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WCAP-12738

LOFTTR2 ANALYSIS FOR A STEAM GENERATOR TUBE RUPTURE FOR BEAVER VALLEY POWER STATION UNIT 2

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OCTOBER 1990

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0644D:1D/102090

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (Cont)

LIST OF FIGURES

LIST OF FIGURES (Cont)

I. INTRODUCTION

An evaluation for a design basis steam generator tube rupture (SGTR) event has been performed for the Beaver Valley Power Station Unit 2 (BVPS Unit 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 BVPS Unit 2 employs a Westinghouse pressurized water reactor (PWR) unit rated at 2660 MWt. The reactor coolant system has three reactor coolant loops with Model 51 steam generators. The SGTR evaluation is based on the current BVPS Unit 2 plant design which reflects the approved changes which have been incorporated since the plant was initially licensed. The evaluation is applicable for BVPS Unit 2 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 BVPS Unit 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 BVPS Unit 2. The LOFTTR1 program was developed as part of the revised SGTR analysis methodology and was used for the SGTR evaluation 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 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 BVPS Unit 2 Emergency Operating Procedures (EOPs), which were developed from the Westinghouse Owners Group Emergency Response Guidelines (ERGs). In subsequent references to the BVPS Unit 2 EOPs throughout the text, the specific BVPS Unit 2 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 those 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 the [ ]^{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 demonstrat s that there is margin to steam generator overfill for BVPS Unit 2.

Since steam generator overfill does not occur, the results of the offsite radiation dose analysis represent the limiting consequences for BVPS Unit 2. For the analysis of the offsite radiation doses, the ruptured steam generator atmospheric steam dump valve 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 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 Standard Technical Specification limit prior to the accident. The results of this evaluation show that the offsite doses for BVPS Unit 2 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 BVPS Unit 2. The analysis was performed using the LOFTTR2 program and the methodology developed in Reference 1, and using the plant specific parameters for BVPS Unit 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 [ ]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.

For the three-loop reference plant in Reference 1, the most limiting single failure with respect to steam generator overfill was determined to be a [ ]a,c The failure of the [ ]a,c causing an increase in total primary to secondary leakage. Consequently, more water will accumulate in the ruptured steam generator.

For BVPS Unit 2, the RCS cooldown can be performed by releasing steam from the intact steam generators using the associated atmospheric steam dump valves or using the residual heat release valve. However, because of the [ ]a,c a single failure of a [ ]a,c Since the [ ]a,c

[ ]a,c the limiting single failure for the BVPS Unit 2 margin to overfill analysis is assumed to be the [ ]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 BVPS Unit 2 parameters in the LOFTTR2 analysis to determine the margin to steam generator overfill for BVPS Unit 2 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. 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, if reactor trip occurs prior to [ ]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 auxiliary feedwater (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 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 BVPS Unit 2 reactor protection system, and turbine runback was simulated [ ]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. 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. AFW System Operation

For the reference plant analysis in Reference 1, 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 motor-driven and turbine-driven AFW pumps. The SI signal will also result in automatic isolation of the main feedwater system and actuation of the motor-driven AFW pumps for the reference plant. For the reference plant analysis, it was conservatively 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 isolate or throttle AFW flow to control steam generator water level in accordance with the emergency procedures.

For the BVPS Unit 2 analysis, reactor trip also occurs on [ ][a,c] and SI is initiated on low pressurizer pressure at approximately [ ][a,c] after reactor trip. The time of reactor trip and SI initiation was determined by modeling the BVPS Unit 2 reactor protection system, and the actuation of the AFW system was based on the time of [ ][a,c] 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. It was assumed that flow from both of the motor-driven AFW pumps and the turbine driven AFW pump is initiated at [ ][a,c] The AFW flow assumed for the analysis is 310 gpm per steam generator, since cavitating venturi flow elements are provided in the AFW supply lines to each steam generator which are designed to limit the flow to 310 gpm.

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 BVPS Unit 2 Plant Operating Manual, Chapter 53A, EOP E-3 (ERG 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 air ejector discharge radiation monitor, steam generator blowdown sample radiation monitor, or main steamline 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 water sample, high radiation indication on a main steamline radiation monitor, or high radiation from a steam generator blowdown line. For an SGTR that results in a high power reactor trip as assumed in this analysis, the steam generator water level will decrease to near the bottom of the narrow range scale for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will increase more rapidly 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 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. For the reference plant analysis in Reference 1, it was assumed that the ruptured steam generator will be isolated when [

   ][a,c] For BVPS Unit 2, the steam generator narrow range level corresponding to being just on span is 5% and the comparable operator action time is 11.75 minutes. Thus, applying the Reference 1 methodology for the BVPS Unit 2 analysis, the ruptured steam generator was assumed to be isolated at 27.5% narrow range level or at 11.75 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 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 atmospheric steam dump valves or the residual heat release valve to release steam from the intact steam generators.

   For BVPS Unit 2, the atmospheric steam dump valves are [ ][a,c] while the residual heat release valve is [ ][a,c] The atmospheric steam dump valves are the first alternative to perform the RCS cooldown, but if the power supply to these valves is not available, the BVPS Unit 2 EOP E-3 (ERG E-3) includes provisions to use the residual heat release valve to perform the cooldown. It is noted that a connection is provided from the steam line for each of the steam generators to the residual heat release valve, with a normally open, manual isolation valve in the connecting line. Thus, if the residual heat release valve is to be used to release steam from only the intact steam generators, an operator must be dispatched to locally close the isolation valve from the ruptured steam generator to the residual heat release valve. The BVPS Unit 2 EOP E-3 (ERG E-3) includes instructions to dispatch an operator for this purpose by the time when the ruptured steam generator is identified and isolated. Since the time required to isolate the ruptured steam generator from the residual heat release valve is less than the time delay to initiate the RCS cooldown, the cooldown can be performed using either the atmospheric steam dump valves or the residual heat release valve without any additional time penalty. Since the residual heat release valve has [ ]a,c than the two intact steam generator atmospheric steam dump valves, the most limiting single failure for the margin to overfill analysis is assumed to be the [ ]a,c Because offsite power is assumed to be lost and a single failure of the [ ]a,c was assumed for the BVPS Unit 2 analysis, the cooldown was performed by dumping steam via the [ ]a,c

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 reinitiation 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. Duquesne Light Company has determined the corresponding operator action times to perform these operations for BVPS Unit 2. The operator actions and the corresponding operator action times used for the BVPS Unit 2 analysis are listed in Table II.1.

TABLE II.1

BVPS UNIT 2 SGTR ANALYSIS

OPERATOR ACTION TIMES FOR DESIGN BASIS ANALYSIS

Action	Time Intervals
Identify and isolate ruptured SG	11.75 min or LOFTTR2 calculated time from event initiation to reach 27.5% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	9 min
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	2.5 min
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	1.25 min
SI termination and pressure equalization	Calculated by LOFTTR2

D. Transient Description

The LOFTTR2 analysis results for the BVPS Unit 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 a low pressurizer pressure trip signal at approximately 154 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 atmospheric steam dump valves (and safety valves if their setpoints are reached) lift to dissipate the energy, as shown in Figure II.3.

The pressurizer level and RCS pressure decrease more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the leak flow continues to deplete primary inventory. The decrease in RCS inventory results in a low pressurizer pressure SI signal at approximately 161 seconds. The main feedwater flow will be isolated and AFW flow will be automatically initiated following SI actuation. 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 subsequently stabilize at 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 temperatures trend 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 cool down the RCS.

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 27.5% on the ruptured steam generator or at 11.75 minutes after initiation of the SGTR, whichever is longer. For the BVPS Unit 2 analysis, the time to reach 27.5% is less than 11.75 minutes, and thus the ruptured steam generator is assumed to be isolated at 11.75 minutes. However, because of the computer program limitations for simulating the operator actions, the ruptured steam generator was isolated three seconds later at 708 seconds.

2. Cool down the RCS to Establish Subcooling Margin

   After isolation of the ruptured steam generator, there is a 9 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, and the limiting single failure is the failure of the [ ]a,c the RCS is cooled by dumping steam to the atmosphere using the [ ]a,c The [ ]a,c are assumed to be opened at 1250 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 2098 seconds, it is assumed that the operator closes the [ ]a,c 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 decreases during this cooldown process due to shrinkage of the reactor coolant and goes offscale low as shown in Figure II.1. The RCS pressure also decreases due to the RCS cooldown as shown in Figure II.2.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a 2.5 minute operator action time is included prior to the RCS depressurization. The actual delay time used in the analysis is two seconds longer because of the computer program limitations for simulating operator actions. 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 2250 seconds and continued until any of the following conditions are satisfied: pressurizer level is greater than 76%, 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 4%. For this case, the RCS depressurization is terminated because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 4%. 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 4%. For the BVPS Unit 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 1.25 minutes was assumed prior to initiation of SI termination. Since the above requirements are satisfied, SI termination actions were performed at this time by closing off the SI flow path. After SI termination, the RCS pressure begins to decrease as shown in Figure II.2. The [ ] a,c are also opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the [ ] a,c 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. 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 less than the total steam generator volume of 5759 ft^3. Therefore, it is concluded that overfill of the ruptured steam generator will not occur for a design basis SGTR for BVPS Unit 2.

TABLE II.2

BVPS UNIT 2 SGTR ANALYSIS

SEQUENCE OF EVENTS

MARGIN TO OVERFILL ANALYSIS

EVENT	Time (sec)
SG Tube Rupture	0
Reactor Trip	154
Safety Injection	161
Ruptured SG Isolated	708
RCS Cooldown Initiated	1250
RCS Cooldown Terminated	2098
RCS Depressurization Initiated	2250
RCS Depressurization Terminated	2364
SI Terminated	2440
Break Flow Terminated	2880

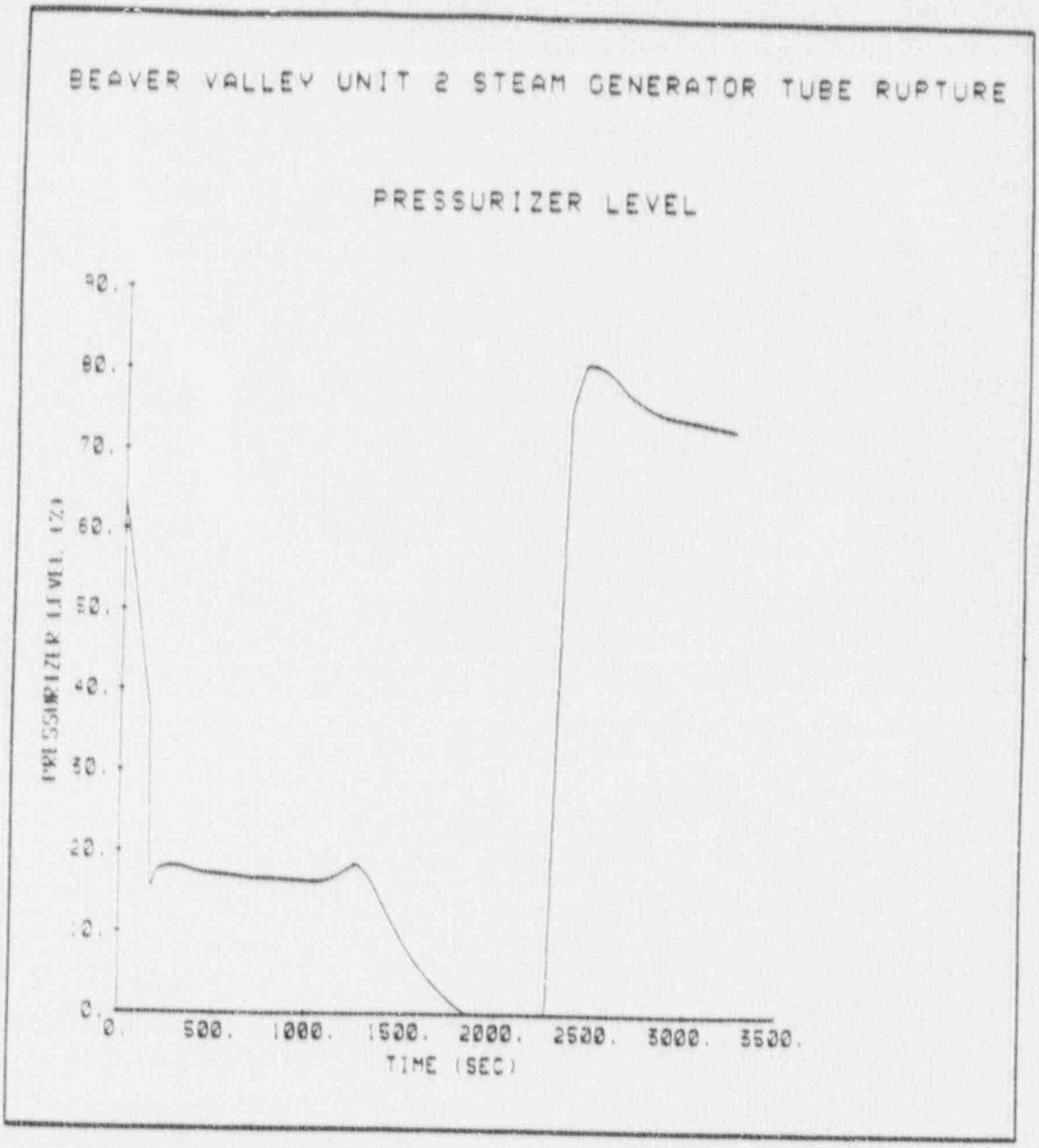

Figure II.1 Pressurizer Level - Margin to Overfill Analysis

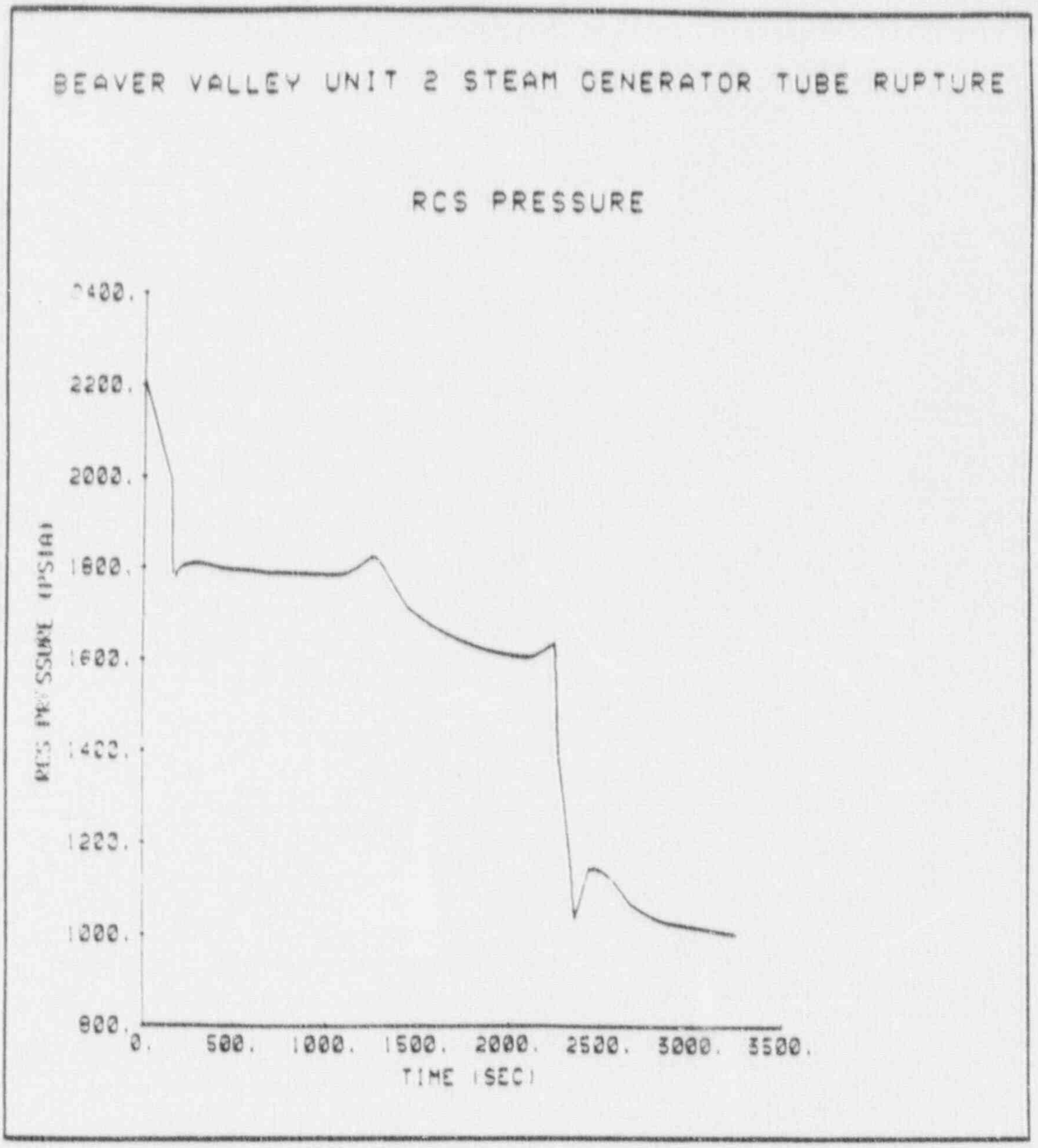

Figure II.2 RCS Pressure - Margin to Overfill Analysis

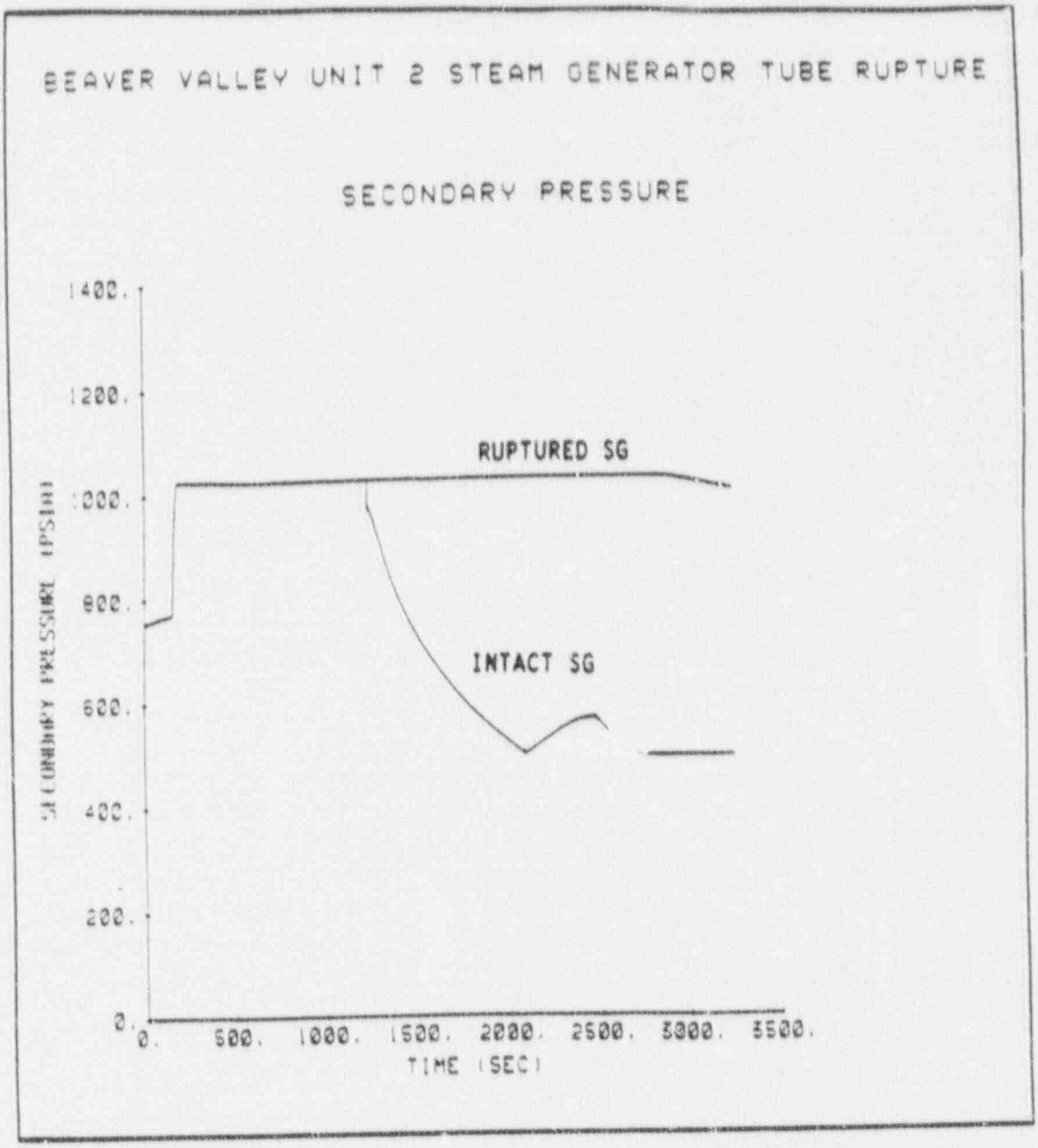

Figure II.3 Secondary Pressure - Margin to Overfill Analysis

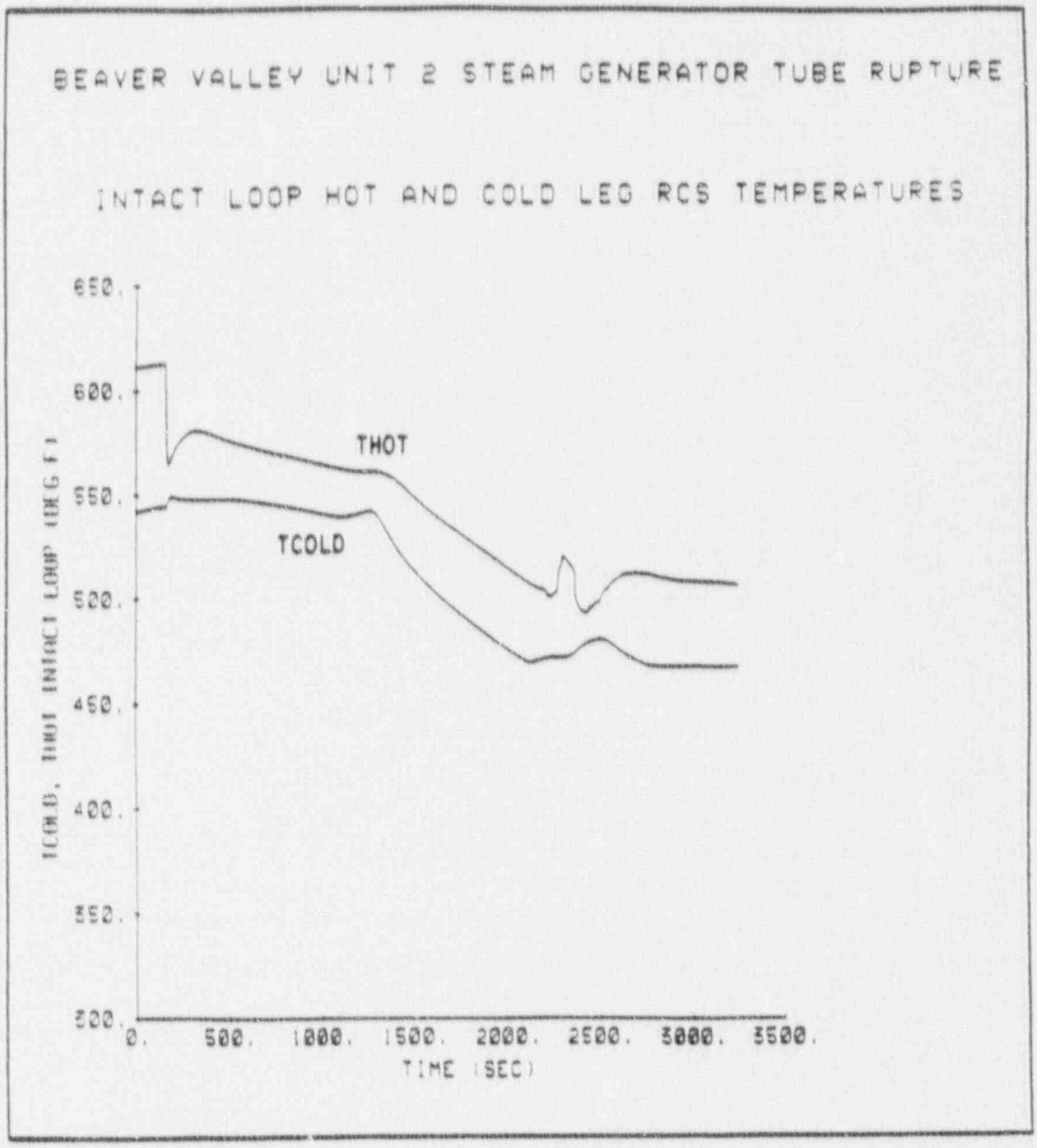

Figure II.4 Intact Loop Hot and Cold Leg RCS Temperatures - Margin to Overfill Analysis

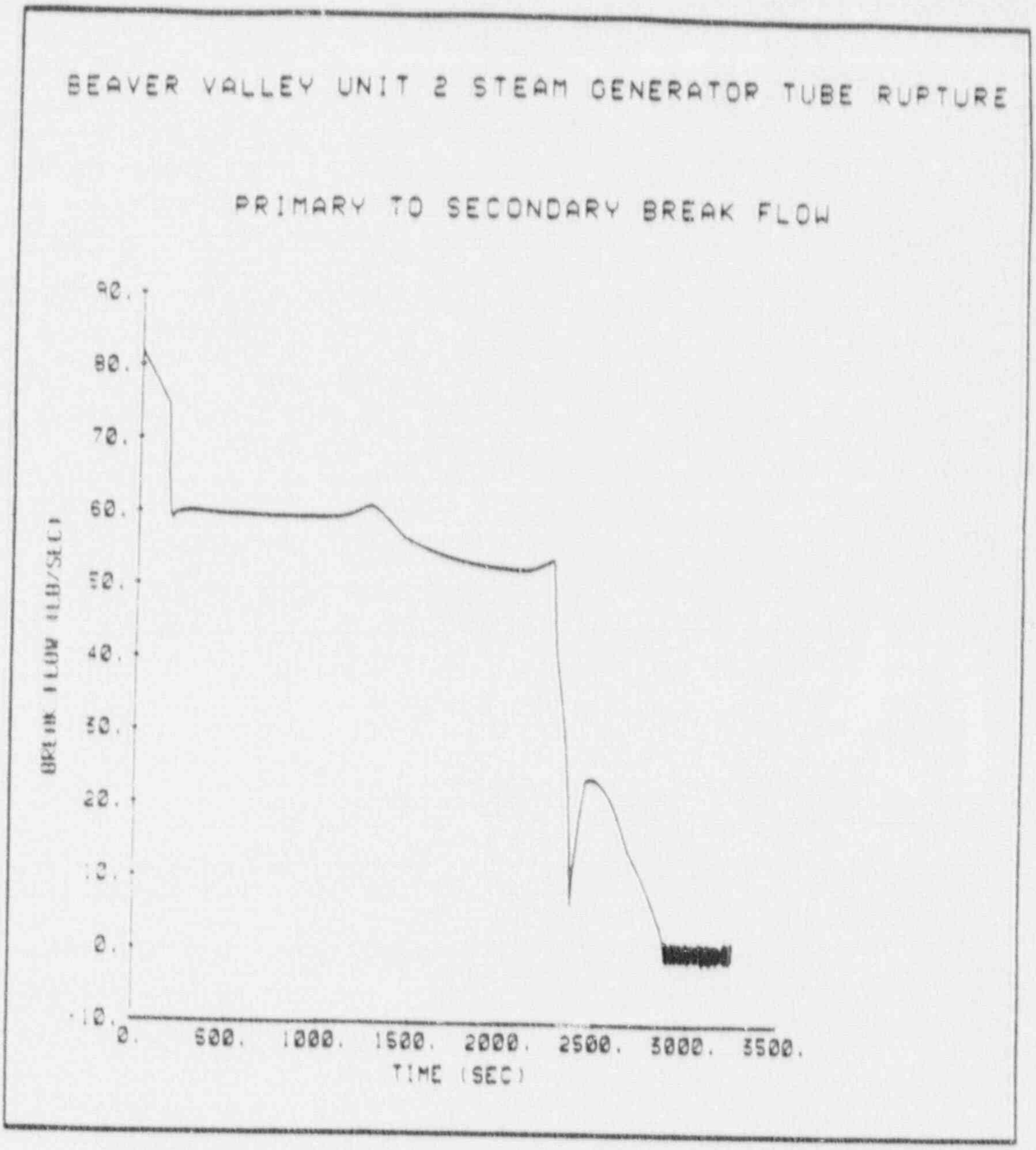

Figure II.5 Primary to Secondary Break Flow Rate - Margin to Overfill Analysis

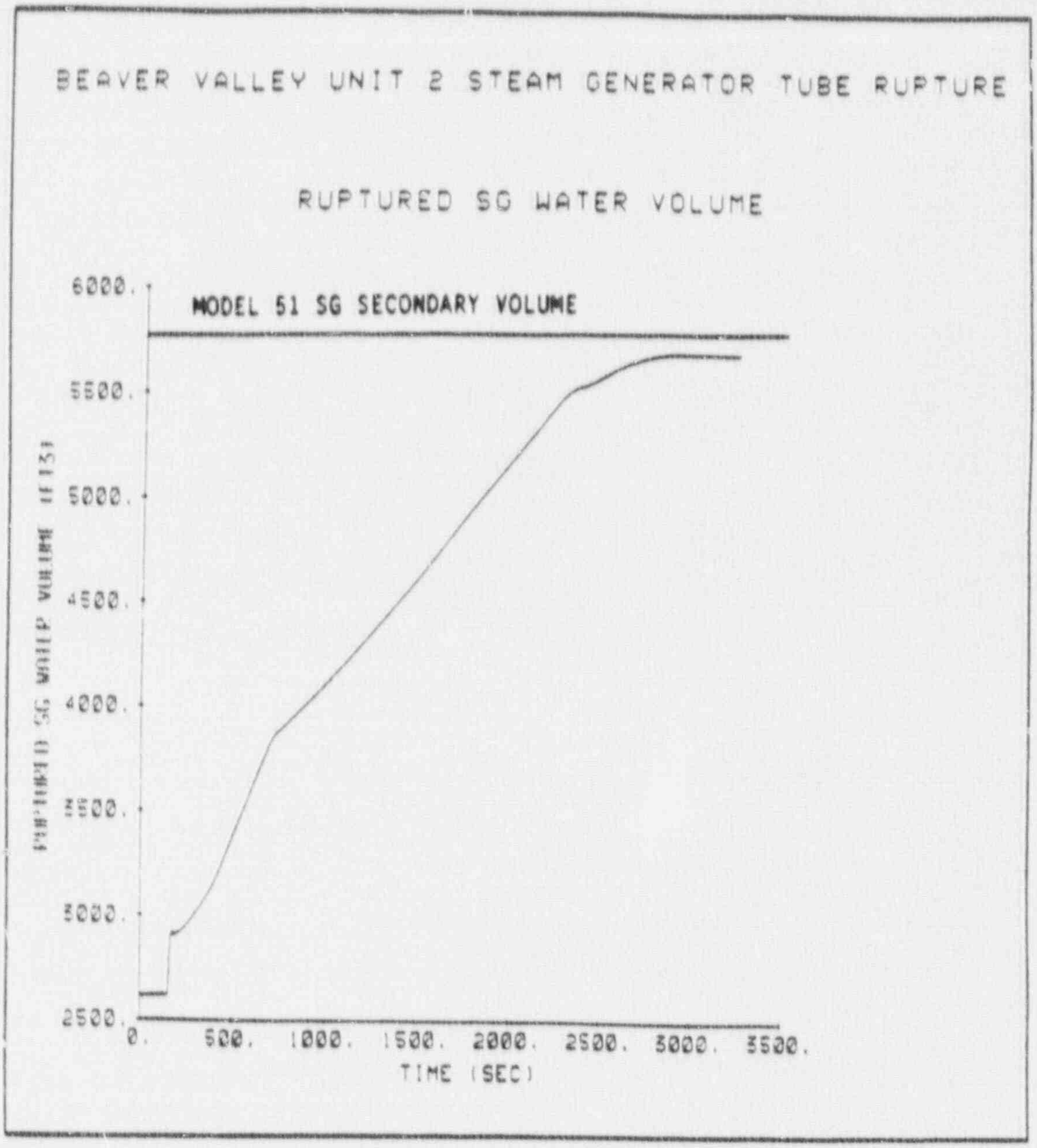

Figure II.6 Ruptured SG Water Volume - Margin to Overfill Analysis

III. ANALYSIS OF OFFSITE RADIOLOGICAL CONSEQUENCES

An analysis was performed to determine the offsite radiological consequences for a design basis SGTR for BVPS Unit 2. The thermal and hydraulic and the offsite radiation dose analyses were performed using the methodology developed in References 1 and 2 and the plant specific parameters for BVPS Unit 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 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.

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 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 BVPS Unit 2 is a failed open atmospheric steam dump valve on the ruptured steam generator. Failure of this atmospheric steam dump valve 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 atmospheric steam dump valve 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 atmospheric steam dump valve fails open and must be locally isolated.

2. <u>Conservative Assumptions</u>

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. <u>Reactor Trip and Turbine Runback</u>

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 atmospheric steam dump valves (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 BVPS Unit 2 reactor protection system, and this time was used for the offsite dose analysis.

[ ]a,c

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

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 to correspond to operation at nominal steam generator mass minus a [ ]^{a,c} allowance for uncertainties. As noted above, [ ]^{a,c}

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

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 atmospheric steam dump valve on the ruptured steam generator, it is not necessary to assume an additional failure in the AFW system. Thus, the turbine-driven pump and both motor-driven pumps were assumed to deliver flow to the three steam generators, and an 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 [ ]^{a,c}

d. 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.

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 assumed for the overfill analysis were also used for the offsite dose analysis. However, for the offsite doses analysis, the atmospheric steam dump valve 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 atmospheric steam dump valve on the ruptured steam generator is assumed to be isolated by locally closing the associated block valve. Duquesne Light Company has determined that an operator can locally close the block valve for the atmospheric steam dump valve on the ruptured steam generator within 6.5 minutes after the failure. Thus, it was assumed that the ruptured steam generator atmospheric steam dump valve is isolated at 6.5 minutes after the valve is assumed to fail open. After the ruptured steam generator atmospheric steam dump valve is isolated, the additional delay time of 9 minutes (Table II.1) was assumed for the operator action time to initiate the RCS cooldown.

4. 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 atmospheric steam dump valve 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 159 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 atmospheric steam dump valve 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 169 seconds.

Major Operator Actions

1. Identify and Isolate the Ruptured Steam Generator

   The ruptured steam generator is assumed to be identified and isolated at 11.75 minutes after the initiation of the SGTR or when the narrow range level reaches 27.5%, whichever time is greater. Since the time to reach 27.5% narrow range level is greater than 11.75 minutes, it was assumed that the ruptured steam generator is isolated when the level reaches 27.5% which occurs at 808 seconds. The ruptured steam generator atmospheric steam dump valve is also assumed to fail open at this time, and the failure is simulated at 810 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 atmospheric steam dump valve is open and to locally close the associated block valve is 6.5 minutes. However, the actual time used in the analysis is 2 seconds longer because of the computer program limitations. Thus, at 1202 seconds the depressurization of ruptured steam generator is terminated and the ruptured steam generator pressure begins to increase as shown on Figure III.2.

2. Cool Down the RCS to establish Subcooling Margin

After the block valve for the ruptured steam generator atmospheric steam dump valve is closed, there is a 9 minute operator action time imposed prior to initiation of cooldown. Thus, the RCS cooldown was initiated at 1742 seconds. By this time, the ruptured steam generator pressure has increased to the intact steam generator pressure and stablized at that value. The RCS cooldown target 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 atmospheric steam dump valves. The cooldown is continued until RCS subcooling at the ruptured steam generator pressure is 20°F plus an allowance for instrument uncertainty. Because the ruptured steam generator pressure has increased to the intact steam generator pressure prior to performing the cooldown, the associated temperature the RCS must be cooled to is not as low, which has the net effect of reducing the time required for cooldown. The cooldown is initiated at 1742 seconds and is completed at 2292 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.

3. Depressurize RCS to Restore Inventory

After the RCS cooldown, a 2.5 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 2442 seconds and continued until any of the following conditions are satisfied: pressurizer level is greater than 76%, 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 4%. For this case, the RCS depressurization is terminated because the RCS pressure is reduced to less than the ruptured steam generator pressure and the pressurizer level is above 4%. 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.

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 4%. For the BVPS Unit 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 1.25 minutes was assumed prior to initiation of SI termination. Since the above requirements are satisfied, SI termination actions were performed at this time 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 atmospheric steam dump valves are also opened to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the atmospheric steam dump valves 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 5759 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

BVPS UNIT 2 SGTR ANALYSIS

SEQUENCE OF EVENTS

OFFSITE RADIATION DOSE ANALYSIS

EVENT	TIME (sec)
SG Tube Rupture	0
Reactor Trip	159
Safety Injection	169
Ruptured SG Isolated	808
Ruptured SG Atmospheric Steam Dump Valve Fails Open	810
Ruptured SG Atmospheric Steam Dump Valve Block Valve Closed	1202
RCS Cooldown Initiated	1742
RCS Cooldown Terminated	2292
RCS Depressurization Initiated	2442
RCS Depressurization Terminated	2558
SI Terminated	2634
Break Flow Terminated	3070

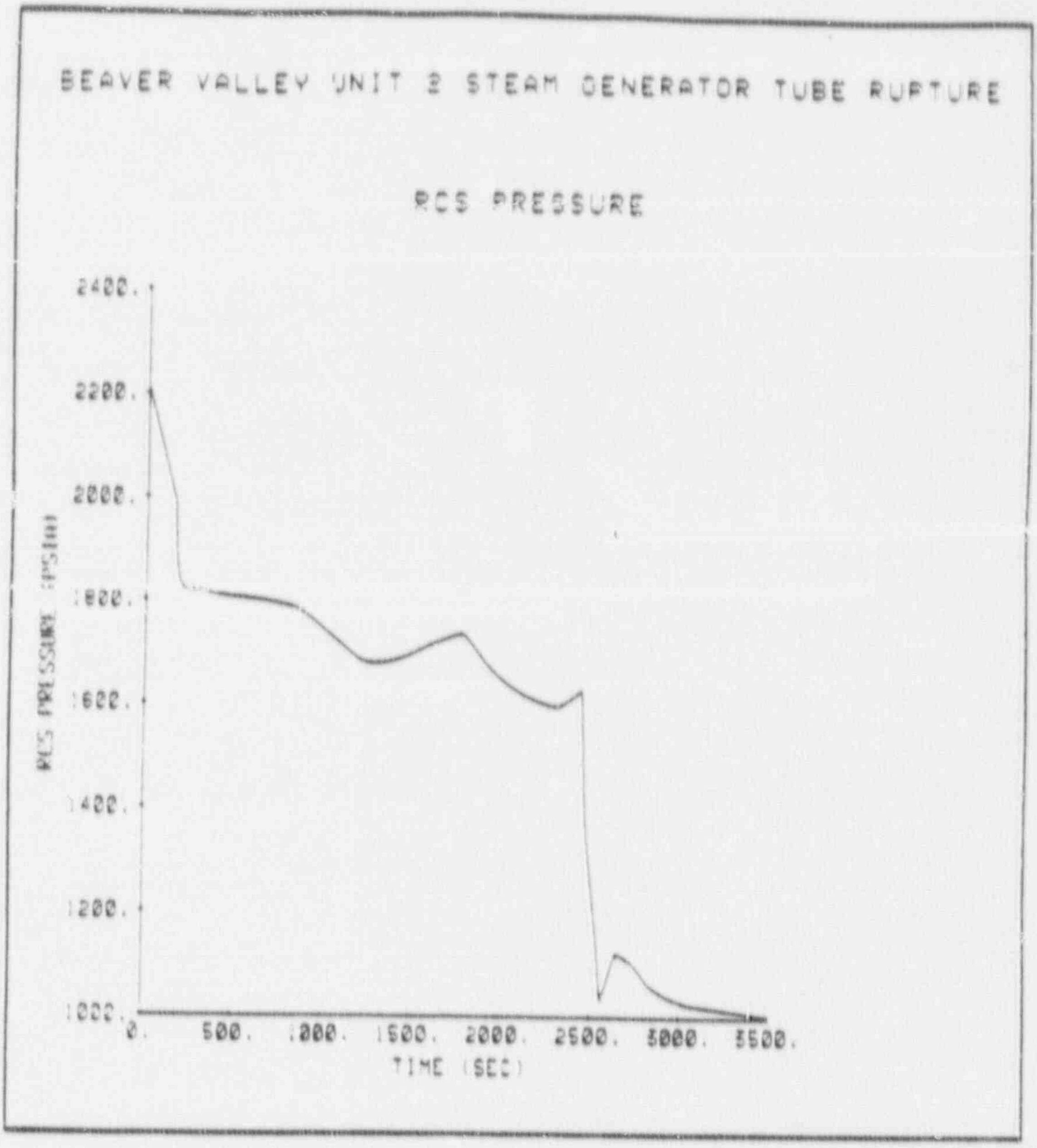

Figure III.1 RCS Pressure - Offsite Radiation Dose Analysis

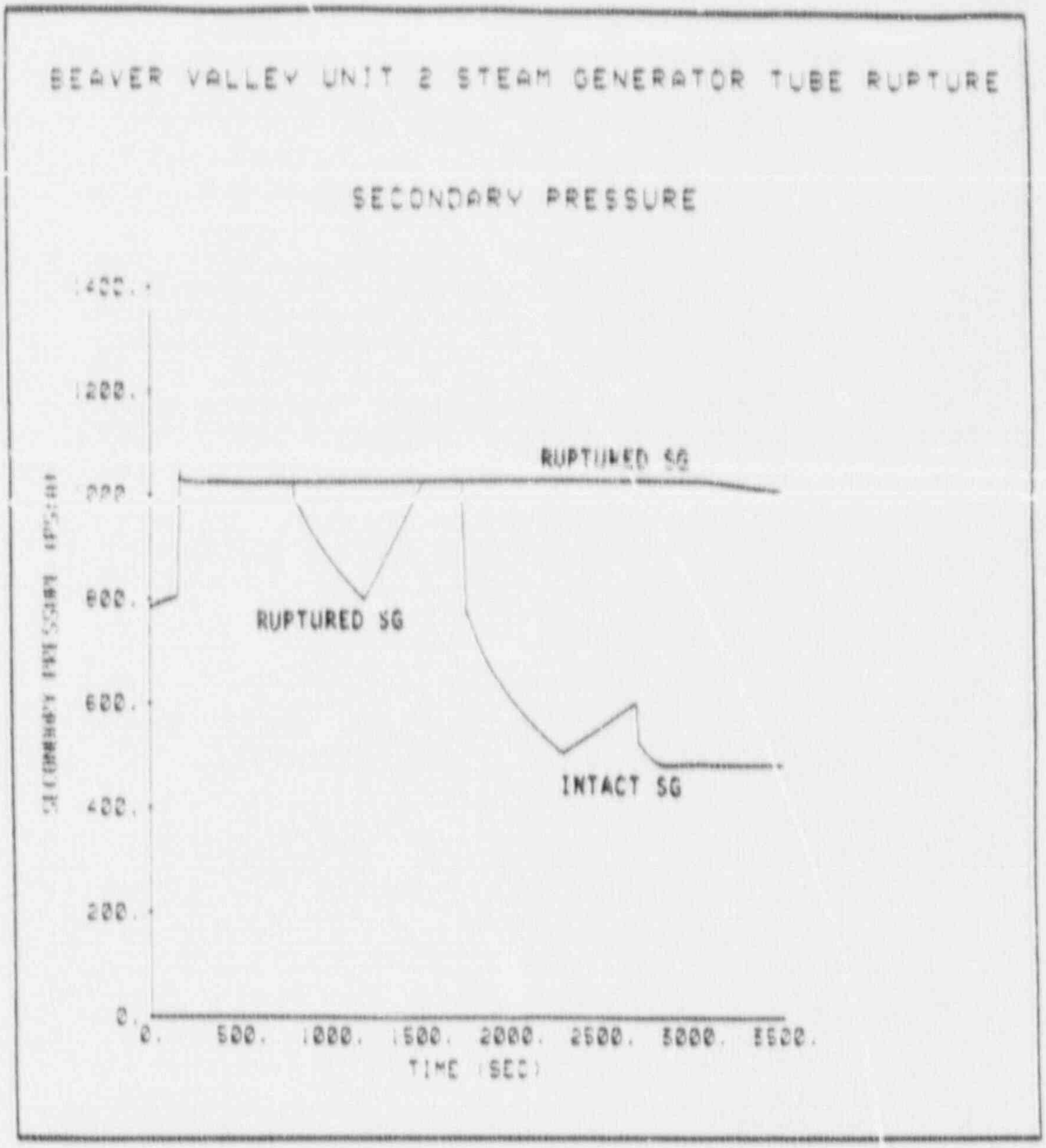

Figure III.2 Secondary Pressure - Offsite Radiation Dose Analysis

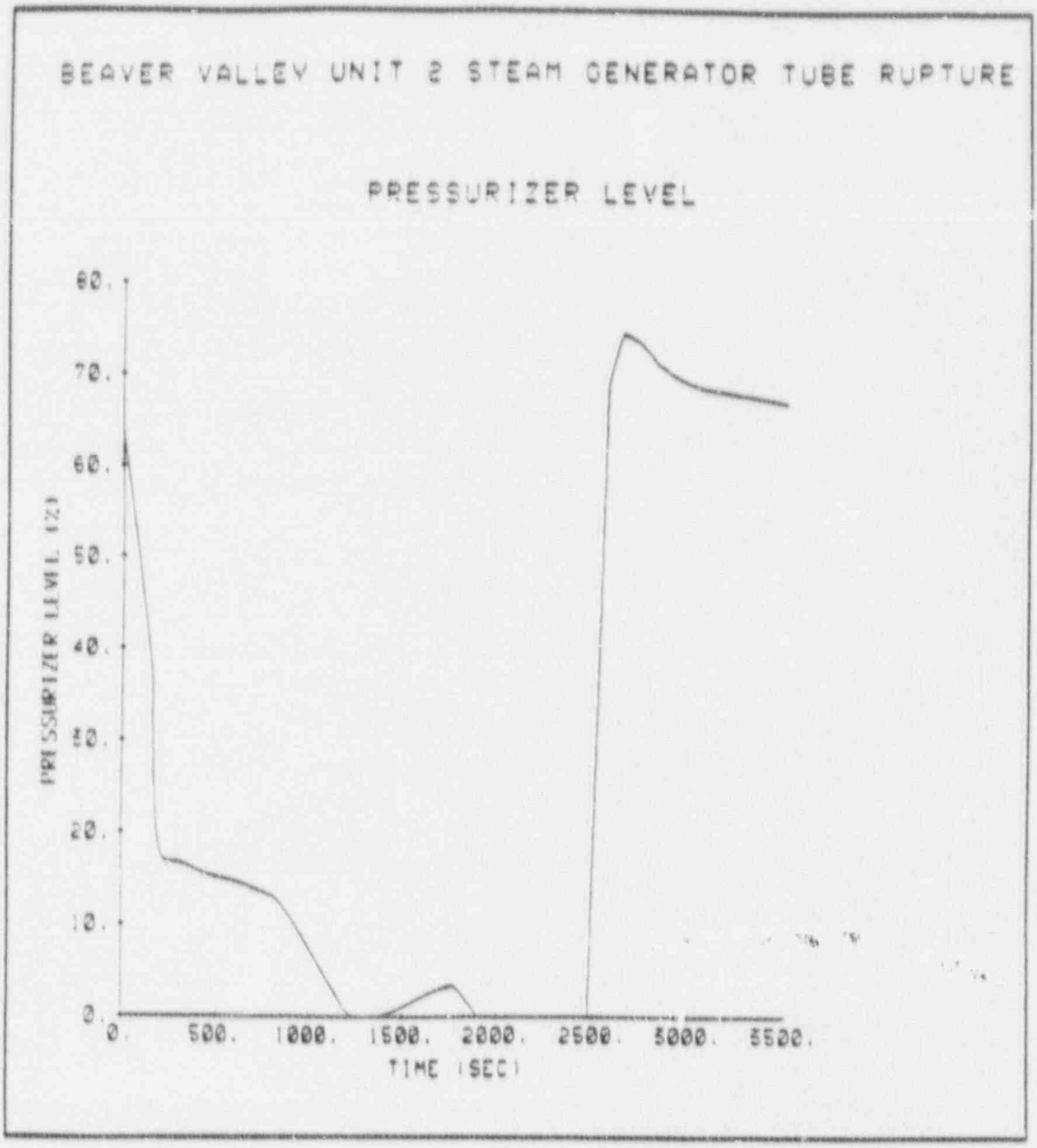

Figure III.3 Pressurizer Level - Offsite Radiation Dose Analysis

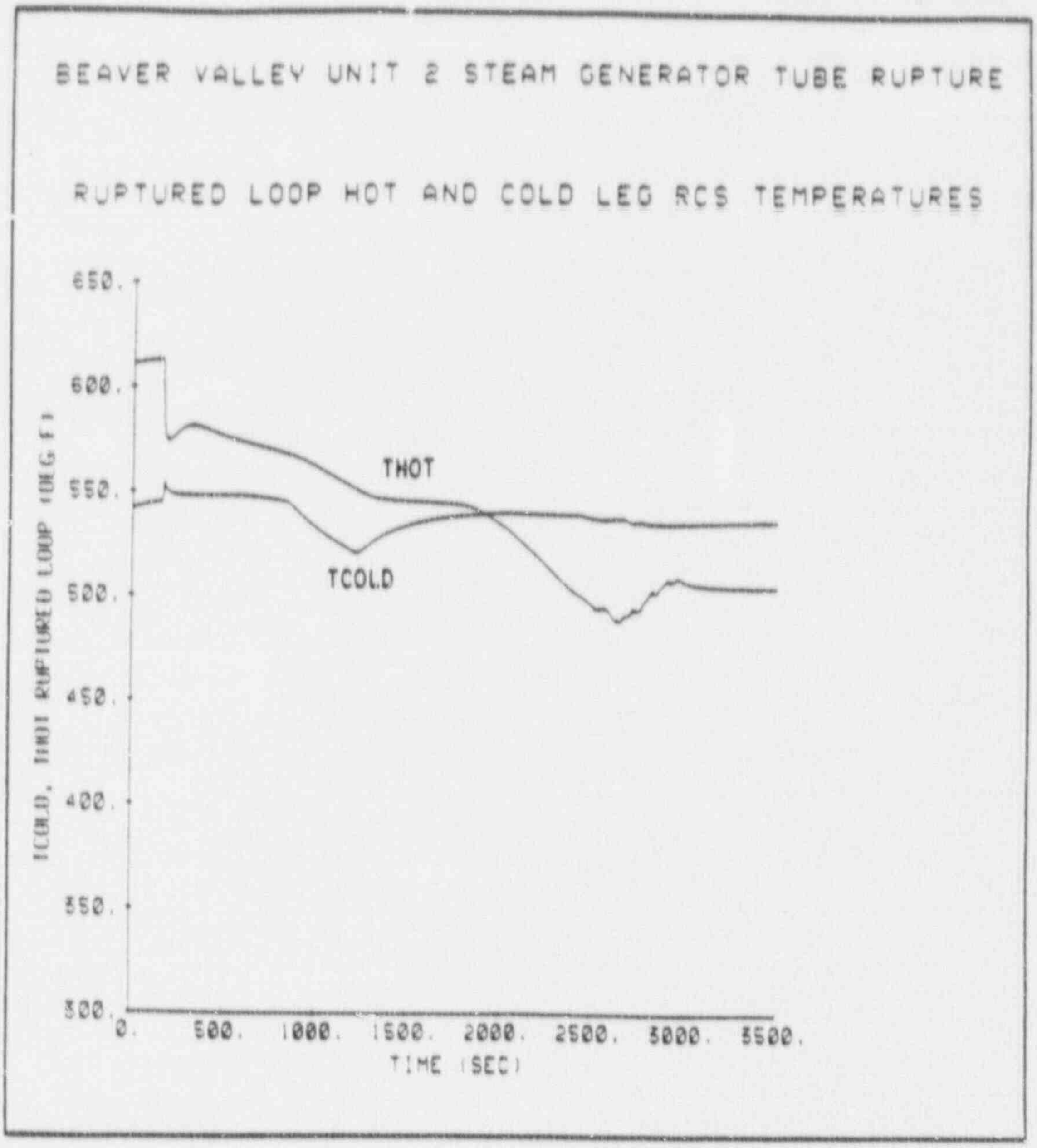

Figure III.4 Ruptured Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

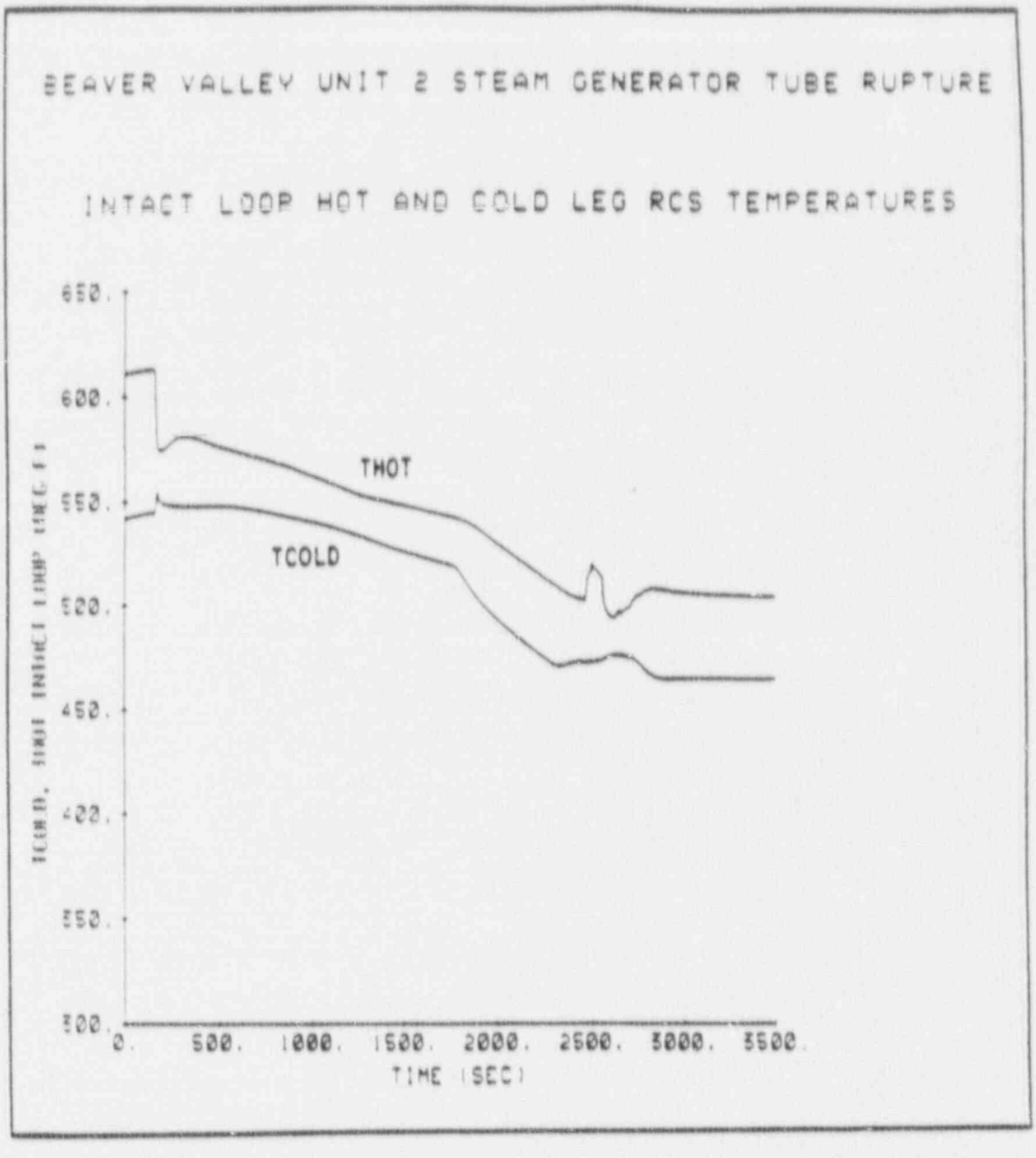

Figure III.5 Intact Loop Hot and Cold Leg RCS Temperatures - Offsite Radiation Dose Analysis

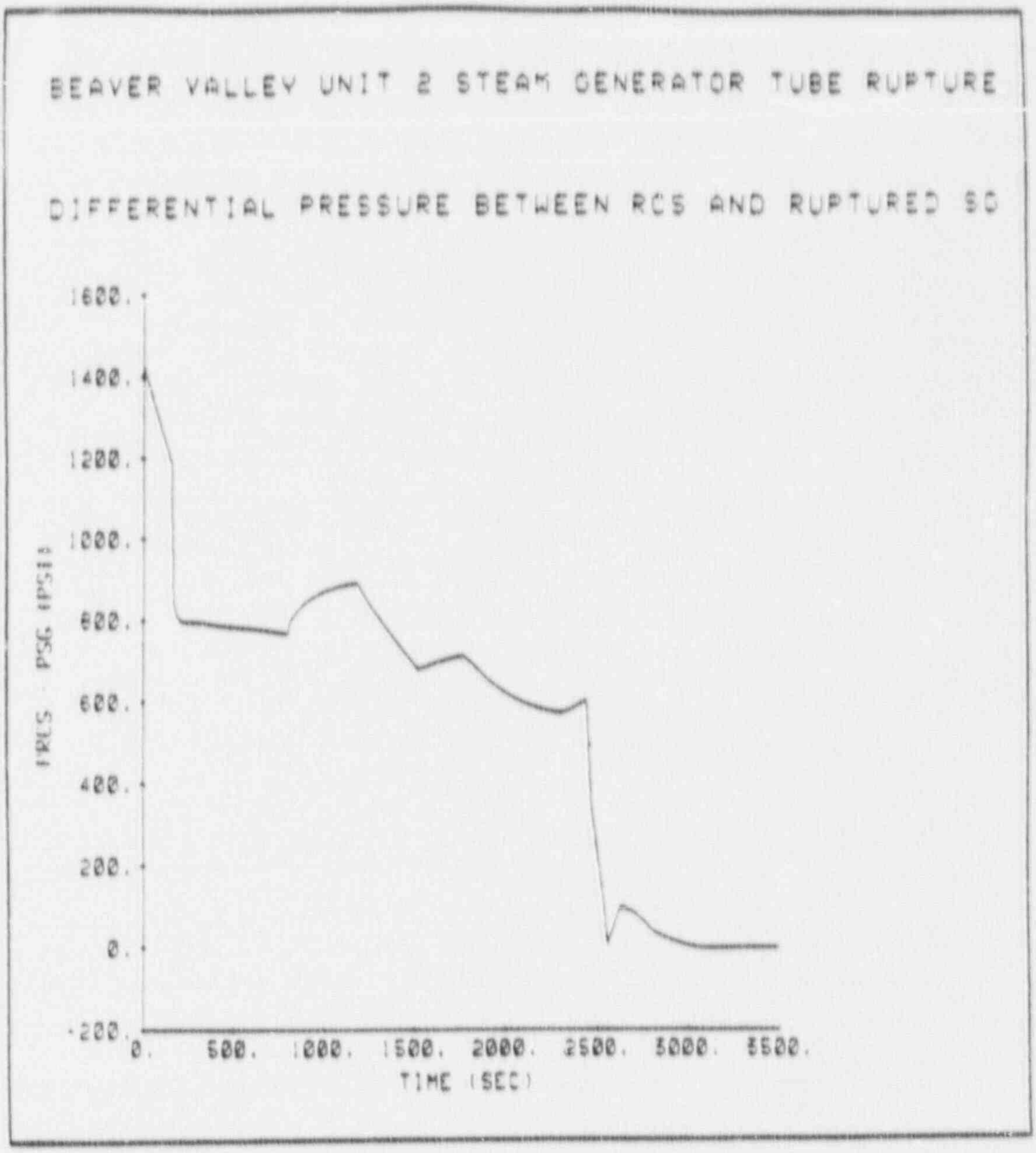

Figure III.6 Differential Pressure Between RCS and Ruptured SG - Offsite Radiation Dose Analysis

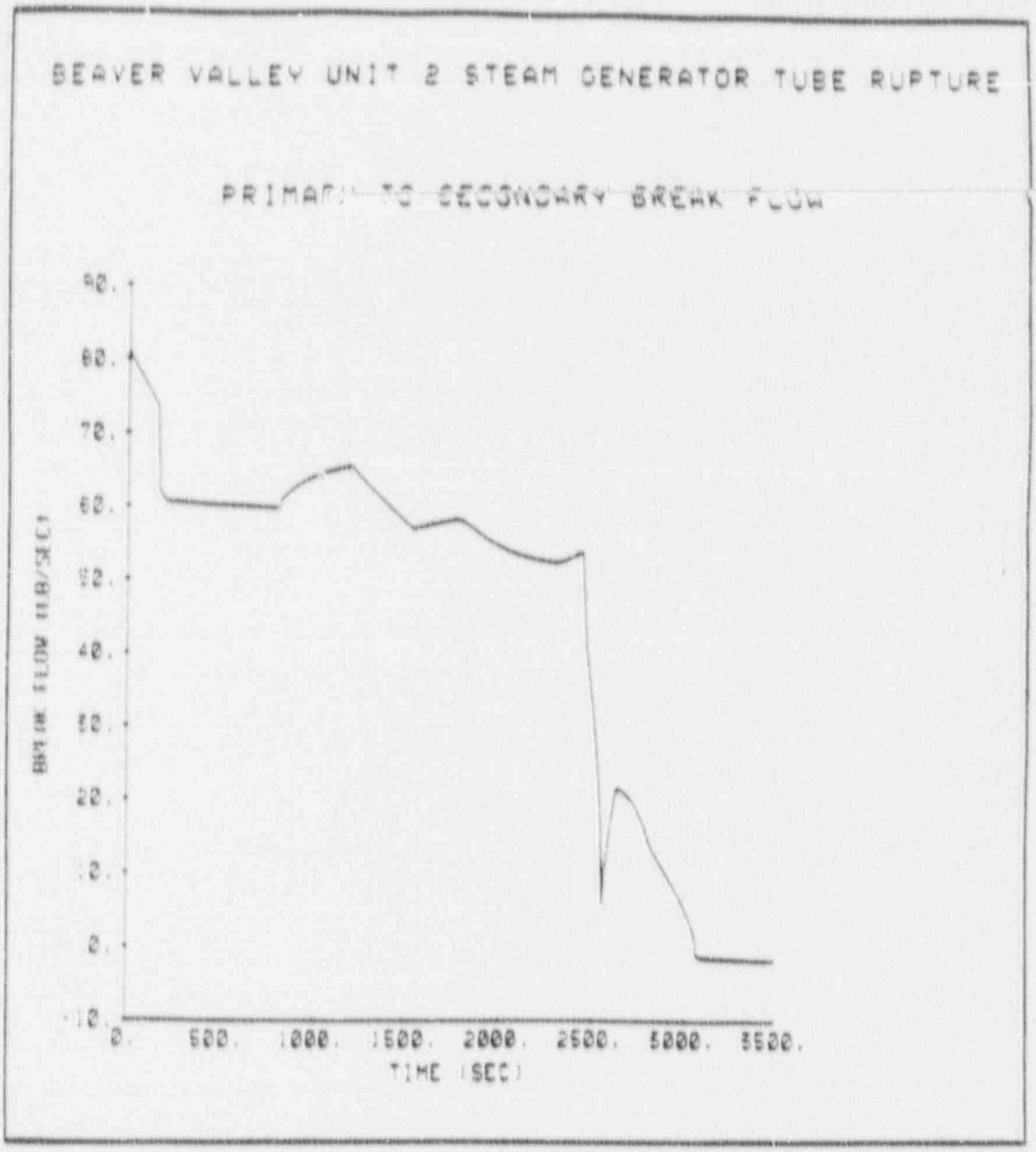

Figure III.7 Primary to Secondary Break Flow Rate - Offsite Radiation Dose Analysis

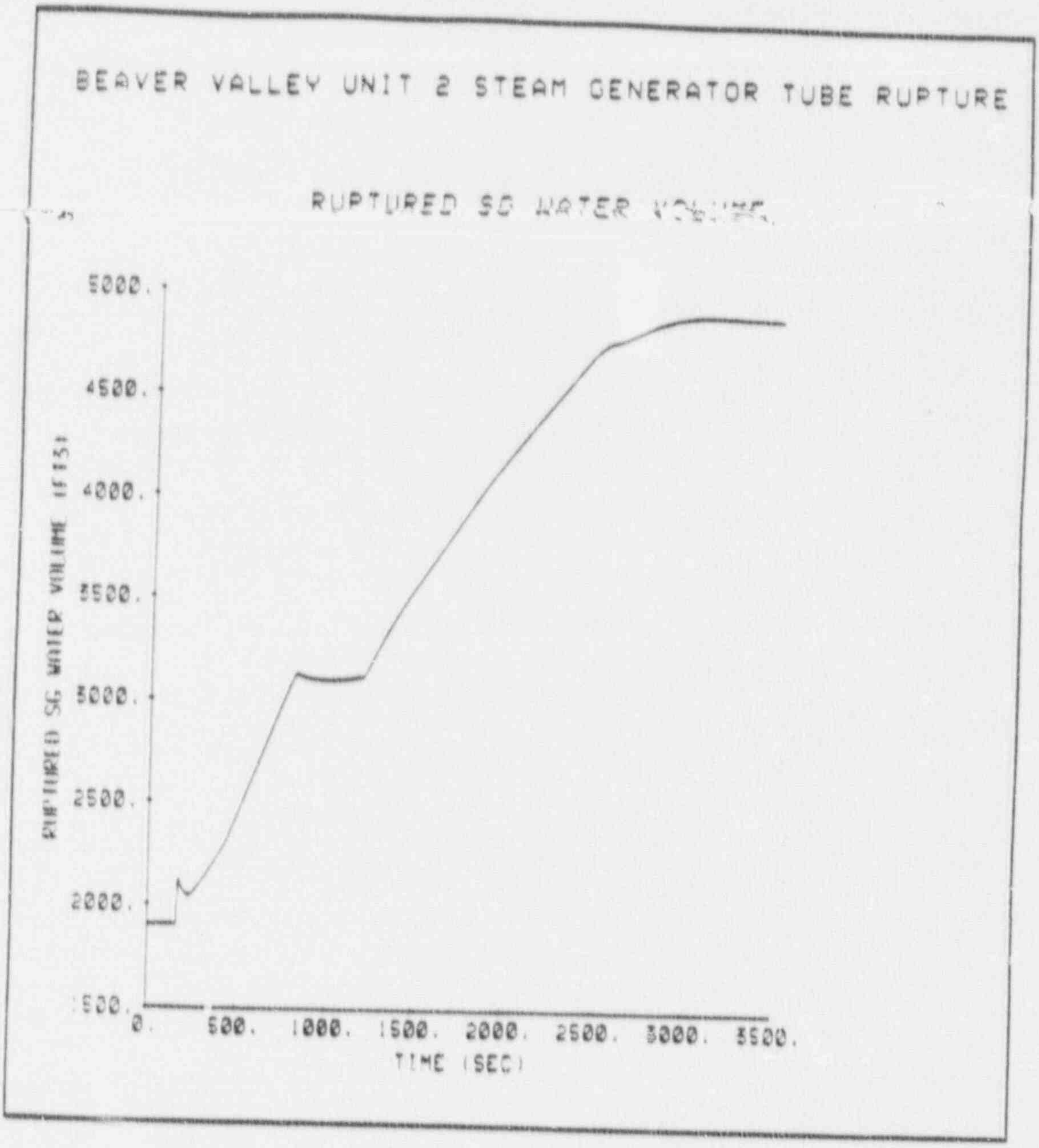

Figure III.8 Ruptured SG Water Volume - Offsite Radiation Dose Analysis

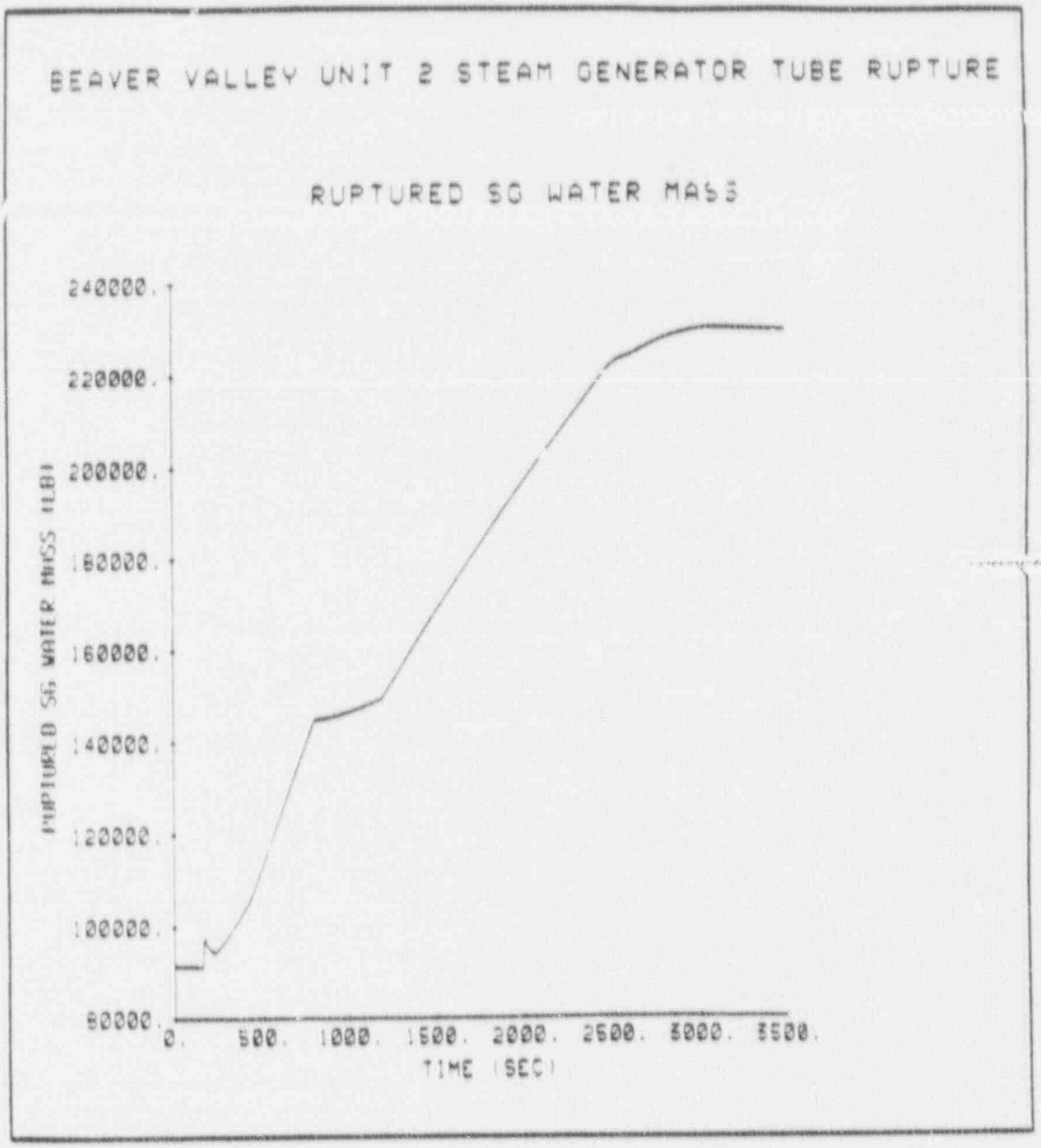

Figure III.9 Ruptured SG Water Mass - Offsite Radiation Dose Analysis

5. <u>Mass Releases</u>

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 analysis, the SGTR recovery actions in BVPS Unit 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 BVPS Unit 2 EOP ES-3.3 (ERG 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 post-SGTR cooldown method using steam dump in BVPS Unit 2 EOP ES-3.3 (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 cooling the RCS is steam dump to the condenser since this minimizes the radiological releases and conserves feedwater supply. The atmospheric steam dump valves 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 BVPS Unit 2 analysis, it was assumed that the cooldown is performed using steam dump to the atmosphere via the intact steam generator atmospheric steam dump valves. 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 atmospheric steam dump valve 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 atmospheric steam dump valve. 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 BVPS Unit 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 for a [ ][a,c] until the cooldown to cold shutdown is initiated. The atmospheric steam dump valves 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 375 psia via steam release from the ruptured steam generator atmospheric steam dump valve, 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}

[

]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 air ejector discharge. After reactor trip, the releases to the atmosphere are assumed to be via the steam generator atmospheric steam dump valves. 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 atmospheric steam dump valves. 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 43,500 lbm of steam are released from the ruptured steam generator to the atmosphere in the first 2 hours. A total of 156,600 lbm of primary water is transferred to the secondary side of the ruptured steam generator before the break flow is terminated.

TABLE III.2

BVPS UNIT 2 SGTR ANALYSIS

MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

TIME PERIOD

	0-TRIP	TRIP - TMSEP	TMSEP - TTBRK	TTBRK - T2HRS	T2HRS - TRHR
Ruptured SG					
- Condenser	177,400	0	0	0	0
- Atmosphere	0	42,200	1,300	0	36,000
- Feedwater	165,000	32,400	0	0	0
Intact SG					
- Condenser	350,800	0	0	0	0
- Atmosphere	0	68,700	30,800	286,900	726,700
- Feedwater	350,800	177,200	75,100	310,200	735,200
Break Flow	12,400	115,800	28,400	0	0

TRIP = Time of reactor trip = 159 sec.

TMSEP = Time when water reaches the moisture separators = 2097 sec.

TTBRK = Time when break flow is terminated = 3070 sec.

T2HRS = Time at 2 hours = 7200 sec.

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

TABLE III.3

BVPS UNIT 2 SGTR ANALYSIS

SUMMARIZED MASS RELEASES

OFFSITE RADIATION DOSE ANALYSIS

TOTAL MASS FLOW (POUNDS)

	0 - TTBRK	TTBRK - 2HRS	2HRS - 8HRS
Ruptured SG			
- Condenser	177,400	0	0
- Atmosphere	43,500	0	36,000
- Feedwater	197,400	0	0
Intact SGs			
- Condenser	350,800	0	0
- Atmosphere	99,500	286,900	726,700
- Feedwater	603,100	310,200	735,200
Break Flow	156,600	0	0

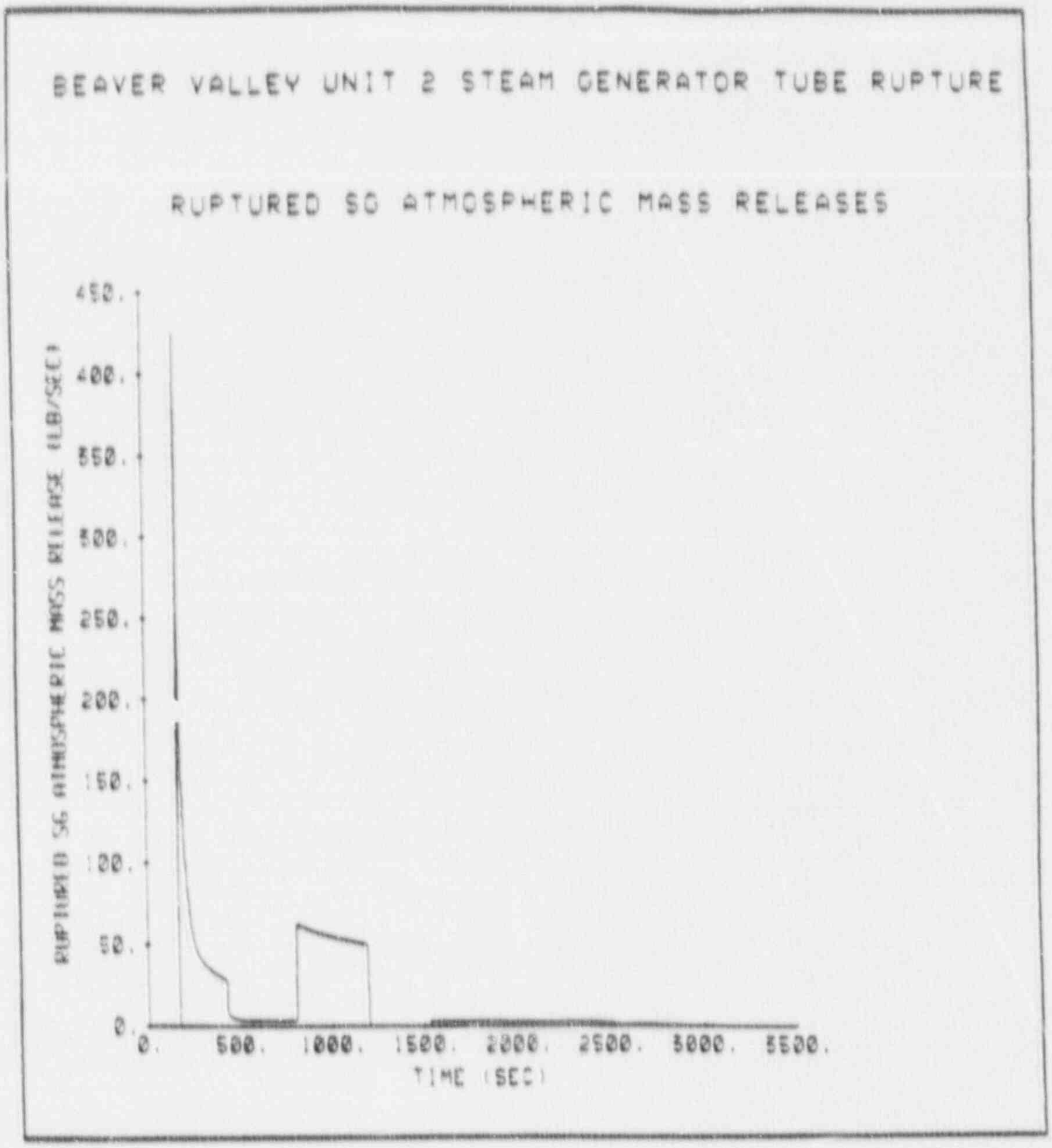

Figure III.10 Ruptured SG Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

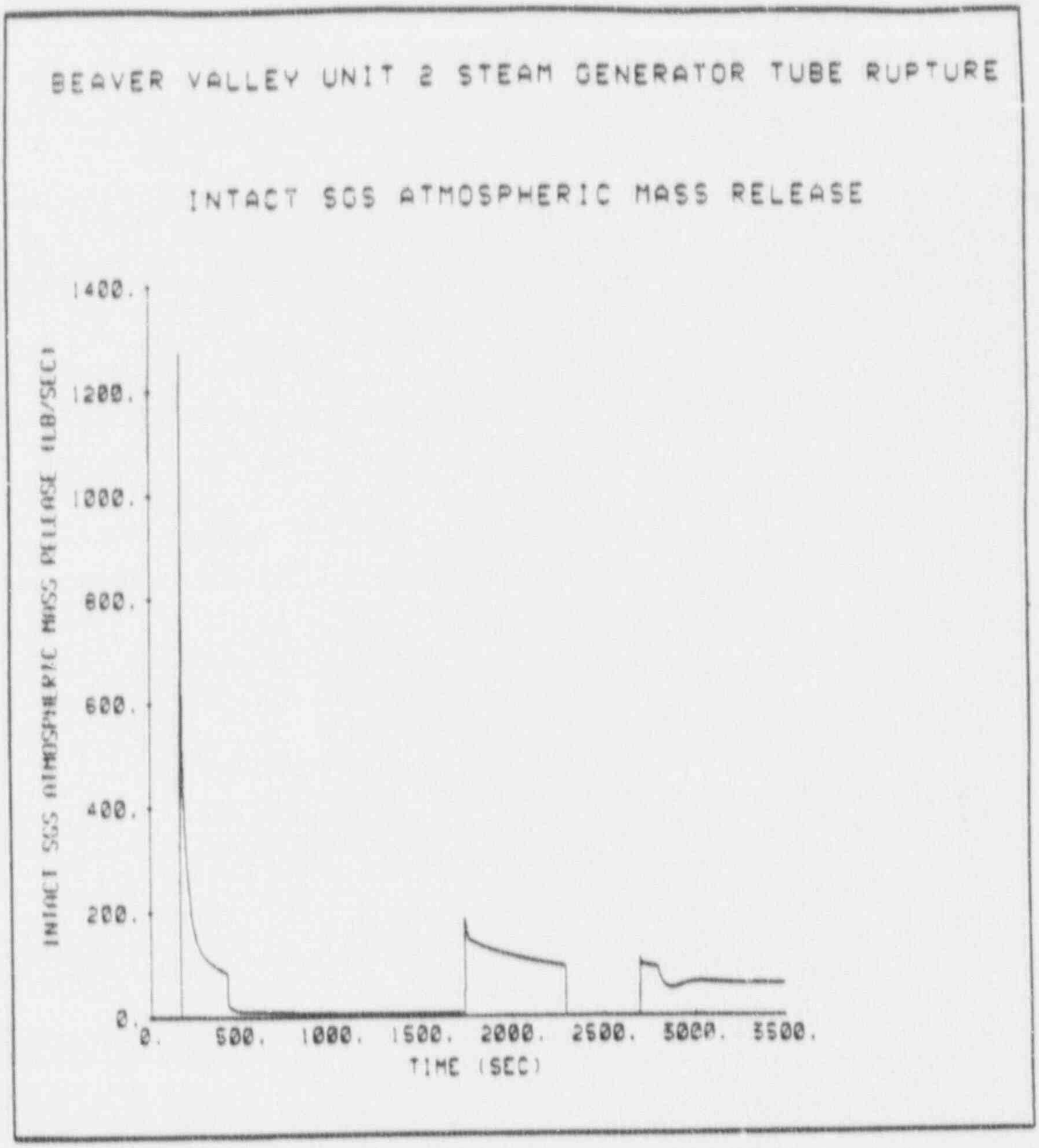

Figure III.11 Intact SGs Mass Release Rate to the Atmosphere - Offsite Radiation Dose Analysis

B. Offsite Radiation Dose Analysis

The evaluation of the radiological consequences of a steam generator tube rupture, assumes that the reactor has been operating at the 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 atmospheric steam dump valves (and safety valves) and via the air ejector discharge.

The quantity of radioactivity released to the environment, due to a 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 for a design basis double ended rupture of a single tube.

1. <u>Design Basis Analytical Assumptions</u>

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

2. <u>Source Term Calculations</u>

   The radionuclide concentrations in the BVPS Unit 2 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 pre-accident 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 that corresponds to the initial primary system iodine concentration. The duration of the spike is 4.0 hours.

ii. Pre-Accident 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 on approximately 0.26% fuel defects.

3. Dose Calculations

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 as a function of the height of the secondary water level above the rupture site. 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 0.01 between the steam generator liquid and steam phases. Droplet removal by the dryers is conservatively 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. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow comes from the hot leg side of the steam generator, whereas the break flow actually comes from both the hot leg and cold leg sides of the steam generator.

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 intersection of the outer tube row and the upper anti-vibration bar (approximately 4 inches below the apex of the tube bundle). However, the tube rupture break flow was conservatively calculated assuming that the break is at the top of the tube sheet. The 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). The iodine scrubbing efficiencies are shown in Figure III.15.

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 BVPS Unit 2 Technical Specifications. The leak rate is assumed to be 0.35 gpm for each of the 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 factor between the liquid and steam of the ruptured and intact steam generators is assumed to be 0.01.

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 (x/Qs) 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 Models

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 isotope 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 second/meter^3 (Table III.8)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for isotope 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 nuclide i released during time interval j in Ci *

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

$\bar{E}_{\gamma i}$ = average gamma energy for noble gas 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:

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

where:

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

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

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

D_β = bata-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. <u>Results</u>

The calculated nuclide releases resulting from an SGTR are presented in Table III.11 for the pre-accident iodine spike case and in Table III.12 for the accident initiated iodine spike case. Thyroid, whole-body gamma, and beta-skin doses at the Exclusion Area Boundary and Low Population Zone are presented in Table III.13. All doses are within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10CFR100.

TABLE III.4

BVPS UNIT 2 SGTR ANALYSIS

PARAMETERS USED IN EVALUATING

RADIOLOGICAL CONSEQUENCES

I. Source Data

A. Core power level, MWt	2766
B. Total steam generator tube leakage, prior to accident, gpm	1.0
C. Reactor coolant iodine 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. Pre-Accident 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 0.26% 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	1.91×10^8
F. Initial steam generator water mass (each), grams	4.5×10^7
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	See Figure III.15
4. Total steam release, lbs	See Table III.2
5. Iodine partition factor	0.01

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 releasc lbs	See Table III.2
3. Iodine partition factor	0.01
C. Condenser	
1. Iodine partition factor	0.01
D. Atmospheric Dispersion Factors	See Table III.8

TABLE III.5

BVPS UNIT 2 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 (ΣμCi/gm)		
	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	0.66	39.9	0.069
I-132	0.23	13.9	0.020
I-133	1.0	62.2	0.098
I-134	0.14	8.7	0.00045
I-135	0.55	33.4	0.044

TABLE III.6

BVPS UNIT 2 SGTR ANALYSIS

IODINE SPIKE APPEARANCE RATES

(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
1.36	2.52	3.08	3.68	2.81

TABLE III.7

BVPS UNIT 2 SGTR ANALYSIS

NOBLE GAS SPECIFIC ACTIVITIES IN THE

REACTOR COOLANT BASED ON 0.26% FUEL DEFECTS

Nuclide	Specific Activity (μCi/gm)
Kr-83m	0.11
Kr-85m	0.55
Kr-85	2.90
Kr-87	0.32
Kr-88	0.84
Kr-89	0.027
Xe-131m	0.028
Xe-133m	0.81
Xe-133	6.9
Xe-135m	0.29
Xe-135	0.85
Xe-137	0.043
Xe-138	0.18

TABLE III.8

BVPS UNIT 2 SGTR ANALYSIS

ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES

Time (hours)	Exclusion Area Boundary χ/Q (Sec/m^3)	Low Population Zone χ/Q (Sec/m^3)	Breathing Rate (m^3/Sec) [5]
0-2	1.44×10^{-3}	7.07×10^{-5}	3.47×10^{-4}
2-8	-	7.07×10^{-5}	3.47×10^{-4}

TABLE III.9

BVPS UNIT 2 SGTR ANALYSIS

THYROID DOSE CONVERSION FACTORS

(Rem/Curie) [Ref. 6]

Nuclide	
I-131	1.48×10^{6}
I-132	5.35×10^{4}
I-133	4.0×10^{5}
I-134	2.5×10^{4}
I-135	1.24×10^{5}

TABLE III.10

BVPS UNIT 2 SGTR ANALYSIS

AVERAGE GAMMA AND BETA ENERGY FOR NOBLE GASES

(Mev/dis) [Ref. 7]

Nuclide	$\bar{E}_\gamma$	$\bar{E}_\beta$
Kr-83m	0.0005	0.042
Kr-85m	0.156	0.253
Kr-85	0.0023	0.251
Kr-87	0.793	1.33
Kr-88	2.21	0.248
Kr-89	2.1	1.2
Xe-131m	0.0029	0.165
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-137	0.19	1.8
Xe-138	1.2	0.66

TABLE III.11

BVPS UNIT 2 SGTR ANALYSIS

ENVIRONMENTAL RELEASES FOR

PRE-ACCIDENT IODINE SPIKE CASE

Nuclide	Total Releases (Ci) 0-2 hr	0-8 hr
Kr-83m	5.7	5.7
Kr-85m	3.1E1	3.1E1
Kr-85	1.7E2	1.7E2
Kr-87	1.6E1	1.6E1
Kr-88	4.6E1	4.6E1
Kr-89	1.7E-1	1.7E-1
Xe-131m	1.7	1.7
Xe-133m	4.8E1	4.8E1
Xe-133	4.1E2	4.1E2
Xe-135m	8.2	8.2
Xe-135	4.9E1	4.9E1
Xe-137	3.2E-1	3.5E-1
Xe-138	5.4	5.4
I-131	6.5E1	6.6E1
I-132	2.1E1	2.1E1
I-133	1.0E2	1.0E2
I-134	1.2E1	1.2E1
I-135	5.3E1	5.4E1

TABLE III.12

BVPS UNIT 2 SGTR ANALYSIS

ENVIRONMENTAL RELEASES FOR

ACCIDENT INITIATED IODINE SPIKE CASE

	Total Releases (Ci)	
Nuclide	0-2 hr	0-8 hr
Kr-83m	5.7	5.7
Kr-85m	3.1E1	3.1E1
Kr-85	1.7E2	1.7E2
Kr-87	1.6E1	1.6E1
Kr-88	4.6E1	4.6E1
Kr-89	1.7E-1	1.7E-1
Xe-131m	1.7	1.7
Xe-133m	4.8E1	4.8E1
Xe-133	4.1E2	4.1E2
Xe-135m	8.2	8.2
Xe-135	4.9E1	4.9E1
Xe-137	3.2E-1	3.5E-1
Xe-138	5.4	5.4
I-131	9.9	1.1E1
I-132	1.6E1	1.7E1
I-133	2.2E1	2.4E1
I-134	2.2E1	2.2E1
I-135	1.9E1	2.1E1

TABLE III.13

BVPS UNIT 2 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	13.4	30
Whole-Body Gamma	0.2	2.5
Beta-Skin	0.2	2.5
Low Population Zone (0-8 hr.)		
Thyroid	0.8	30
Whole-Body Gamma	0.009	2.5
Beta-Skin	0.007	2.5
2. Pre-Accident Iodine Spike		
Exclusion Area Boundary (0-2 hr.)		
Thyroid	71.6	300*
Whole-Body Gamma	0.2	25*
Beta-Skin	0.1	25*
Low Population Zone (0-8 hr.)		
Thyroid	3.6	300*
Whole-Body Gamma	0.007	25*
Beta-Skin	0.005	25*

*Doses should be appropriately within the guideline values.

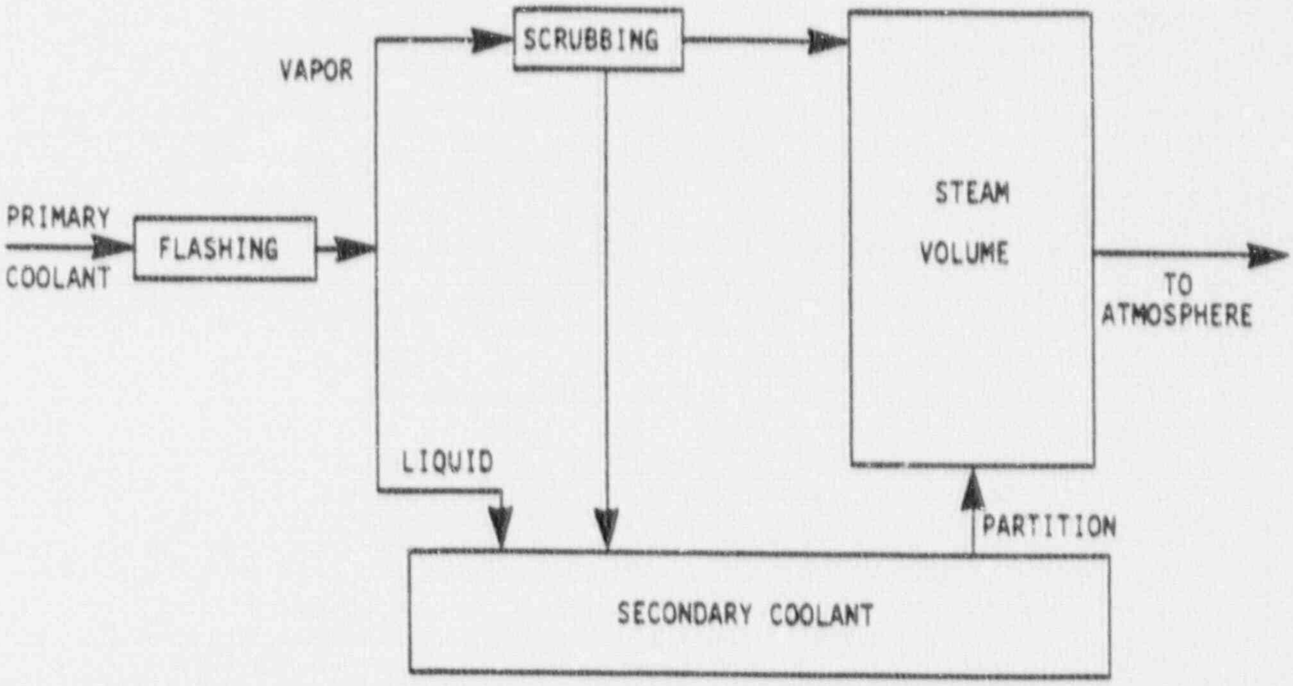

Figure III.12 Iodine Transport Model - Offsite Radiation Dose Analysis

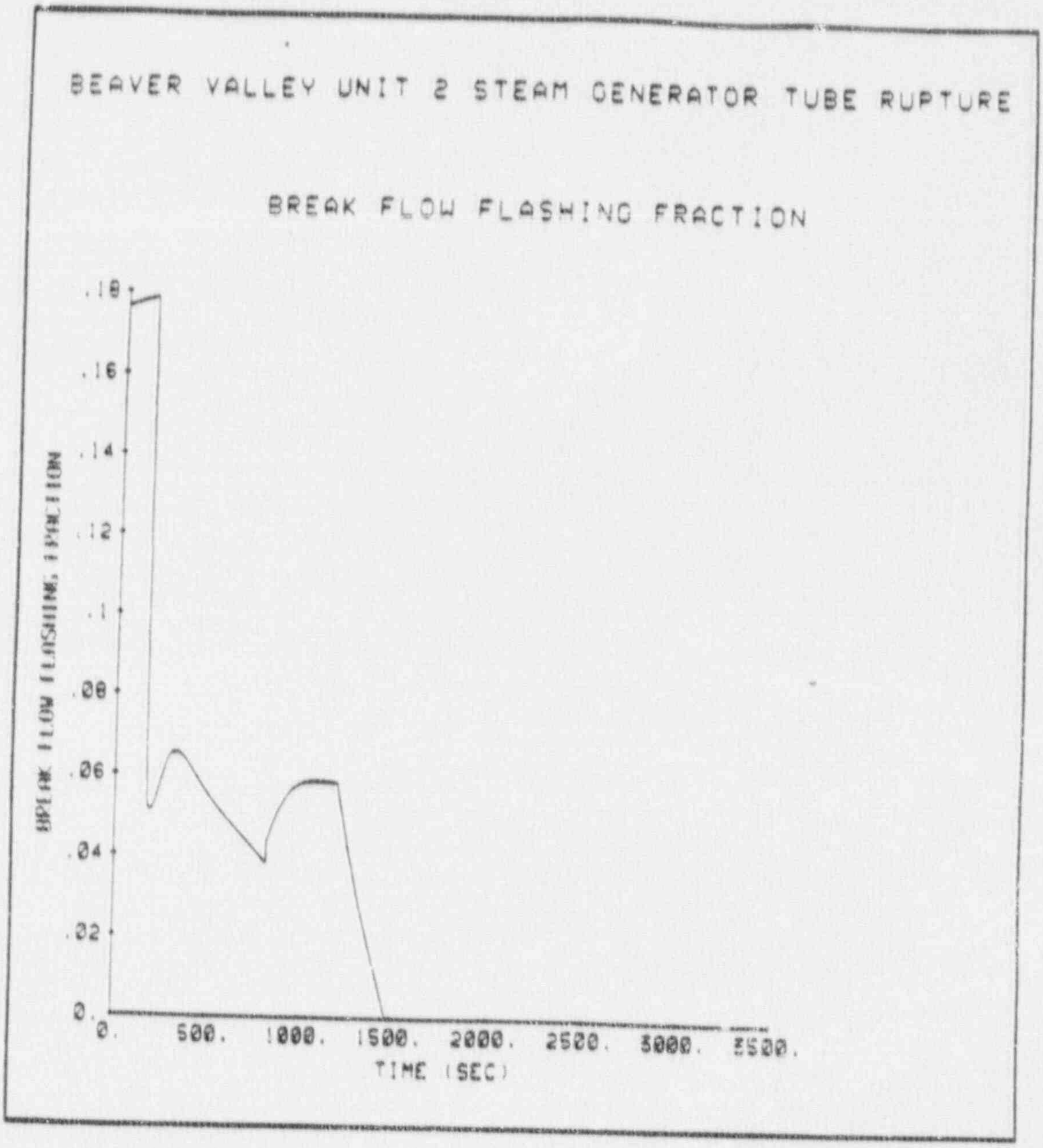

Figure III.13 Break Flow Flashing Fraction - Offsite Radiation Dose Analysis

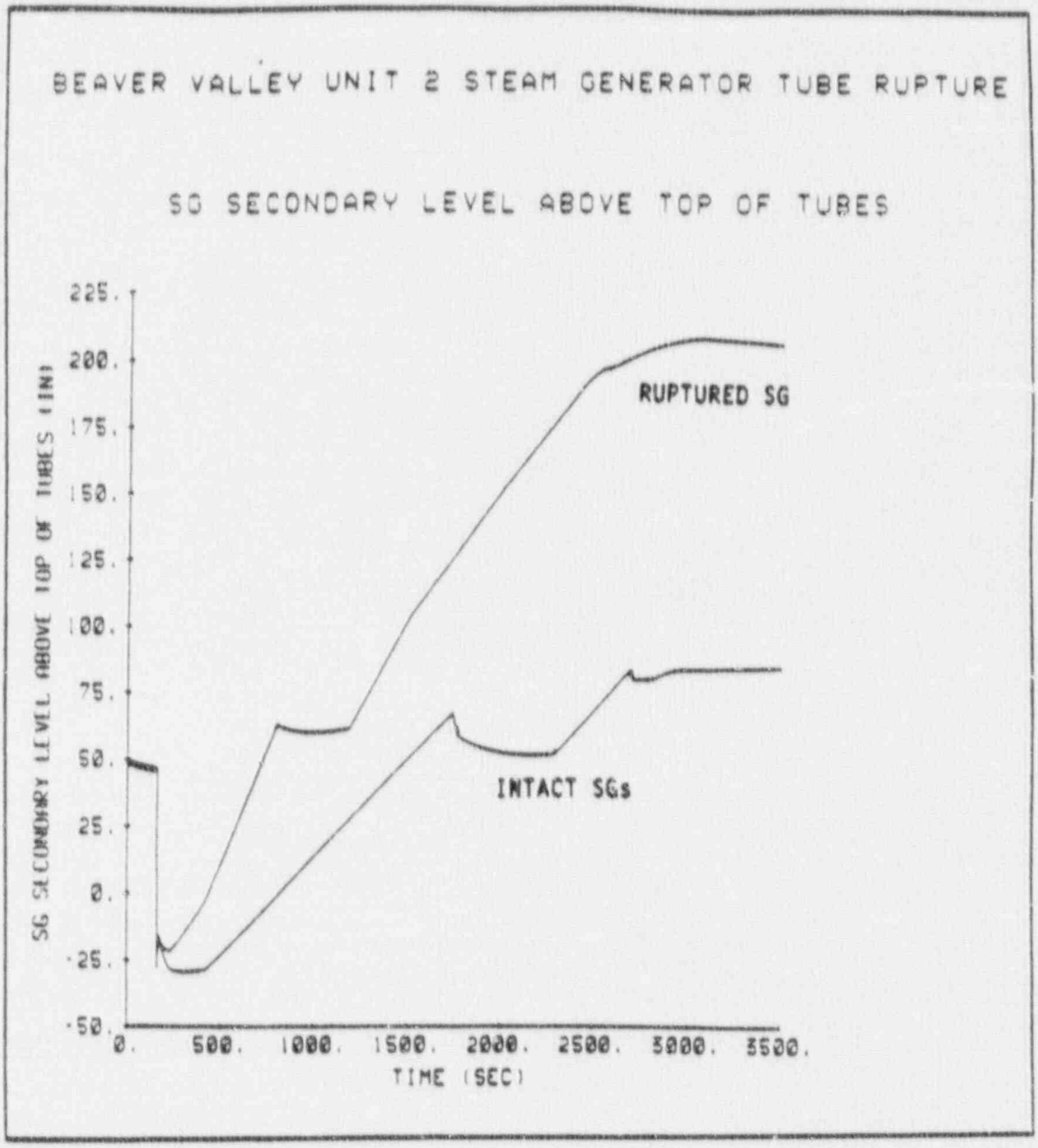

Figure III.14 SG Water Level Above Top of Tubes - Offsite Radiation Dose Analysis

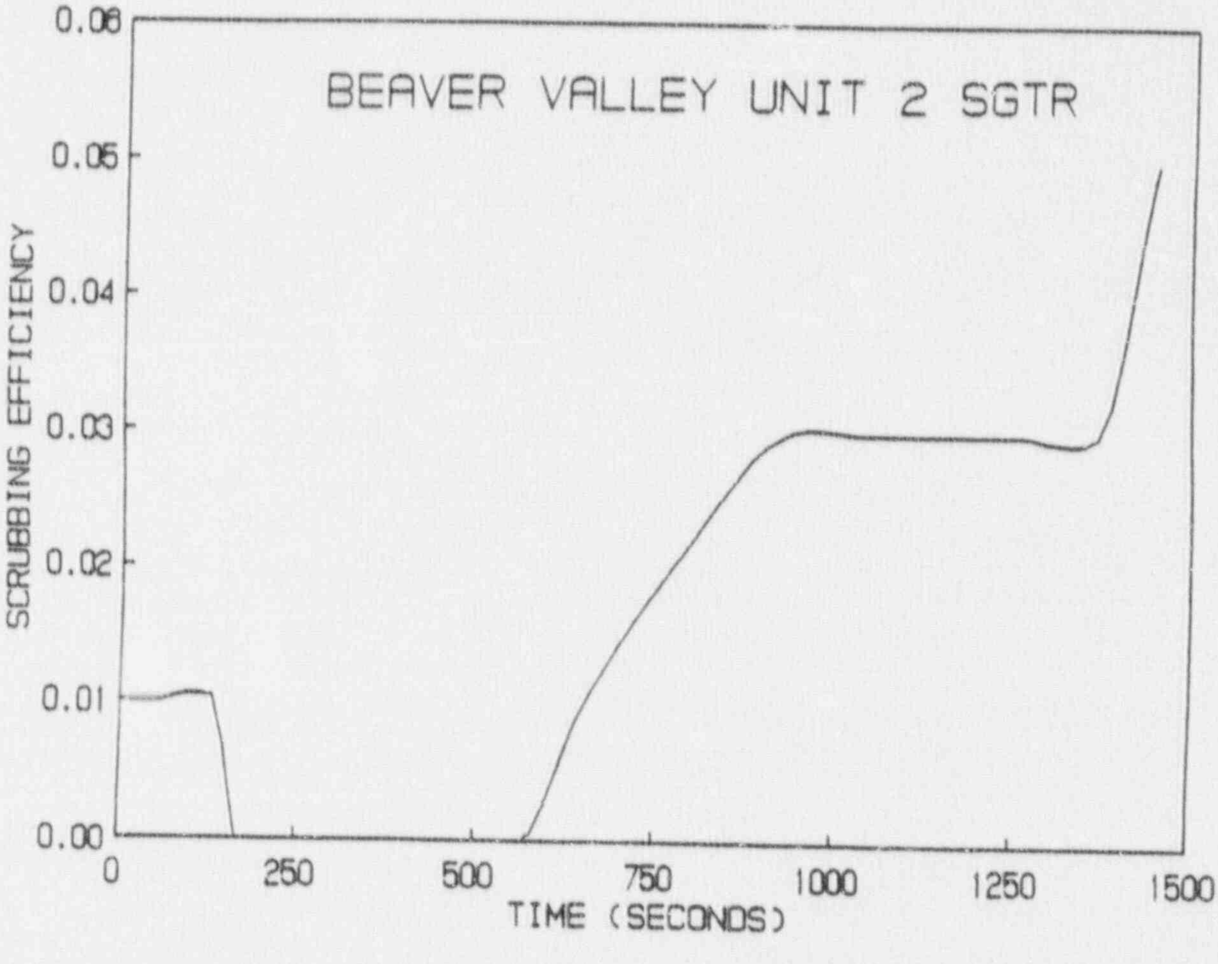

Figure III.15 Iodine Scrubbing Efficiency - Offsite Radiation Dose Analysis

IV. CONCLUSION

An evaluation has been performed for a design basis SGTR for Beaver Valley Power Station Unit 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 the [ ]^{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 the ruptured steam generator atmospheric steam dump valve 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, 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 Beaver Valley Power Station Unit 2 would be acceptable.

V. REFERENCES

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