG		
- Condenser	102,100	0
- Atmosphere	104,900	31,600
- Feedwater	131,200	0
Intact SG		
- Condenser	303,500	0
- Atmosphere	516,400	935,300
- Feedwater	981,200	943,800
Break Flow	176,700	0

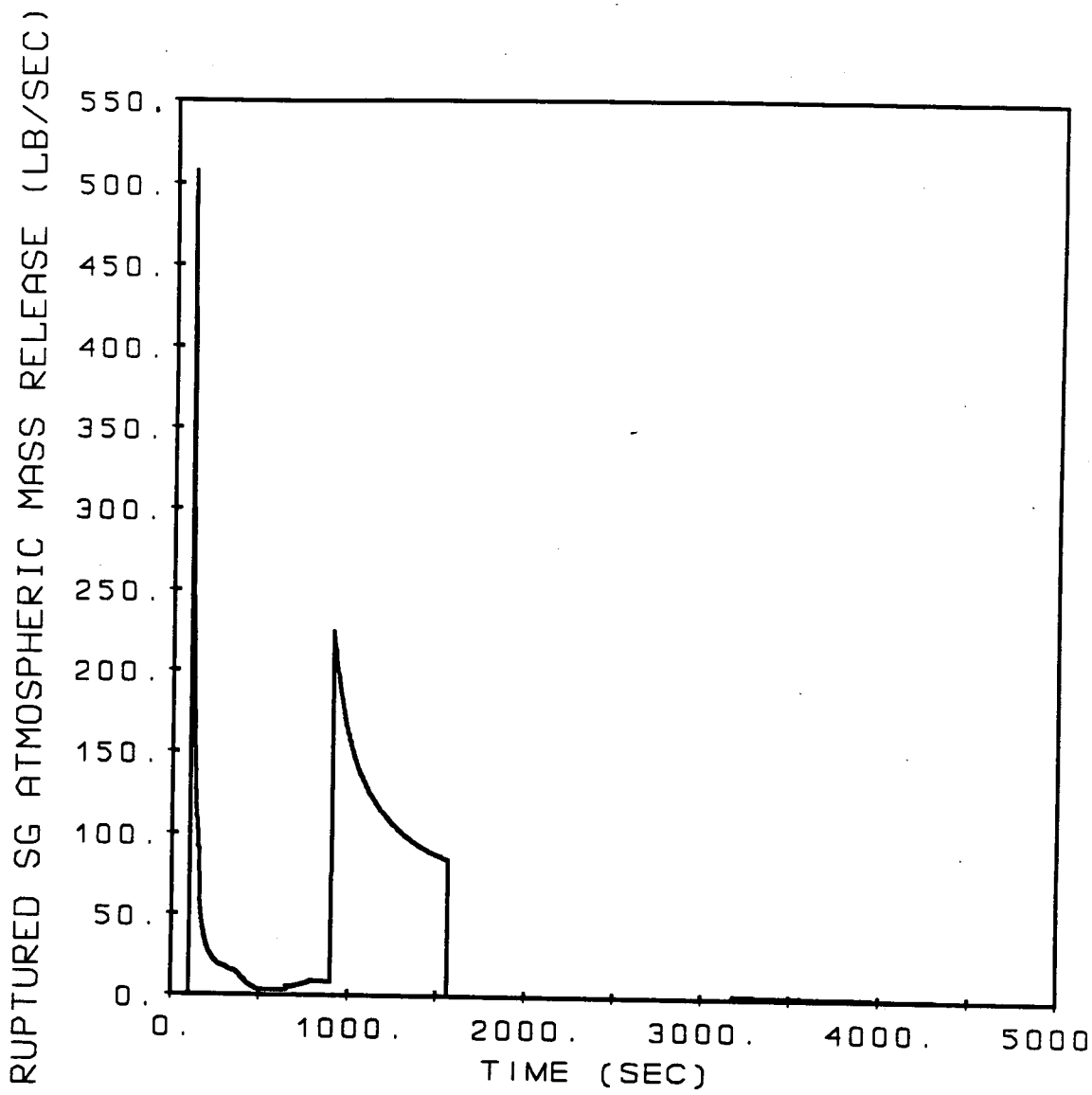

Figure III.10 Ruptured SG Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

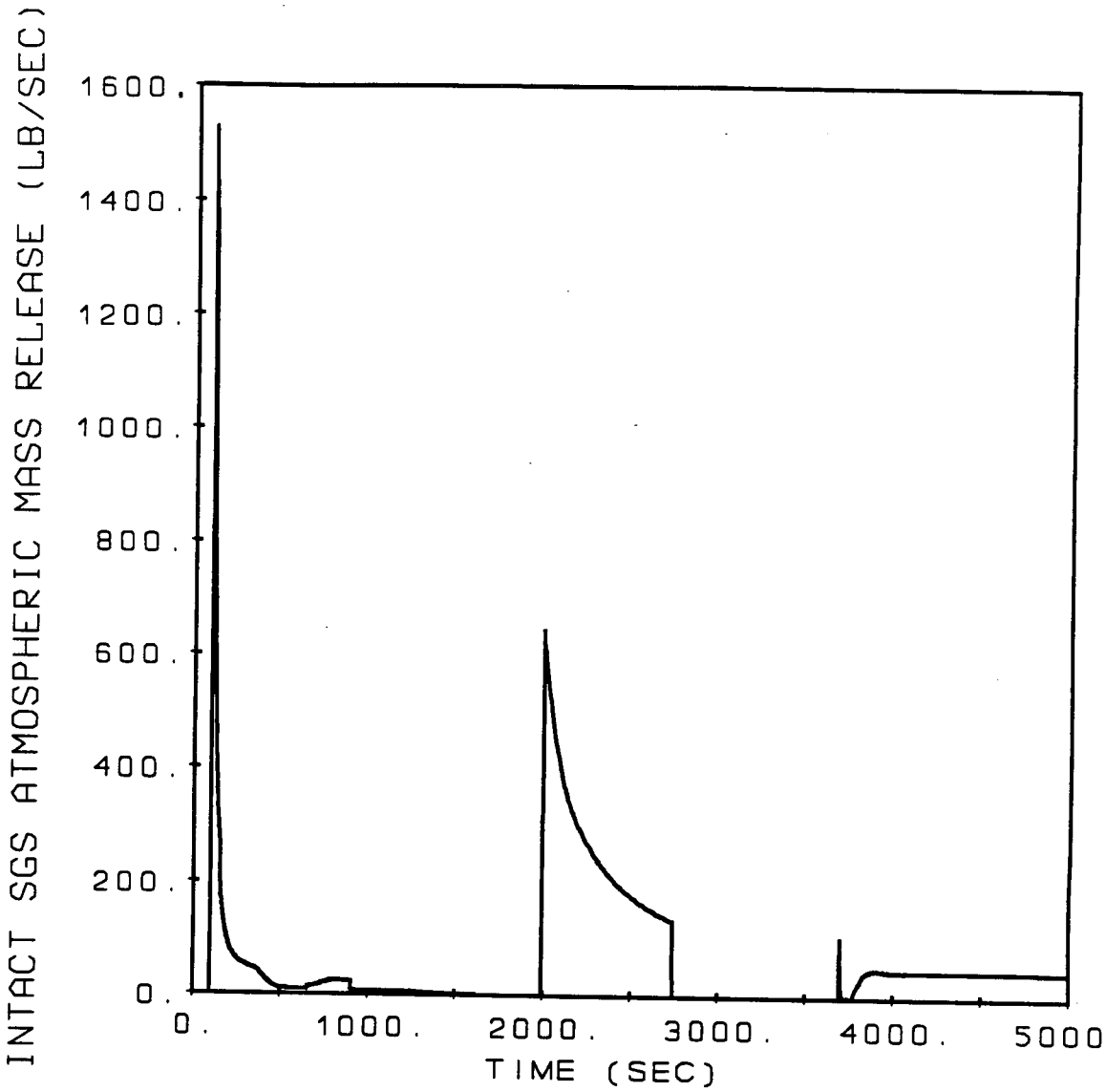

Figure III.11 Intact SGs Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

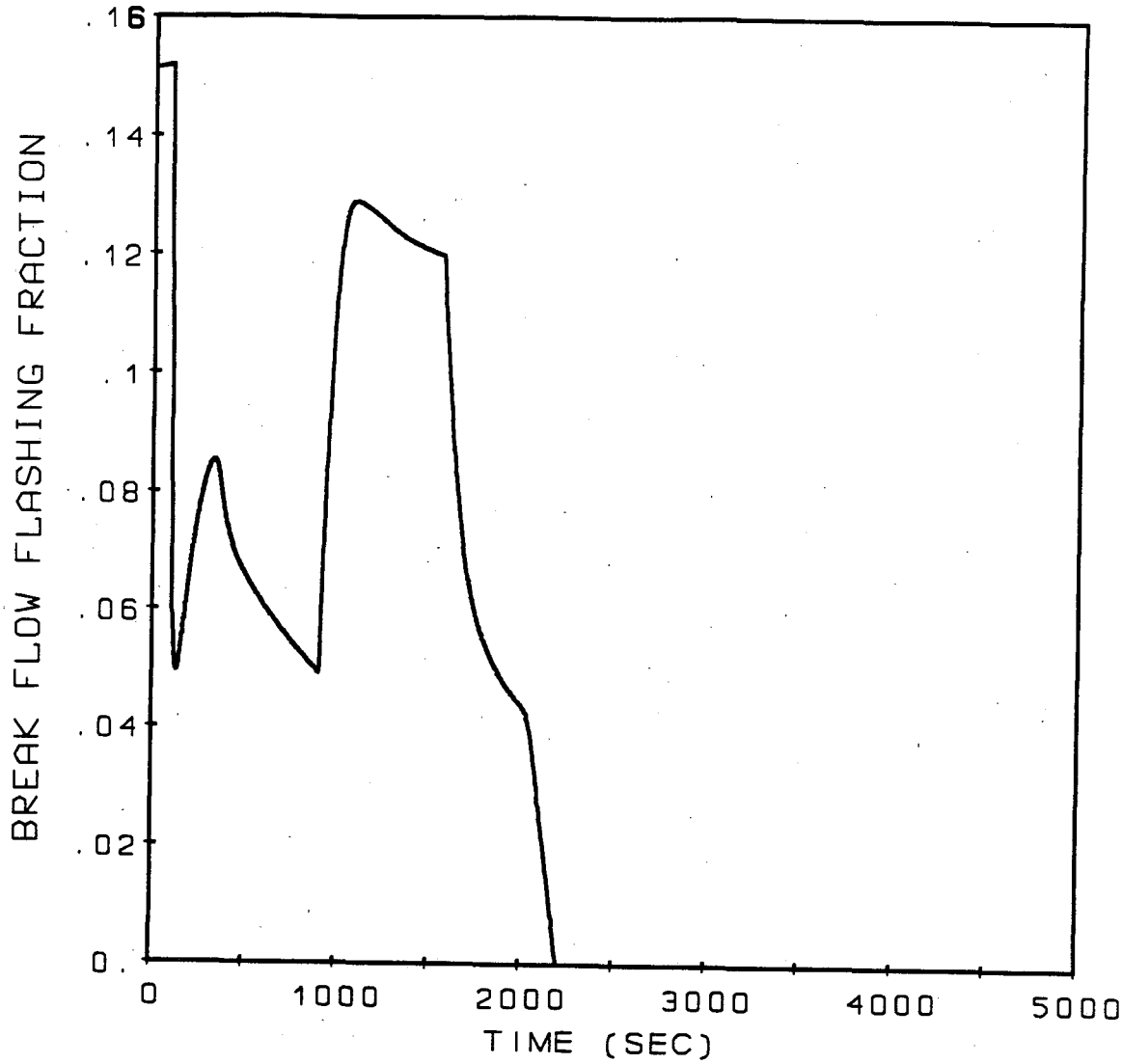

Figure III.12 Break Flow Flashing Fraction -
Offsite Radiation Dose Analysis

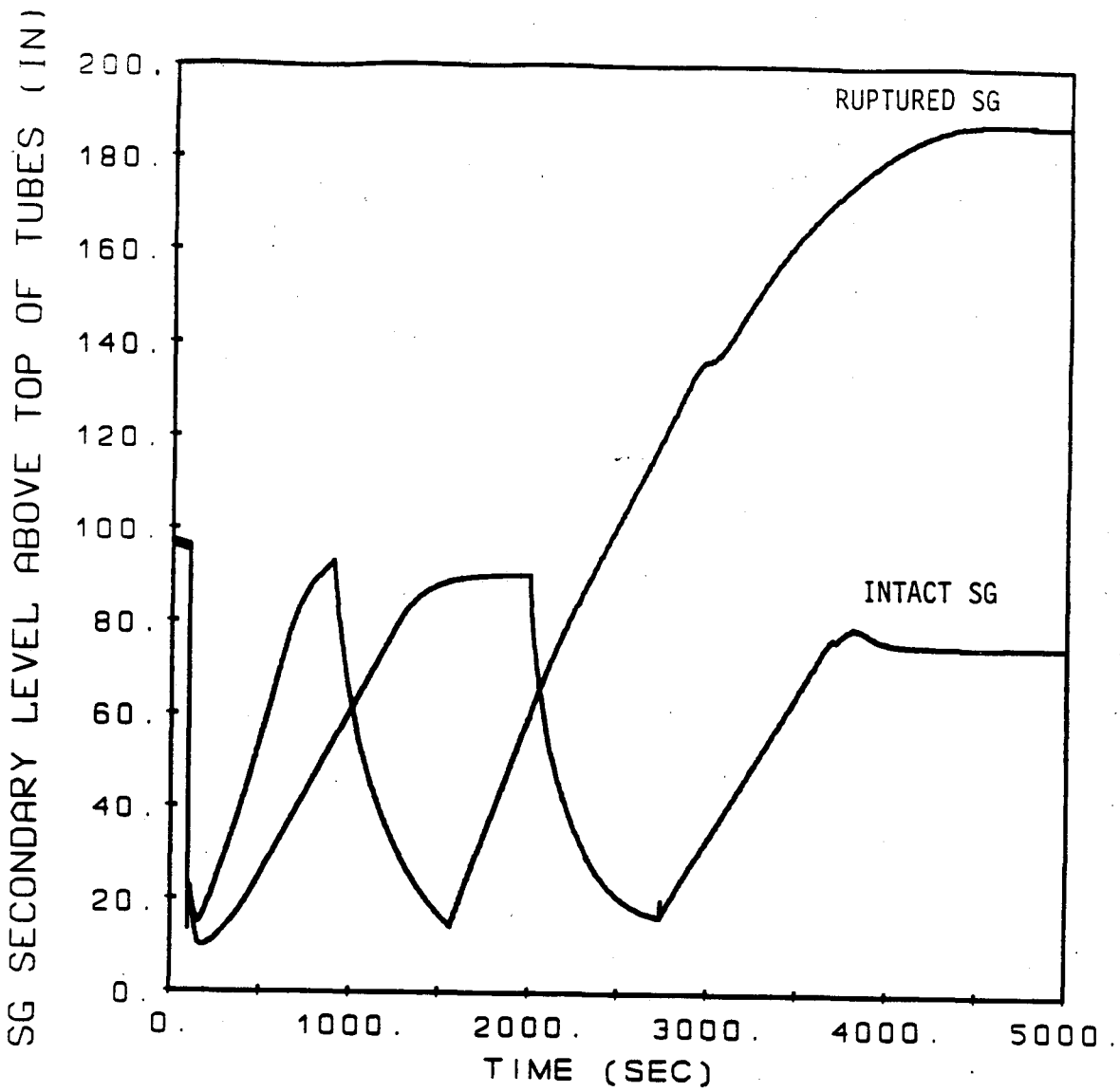

Figure III.13 SG Water Level Above Top of Tubes - Offsite Radiation Dose Analysis

IV. CONCLUSION

An evaluation has been performed for a design basis SGTR for Watts Bar Nuclear Power Plant Units 1 and 2 to demonstrate that the potential consequences are acceptable. An analysis was performed to demonstrate margin to steam generator overfill with 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, a thermal and hydraulic analysis was also performed to determine the input for the offsite radiation doses assuming the limiting single failure for offsite doses. For this analysis, it was assumed that the ruptured steam generator PORV fails open at the time the ruptured steam generator is isolated, and that the failed-open valve must be isolated by locally closing the associated block valve. The primary to secondary break flow and the mass releases to the atmosphere were determined for this case for use in calculating the offsite radiation doses for a design basis SGTR.

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.

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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.