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PG&E Letter DCL-01-115

U.S. Nuclear Regulatory Commission
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Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 01-05
Revision to Technical Specification 1.1, "Definitions, Dose Equivalent I-131," and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant respectively. This License Amendment Request (LAR) proposes to modify Technical Specification (TS) 1.1, "Definitions, Dose Equivalent I-131," to allow use of thyroid dose conversion factors listed in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979, in the Steam Generator Tube Rupture (SGTR) and Main Steam Line Break (MSLB) radiological consequences analyses.

Revised SGTR and MSLB radiological consequences analyses are proposed which have utilized the thyroid dose conversion factors listed in ICRP Publication 30 based on guidance contained in NRC Regulatory Guide (RG) 1.183 Appendix F. The current SGTR and MSLB analyses assumed thyroid dose conversion factors listed in NRC RG 1.109, consistent with the current TS 1.1. The use of thyroid dose conversion factors listed in ICRP Publication 30 is considered a change in analysis methodology that requires prior NRC review and approval. In addition, the revised SGTR radiological consequences analysis has resulted in more than a minimal increase in consequences that requires prior NRC review and approval.

The revised SGTR radiological consequences analysis has also utilized a revised iodine spiking factor of 335 based on NRC RG 1.183 Appendix F. The current SGTR analysis assumed a spiking factor of 500 based on NUREG-0800 Standard Review Plan section 15.6.3. The use of the iodine spiking factor from RG 1.183 Appendix F is considered a change in the analysis methodology that requires prior NRC review and approval.

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The revised SGTR and MSLB radiological consequences analyses incorporate revised accident-initiated iodine release rates. The accident-initiated iodine spiking rates are derived from the equilibrium iodine appearance rates. Nonconservatism in the calculation of the iodine appearance rates was identified by the vendor that performed the analyses.

To ensure that the current SGTR and MSLB radiological consequence analyses remain bounding until NRC approves the revised analyses, the primary coolant iodine activity is being administratively controlled to a value lower than the 1.0 μCi/g of dose equivalent I-131 allowed by TS 3.4.16, "RCS Specific Activity." This ensures that the current SGTR and MSLB offsite doses are less than the previously calculated values with correction of the nonconservatisms in the iodine appearance rates.

Enclosure 1 provides a description of the proposed changes, the supporting technical analyses, and the significant hazards determination. Enclosures 2 and 3 provided marked-up and revised TS pages, respectively. Enclosure 4 contains the revised SGTR analysis Figures and Enclosure 5 provides the marked-up Final Safety Analysis Report Update sections. The change to TS 1.1, "Definitions, Dose Equivalent I-131," proposed in this LAR has also been proposed in PG&E Letter DCL-01-104, "License Amendment Request 01-04, Revision to Technical Specifications 3.9.4 Containment Penetrations," dated October 17, 2001.

The changes in this LAR are not required to address an immediate safety concern. However, since administrative controls are being applied to TS 3.4.16, "RCS Specific Activity," to ensure conservatism of the current SGTR and MSLB radiological consequences analyses, PG&E requests that the NRC staff review this LAR on a medium priority. PG&E desires approval of this LAR by December 1, 2002, and requests the LAR be made effective upon NRC issuance, to be implemented within 30 days from the date of issuance.

Sincerely,

Gregory M. Rueger
Senior Vice President - Generation and Chief Nuclear Officer

cc: Edgar Bailey, DHS
Ellis W. Merschoff
David L. Proulx
Girija S. Shukla
Diablo Distribution (w/o Enclosure 5)

Enclosures

KJS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

Gregory M. Rueger, of lawful age, first being duly sworn upon oath says that he is Senior Vice President - Generation and Chief Nuclear Officer of Pacific Gas and Electric Company; that he has executed LAR 01-05 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Gregory M. Rueger
Senior Vice President - Generation and Chief Nuclear Officer

Subscribed and sworn to before me this 16th day of November, 2001.

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1204640
Notary Public - California
San Luis Obispo County
My Comm. Expires Dec 12, 2002

REVISED TECHNICAL SPECIFICATION 1.1 THYROID DOSE CONVERSION FACTORS AND REVISED STEAM GENERATOR TUBE RUPTURE AND MAIN STEAM LINE BREAK ANALYSES

1.0 DESCRIPTION

This license amendment request (LAR) proposes to revise Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed change will modify Technical Specification (TS) 1.1, "Definitions, Dose Equivalent I-131," to allow use of thyroid dose conversion factors listed in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers", 1979.

Revised steam generator tube rupture (SGTR) and main steam line break (MSLB) radiological consequences analyses are proposed which have utilized the thyroid dose conversion factors listed in ICRP Publication 30. The current SGTR and MSLB analyses assumed thyroid dose conversion factors listed in NRC Regulatory Guide (RG) 1.109, consistent with the current TS 1.1. The use of thyroid dose conversion factors listed in ICRP Publication 30 is considered a change in analysis methodology that requires prior NRC review and approval. In addition, the revised SGTR radiological consequences analysis has resulted in more than a minimal increase in consequences that requires prior NRC review and approval.

The revised SGTR radiological consequences analysis has also utilized a revised iodine spiking factor of 335 for the exposure to individuals at the exclusion area boundary based on NRC RG 1.183 Appendix F. The current SGTR analysis assumed a spiking factor of 500 based on NUREG-0800 Standard Review Plan (SRP) section 15.6.3. The use of the iodine spiking factor from RG 1.183 Appendix F is considered a change in the analysis methodology that requires prior NRC review and approval.

2.0 PROPOSED CHANGE

This LAR proposes to revise the second sentence of TS 1.1, "Definitions, Dose Equivalent I-131," from

> "The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977"

to

"The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977, or those listed in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979."

Enclosures 2 and 3 provide marked-up and revised TS pages, respectively.

In addition, the SGTR thermal and hydraulic analysis, SGTR radiological consequences analysis, and MSLB radiological consequences analysis are being revised. The revised analyses use a methodology not previously approved at DCPP and result in more than a minimal increase in consequences.

The revised SGTR thermal and hydraulic analysis input changes include reduced maximum emergency core cooling system (ECCS) injection flow, increased operator action time for safety injection (SI) initiation, and conservative modeling to bound a 5 percent reactor coolant system (RCS) flow asymmetry.

The SGTR offsite and control room radiological consequences analysis input changes include the use of thyroid dose conversion factors listed in ICRP Publication 30, a reduced iodine spiking factor listed in RG 1.183 Appendix F, increased steady-state primary-to-secondary flow to the intact steam generators (SG), equilibrium iodine appearance rates based on increased maximum RCS letdown flow, and revised initial reactor coolant fission product specific activity.

The revised MSLB radiological consequences analysis input changes include the use of thyroid dose conversion factors listed in ICRP Publication 30, equilibrium iodine appearance rates based on increased maximum RCS letdown flow, and a reduction in the steam line break accident-induced SG tube leakage rate.

Enclosure 4 contains the revised SGTR thermal and hydraulic analysis figures and Enclosure 5 provides the marked-up Final Safety Analysis Report (FSAR) Update sections.

3.0 BACKGROUND

The changes proposed in this LAR are required to address errors identified by Westinghouse in the SGTR radiological consequences analyses for DCPP. This was identified in Westinghouse letter NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations," dated March 7, 2000. The issues identified in NSAL-00-004 also apply to the MSLB radiological consequences analysis. Significant time was required to develop appropriate revised analysis inputs, to

perform the revised analyses, and to perform a quality review of the revised analyses.

3.1 SGTR Analyses

An SGTR accident results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of some radioactivity to the atmosphere. Therefore, an analysis is performed to assure that the offsite and control room radiological consequences resulting from an SGTR meet the allowable 10 CFR Part 100 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 limits.

The SGTR analysis consists of a thermal and hydraulic analysis to determine the mass releases during a SGTR and a dose analysis to determine the radiological consequences. The SGTR thermal and hydraulic analysis is described in FSAR Update section 15.4.3. The SGTR dose analysis is described in FSAR Update section 15.5.20. These analyses are discussed below.

3.2 SGTR Thermal and Hydraulic Analysis to Demonstrate Margin to SG Overfill

The possibility of SG overfill is a concern since it could potentially result in a significant increase in offsite radiological consequences. Therefore a thermal and hydraulic analysis was performed to demonstrate there is margin to SG overfill, assuming the limiting single failure relative to SG overfill. This analysis was performed in accordance with the methodology in NRC approved WCAP-10698, which is the currently approved method at DCPP. Since the revised analysis uses the currently approved methodology and only changes input assumptions to be consistent with actual plant limits and parameters, this analysis will be incorporated into the DCPP licensing basis via 10 CFR 50.59. The results of this analysis demonstrate there is margin to SG overfill for DCPP.

3.3 SGTR Thermal and Hydraulic Analysis for Radiological Consequences

A thermal and hydraulic analysis was performed to determine the offsite radiological consequences, assuming the limiting single failure relative to offsite doses without SG overfill. Since SG overfill does not occur in the SGTR overfill analysis, the results of this analysis represent the limiting radiological consequences for a SGTR for DCPP. This analysis determines the integrated primary-to-secondary break flow and the mass releases from the ruptured and intact SGs to the condenser and to the atmosphere during a SGTR accident. This information is then used to calculate the quantity of radioactive material released to the environment

and the resulting radiological consequences. The results of the thermal and hydraulic analysis for radiological consequences will be described in FSAR Update section 15.4.3.3.

The licensing basis SGTR thermal and hydraulic analysis for DCPP was documented in WCAP-11723. The WCAP-11723 analysis used the SGTR thermal and hydraulic analysis methodology contained in WCAP-10698, Supplement 1, which was approved by the NRC in a letter dated December 17, 1985. The revised SGTR thermal and hydraulic analysis uses the NRC approved methodology of WCAP-10698, Supplement 1 and incorporates changes to several of the analysis input assumptions to be consistent with actual plant limits and parameters. This analysis will be incorporated into the DCPP licensing basis via 10 CFR 50.59.

3.4 SGTR Radiological Consequences

The radiological consequences of an SGTR were determined based on the thermal and hydraulic analysis described in section 3.3 of this LAR and that will be described in FSAR Update section 15.4.3.3. Both offsite exposures and control room exposures were calculated and will be described in FSAR Update sections 15.5.20.1 and 15.5.20.2, respectively.

The licensing basis SGTR radiological consequences analysis for DCPP was documented in WCAP-11723. The WCAP-11723 analysis used the SGTR offsite radiological consequences methodology contained in WCAP-10698, Supplement 1, which was approved by the NRC in a letter dated December 17, 1985. The revised SGTR radiological consequences analysis is based on the NRC approved methodology of WCAP-10698, Supplement 1. The revised analysis includes some minor changes to the WCAP-10698, Supplement 1, methodology and incorporates changes to several of the analysis input parameters. These changes are described below.

The revised SGTR radiological consequences analysis, based on the thermal and hydraulic analysis described in section 3.3 of this LAR, utilized the thyroid dose conversion factors listed in ICRP Publication 30, 1979. The current analysis used thyroid dose conversion factors listed in NRC RG 1.109, consistent with TS 1.1.

The revised SGTR offsite radiological consequences analysis utilized a revised iodine spiking factor of 335 based on NRC RG 1.183 Appendix F. The current analysis used a spiking factor of 500 based on NUREG-0800 SRP section 15.6.3.

The revised SGTR offsite radiological consequences analysis incorporates revised accident-initiated iodine release rates. The accident-initiated iodine release rates are derived from the equilibrium iodine appearance rates. Nonconservatisms which affect the calculation of iodine appearance rates used in the current SGTR offsite radiological consequences analysis were identified by Westinghouse in letter NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations," dated March 7, 2000. NSAL-00-004 identified nonconservatisms in the values assumed for letdown flow rate, demineralizer iodine removal efficiency, and primary coolant leakage resulting in underestimation of the equilibrium iodine appearance rates, and subsequent underestimation of the accident-initiated iodine release rates.

The revised SGTR offsite radiological consequences incorporates other input changes including revised initial reactor coolant fission product specific activity, revised disintegration energies, a revised iodine transport model, and revised whole body and skin offsite dose calculations. The analysis now accounts for the nonconservatisms in the accident initiated iodine release rates.

A revised SGTR control room radiological consequences analysis is also proposed which incorporates most of the input changes made for the offsite radiological consequences calculation.

3.5 MSLB Radiological Consequences

An MSLB accident results in a small amount of leakage of contaminated reactor coolant into the secondary system and a release of secondary activity to the atmosphere. Therefore, an analysis is performed to assure that the offsite and control room radiological consequences resulting from an MSLB meet the allowable 10 CFR Part 100 and 10 CFR Part 50, Appendix A, GDC 19 limits respectively.

A revised MSLB radiological consequences analysis, based on the steam releases expected following a major steam line break without condenser bypass contained in the current FSAR Update Table 15.5-34, is proposed which has utilized the thyroid dose conversion factors listed in ICRP Publication 30, 1979.

The revised MSLB radiological consequences analysis incorporates revised accident-initiated iodine release rates to correct nonconservatisms in the equilibrium iodine appearance rates identified in Westinghouse letter NSAL-00-004. A revised letdown flow rate, letdown flow rate uncertainty, and demineralizer iodine removal efficiency has been used to correct these nonconservatisms.

To compensate for the increases in the accident-initiated iodine release rates due to correction of the nonconservative equilibrium iodine appearance rates, a reduced accident-induced primary-to-secondary leakage rate has been used. The accident-induced primary-to-secondary leakage rate is part of the basis for the maximum primary-to-secondary leakage limit for SG indications which are left in service under the TS 5.5.9 SG tube alternate repair criteria (ARC) for DCPP Units 1 and 2. To support the TS 5.5.9 SG ARC, the MSLB radiological consequences analysis must account for the total accident-induced leakage due to all indications left in service under the TS 5.5.9 SG ARC in the determination of the dose consequences. The current TS 5.5.9 SG tube ARC applies to indications due to outer diameter stress corrosion cracking (ODSCC) at tube support intersections and indications in the Westinghouse Explosive Tube Expansion (WEXTEX) Region.

The revised MSLB radiological consequences analysis also incorporates revised initial reactor coolant and secondary coolant activity.

4.0 TECHNICAL ANALYSES

4.1 SGTR Thermal and Hydraulic Analysis

4.1.1 SGTR Thermal and Hydraulic Analysis Input Changes

A revised SGTR thermal and hydraulic analysis was performed that used the NRC approved methodology of WCAP-10698, Supplement 1 and incorporated changes to the analysis input parameters including reduced maximum ECCS injection flow, increased operator action time to terminate SI, and conservative modeling to bound a 5 percent RCS flow asymmetry.

For this analysis, the single failure assumed is the power-operated relief valve (PORV) on the ruptured SG fails open at the time the ruptured SG is isolated. This is consistent with the analysis in WCAP-11723 and WCAP-10698, Supplement 1. Before proceeding with the recovery operations, the failed-open PORV on the ruptured SG was assumed to be isolated by locally closing the associated block valve. It was assumed that the ruptured SG PORV is isolated at 30 minutes after the valve is assumed to fail open consistent with the current licensing basis analysis in WCAP-11723. After the ruptured SG PORV is isolated, an additional delay time of 5 minutes is assumed for the operator to initiate the RCS cooldown.

The details of the input changes for the revised analysis are described below.

Reduced Maximum ECCS Injection Flow

A DCPP specific maximum ECCS flow calculation has been performed using the PROTOFLOW™ code and DCPP specific piping data in order to reduce the conservatisms in the maximum ECCS flow values. This calculation resulted in reduced maximum ECCS flow.

At a pressure of 1000 psig, the revised ECCS flow assuming two centrifugal charging pumps and 2 safety injection pumps operating is approximately 1119 gallons per minute, which is 53 gallons per minute lower than the previously calculated value.

Increased Operator Action Time to Terminate SI

The revised analysis incorporates a new 2 minute operator action time from the end of depressurization to terminate the SI flow. The licensing basis analysis assumed a 1 minute operator action time for this action. However, as indicated in PG&E letter DCL-91-009, dated January 17, 1991, the 1990 plant specific operator action time data identified a delay time of 1.4 minutes to terminate the SI flow. The 2 minute operator action time to terminate the SI flow is greater than the plant specific operator action time data, and therefore, is conservative for the SGTR thermal and hydraulic analysis. The revised analysis assumes the same operator action times as the licensing basis analysis for the actions to isolate auxiliary feedwater (AFW) flow from the turbine driven AFW pump (5.54 minutes after reactor trip), isolate AFW flow to the ruptured SG and steam flow from the ruptured SG (10 minutes from accident initiation), initiate cooldown (5 minutes from steam line isolation), and initiate depressurization (4 minutes from end of cooldown).

Conservative Modeling to Bound a 5 Percent RCS Flow Asymmetry

The revised analysis also incorporates conservative modeling to bound the potential asymmetric effects on the SG initial conditions associated with a 5 percent flow imbalance between RCS loops. A 5 percent flow imbalance between RCS loops will bound a maximum of 10 percent difference in the loop-to-loop SG tube plugging. The flow asymmetry was modeled by reducing SG initial secondary side mass based on steady state SG secondary mass calculations performed with the GENF code at a total RCS flow 5 percent different from the thermal design flow.

4.1.2 SGTR Thermal and Hydraulic Analysis Results

Transient Results

The sequence of events for the SGTR thermal and hydraulic analysis is presented in Table 1. The transient pressurizer pressure, secondary pressure, pressurizer level, ruptured loop temperatures, intact loop temperatures, primary-to-secondary break flow, break flow flashing fraction, total flashed break flow, ruptured SG mass release rate, intact SGs mass release rate, ruptured SG water volume, and ruptured SG water mass are contained in Figures 1 through 12 in Enclosure 4 of this LAR.

Following the SGTR, the RCS pressure decreases as shown in Figure 1 due to the primary to secondary leakage. In response to this depressurization, the reactor trips on overtemperature -ΔT at approximately 112 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 2. The RCS pressure and pressurizer level also decrease more rapidly following reactor trip as shown in Figures 1 and 3. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 152 seconds. The AFW flow from the turbine driven (TD) AFW pump is isolated at 5.54 minutes after reactor trip.

Recovery actions to isolate the ruptured SG begin by throttling the AFW flow to the ruptured SG and isolating steam flow from the ruptured SG. AFW flow to the ruptured SG and steam flow from the ruptured SG are assumed to be identified and isolated when the narrow range level reaches 28 percent on the ruptured SG or at 10 minutes after initiation of the SGTR, whichever time is greater. The time to reach 28 percent is approximately 10.8 minutes, and thus the ruptured SG is assumed to be isolated at that time. The ruptured SG PORV is also assumed to fail open at this time. The failure causes the SG to rapidly depressurize, which results in an increase in primary to secondary leakage. The ruptured SG depressurization causes a cooldown in the intact SG loops. It is assumed that the time required for the operator to identify that the ruptured SG PORV is open and to locally close the associated block valve is 30 minutes. At 2452 seconds the depressurization of the ruptured SG is

terminated and the ruptured SG pressure begins to increase as shown in Figure 2.

Following isolation of the ruptured SG, the operators cooldown the RCS to establish RCS subcooling margin. After the block valve for the ruptured SG PORV is closed, there is a 5 minute operator action time imposed prior to initiation of cooldown. The depressurization of the ruptured SG due to the failed-open PORV affects the RCS cooldown target temperature since the temperature is determined based upon the pressure in the ruptured SG at that time. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the intact SG PORVs. The cooldown is continued until RCS subcooling at the ruptured SG pressure is 36°F. The cooldown begins at 2752 seconds and is completed at 3612 seconds.

The reduction in the intact SGs pressure during the cooldown is shown in Figure 2 and the effect of the cooldown on the ruptured loop and intact loop RCS temperature is shown in Figures 4 and 5. The RCS pressure and pressurizer level also decrease during this cooldown process due to shrinkage of the reactor coolant as shown in Figures 1 and 3. The break flow flashing fraction is calculated throughout the transient based on the difference between the enthalpy of the break flow and the saturation enthalpy at the ruptured SG pressure as shown in Figure 7. Break flow is calculated to stop flashing at approximately 2980 seconds as a result of the reduction in primary coolant temperature associated with the cooldown (Figure 4) and the increase in ruptured SG pressure following isolation of the failed open PORV (Figure 2).

After the RCS cooldown is completed, a 240 second operator action time is included prior to the RCS depressurization. The RCS depressurization is performed to assure adequate reactor coolant inventory prior to terminating SI flow. With the reactor coolant pumps stopped, normal pressurizer spray is not available and thus the RCS is depressurized by opening a pressurizer PORV. The RCS depressurization is initiated at 3860 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured SG pressure and pressurizer level is greater than the allowance of 12 percent for pressurizer level uncertainty, or pressurizer level is greater than 74 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. For this case, the RCS depressurization is terminated at 3988 seconds because the RCS pressure is reduced to less

than the ruptured SG pressure and the pressurizer level is above 12 percent. The RCS depressurization reduces the break flow as shown in Figure 6 and increases SI flow to refill the pressurizer, as shown in Figure 3.

After depressurization is completed, an operator action time of 2 minutes was assumed prior to SI termination. The SI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated when RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 12 percent allowance for uncertainty. These requirements were satisfied and SI termination actions were performed at 4108 seconds by closing off the SI flow path. After SI termination the RCS pressure begins to decrease as shown in Figure 1.

The intact SG PORVs also automatically open to dump steam to maintain the prescribed RCS temperature to ensure that subcooling is maintained. When the PORVs are opened, the increased energy transfer from primary to secondary also aids in the depressurization of the RCS to the ruptured SG pressure. The ruptured SG pressure increases to the PORV setpoint and steam release is reinitiated. SG pressure is maintained at the SG PORV setpoint rather than the safety valve setpoint for modeling efficiency. This modeling is conservative since it delays break flow termination by requiring the RCS pressure to drop further, maximizes the break flow rate by maintaining a larger primary-to-secondary pressure differential, and results in more steam release from the ruptured SG. The primary to secondary leakage continues after the SI flow is terminated until the RCS and ruptured steam generator pressures equalize at 5040 seconds.

Mass Releases

The steam releases from the ruptured and intact SGs, the feedwater flows to the ruptured and intact SGs, and the primary-to-secondary leakage into the ruptured SG were determined from the LOFTTR2 code results for the period from the initiation of the accident until the primar-to-secondary leakage is terminated.

Following the termination of leakage, it was assumed that the RCS and intact SGs conditions are maintained stable for a 20 minute period until

the cooldown to cold shutdown is initiated. The PORVs for the intact SGs were then assumed to be used to cool down the RCS to the residual heat removal (RHR) system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hour. The RCS and the intact SGs temperatures at 2 hours were then determined using the RCS and intact SGs parameters at the time of leakage termination and the RCS cooldown rate. The steam releases and the feedwater flows for the intact SGs for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact SGs conditions at the time of leakage termination and at 2 hours. Since the ruptured SG is isolated, no change in the ruptured SG 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 SG was then assumed to be performed immediately following the completion of the RCS cooldown. The ruptured SG was assumed to be depressurized to the RHR in-service pressure of 405 psia via steam release from the ruptured SG PORV, since this maximizes the steam release from ruptured SG 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 SG 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 SGs from a mass and energy balance using the RCS and SG conditions at 2 hours and at the RHR system in-service conditions. The steam released from the ruptured SG from 2 to 8 hours was determined based on a mass and energy balance for the ruptured SG using the conditions at the time of leakage termination and saturated conditions at the RHR in-service pressure.

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.

Radioactivity released to the atmosphere prior to the reactor trip from the condenser will be through the condenser air ejector. After the

reactor trip, the releases to the atmosphere are assumed to be via the SG PORVs.

The mass releases for the SGTR accident assuming failure and isolation of the ruptured SG PORV are presented in Table 2. The results indicate that approximately 151,500 pounds mass (lbm) of steam are released to the atmosphere from the ruptured SG within the first 2 hours for consideration of the SRP 15.6.3 2 hour exclusion area boundary dose. After 2 hours, 41,400 lbm of steam are released to the atmosphere from the ruptured SG. A total of 272,400 lbm of primary water is transferred to the secondary side of the ruptured SG before break flow is terminated. A total of 17,904 lbm of this break flow is assumed to flash to steam upon entering the SG.

The SGTR mass results of the thermal and hydraulic analysis are used as input to the SGTR radiological consequences analysis presented in Section 4.2 of this LAR. The SGTR thermal and hydraulic analysis results are also discussed in the marked-up FSAR Update section 15.4.3.3 contained in Enclosure 5 of this LAR.

Table 1
Sequence of Events
SGTR Thermal and Hydraulic Analysis

Event	Time (seconds)
Steam Generator Tube Rupture	0
Reactor Trip	112
SI Actuation	152
TDAFW Pump Flow Isolated	444
Ruptured SG Isolated	650
Ruptured SG PORV Fails Open	652
Ruptured SG Block Valve Closed	2452
RCS Cooldown Initiated	2752
RCS Cooldown Terminated	3612
RCS Depressurization Initiated	3860
RCS Depressurization Terminated	3988
SI Terminated	4108
Break Flow Terminated	5040

Table 2
SGTR Mass Releases
Total Mass Flow (Pounds)

	Time Period			
	Time Zero to Time of Reactor Trip*	**Time of Reactor Trip to Time at Which Break Flow is Terminated***	**Time at Which Break Flow is Terminated to 2 Hours**	**2 Hours to Time at Which RCS Reaches RHR In-Service Conditions***
Ruptured SG - Condenser - Atmosphere - Feedwater	 119,500 0 110,700	 0 151,500 32,700	 0 0 0	 0 41,400 0
Intact SGs - Condenser - Atmosphere - Feedwater	 354,400 0 354,400	 0 216,100 429,500	 0 163,300 195,600	 0 973,400 1,037,900
Break Flow	8,800	263,600	0	0
Flashed Break Flow	1,570	16,334	0	0

* Reactor trip occurs at 112 seconds; break flow is terminated at 5040 seconds; Residual Heat Removal (RHR) system conditions are reached at 8 hours.

4.2 SGTR Radiological Consequences

4.2.1 SGTR Offsite Radiological Consequences Analysis Input Changes

A revised SGTR radiological consequences analysis was performed. The analysis is based on the NRC approved methodology of WCAP-10698, Supplement 1. The revised analysis includes some minor changes to the WCAP-10698, Supplement 1, methodology and incorporates changes to several of the analysis input parameters.

The revised SGTR offsite radiological consequences analysis is based on the thermal and hydraulic analysis described in section 4.1, and uses revised initial reactor coolant fission product specific activity, revised thyroid dose conversion factors, a revised iodine spiking factor, revised accident-initiated iodine release rates, revised disintegration energies, revised iodine transport model, and revised whole body and skin offsite dose calculations. These input changes are described below. The remaining offsite dose analysis assumptions and methodology are the same as documented in WCAP-11723.

Revised Initial Reactor Coolant Fission Product Specific Activity

The revised SGTR offsite radiological consequences analysis incorporates revised steady state initial RCS iodine concentrations at 1 micro Curie per gram (μCi/gm) and 60 μCi/gm of I-131 dose equivalent concentration (DEC) for the accident-initiated iodine spike case and the pre-existing iodine spike case respectively. The initial iodine concentrations were calculated assuming 1 percent fuel defect, a 1 gpm allowable primary-to-secondary leak rate prior to the SGTR accident, a TS 3.4.13.b maximum allowable unidentified RCS leakage of 1 gpm, and a TS 3.4.13.c maximum allowable identified RCS leakage of 10 gpm, a two year fuel cycle at 3580 megawatts thermal (MWt), a 75 gpm letdown flow rate, and a 90 percent demineralizer iodine removal efficiency. The initial noble gas nuclides were also calculated based on a 1 percent fuel defect.

The initial reactor coolant fission product specific activity used in the SGTR offsite radiological consequences analysis are contained in Table 3. The initial RCS iodine concentration activities (Table 3) were converted to activities based on a letdown flow rate of 143 gpm and a 100 percent demineralizer iodine removal efficiency to determine the dose equivalent I-131 values for primary coolant based on 1.0 μCi/gm of dose equivalent I-131, primary coolant based on 60 μCi/gm of dose equivalent I-131, and secondary coolant based on 0.1 μCi/gm of dose equivalent I-131. These values are contained in Table 5.

Revised Thyroid Dose Conversion Factors

The revised SGTR offsite radiological consequences analysis utilizes the thyroid dose conversion factors listed in ICRP Publication 30, 1979. The use of the ICRP Publication 30 thyroid dose conversion factors is consistent with current analysis techniques as indicated in RG 1.183, "Alternate Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Plants," July 2000. The NRC staff used the ICRP Publication 30 thyroid dose conversion factors in the independent verification of the DCPP licensing basis SGTR radiological consequences analysis as documented in the NRC letter dated April 3, 1991. The dose conversion values used for the revised analysis are contained in Table 4.

The ICRP Publication 30 dose conversion values were used to convert the one percent fuel defect initial RCS iodine concentration activities (Table 3) to dose equivalent iodine-131 (I-131) values and were used to calculate the dose resulting from the iodine releases. The Dose equivalent I-131 values for primary coolant based on 1.0 μCi/gm of dose equivalent I-131, primary coolant based on 60 μCi/gm of dose equivalent I-131, and secondary coolant based on 0.1 μCi/gm of dose equivalent I-131, are contained in Table 5.

Revised Iodine Spiking Factor

For the accident-initiated iodine spike case where the SGTR causes an iodine spike in the primary system, the revised SGTR radiological consequences analysis incorporates an iodine spiking factor of 335 for the exclusion area boundary dose calculation. The iodine spiking factor is used to determine the increase in primary coolant iodine concentration. The current SGTR analysis assumed a spiking factor of 500 based on SRP section 15.6.3. The use of the iodine spiking factor of 335 is consistent with current analysis techniques as indicated in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident." For the accident-initiated iodine spike case low population zone and control room dose calculations, a spiking factor of 500 was used for additional conservatism.

Revised Accident-initiated Iodine Release Rates

The revised SGTR offsite radiological consequences analysis incorporates revised accident-initiated iodine release rates that are derived from equilibrium iodine appearance rates. Nonconservatisms in the assumptions for letdown flowrate, demineralizer iodine removal efficiency, and primary coolant leakage, which affect the calculation of

iodine appearance rates used in the current SGTR offsite radiological consequences analysis, were identified by Westinghouse in letter NSAL-00-004.

In the licensing basis analysis calculation of the equilibrium iodine appearance rates, nonconservatisms in the values assumed for letdown flow rate (75 gpm), demineralizer iodine removal efficiency (decontamination factor of 10), and primary coolant leakage (0 gpm unidentified, 0 gpm identified primary-to-secondary) resulted in underestimation of the equilibrium iodine appearance rates and the subsequent underestimation of the accident-initiated iodine release rates. A revised calculation of the equilibrium iodine appearance rates assumed bounding values for letdown flow rate (120 gpm), letdown flow rate uncertainty (10 percent), demineralizer iodine removal efficiency (infinite decontamination factor), and primary coolant leakage (10 gpm unidentified, 1 gpm identified primary-to-secondary) which maximize the equilibrium iodine appearance rates and result in conservative accident-initiated iodine release rates. The total effective letdown flow assumed in the revised calculation of the equilibrium iodine appearance rates is 143 gpm which consists of 120 gpm letdown flow, 12 gpm flow rate uncertainty, 10 gpm identified leakage from the RCS, and 1 gpm unidentified leakage from the RCS. The revised accident-initiated iodine release rates are contained in Table 6.

The accident-initiated iodine spike was allowed to continue until 8 hours after the start of the accident. In the licensing basis analysis calculation, the spike was assumed to be terminated at 3.1 hours. The spike duration was extended in response to NRC comments on recent analyses performed for other plants.

Revised Disintegration Energies

Revised beta and gamma disintegration energies were used to calculate the control room beta doses and the offsite gamma doses based on ENDF-223, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," dated October 1975. The disintegration energies used in the current licensing basis analysis were based on ORNL-4628, "ORIGEN - The ORNL Isotope Generation and Depletion Code," dated May 1973. The revised beta and gamma disintegration energies are described in Table 7.

Revised Iodine Transport Model

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 was not calculated and was conservatively neglected. This iodine removal was calculated and credited in the current licensing basis analysis, however, it is no longer considered in standard Westinghouse dose analyses and is conservatively neglected in this new analysis.

No SG tube uncovery was assumed for the revised offsite dose consequences analysis. The issue of SG tube bundle uncovery was considered in a Westinghouse Owners Group (WOG) program WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," dated March 1992. The WOG program concluded that the effect of SG tube uncovery is essentially negligible for the limiting SGTR transient. The WOG program concluded that the SG tube uncovery issue could be closed without any further investigation or generic restrictions. The NRC review of the WOG submittal was documented in a NRC letter dated March 10, 1993 and concluded "*... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved.*" The current licensing basis analysis was completed prior to the resolution of this tube uncovery issue and conservatively modeled the direct release of all iodine transferred to the ruptured SG in the break flow when the tubes were assumed to be uncovered.

Since there is no penalty taken for tube uncovery and no iodine scrubbing was credited in the revised radiological consequences analysis, the location of the tube rupture is not significant for the radiological analysis. However, the thermal and hydraulic analysis presented in section 4.1.1 conservatively addressed the issue of the location of the tube rupture in the calculations of break flow and flashing of break flow.

Revised Whole Body and Skin Offsite Dose Calculation

The revised offsite dose calculation did not consider the contribution due to beta doses. Although these were calculated and reported in the current licensing basis analysis in WCAP-11723, the current interpretation of the acceptance criteria for offsite doses contained in SRP 15.6.3 is that the beta dose is not considered since it has a negligible impact on the dose results.

The whole body dose via cloud immersion was calculated combining the dose from the released noble gases with the dose from the iodine

releases. This is more conservative than the current licensing basis analysis calculations, which only considered the contribution from noble gases.

The offsite whole body gamma doses were calculated using the following equation:

$$D_{WB} = 0.25 \sum_{i} \left[E_{\gamma i} \left(\sum_{j} (IAR)_{ij} (\chi / Q)_{j} \right) \right]$$

where:

D_{WB} = whole body dose via cloud immersion (Rem)

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis)

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

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/mr^3)

Break flow, flashing break flow, and steam releases from the intact and ruptured SGs are modeled using data from the thermal and hydraulic analysis in section 4.1.2 of this LAR.

A total primary-to-secondary leak rate prior to the accident is assumed to be 1.0 gpm. The leakage to the intact SGs is assumed to persist for the duration of the accident. Atmospheric conditions are assumed in determining the density for this leakage.

The offsite doses were calculated for the thermal hydraulic analysis presented in section 4.1.1 of this LAR. Mass transfer data used in the analysis was taken from the primary-to-secondary break flow in Figure 6, the flashed break flow in Figure 8, the ruptured SG mass release in Figure 9, and the intact SG mass releases in Figure 10.

Table 3
Reactor Coolant Fission Product Specific Activity Based on 1 Percent Fuel Defects

Nuclide	Specific Activity (μCi/gm)
I-131	2.744
I-132	0.703
I-133	3.845
I-134	0.481
I-135	2.036
Kr-85m	2.141
Kr-85	6.209
Kr-87	1.232
Kr-88	3.907
Xe-131m	2.523
Xe-133m	3.911
Xe-133	256.3
Xe-135m	0.449
Xe-135	8.663
Xe-138	0.568

Table 4
Thyroid Dose Conversion Factors

Nuclide	DCF (Rem/Curie)*
I-131	1.07×10^6
I-132	6.29×10^3
I-133	1.81×10^5
I-134	1.07×10^3
I-135	3.14×10^4

* ICRP Publication 30 provides the dose conversion factors in units of sievert/becquerel.

Table 5
Iodine Specific Activities (μCi/gm) in the Primary Coolant

	Primary Coolant		Secondary Coolant
Nuclide	1.0 μCi/gm	60 μCi/gm	0.1 μCi/gm
1-131	0.793	47.58	0.0793
1-132	0.204	12.24	0.0204
1-133	1.113	66.78	0.1113
1-134	0.139	8.34	0.0139
1-135	0.589	35.34	0.0589

Table 6
Iodine Spike Appearance Rates (Curies/Minute)
Based on 1.0 μCi/gm of Dose Equivalent I-131 Primary Coolant Activity

I-131	I-132	I-133	I-134	I-135
221.5	178.0	375.5	256.0	282.5

Table 7
Disintegration Energies

Nuclide	Gamma Disintegration Energy (Mev/Dis)	Beta Disintegration Energy (Mev/Dis)
I-131	0.38	0.19
I-132	2.2	0.52
I-133	0.6	0.42
I-134	2.6	0.69
I-135	1.4	0.43
Kr-85m	0.16	0.25
Kr-85	0.0023	0.25
Kr-87	0.79	1.3
Kr-88	2.2	0.25
Xe-131m	0.0029	0.16
Xe-133m	0.02	0.21
Xe-133	0.03	0.15
Xe-135m	0.43	0.099
Xe-135	0.25	0.32
Xe-138	1.2	0.66

4.2.2 SGTR Offsite Radiological Consequences Analysis Results

The pre-accident iodine spike thyroid doses for the SGTR analysis are calculated at the exclusion area boundary (EAB) and low population zone (LPZ) and are summarized in Table 8. The table includes the results of the current licensing basis analysis reported in WCAP-11723 and the applicable NUREG-0800 SRP section 15.6.3 limits. The applicable limits are met.

The accident-initiated iodine spike thyroid doses are calculated at the EAB and LPZ and are summarized in Table 9. The table includes the results of the current licensing basis analysis based on WCAP-11723 and the applicable NUREG-0800 SRP section 15.6.3 limits. The current licensing basis dose for the EAB (29.5 Rem) includes an increase to the WCAP-11723 reported value as the result of an evaluation for increased pressurizer pressure uncertainty. The current licensing basis dose for the LPZ (1.3 Rem) includes an increase to the WCAP-11723 reported value as the result of a typographical error in the WCAP-11723 reported value.

The results in Table 9 demonstrate that, with the exception of the 2 hour EAB thyroid dose, the applicable SRP guideline values are met. The calculated 2 hour EAB thyroid dose is 30.5 Rem, which is less than 1.5 percent above the SRP 15.6.3 guideline value.

In the RG 1.183 methodology, the total effective dose equivalent (TEDE) is calculated rather than a thyroid and whole body dose. The RG 1.183 TEDE limit is 2.5 Rem at the EAB for the SGTR accident-initiated iodine spike. The TEDE dose is the combination of the committed effective dose equivalent (CEDE) dose and the whole-body dose. The CEDE dose from iodine is approximately 3 percent of the thyroid dose. Therefore, under the RG 1.183 methodology, the 30.5 Rem thyroid dose would be equivalent to approximately 0.92 Rem CEDE and combined with the 0.39 Rem whole-body dose (Table 10) to obtain a TEDE dose of 1.25 Rem. This is well below the RG 1.183 limit of 2.5 Rem TEDE for the accident-initiated iodine spike case.

In conclusion, the 2 hour EAB thyroid dose has been calculated to be 30.5 Rem, which is 1.5 percent above the SRP 15.6.3 guideline value. The 2 hour EAB thyroid dose has been compared against the conservative accident-initiated iodine spike thyroid dose SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB dose thyroid dose would be equivalent to a RG 1.183 methodology TEDE of 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem, is considered to be acceptable. The 2 hour EAB thyroid dose of 30.5 Rem meets the 10 CFR 100 dose limit of 300 Rem for the first 2 hours at the EAB.

The whole body doses, including the contribution from iodines, have been calculated at the EAB and LPZ and are summarized in Table 10. The iodine contribution from the limiting iodine spike case was used. The table includes the results of the current licensing basis analysis reported in WCAP-11723 and the applicable SRP section 15.6.3 limits. The applicable limits are met. The SGTR offsite radiological consequences

analysis results are discussed in the marked-up FSAR Update section 15.5.20.1 contained in Enclosure 5.

Table 8
Pre-Accident Iodine Spike Thyroid Doses

	Calculated Doses (Rem)	WCAP-11723 Doses (Rem)	SRP 15.6.3 Guideline Value
Exclusion Area Boundary (0-2 hr.)	74	192.4	300
Low Population Zone (0-8 hr.)	3.2	8.0	300

Table 9
Accident-Initiated Iodine Spike Thyroid Doses

	Calculated Doses (Rem)	WCAP-11723 Doses (Rem)	SRP 15.6.3 Guideline Value
Exclusion Area Boundary (0-2 hr.)	30.5	29.5	30
Low Population Zone (0-8 hr.)	2.1	1.3	30

Table 10
Total Whole Body Gamma Doses

	Calculated Doses (Rem)	WCAP-11723 (Rem)	SRP 16.6.3 Guideline Value
Exclusion Area Boundary (0-2 hr.)	0.39	0.23	2.5
Low Population Zone (0-8 hr.)	0.02	0.01	2.5

4.2.3 SGTR Control Room Radiological Consequences Analysis Input Changes

The licensing basis SGTR control room radiological consequences analysis for DCPP is documented in FSAR Update section 15.5.20.2.

A revised SGTR control room radiological consequences analysis, based on the thermal and hydraulic analysis described in section 4.1.2, is proposed which has utilized revised initial reactor coolant fission product specific activity, revised thyroid dose conversion factors, revised accident-initiated iodine release rates, revised disintegration energies, and a revised iodine transport model. These input changes are the same as those described in section 4.2.1.

Thyroid, whole body gamma, and beta skin doses are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation was continued to account for additional doses due to continued occupancy.

The control room was modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake were used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow were used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 11.

Control room thyroid doses were calculated using the following equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right]$$

where:

D_{Th} = thyroid dose via inhalation (Rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (Rem/Ci)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

$(BR)_j$ = breathing rate during time interval j (m^3/sec)

The breathing rate used is contained in Table 12.

Control room whole body doses are calculated using the following equation:

$$D_{WB} = 0.25 * \left(\frac{1}{GF}\right) * \sum_i E_{\gamma i}\left(\sum_j Conc_{ij}\right)$$

where:

D_{WB} = whole body dose via cloud immersion in Rem.

GF = geometry factor, calculated based on Reference 12, using the equation $GF = \frac{1173}{V^{0.338}}$ where V is the control room volume in ft^3

$E_{\gamma I}$ = average gamma disintegration energy for isotope i (Mev/dis)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow ($Ci\text{-}sec/m^3$)

Control room skin doses are calculated using the following equation:

$$D_{\beta} = 0.23 * \sum_i E_{\beta i}\left(\sum_j Conc_{ij}\right)$$

where:

D_{β} = whole body dose via cloud immersion (Rem)

$E_{\beta I}$ = average beta disintegration energy for isotope i (Mev/dis)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow ($Ci\text{-}sec/m^3$)

Break flow, flashing break flow, and steam releases from the intact and ruptured SGs are modeled using data from the thermal and hydraulic analysis in section 4.1.2.

The total primary-to-secondary leak rate prior to the accident was assumed to be 1.0 gpm. The leakage to the intact SGs was assumed to persist for the duration of the accident. Atmospheric conditions are assumed in determining the density for this leakage.

The control room doses are calculated for the thermal hydraulic analysis presented in section 4.1.2. Mass transfer data was taken from the primary-to-secondary break flow in Figure 6, the flashed break flow in Figure 8, the ruptured SG mass release in Figure 9, and the intact SG mass releases in Figure 10.

The calculations determined the thyroid doses based on a pre-accident iodine spike and based on an accident-initiated iodine spike. Both spike assumptions consider 0.1 µCi/gm D.E. I-131 secondary activity. The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases. The whole body doses are calculated with the limiting iodine releases (either pre-accident spike or accident-initiated iodine spike).

Table 11
Control Room Model

Control Room Isolation Signal Generated	Time of SI signal from Section 1.2
Delay in Control Room Isolation After Isolation Signal is Generated	35 Seconds
Control Room Volume	170,000 ft^3
Control Room Unfiltered In-Leakage	10 cfm
Control Room Unfiltered Inflow	
Normal Mode	4200 cfm
Emergency Mode	0 cfm
Control Room Filtered Inflow	
Normal Mode	0 cfm
Emergency Mode	2100 cfm
Control Room Filtered Recirculation	
Normal Mode	0 cfm
Emergency Mode	2100 cfm
Control Room Filter Efficiency	95%

Table 12
Additional SGTR Control Room Dose Related Inputs

Time (hours)	Control Room χ/Q Filtered Pressurization (sec/m^3)	Control Room χ/Q Unfiltered Infiltration (sec/m^3)	Control Room Breathing Rate (m^3/sec)	Control Room Occupancy Factor
0 - 8	7.05×10^{-5}	1.96×10^{-4}	3.47×10^{-4}	1.0
8 - 24	5.38×10^{-5}	1.49×10^{-4}	3.47×10^{-4}	1.0
24 - 96	3.91×10^{-5}	1.08×10^{-4}	3.47×10^{-4}	0.6
> 96	2.27×10^{-5}	6.29×10^{-5}	3.47×10^{-4}	0.4

4.2.4 SGTR Control Room Radiological Consequences Analysis Results

The pre-accident iodine spike thyroid, accident-initiated iodine spike thyroid, whole body gamma, and beta skin doses for SGTR were calculated for the control room and are summarized in Table 13. The iodine contribution from the limiting iodine spike case was used. The control room dose guidelines are specified in NUREG-0800 SRP section 6.4 based on GDC 19. Doses in the control room must be less than 30 Rem thyroid, 5 Rem whole body, and 30 Rem beta-skin. The table includes the applicable SRP section 6.4 limits and the current licensing basis analysis FSAR Update section 15.5.20.2 values. The results in Table 13 demonstrate that the applicable SRP 6.4 guideline values are met. The SGTR control room radiological consequences analysis results are discussed in the marked-up FSAR Update section 15.5.20.2 contained in Enclosure 5.

Table 13
SGTR Control Room Doses

	Calculated Doses (Rem)	FSAR Section 5.5.20.2	SRP 6.4 Guideline Value
Pre-Accident Iodine Spike - Thyroid (0-30 Days)	2.3	1.59	30
Accident-Initiated Iodine Spike - Thyroid (0-30 Days)	1.4	0.24	30
Accident-Initiated Iodine Spike - Whole Body Gamma (0-30 Days)	2.7E-4	0.03	5
Accident-Initiated Iodine Spike - Beta Skin Dose (0-30 Days)	2.0E-2	2.7E-2	30

4.3 MSLB Radiological Consequences

4.3.1 MSLB Radiological Consequences Analysis Input Changes

The licensing basis MSLB radiological consequences analysis for DCPP was submitted to the NRC in letter DCL-97-034 dated February 26, 1997 in support of the voltage-based SG ARC for ODSSC at tube support plate intersections. The NRC approved the MSLB radiological consequences analysis in license amendments 124 for DCPP Unit 1 and 122 for DCPP Unit 2 in a letter dated March 12, 1998. The licensing basis analysis was performed based on the methodology in SRP 15.1.5, Appendix A, and in accordance with the primary coolant specific activity limits in Generic Letter 95-05.

A revised MSLB radiological consequences analysis, based on the steam releases expected following a major steam line break without condenser bypass contained in FSAR Update Table 15.5-34, is proposed which has utilized revised thyroid dose conversion factors, revised accident-initiated iodine release rates, revised primary and secondary activities, and revised accident-induced primary-to-secondary leakage. The details of the revised thyroid dose conversion factors and revised accident-initiated iodine release rates are the same as described in section 4.2.1. The revised primary and secondary activities and accident-induced primary-to-secondary leakage are described below. The remaining dose analysis

assumptions and methodology are the same as those documented in letter DCL-97-034.

Revised Primary and Secondary Activities

The revised MSLB radiological consequences analysis incorporated steady state initial RCS iodine concentrations at 1 μCi/gm and 60 μCi/gm of I-131 DEC for the accident-initiated iodine spike case and the pre-existing iodine spike case respectively. The initial primary iodine concentrations were calculated assuming 1 percent fuel defect, a 1 gpm maximum allowable primary-to-secondary leak rate prior to the MSLB accident, a TS 3.4.13.b maximum allowable unidentified RCS leakage of 1 gpm, and a TS 3.4.13.c maximum allowable identified RCS leakage of 10 gpm, a two year fuel cycle at 3580 megawatts thermal (MWt), a 143 gpm letdown flow rate, and a 100% demineralizer iodine removal efficiency. The initial noble gas nuclides were also calculated based on a 1 percent fuel defect. The initial primary and secondary coolant activities are contained in section 4.3.2.

Revised Accident-Induced Primary-to-Secondary Leakage

To compensate for the increases in the accident-initiated iodine release rates due to correction of the nonconservative equilibrium iodine appearance rates, a reduced accident-induced primary-to-secondary leakage rate of 10.5 gpm was used. The current licensing basis MSLB dose analysis assumed a 12.8 gpm primary-to-secondary leak in the ruptured SG. The primary-to-secondary leak is postulated to be induced by the MSLB accident. The 12.8 gpm accident-induced primary-to-secondary leakage rate provides the basis for the maximum allowable primary-to-secondary leakage limit for the TS 5.5.9 voltage-based SG tube ARC for ODSCC at tube support intersections and the TS 5.5.9 SG ARC for indications in the Westinghouse Explosive Tube Expansion (WEXTEX) Region. The TS 5.5.9 SG ARC for indications in the WEXTEX Region was requested in letter DCL-97-038 dated March 10, 1997 and was approved by the NRC in license amendments 129 for DCPP Unit 1 and 127 for DCPP Unit 2 in a letter dated June 4, 1999.

4.3.2 <u>MSLB Radiological Consequences Analysis Assumptions</u>

The radiological consequences analysis was performed to establish the limiting maximum primary-to-secondary post-MSLB leak rates in the DCPP SG in the faulted loop and the intact loops. These leak rates were calculated to be 10.5 gpm for the ruptured loop and 0.3124 gpm (150 gallons per day) for each intact loop.

Consistent with the current licensing basis analysis described in FSAR Update section 15.5.18.1 and SRP 15.1.5, two cases were analyzed:

- An accident-initiated iodine spike of 500 times the release rate corresponding to the Technical Specification 3.4.16 limit of 1 μCi/g I-131 DEC in the RCS.
- A pre-existing iodine spike of 60 μCi/g I-131 DEC in the reactor coolant system and 0.1 μCi/g I-131 DEC in the secondary system.

  The FAR Update Section 15.5.18.1 only contains a brief description of the MSLB radiological analysis assumptions. Therefore the MSLB radiological analysis assumptions are summarized below:

- The pre-MSLB primary-to-secondary leak rate was assumed to be 1 gpm, which is greater than the TS 3.4.13.d leak rate limit of 150 gallons per day per SG, to yield a conservatively high isotopic concentration in the secondary system.
- During the MSLB accident, the primary-to-secondary leak rate in each intact SG was assumed to be at the TS 3.4.13.d limit of 150 gpd. Therefore, the total leakage to the three intact SGs during the MSLB accident was 450 gpd, or 0.3125 gpm. The primary-to-secondary leak rate in the faulted SG was assumed at the maximum rate of 10.5 gpm.
- The MSLB occurred in the section of piping between the containment building and the main steam line isolation valves (MSIVs). Prior to control room isolation and pressurization, the control room HVAC intake χ/Q is the unfiltered χ/Q taken from the loss-of-coolant accident condition outside containment.
- Loss of offsite power was assumed to occur coincident with MSLB accident.
- The control room was conservatively assumed to be isolated in 2 minutes when the control room isolation dampers are fully closed.
- For the first 8 hours, the condenser is not available and steam release occurs. All steam releases were assumed to end after 8 hours, when the plant is placed on the RHR system.
- For a pre-existing iodine spike, the activity in the reactor coolant was based upon an iodine spike which has raised the reactor coolant

concentration to 60 μCi/g of I-131 DEC, based on DCPP TS Figure 3.4-1. The secondary coolant activity was 0.1 μCi/g of I-131 DEC, based on DCPP TS 3.7.18. Noble gas activity was based on 1 percent failed fuel.

- For an accident-initiated (concurrent) iodine spike, the accident initiates an iodine spike in the RCS which increases the iodine release rate from the fuel to a value 500 times greater than the release rate corresponding to an RCS concentration of 1 μCi/g I-131 DEC. 1 μCi/g I-131 DEC was based on DCPP TS 3.4.16. The iodine activity released to the RCS for the duration of the accident was conservatively assumed to mix instantaneously and uniformly in the RCS. Noble gas activity was based on 1 percent failed fuel.

- Following the pipe rupture, auxiliary feedwater to the faulted loop was isolated and the SG is allowed to steam dry. The iodine partition factor for the faulted SG was assumed to be 1.0. Also, the partition factor for the intact SGs was conservatively assumed to be 1.0, i.e., no credit was taken for iodine partition.

- All activity in the SGs was released to the atmosphere in accordance with the release rates in FSAR Update Table 15.5-34, with added releases from primary-to-secondary leaks in the faulted SG and intact SGs.

- Atmospheric steam releases (not included primary-to-secondary leaks):

Ruptured loop	162,784 pounds (lb) at 45.0 pounds/cubic feet (lb/ft^3) (0-2 hr)
	0 lb (2-8 hr)
Intact loops	393,464 lb at 45.0 lb/ft^3 (0-2 hr)
	860,461 lb at 50.0 lb/ft^3 (2-8 hr)

- The source term was based on a composite source term of 3.5 percent and 4.5 percent fuel enrichment. An evaluation has been performed and concluded that the current source term bounds the 5 percent enrichment fuel up to 50,000 megawatt days per metric ton of uranium for a 21 month operating cycle.

- To maximize the accident-initiated iodine release rates, an RCS letdown rate of 143 gpm with 100% iodine removal through the filters in the demineralizers was assumed.

- The thyroid dose conversion factors based on ICRP Publication 30 and contained in Table 4 were used.

- Atmospheric Dispersion Factors (sec/m^3) per DCPP FSAR Update Tables 15.5-3 and 15.5-6

Time	EAB	LPZ	Control Room	
			Pressurized	Infiltration
0 -2 hr	5.29E-4	2.20E-5	7.05E-5	1.96E-4
2 - 8 hr		2.20E-5	7.05E-5	1.96E-4
8 - 24 hr		4.75E-6	5.38E-5	1.49E-4
24-96 hr		1.54E-6	3.91E-5	1.08E-4
96-720 hr		3.40E-7	2.27E-5	6.29E-5

- Reactor coolant iodine activity based on 1 percent failed fuel

Isotope	Gap Activity (μCi/gm)	500 Iodine Spike Activity Release Rate (Ci/hr)
I-131	1.351	1.289E+4
I-132	0.60	1.856E+4
I-133	2.25	2.629E+4
I-134	0.45	2.968E+4
I-135	1.45	2.449E+4
Kr-83m	0.38	3.701E+1
Kr-85m	2.14	8.585E+1
Kr-85	6.21	2.439E+0
Kr-87	1.23	1.729E+2
Kr-88	3.91	2.446E+2
Kr-89	0.09	3.096E+2
Xe-131m	2.52	2.557E+0
Xe-133m	3.87	1.439E+1
Xe-133	255.0	4.513E+2
Xe-135m	0.36	9.127E+1
Xe-135	8.09	1.303E+2
Xe-137	0.15	4.128E+2
Xe-138	0.57	4.280E+2

- Secondary coolant activity based on 1 percent failed fuel

Isotope	Secondary Activity (μCi/gm)
I-131	1.742E-4
I-132	7.412E-5
I-133	2.893E-4
I-134	5.289E-5
I-135	1.851E-4

Kr-83m	4.691E-5
Kr-85m	2.712E-4
Kr-85	8.001E-4
Kr-87	1.499E-4
Kr-88	4.905E-4
Kr-89	4.923E-6
Xe-131m	3.238E-4
Xe-133m	4.982E-4
Xe-133	3.285E-2
Xe-135m	4.355E-5
Xe-135	1.043E-3
Xe-137	8.743E-6
Xe-138	5.553E-5

- Reactor coolant iodine activity based on 60 μCi/gm I-131 DEC

Isotope	Gap Activity (μCi/g)
I-131	45.74
I-132	20.11
I-133	76.22
I-134	15.07
I-135	49.14
Kr-83m	0.38
Kr-85m	2.14
Kr-85	6.21
Kr-87	1.23
Kr-88	3.91
Kr-89	0.09
Xe-131m	2.52
Xe-133m	3.91
Xe-133	256.3
Xe-135m	0.45
Xe-135	8.66
Xe-137	0.15
Xe-138	0.57

- Secondary coolant activity based on 0.1 μCi/gm I-131 DEC

Isotope	Gap Activity (μCi/g)
I-131	7.623E-2
I-132	3.351E-2
I-133	1.270E-1
I-134	2.512E-2
I-135	8.190E-2
Kr-83m	3.791E-2

Kr-85m	2.141E-1
Kr-85	6.209E-1
Kr-87	1.232E-1
Kr-88	3.907E-1
Kr-89	9.223E-3
Xe-131m	2.523E-1
Xe-133m	3.911E-1
Xe-133	2.563E+1
Xe-135m	4.491E-2
Xe-135	8.663E-1
Xe-137	1.477E-2
Xe-138	5.679E-2

- Control Room HVAC Flow Rates and Filtration Efficiencies:

Filtered Intake Flow	2100 cfm
Unfiltered Intake Flow	10 cfm
Exhaust Flow	2110 cfm
Filtered Recirculation Flow	2100 cfm

Charcoal Filter Iodine Removal Efficiency

Elemental	95%
Organic	95%
Particulate	95%

- RCS and Secondary Water Volume and Water Mass

RCS water volume	94,000 gallons
RCS water mass	566,000 pounds
Water in SGs	6735.54 ft^3 at 45.0 lb/ft^3 (0-2 hr) and 50.0 lb/ft^3 (2-8 hr)
	Loop 1 1683.88 ft^3
	Loops 2,3,4 5051.65 ft^3
Water in Condensers	27243.59 ft^3 at 62.4 lb/ft^3
Water in SGs and Condensers	33979.13 ft^3

4.3.3 MSLB Radiological Consequences Results

The accident-initiated iodine spike and pre-accident iodine spike thyroid, beta skin, and whole body doses were calculated at the EAB for 2 hours and at the LPZ for 30 days and are summarized in Table 14. The table includes applicable SRP section 15.1.5 Appendix A offsite dose guideline values.

The accident-initiated iodine spike and pre-accident iodine spike control room thyroid, beta skin, and whole body doses for 30 days are also summarized in Table 14. The table contains the applicable SRP section 6.4 guideline values which are based on GDC 19.

The results in Table 14 demonstrate that the applicable SRP 15.1.5 and 6.4 guideline values are met. The limiting case was the thyroid dose for the accident-initiated spike at the EAB which was equal to the guideline value of 30 Rem.

The MSLB offsite and control room radiological consequences analysis results are discussed in the marked-up FSAR Update section 15.5.18.1 contained in Enclosure 5.

Table 14
MSLB Radiological Consequences Results

Location	Dose (Rem)		
	Thyroid	Beta Skin	Whole Body
Accident-Initiated Spike			
EAB (0-2 hr)	30.0	1.50E-1	9.40E-2
LPZ (30 days)	6.48	1.91E-2	1.18E-2
SRP 15.1.5 Guideline Value	30.0	2.5	2.5
Control Room (30 days)	6.66E-1	7.09E-3	1.49E-4
SRP 6.4 Guideline Value	30.0	5	5
Pre-Existing Spike			
EAB (0-2 hr)	53.05	1.25E-1	7.26E-2
LPZ (30 days)	4.58	9.80E-3	5.56E-3
SRP 15.1.5 Guideline Value	300	25	25
Control Room (30 days)	5.53E-1	6.70E-3	1.27E-4
SRP 6.4 Guideline Value	30	5	5

4.4 Summary of Technical Analyses

SGTR Thermal and Hydraulic Analysis

A revised SGTR thermal and hydraulic analysis for radiological consequences was performed using the NRC approved WCAP-10698, Supplement 1, methodology. The limiting single failure was assumed to be that the PORV on the ruptured SG fails open at the time the ruptured SG was isolated as previously determined to be limiting in the current licensing basis analysis in WCAP-11723 and WCAP-10698, Supplement 1. The revised analysis was performed with input changes for reduced maximum ECCS injection flow, increased operator action time for SI initiation, and conservative modeling to bound a 5 percent RCS flow asymmetry. The mass release results for this analysis were used to determine the SGTR offsite and control room radiological consequences.

SGTR Offsite and Control Room Radiological Consequences

A revised SGTR offsite and control room radiological consequences analysis was performed based on the NRC approved WCAP-10698, Supplement 1, methodology. The analysis was based on an iodine spiking factor of 335 for the accident-initiated iodine spike case EAB dose. A conservative iodine spiking factor of 500 was used for the remaining offsite dose and control room calculations. The revised analyses were performed with input changes for revised initial reactor coolant fission product specific activity, revised thyroid dose conversion factors based on ICRP Publication 30, a revised accident-initiated iodine spiking factor, revised accident-initiated iodine release rates, revised disintegration energies, a revised iodine transport model, and revised whole body and skin offsite dose calculations, and revised SG mass releases. The use of the ICRP Publication 30 thyroid dose conversion factors requires a change to TS 1.1, "Definitions," term "Dose Equivalent I-131." Use of ICRP Publication 30 thyroid dose conversion factors is endorsed by RG 1.183.

The results of the analysis showed the pre-accident iodine spike thyroid doses at the EAB and LPZ are less than the SRP section 15.6.3 guideline values and that the whole body doses, including the contribution from iodine, at the EAB and LPZ are less than the SRP section 15.6.3 guideline values.

The analysis showed that the accident-initiated iodine spike thyroid doses at the EAB and LPZ are less than the SRP section 15.6.3 guideline values with the exception of the 2 hour EAB thyroid dose. The calculated 2 hour EAB thyroid dose of 30.5 Rem is 1.5% above the SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB thyroid dose has been compared against the conservative thyroid dose SRP 15.6.3 guideline value of 30 Rem. The

2 hour EAB thyroid dose would be equivalent to a RG 1.183 methodology TEDE of 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem is considered to be acceptable. The 2 hour EAB thyroid dose of 30.5 Rem meets the 10 CFR 100 dose limit of 300 Rem for the first 2 hours at the EAB.

The analysis for the control room showed that the pre-accident iodine spike thyroid and the accident-initiated iodine spike thyroid, whole body gamma, and beta skin doses for a SGTR are less than the control room dose limits specified in SRP section 6.4.

MSLB Offsite and Control Room Radiological Consequences

A MSLB radiological consequences analysis was performed using the methodology in SRP 15.1.5, Appendix A, and in accordance with the primary coolant specific activity limits in Generic Letter 95-05. The revised MSLB radiological consequences analysis was performed for input changes for revised thyroid dose conversion factors based on ICRP Publication 30, revised equilibrium iodine appearance rates, and a reduction in the steam line break accident-induced SG tube leakage rate from 12.8 gpm to 10.5 gpm. The use of the ICRP Publication 30 dose conversion factors requires a change to TS 1.1, "Definitions," term "Dose Equivalent I-131," and is endorsed by RG 1.183. The use of the reduced accident-induced primary-to-secondary leakage rate of 10.5 gpm will become the new basis for the maximum allowable primary-to-secondary leakage limit for the TS 5.5.9 SG tube ARC.

The results of the analysis showed the applicable SRP section 6.4 and SRP section 15.1.5 guideline values were met. The limiting case is the thyroid dose for the accident-initiated spike at the EAB which is equal to the SRP section 15.1.5 guideline value of 30 Rem.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The revision of Technical Specification (TS) 1.1, Definitions, "Dose Equivalent I-131," to allow use of the iodine thyroid dose conversion factors from the International Commission on Radiological Protection (ICRP) Publication 30, 1979, and the revised steam generator tube rupture (SGTR) and main steam line break (MSLB) radiological consequences analyses are used to determine post-accident dose. They are not related to any accident initiator. Therefore, this change cannot increase the probability of an accident.

The revised SGTR thermal and hydraulic analysis input assumptions are consistent with actual plant limits and parameters.

The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR Part 100 limits and the NUREG-0800 Standard Review Plan (SRP) section 15.1.5 and section 6.4 guideline values.

The revised SGTR control room radiological consequences analysis dose results are within the SRP section 6.4 guideline values.

The revised SGTR offsite radiological consequences analysis dose results are within the 10 CFR Part 100 dose limits. The SGTR offsite dose results also meet the SRP section 15.6.3 and section 6.4 guideline values, with the exception of the 2 hour Exclusion Area Boundary (EAB) thyroid dose. The calculated 2 hour EAB thyroid dose of 30.5 Rem is 1.5 percent above the SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB thyroid dose has been compared against the conservative thyroid dose SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB dose thyroid dose would be equivalent to a Regulatory Guide (RG) 1.183 methodology Total Effective Dose Equivalent (TEDE) of approximately 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated

iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem is not considered to be a significant increase in dose.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: No.

The use of the iodine thyroid dose conversion factors from ICRP Publication 30 and the revised SGTR and main steam line break MSLB radiological consequences analyses do not involve any physical plant changes. The change does not involve changes in operation of the plant that could introduce a new failure mode for creating an accident or affect the mitigation of an accident.

Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The use of ICRP Publication 30 thyroid dose conversion factors to calculate the radiological consequences for a SGTR and MSLB accident is endorsed by RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," US Nuclear Regulatory Commission," July 2000. Therefore, the revision of TS 1.1, Definitions, "Dose Equivalent I-131," to allow use of the iodine thyroid dose conversion factors from ICRP Publication 30 does not result in a significant reduction in the margin provided by TS 1.1. The revised SGTR thermal and hydraulic analysis input assumptions are consistent with actual plant limits and parameters.

The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR Part 100 limits and the NUREG-0800 SRP section 15.1.5 and section 6.4 guideline values.

The revised SGTR control room radiological consequences analysis dose results are within the SRP section 6.4 guideline values.

The revised SGTR radiological consequences analysis dose results are within the 10 CFR Part 100 dose limits. The SGTR dose results also meet the SRP section 15.6.3 and section 6.4 guideline values, with the exception of the 2 hour EAB thyroid dose. The calculated 2 hour EAB thyroid dose of 30.5 Rem is 1.5 percent above the SRP 15.6.3 guideline value of 30 Rem. The 2 hour EAB dose thyroid dose would be equivalent to a RG 1.183 methodology TEDE of approximately 1.25 Rem, which is well below the RG 1.183 TEDE limit of 2.5 Rem for the accident-initiated iodine spike case. Therefore, the 2 hour EAB thyroid dose of 30.5 Rem is not a significant reduction in the margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, PG&E concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendments and determined the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

7.0 REFERENCES

1. Westinghouse Nuclear Safety Advisory Letter (NSAL), NSAL-99-006, "GENF Code Error," dated June 1, 1999.
2. Westinghouse Nuclear Safety Advisory Letter (NSAL), NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations," dated March 7, 2000.
3. WCAP-11723, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for the Diablo Canyon Power Plant Units 1 and 2," dated February 1988. [Proprietary]
4. Letter from PG&E to NRC, DCL-88-114, "Steam Generator Tube Rupture (SGTR) Analysis," dated April 29, 1988.
5. Letter from PG&E to NRC, DCL-90-052, "Supplemental Information for Steam Generator Tube Rupture (SGTR) Analysis," dated February 22, 1990.

6. Letter from PG&E to NRC, DCL-91-009, "Steam Generator Tube Rupture (SGTR) Analysis Operator Action Times," dated January 17, 1991.
7. Letter from NRC to PG&E, "Closeout of Steam Generator Tube Rupture Analysis Issue for Diablo Canyon Power Plant, and Finding of Compliance with Condition 2.C.(9) of Unit 2 Operating License DPR-82 (TAC Nos. 68346 and 68347).," dated April 3, 1991.
8. WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," dated December 1984. [Proprietary]
9. Letter from NRC to Westinghouse Owners Group SGTR Subgroup, "Acceptance for Referencing of Licensing Topical Report WCAP-10698," dated March 30, 1987.
10. Letter from NRC to Westinghouse Owners Group SGTR Subgroup, "Acceptance of Referencing of Licensing Topical Report WCAP-10698 Supplement 1," dated December 17, 1985.
11. WCAP-10698, Supplement 1, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," dated May 1985. [Proprietary]
12. International Commission on Radiological Protection (ICRP), "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979.
13. ENDF-223, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," T. R. England and R. E. Schenter, October 1975
14. ORNL-4628, "ORIGEN - The ORNL Isotope Generation and Depletion Code," Oak Ridge National Laboratory, dated May 1973
15. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," dated March 1992.
16. Letter from NRC to Lawrence A. Walsh, "Westinghouse Owners Group-Steam Generator Tube Uncovery Issue," dated March 10, 1993.
17. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," US Nuclear Regulatory Commission, July 2000.
18. NUREG-0800, Standard Review Plan Section 15.6.3, "Radiological Consequences of a Steam Generator Tube Failure (PWR)," Rev. 2, July 1981.
19. NUREG-0800, Standard Review Plan Section 6.4, "Control Room Habitability System," Revision 2, July 1981.
20. Letter from PG&E to NRC, DCL-97-034, "License Amendment Request 97-03, Voltage-Based Alternate Steam Generator Tube Repair Limit for Outside Diameter Stress Corrosion Cracking at Tube Support Plate Intersections," dated February 26, 1997.
21. Letter from NRC to PG&E, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit 1 (TAC No. M97254) and Unit No. 2 (TAC No. M97255)," dated March 12, 1998.
22. PG&E Letter DCL-97-038, "License Amendment Request 97-04, Steam Generator Tube Alternate Repair Criteria for Indications in the

Westinghouse Explosive Tube Expansion (WEXTEX) Region," dated March 10, 1997.

23. Letter from NRC to PG&E, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit No. 1 (TAC No. M98283) and Unit No. 2 (TAC No. M98284)," dated February 19, 1999.
24. Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995.

MARKED-UP TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
1.1-3	1.1-3

or those listed in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.

1.1 Definitions (continued)

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977~~.~~,

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

   1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
   2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

PROPOSED TECHNICAL SPECIFICATIONS PAGE

1.1 Definitions (continued)

DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977, or those listed in International Commission on Radiological Protection Publication 30, "Limits for Intakes of Radionuclides by Workers," 1979.
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

(continued)

STEAM GENERATOR TUBE RUPTURE THERMAL AND HYDRAULIC ANALYSIS FIGURES

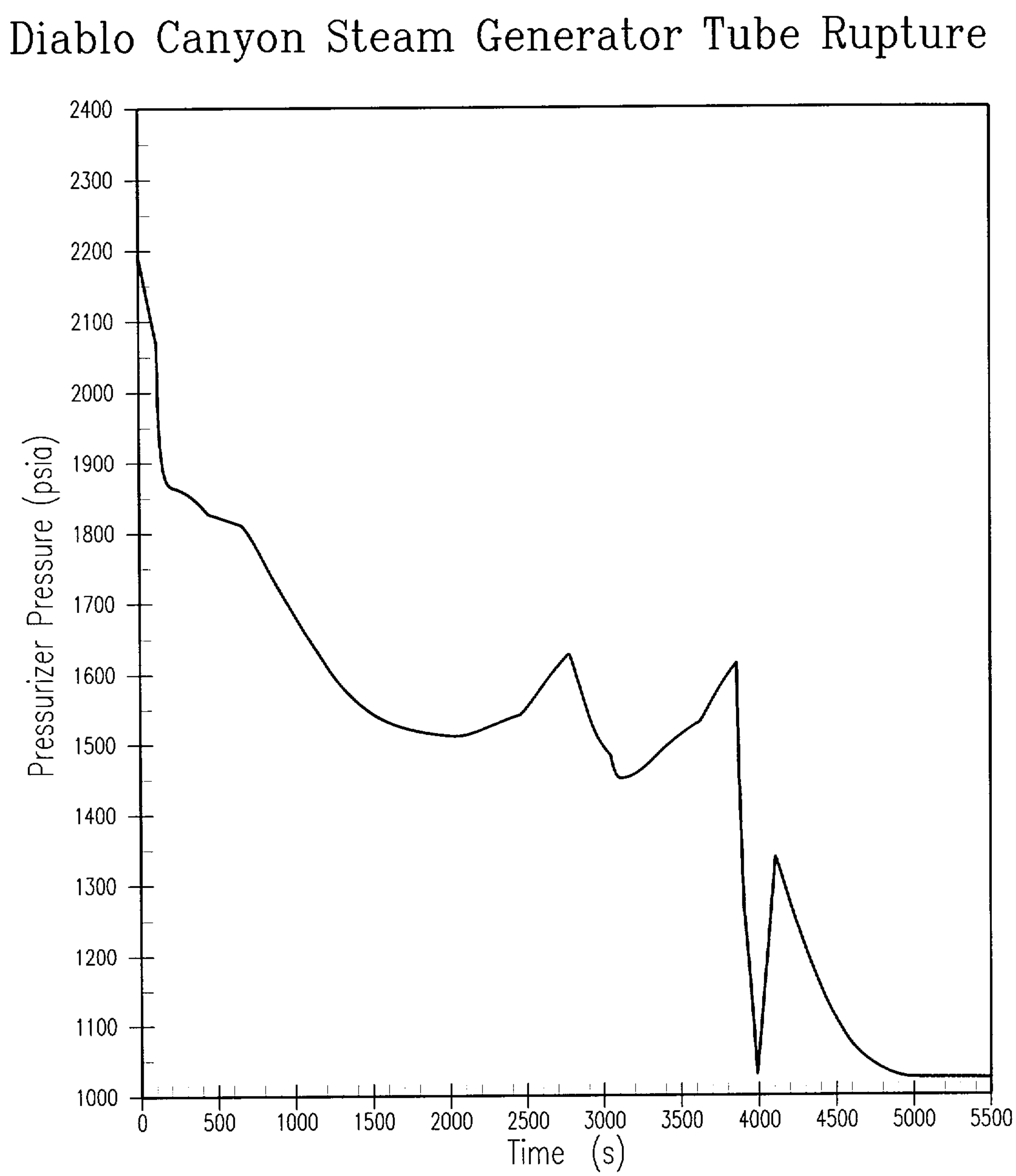

Figure 1, Pressurizer Pressure
Thermal and Hydraulic Analysis

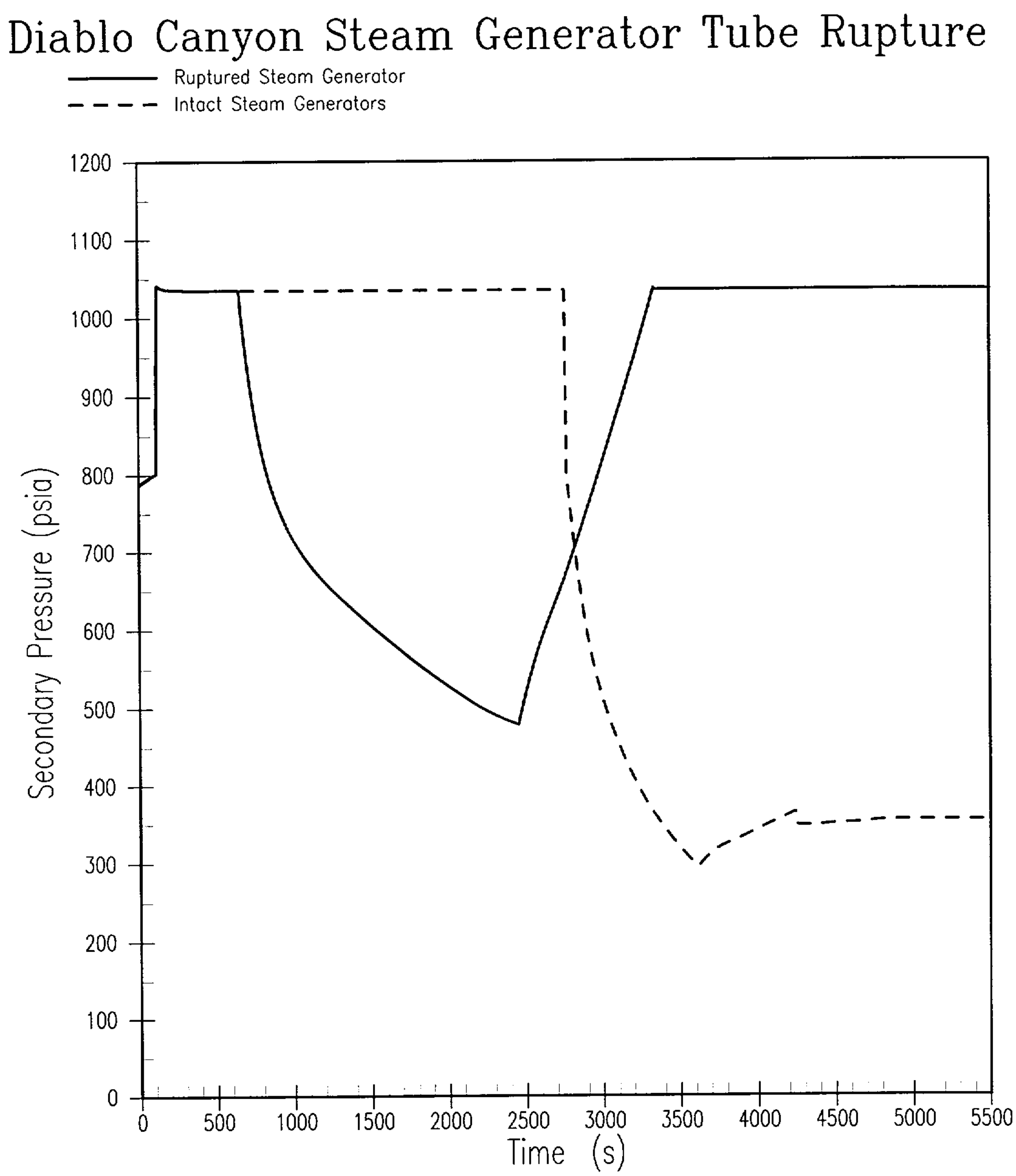

Figure 2, Secondary Pressure
Thermal and Hydraulic Analysis

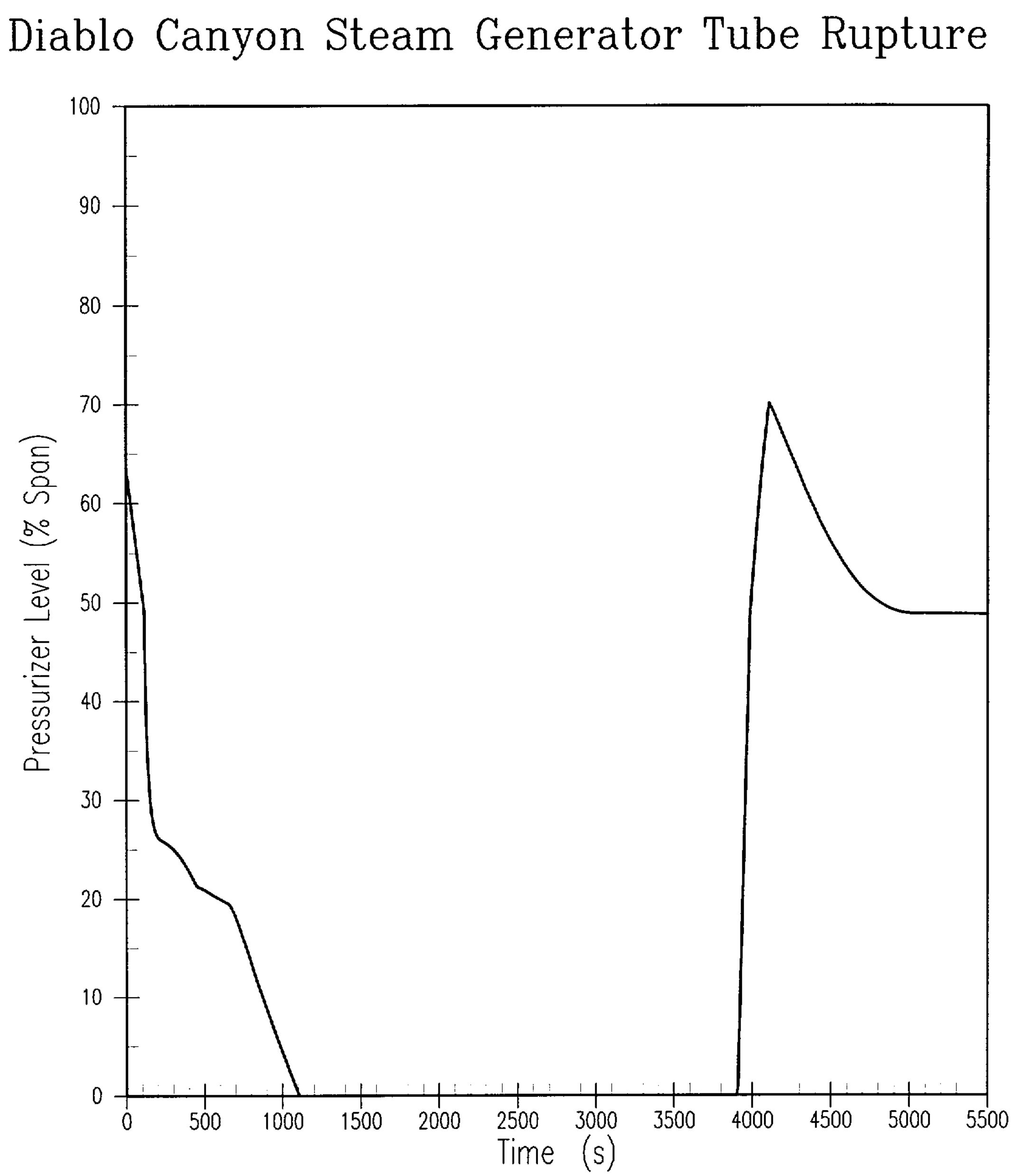

Figure 3, Pressurizer Level
Thermal and Hydraulic Analysis

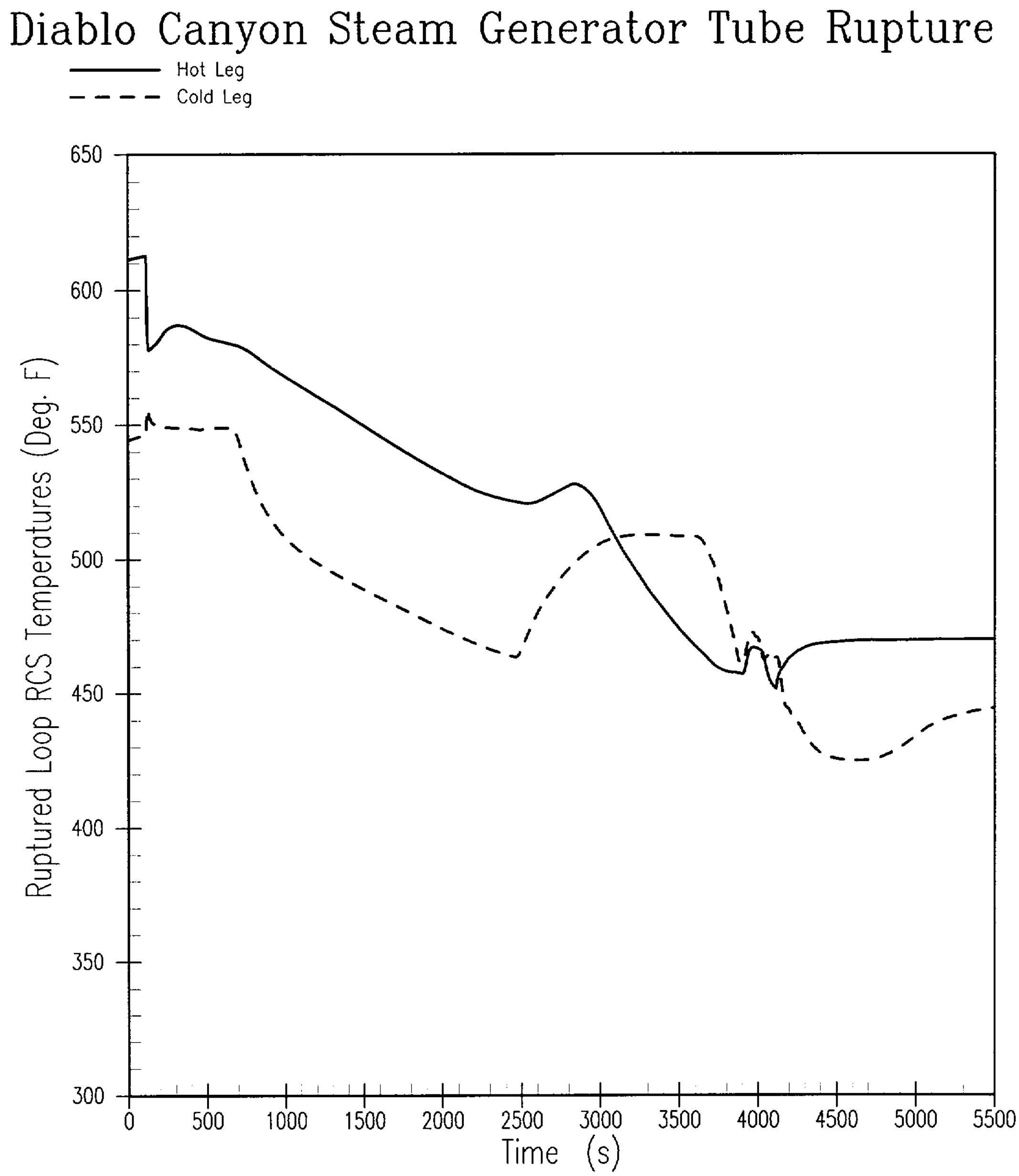

Figure 4, Ruptured Loop Hot and Cold Leg Temperatures
Thermal and Hydraulic Analysis

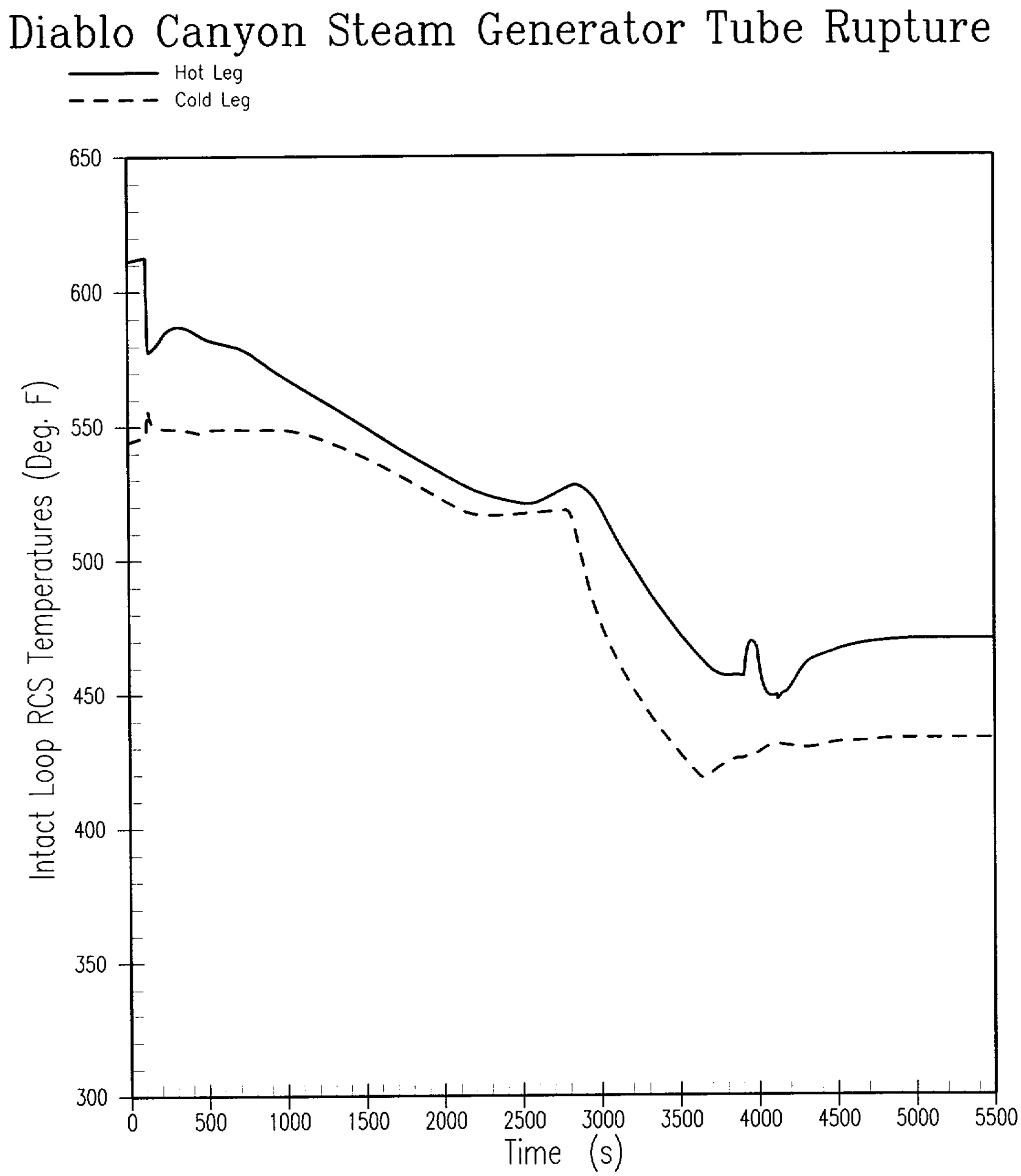

Figure 5, Intact Loop Hot and Cold Leg Temperatures
Thermal and Hydraulic Analysis

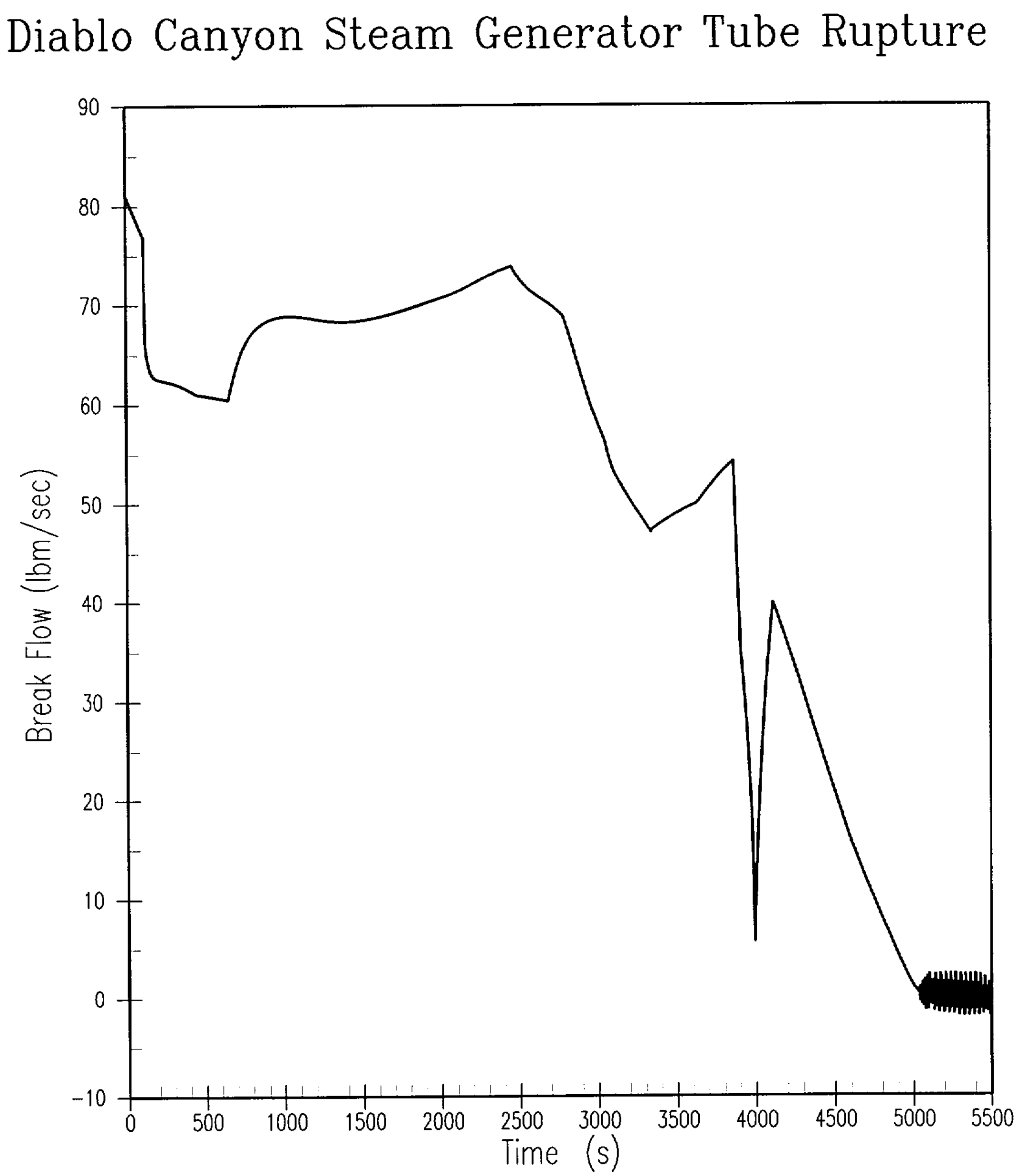

Figure 6, Primary-to-secondary Break Flow
Thermal and Hydraulic Analysis

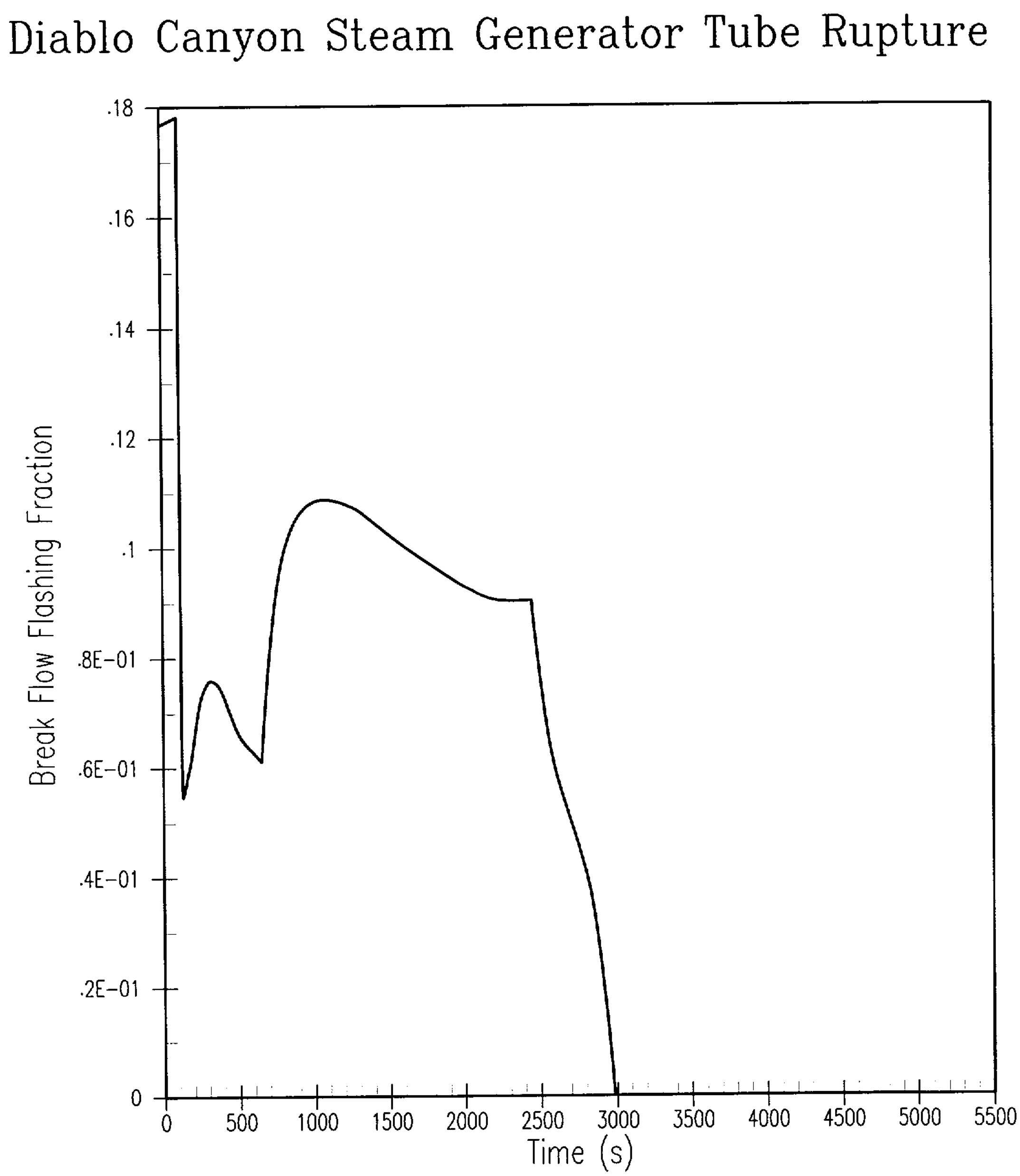

Figure 7, Break Flow Flashing Fraction
Thermal and Hydraulic Analysis

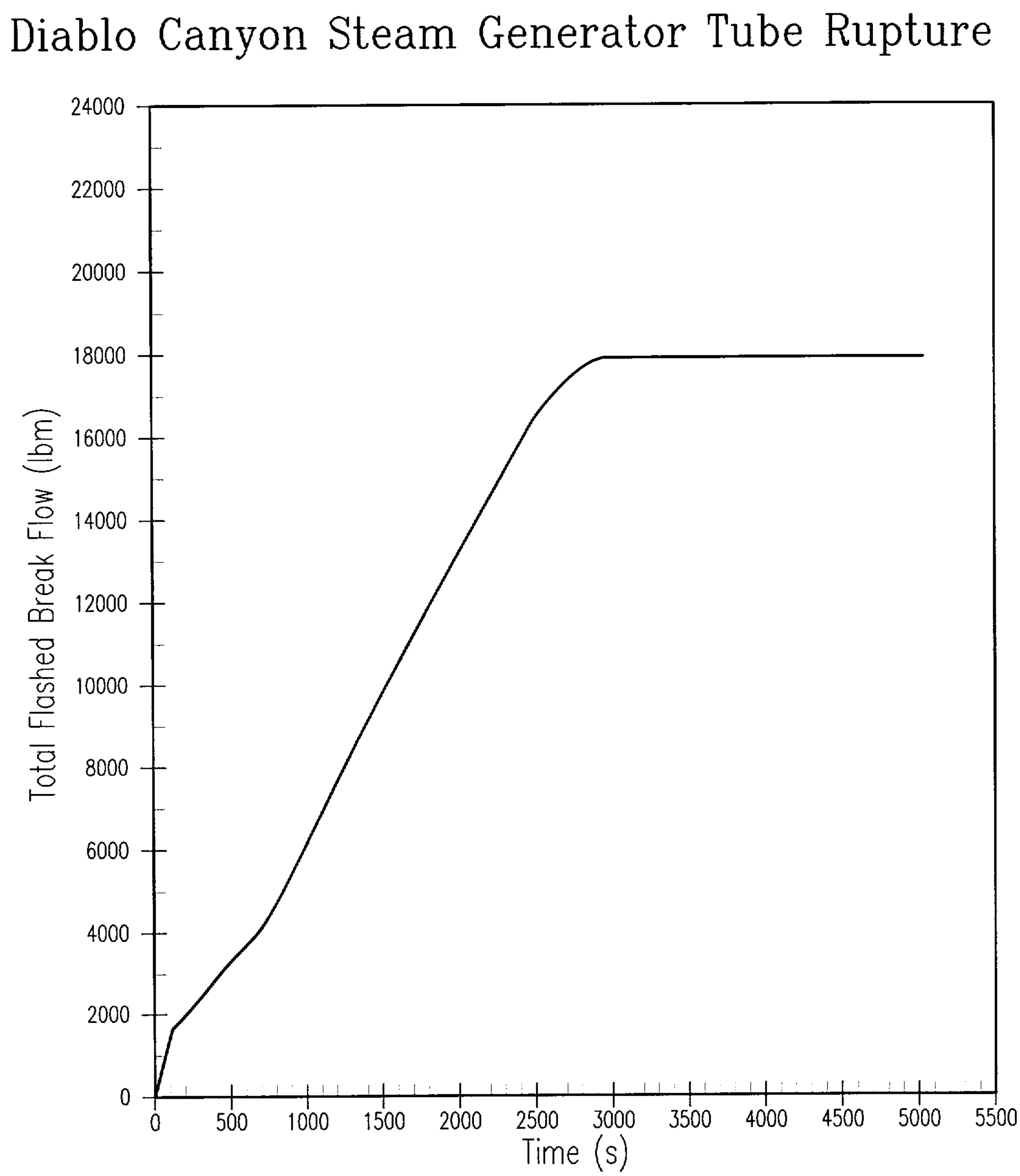

Figure 8, Total Flashed Break Flow
Thermal and Hydraulic Analysis

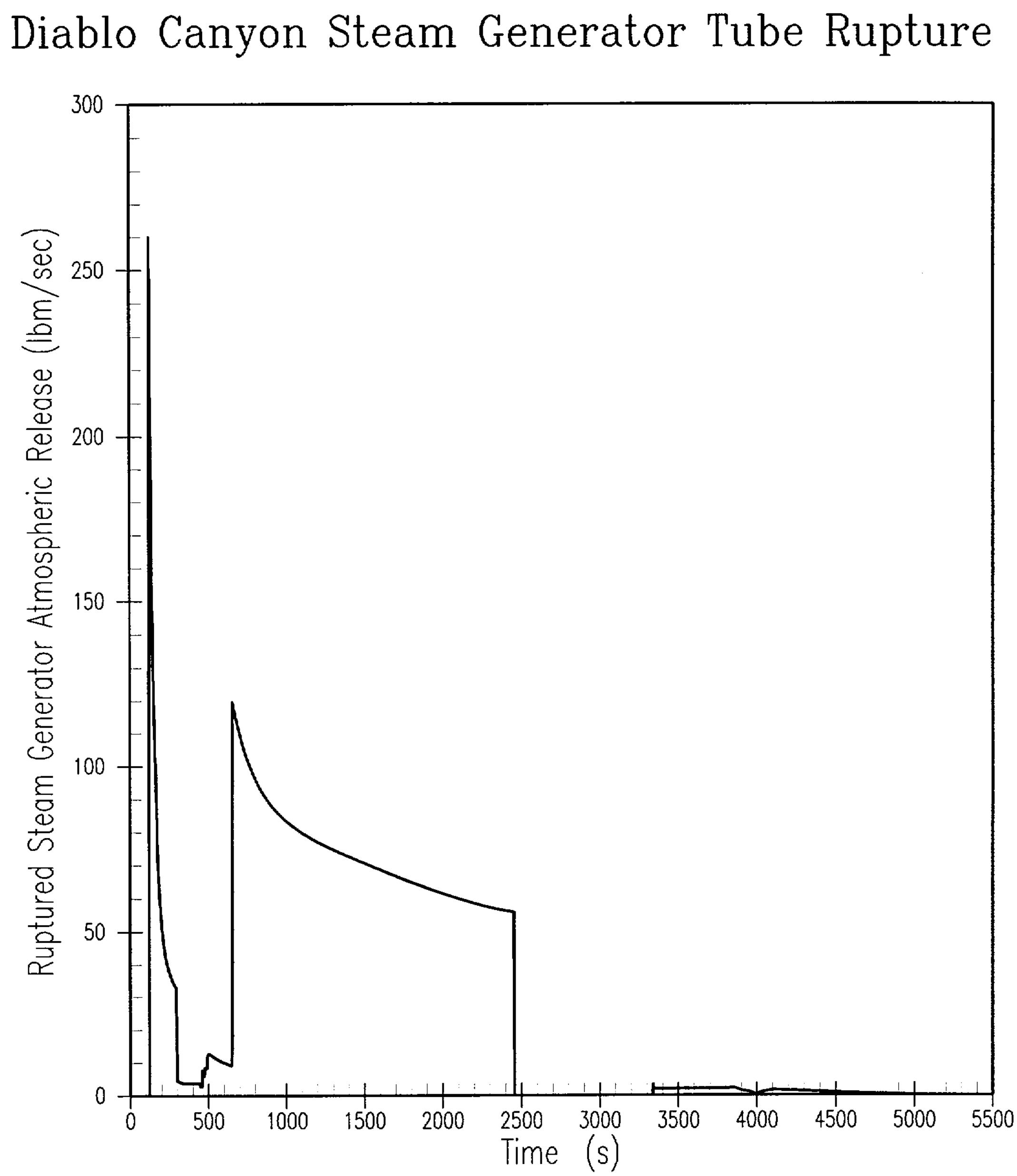

Figure 9, Ruptured SG Mass Release Rate to the Atmosphere
Thermal and Hydraulic Analysis

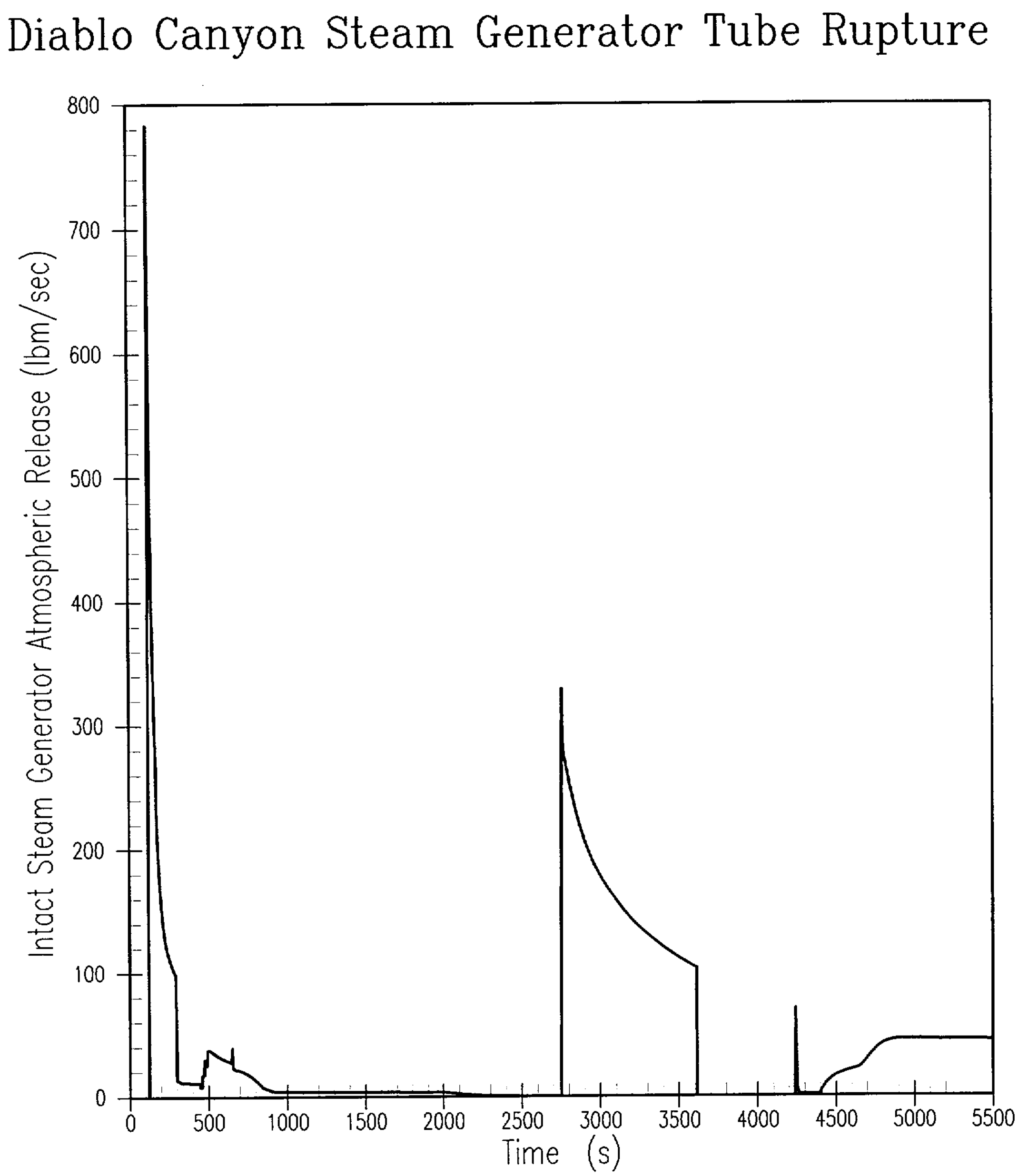

Figure 10, Intact SGs Mass Release Rate to the Atmosphere
Thermal and Hydraulic Analysis

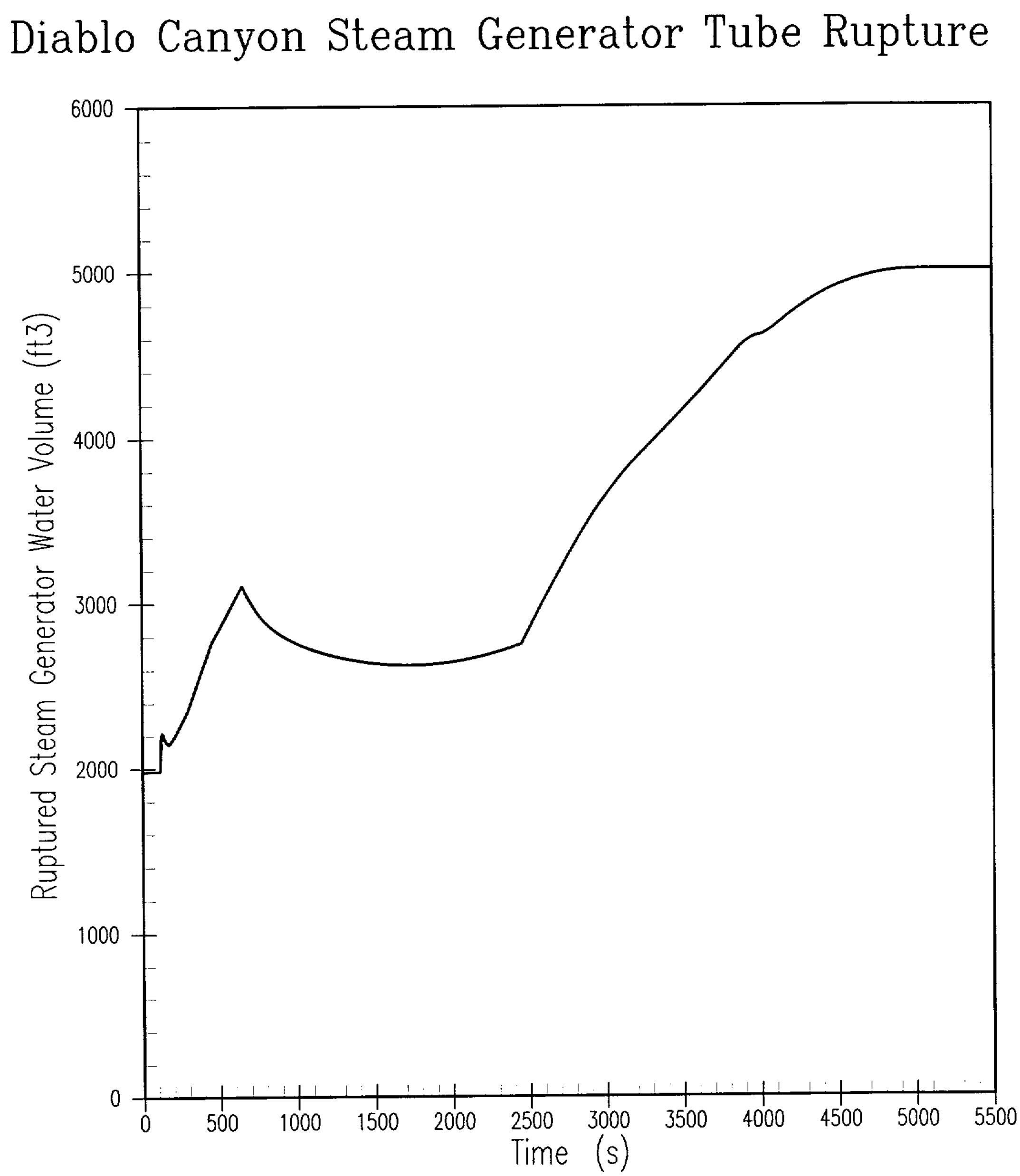

Figure 11, Ruptured SG Water Volume
Thermal and Hydraulic Analysis

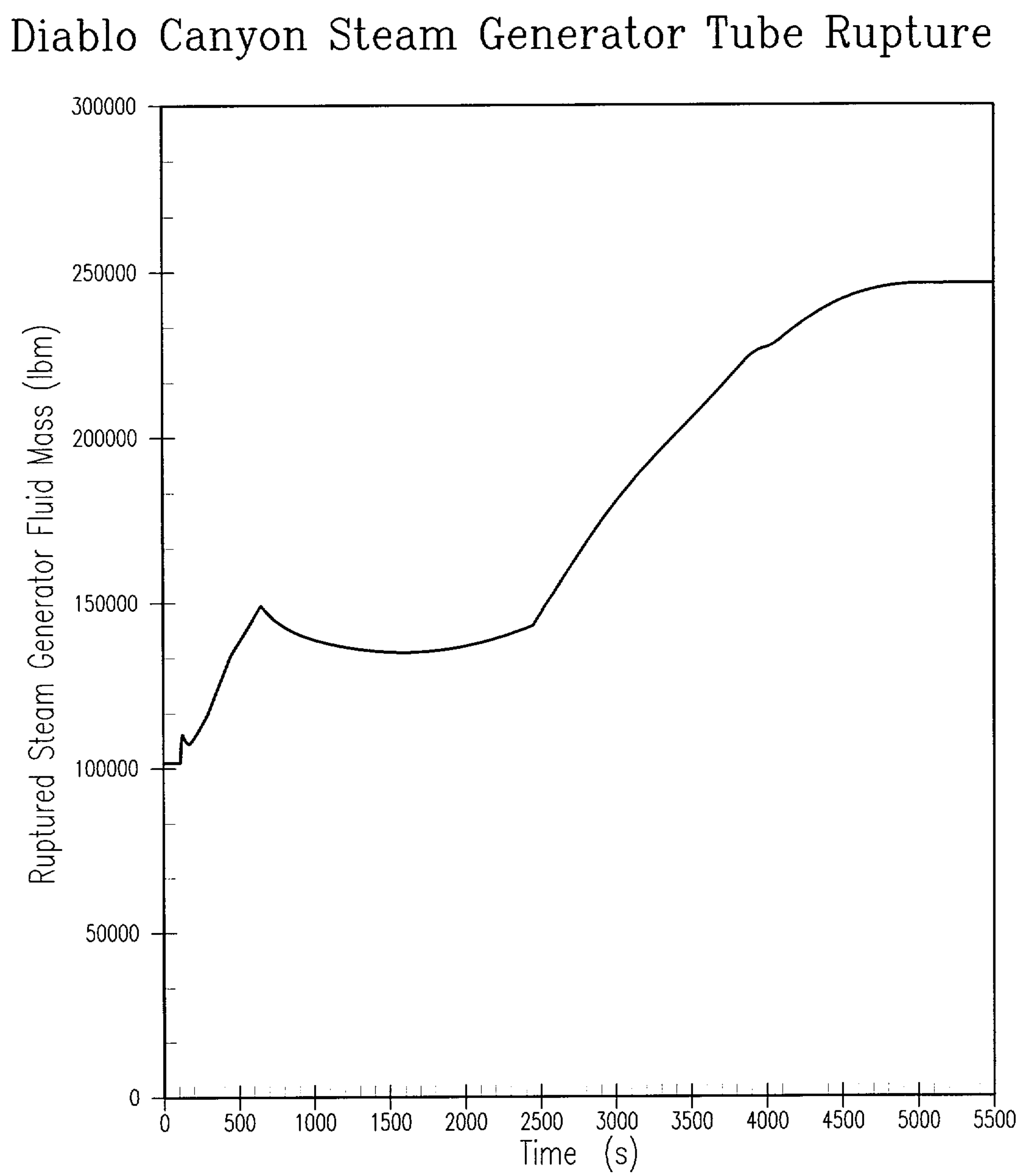

Figure 12, Ruptured SG Water Mass
Thermal and Hydraulic Analysis

MARKED-UP FINAL SAFETY ANALYSIS REPORT UPDATE PAGES
(FOR INFORMATION ONLY)

 No changes

(5) Protection System - The analysis only models those reactor protection system features, which would be credited for at power conditions and up to the time a reactor trip is initiated. The FSAR Update, Section 15.4.2.1, presents the analysis of the bounding transient following reactor trip, where engineered safety features are actuated to mitigate the effects of a steam line break.

(6) Control Systems - The results of a main steam pipe rupture at power would be made less severe as a result of control system actuation. Therefore, the mitigation effects of control systems have been ignored in the analysis.

15.4.2.3.3 Results

A spectrum of steam line break sizes was analyzed. The results show that for break sizes up to 0.53 ft^2, a reactor trip is not generated. In this case, the event is similar to an excessive load increase event as described in FSAR Update, Section 15.2.11. The core reaches a new equilibrium condition at a higher power equivalent to the increased steam flow. For break sizes larger than 0.53 ft^2, a reactor trip is generated within a few seconds of the break on the safety injection signal from low steam line pressure.

The limiting case for demonstrating DNB protection is the 0.53-ft^2 break, the largest break size that does not result in an early trip on low steam pressure SI actuation. The time sequence of events for this case is shown on Table 15.4-8. Figures 15.4.2-15 through 15.4.2-18 show the transient response.

15.4.2.3.4 Conclusions

The analysis concludes that the DNB design basis is met for the limiting case. Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the minimum DNBR remains above the DNBR limit value for the most limiting break occurring from an at-power condition.

15.4.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

15.4.3.1 Identification of Causes and Accident Description

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system (RCS). In the event of a coincident loss of offsite power, or failure of the condenser steam dump system, discharge of activity to the atmosphere takes place via the steam generator power-operated relief valves (and safety valves if their setpoint is reached).

Although the steam generator tube material is Inconel 600, a highly ductile material, it is assumed that complete severance could occur. The more probable mode of tube failure would

 No Changes

be one or more minor leaks of undetermined origin. Activity in the steam and power conversion system is subject to continual surveillance and an accumulation of minor leaks that exceeds the limits established in the Technical Specifications (Reference 30) is not permitted during the unit operation.

The operator is expected to determine that a steam generator tube rupture has occurred, to identify and isolate the faulty steam generator, and to complete the required recovery actions to stabilize the plant and terminate the primary to secondary break flow. These actions should be performed on a restricted time scale in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. Consideration of the indications provided at the control board, together with the magnitude of the break flow, leads to the conclusion that the recovery procedure can be carried out on a time scale that ensures that break flow to the secondary system is terminated before water level in the affected steam generator rises into the main steam pipe. Sufficient indications and controls are provided to enable the operator to carry out these functions satisfactorily.

Assuming normal operation of the various plant control systems, the following sequence of events is initiated by a tube rupture:

(1) Pressurizer low pressure and low-level alarms are actuated and charging pump flow increases in an attempt to maintain pressurizer level. On the secondary side there is a steam flow/feedwater flow mismatch before trip as feedwater flow to the affected steam generator is reduced due to the break flow that is now being supplied to that unit.

(2) The main steamline radiation monitors, the air ejector radiation monitor and/or the steam generator blowdown radiation monitor will alarm, indicating a sharp increase in radioactivity in the secondary system, and steam generator blowdown will be automatically terminated.

(3) Continued loss of reactor coolant inventory leads to a reactor trip signal generated by low pressurizer pressure or overtemperature ΔT. A safety injection (SI) signal, initiated by low pressurizer pressure, follows soon after the reactor trip. The SI signal automatically terminates normal feedwater supply and initiates AFW addition.

(4) The reactor trip automatically trips the turbine and, if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a coincident loss of offsite power, the steam dump valves would automatically close to protect the condenser. The steam generator pressure would rapidly increase resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves (PORVs) (and safety valves if their setpoint is reached).

 No Changes

(5) Following reactor trip and SI actuation, the continued action of AFW supply and borated SI flow (supplied from the refueling water storage tank) provides a heat sink that absorbs some of the decay heat. This reduces the amount of steam bypass to the condenser, or in the case of loss of offsite power, steam relief to the atmosphere.

(6) SI flow results in stabilization of the RCS pressure and pressurizer water level, and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate.

In the event of an SGTR, the plant operators must diagnose the SGTR and perform the required recovery actions to stabilize the plant and terminate the primary to secondary leakage. The operator actions for SGTR recovery are provided in the Emergency Operating Procedures (Reference 42). The major operator actions 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:

(a) Identify the ruptured steam generator.

High secondary side activity, as indicated by the main steamline radiation monitors, the air ejector radiation monitor, or steam generator blowdown radiation monitor typically will provide the first indication of an SGTR event. The ruptured steam generator can be identified by an unexpected increase in steam generator level, or a high radiation indication on the corresponding main steamline monitor, or from a radiation survey of the main steamlines. For an SGTR that results in a reactor trip at high power, the steam generator water level will decrease off-scale on the narrow range for all of the steam generators. The AFW flow will begin to refill the steam generators, distributing approximately equal flow to each of the steam generators. Since primary to secondary leakage adds additional liquid inventory to the ruptured steam generator, the water level will return to the narrow range earlier in that steam generator and will continue to increase more rapidly. This response, as indicated by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

(b) 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 (a) minimizing the accumulation of feedwater flow and (b)

 No Changes

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.

(c) *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 PORVs on the intact steam generators.

(d) *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, if offsite power is lost or the RCPs are not running for some other reason, normal pressurizer spray is not available. In this event, RCS depressurization can be performed using a pressurizer PORV or auxiliary pressurizer spray.

(e) *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 the 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.

Following SI termination, the plant conditions will be stabilized, the primary to secondary break flow will be terminated and all immediate safety concerns will have been addressed. At this time a series of operator actions are performed to prepare the plant for cooldown to cold shutdown conditions. Subsequently, actions are performed to cooldown and depressurize the RCS to cold shutdown conditions and to depressurize the ruptured steam generator.

15.4.3.2 Analysis of Effects and Consequences

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 radiological consequences resulting from an SGTR are within allowable guidelines. Another concern for SGTR consequences is the possibility of steam generator overfill since this could potentially result in a significant increase in the offsite radiological consequences. Overfill could result in water entering the main steamline. If water continues to leak into the main steamlines, the release of liquid through the steam generator safety valves could result in an increase in radiological doses. Therefore, an analysis was performed to demonstrate margin to steam generator overfill, assuming the limiting single failure relative to overfill. The results of this analysis demonstrate that there is margin to steam generator overfill for DCPP.

The overfill analysis is presented in Reference 41 and the major assumptions include:

(1) Complete severance of a single tube located at the top of the tube sheet on the outlet side of the steam generator, resulting in double ended flow

(2) Initiation of the event from 100 percent power

(3) A loss of offsite power coincident with reactor trip

(4) Failure of an AFW control valve to close (limiting single failure)

(5) The PORVs on all three intact steam generators are fully opened during the RCS cooldown

(6) Operator actions are consistent with the times shown in Table 15.4-12

An analysis was also performed to determine the offsite radiological consequences, assuming the limiting single failure relative to offsite doses without steam generator overfill. Since steam generator overfill does not occur, the results of this analysis represent the limiting consequences for an SGTR for DCPP. The analysis to demonstrate margin to overfill is
41 presented in Reference ~~46~~; issues affecting the overfill margin are summarized in Reference 53. The analysis to determine the offsite radiological consequences is presented in Reference 41. The results of the offsite radiological consequences analysis are discussed below.

A thermal and hydraulic analysis was performed to determine the plant response for a design basis SGTR, and to determine 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. This information was then used to calculate the quantity of radioactivity released to the environment and the resulting radiological consequences. The results of the thermal and hydraulic analysis are discussed in Section 15.4.3.3 and the results of the environmental consequences analysis are discussed in Section 15.5.20.

15.4.3.3 Thermal and Hydraulic Analysis

The plant response following an SGTR was analyzed with the LOFTTR2 program until the primary to secondary break flow is terminated. The reactor protection system and the automatic actuation of the engineered safeguards systems were modeled in the analysis. The major operator actions which are required to terminate the break flow for an SGTR were also simulated in the analysis.

Analysis Assumptions

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. However, as indicated subsequently, the break flow flashing fraction was conservatively calculated assuming that all of the break flow comes from the hot leg side of the steam generator. In addition, the iodine scrubbing effectiveness of the steam generator water was calculated based on the conservative assumption that the rupture is located near the top of the tube bundle at the intersection of the outer tube row and the upper anti-vibration bar. The combination of these conservative assumptions regarding the break flow location results in a very conservative calculation of the offsite radiation doses. It was assumed that the reactor is operating at full power at the time of the accident and the secondary mass was assumed to correspond to operation at the bottom of the steam generator level control band with an allowance for uncertainties. It was also assumed that a loss of offsite power occurs at the time of reactor trip and the highest worth control assembly was assumed to be stuck in its fully withdrawn position at reactor trip.

The limiting single failure was assumed to be the failure of the PORV on the ruptured steam generator. Failure of this PORV in the open position will cause an uncontrolled depressurization of the ruptured steam generator which will increase primary to secondary leakage and the mass release to the atmosphere. It was assumed that the ruptured steam generator PORV fails open when the ruptured steam generator is isolated, and that the PORV was isolated by locally closing the associated block valve.

The major operator actions required for the recovery from an SGTR are discussed in Section 15.4.3.1 and these operator actions were simulated in the analysis. The operator action times which were used for the analysis are presented in Table 15.4-12. It is noted that the PORV on the ruptured steam generator was assumed to fail open at the time the ruptured steam generator was isolated. It was assumed that the operators isolate the failed open PORV by locally closing the associated block valve to complete the isolation of the ruptured steam generator before proceeding with the subsequent recovery operations. It was assumed that the ruptured steam generator PORV was isolated at 30 minutes after the valve was assumed to fail

open. After the ruptured steam generator PORV was isolated, an additional delay time of 5 minutes (Table 15.4-12) was assumed for the operator action time to initiate the RCS cooldown.

Transient Description

The LOFTTR2 analysis results are described below. The sequence of events for this transient is presented in Table 15.4-13.

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 15.4.3-2
Figure ~~15.4-99~~. The ~~RCS~~ pressurizer pressure also decreases as shown in Figure ~~15.4-100~~ as the steam
15.4.3-1 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 ΔT trip signal.

After reactor trip, core power rapidly decreases to decay heat levels. The turbine stop valves close and steam flow to the turbine is terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remain closed due to the loss of condenser vacuum resulting from the assumed loss of offsite power at the time of reactor trip. Thus, the energy transfer from the primary system causes the secondary side pressure to increase rapidly after reactor trip until the steam generator PORVs (and safety valves if their setpoints are reached) lift to dissipate the energy,
15.4.3-3 as shown in Figure ~~15.4-101~~. The main feedwater flow will be terminated and AFW flow will be automatically initiated following reactor trip and the loss of offsite power.

The RCS pressure decreases more rapidly after reactor trip as energy transfer to the secondary shrinks the reactor coolant and the tube rupture break flow continues to deplete primary inventory. Pressurizer level also decreases more rapidly following reactor trip. The decrease in RCS inventory results in a low pressurizer pressure SI signal. After SI actuation, the SI flow rate maintains the reactor coolant inventory and the pressurizer level begins to stabilize. The RCS pressure also trends toward the equilibrium value where the SI flow rate equals the break flow rate.

Since offsite power was 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 Figures ~~15.4-102~~ and ~~15.4-103~~), however, 15.4.3-4
the temperature differential subsequently increases as natural circulation flow develops. The 15.4.3-5
cold leg temperatures trend toward the steam generator temperature as the fluid residence time in the tube region increases. The intact steam generator loop temperatures slowly decrease due to the continued AFW flow until operator actions are taken to control the AFW flow to maintain the specified level in the intact steam generators. The ruptured steam generator loop temperatures also continue to slowly decrease until the ruptured steam generator is isolated; the PORV is assumed to fail open.

Major Operator Actions

(a) *Identify and Isolate the Ruptured Steam Generator*

As indicated in Table 15.4-12, it was assumed that the ruptured steam generator is identified and isolated at 10 minutes after the initiation of the SGTR or when the narrow range level reaches ~~27~~ 28 percent, whichever time is longer. Since the time to reach ~~27~~ percent narrow range level was ~~636~~ 650 seconds, it was assumed that the actions to isolate the ruptured steam generator are performed at this time.

The ruptured steam generator PORV was also assumed to fail open at this time, and the failure was simulated at ~~638~~ 652 seconds. 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 ~~15.4-103~~ 15.4.3-5. As noted previously, the intact steam generator loop temperatures also decrease, as shown in Figure ~~15.4-102~~ 15.4.3-4, until the AFW flow to the intact steam generators is throttled. These effects result in a decrease in the RCS pressure and pressurizer level, and the pressurizer level goes offscale low. However, the increased SI flow subsequently causes the RCS pressure and pressurizer level to increase again.

It was 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 30 minutes. Thus, the isolation of the ruptured steam generator was completed at ~~2440~~ 2452 seconds and the depressurization of the ruptured steam generator was terminated. At this time, the ruptured steam generator pressure increases rapidly to the PORV setpoint and the primary to secondary break flow begins to decrease. Because the SI flow rate exceeds the break flow rate, the rate of RCS repressurization increases and the pressurizer level increases and returns onscale.

(b) *Cool Down the RCS to establish Subcooling Margin*

After the ruptured steam generator PORV block valve was closed, a 5 minute operator action time was imposed prior to initiation of cooldown. The depressurization of the ruptured steam generator affects the RCS cooldown target temperature since the temperature is dependent upon the pressure in the ruptured steam generator. Since offsite power was lost, the RCS was cooled by dumping steam to the atmosphere using the intact steam generator PORVs. The cooldown was continued until RCS subcooling at the ruptured steam generator pressure is ~~20~~ 36°F ~~plus~~ including an allowance ~~of 20°F~~ for instrument uncertainty. Because of the lower pressure in the ruptured steam generator, the associated temperature the RCS must be cooled to is also lower, which has the net effect of extending the time for cooldown. The cooldown was initiated at ~~2742~~ 2752 seconds and was completed at ~~3664~~ 3612 seconds.

pressurizer

15.4.3-3 The reduction in the intact steam generator pressures required to accomplish the cooldown is shown in Figure ~~15.4-101~~, and the effect of the cooldown on the
15.4.3-4 RCS temperature is shown in Figure ~~15.4-102~~. The pressurizer level and ~~RCS~~ pressure also decrease during this cooldown process due to shrinkage of the reactor coolant, as shown in Figures ~~15.4-99~~ and ~~15.4-100~~. 15.4.3-2 15.4.3-1

(c) *Depressurize to Restore Inventory*

After the RCS cooldown, a 4 minute operator action time was included prior to depressurization. The RCS was depressurized at ~~3904~~ 3860 seconds to assure adequate coolant inventory prior to terminating SI flow. With the RCPs stopped, normal pressurizer spray is not available and thus the RCS was depressurized by opening a pressurizer PORV. The depressurization was continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of ~~4~~ 12 percent for pressurizer level uncertainty, or pressurizer level is greater than ~~77~~ 74 percent, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. The RCS depressurization reduces the break flow as shown in Figure ~~15.4-103~~ 15.4.3-6, and increases SI flow to refill the pressurizer as shown in Figure ~~15.4-99~~ 15.4.3-1.

(d) *Terminate SI to Stop Primary to Secondary Leakage*

The previous actions have established adequate RCS subcooling, verified a secondary side heat sink, and restored the reactor coolant inventory to ensure that SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to prevent repressurization of the RCS and to terminate primary to secondary leakage. The SI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for uncertainty, minimum AFW flow is available or at least one intact steam generator level is in the narrow range, the RCS pressure is Stable or increasing, and the pressurizer level is greater than the ~~4~~ 12 percent allowance for uncertainty. ~~To assure that the RCS pressure is increasing, SI was not terminated until the RCS pressure increased by at least 50 psi.~~

After depressurization was completed, an operator action time of ~~1~~ 2 minutes was assumed prior to SI termination. Since the above requirements are satisfied, SI termination was performed at this time. After SI termination, the ~~RCS~~ pressurizer pressure decreases as shown in Figure ~~15.4-100~~ 15.4.3-2. ~~The differential pressure between the RCS and the ruptured steam generator is shown in Figure 15.4-104.~~ Figure ~~15.4-105~~ 15.4.3-6 shows that the primary to secondary leakage continues after the SI flow was stopped until the RCS and ruptured steam generator pressures equalize.

The ruptured steam generator water volume is shown in Figure ~~15.4-106~~ 15.4.3-7. It is noted that the water volume in the ruptured steam generator is significantly less than the total steam generator volume of ~~5759~~ 5730 ft^3 when the break flow is terminated. The mass of water in the ruptured steam generator is also shown as a function of time in Figure ~~15.4-107~~ 15.4.3-8.

No Changes

Mass Releases

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

The operator actions for the SGTR recovery up to the termination of primary to secondary leakage were simulated in the LOFTTR2 analysis. 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 for the accident until the leakage was terminated.

Following the termination of leakage, it was assumed that the actions are taken to cool down the plant to cold shutdown conditions. The PORVs for the intact steam generators were 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 steam releases and the feedwater flows for the intact steam generator for the period from leakage termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact steam generator conditions at the time of leakage termination and at 2 hours. 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 to the RHR in-service pressure of 405 psia via steam release from the ruptured steam generator PORV. The RCS pressure was also assumed to be reduced concurrently as the ruptured steam generator is depressurized. It was 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 and ruptured steam generators from a mass and energy balance using conditions at 2 hours and at the RHR system in-service conditions.

After 8 hours, it was assumed that further plant cooldown to cold shut down 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.

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 would be through the condenser air ejector. After reactor trip, the releases to the atmosphere were assumed to be via the steam generator PORVs. The mass release rates to the atmosphere from the LOFTTR2 analysis are presented in

Figures ~~15.4-108~~ 15.4.3-9 and ~~15.4-109~~ 15.4.3-10 for the ruptured and intact steam generators, respectively, for the time period until leakage termination. Insert 15.4.3-1 The mass releases calculated ~~form~~ to the time of leakage termination until 2 hours and from 2-8 hours were also assumed to be released to the atmosphere via the steam generator PORVs. The mass releases for the SGTR event for the 0-2 hour and 2-8 hour time intervals are presented in Table 15.4-14.

15.4.4 SINGLE REACTOR COOLANT PUMP LOCKED ROTOR

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell-side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in that sequence. The three power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

Three digital computer codes are used to analyze this transient. The LOFTRAN[26] code is used to calculate the resulting loop and core coolant flow following the pump seizure. The LOFTRAN code is also used to calculate the time of reactor trip based on the calculated flow, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[17] code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient. The THINC[31] code is used to calculate the DNBR during the transient based on flow calculated by LOFTRAN and heat flux calculated by FACTRAN.

The following case is analyzed:

- All loops operating, one locked rotor

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the RCPs is assumed to seize, the plant is assumed to be operating under steady state operating conditions with respect to the margin to DNB, i.e., normal steady state power level, nominal

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.4.3

INSERT 15.4.3-1

The total flashed break flow from the LOFTTR2 analysis is presented in Figure 15.4.3-11.

No Changes

15.4.9.2 Conclusions

The probability of a volume control tank rupture is small, but the probability of the release of all or part of the contents of a tank through operator error or valve failure should be considered somewhat greater. The release of the total contents of a volume control tank is taken as the postulated accident. Smaller leaks and spills from the volume control tank were found to have negligible environmental consequences, and therefore are not included. The analysis of the radiological effects of this accident is contained in Section 15.5.

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32. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, 1979.

33. Deleted in Revision 12.

34. Deleted in Revision 12.

35. Deleted in Revision 12.

36. Deleted in Revision 12

37. Deleted in Revision 12.

38. Deleted in Revision 12.

39. Deleted in Revision 12.

40. W. J. Johnson and C. M. Thompson, Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version, WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.

Insert 15.4.10-1

41. D.F. Holderbaum, R. N. Lewis, T. A. Miller and K. Rubin, LOFTR2 Analysis For A Steam Generator Tube Rupture For The Diablo Canyon Power Plant Units 1 and 2. WCAP-11723 (Proprietary)/WCAP-11724 (Nonproprietary). February, 1988.

42. Plant Manual, Volume 3A, Emergency Operating Procedures, Diablo Canyon Power Plant Units 1 and 2.

43. Deleted in Revision 12.

44. Deleted in Revision 12.

45. Deleted in Revision 12.

46. M. Cerrone, R. N. Lewis and J. S. Monahan, Steam Generator Tube Rupture Margin to Overfill Analysis for the Diablo Canyon Power Plant, Units 1 and 2. WCAP-11723 Update, PGE-92-685, September 1992.

47. Diablo Canyon Unit 1 10 CFR 50.46 30-Day Report of Significant ECCS Evaluation Model Changes, Letter from G. M. Rueger (PG&E) to NRC, DCL-93-029, February 2, 1993.

48. Deleted in Revision 12.

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.4.10

INSERT 15.4.10-1

Steam Generator Tube Rupture (SGTR) Re-analysis Report, Letter from W. R. Rice (Westinghouse) to M. Mayer (PG&E), PGE-01-535, October 26, 2001.

TABLE 15.4-12

OPERATOR ACTION TIMES FOR DESIGN BASIS SGTR ANALYSIS

Action	Time (min)
Identify and isolate ruptured SG	10 min or LOFTTR2 calculated time to reach ~~27~~ 28% narrow range level in the ruptured SG, whichever is longer
Operator action time to initiate cooldown	5
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4
Depressurization	Calculated by LOFTTR2
Operator action time to initiate SI termination	~~1~~ 2
SI termination and pressure equalization	Calculated time for SI termination and equalization of RCS and ruptured SG pressures

TABLE 15.4-13

TIMED SEQUENCE OF EVENTS - SGTR ANALYSIS

Event	Time (sec)
SG Tube Rupture	0
Reactor Trip	~~116~~ 112
SI Actuated	~~174~~ 152
Turbine Driven AFW Pump Flow Isolated	444
Ruptured SG Isolated	~~636~~ 650
Ruptured SG PORV Fails Open	~~638~~ 652
Ruptured SG PORV Block Valve Closed	~~2440~~ 2452
RCS Cooldown Initiated	~~2742~~ 2752
RCS Cooldown Terminated	~~3664~~ 3612
RCS Depressurization Initiated	~~3904~~ 3860
RCS Depressurization Terminated	~~4034~~ 3988
SI Terminated	~~4094~~ 4108
Break Flow Terminated	~~4718~~ 5040

TABLE 15.4-14

MASS RELEASE RESULTS - SGTR ANALYSIS

	0 - 2 Hrs, lbm	2 - 8 Hrs, lbm
Ruptured SG		
- Condenser	~~127,100~~ 119,500	0
- Atmosphere	~~146,700~~ 151,500	41,400 ~~37,800~~
- Feedwater	~~149,600~~ 143,400	0
Intact SGs		
- Condenser	~~375,700~~ 354,400	0
- Atmosphere	~~444,800~~ 379,400	973,400 ~~901,400~~
- Feedwater	~~1,070,900~~ 979,500	1,037,900 ~~906,200~~
Break Flow	~~291,800~~ 272,400	0
Flashed Break Flow	17,904	0

FIGURES (Continued)

FIGURES (Continued)

CHAPTER 15

FIGURES (Continued)

Replace with Figure 15.4.3-1

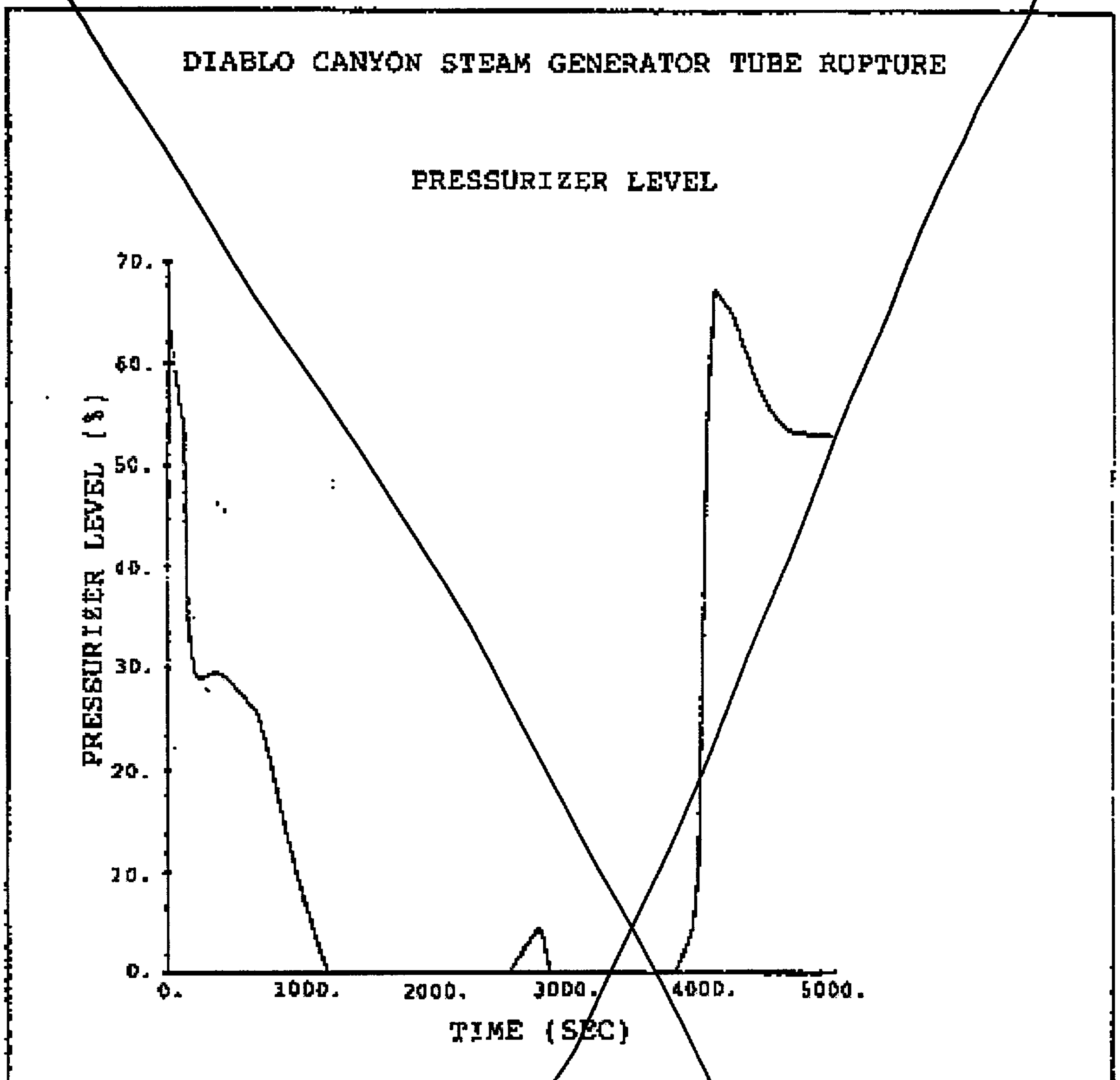

Figure 15.4-99 Pressurizer Level

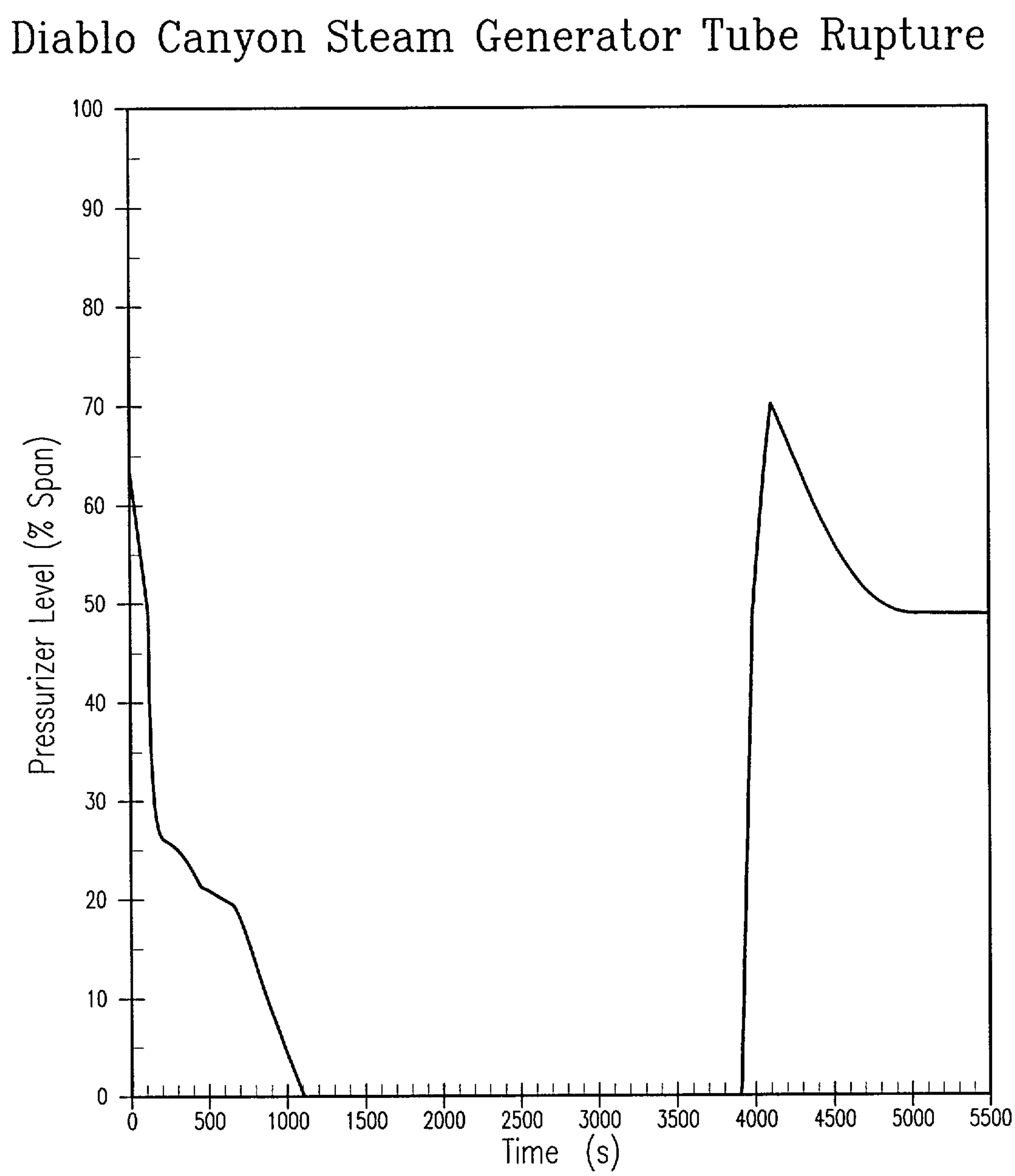

Figure 15.4.3-1 Pressurizer Level

Replace with Figure 15.4.3-2

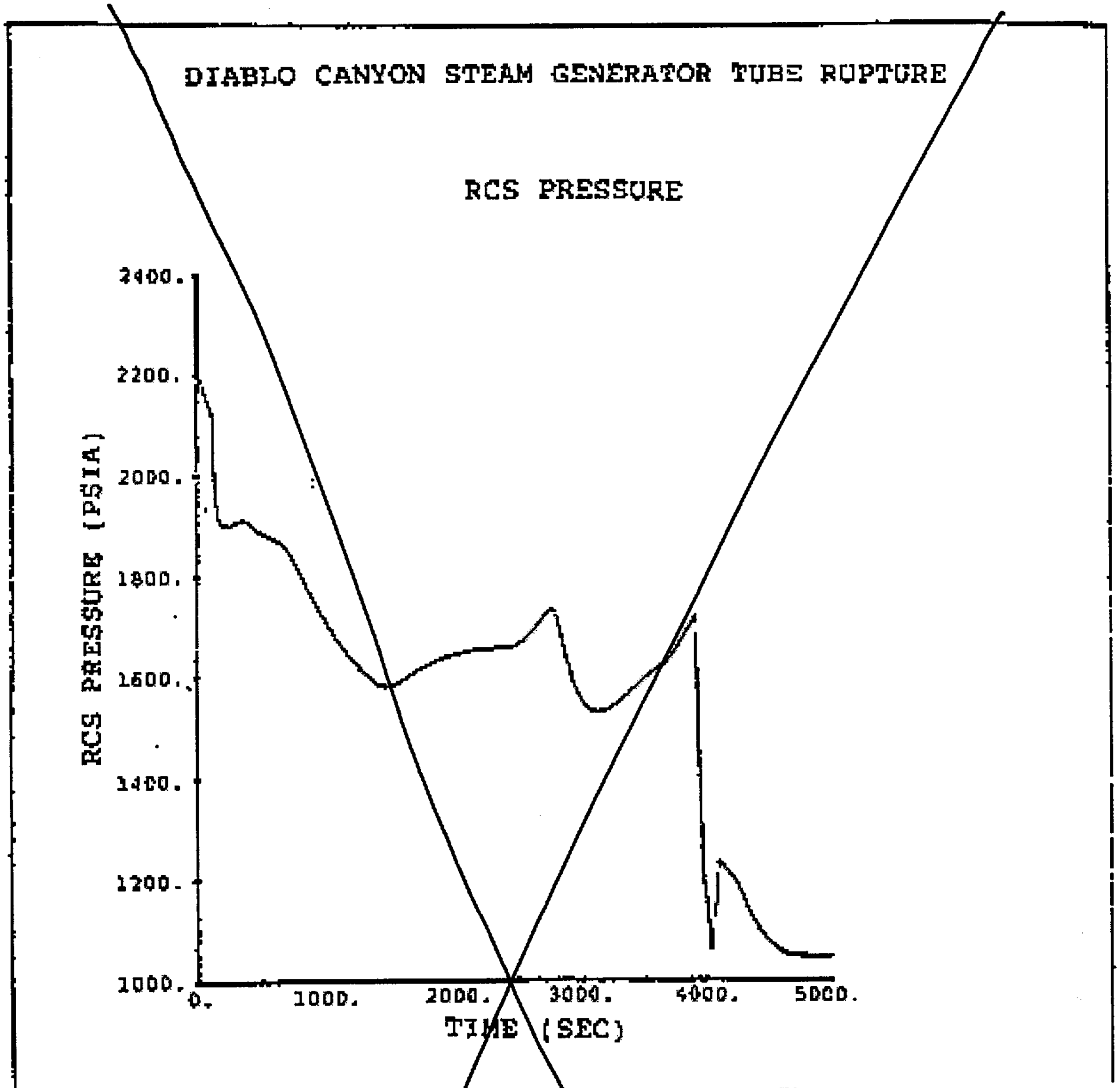

Figure 15.4-100 RCS Pressure

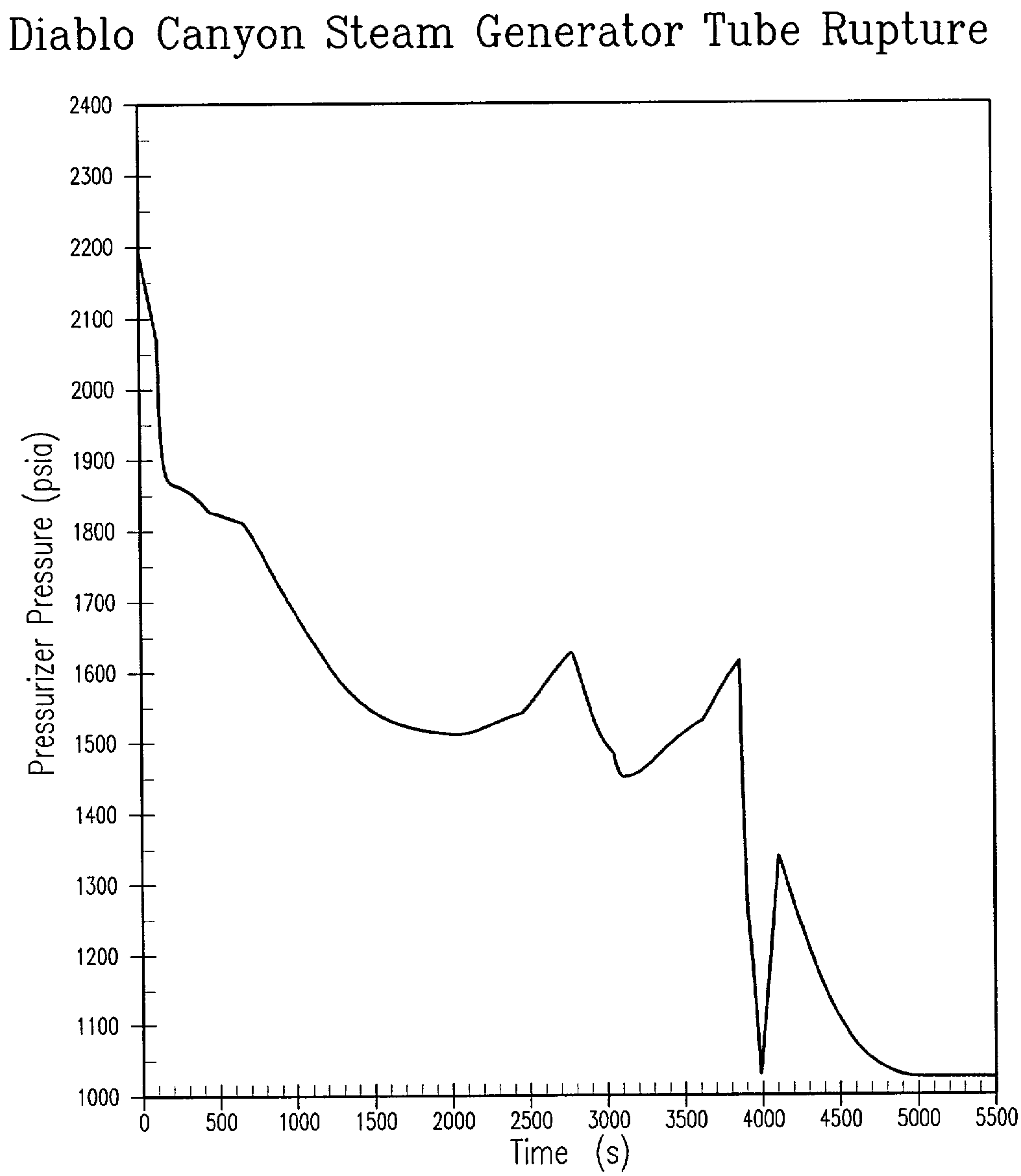

Figure 15.4.3-2 Pressurizer Pressure

Replace with Figure 15.4.3-3

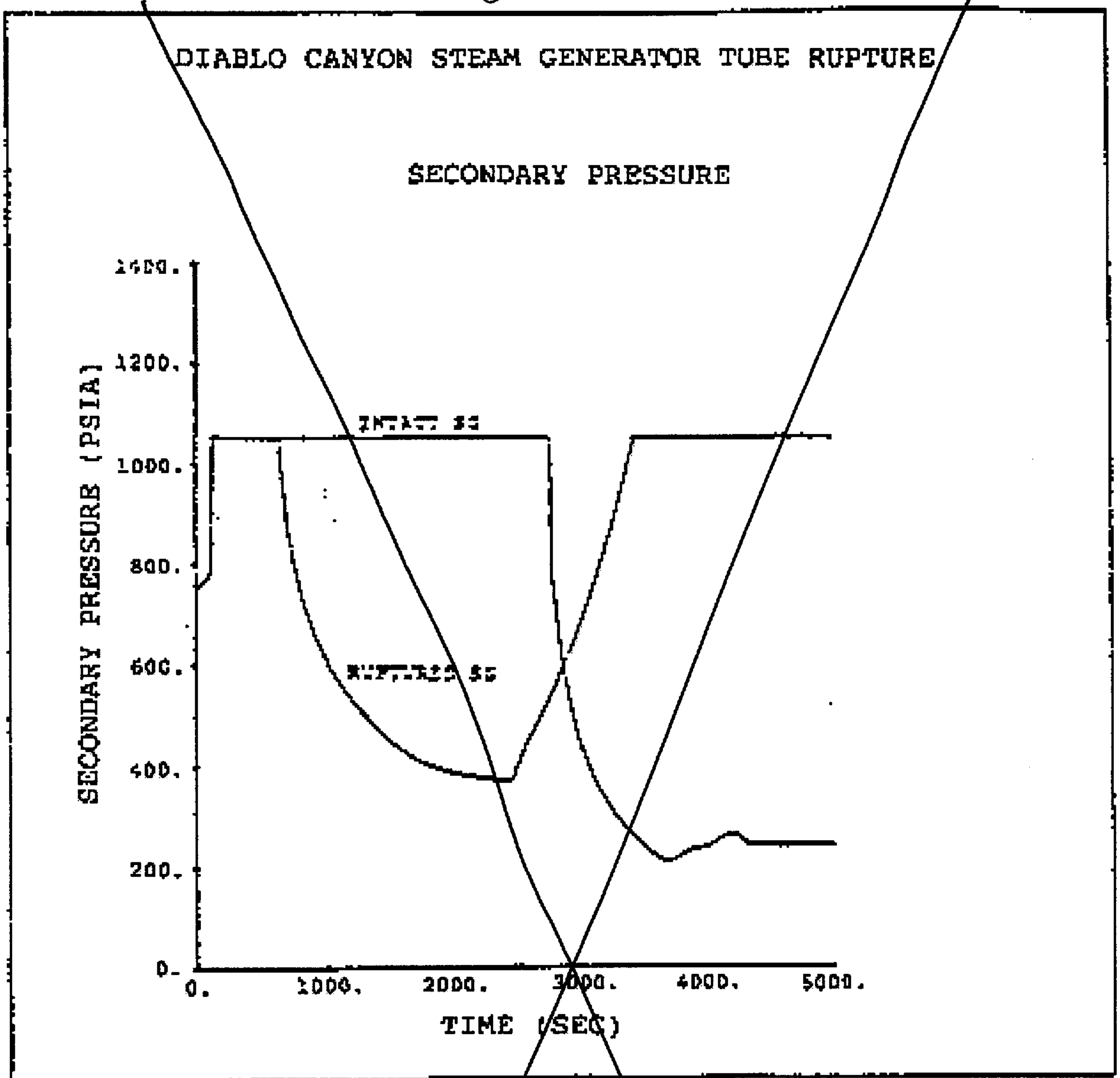

Figure 15.4-101 Secondary Pressure

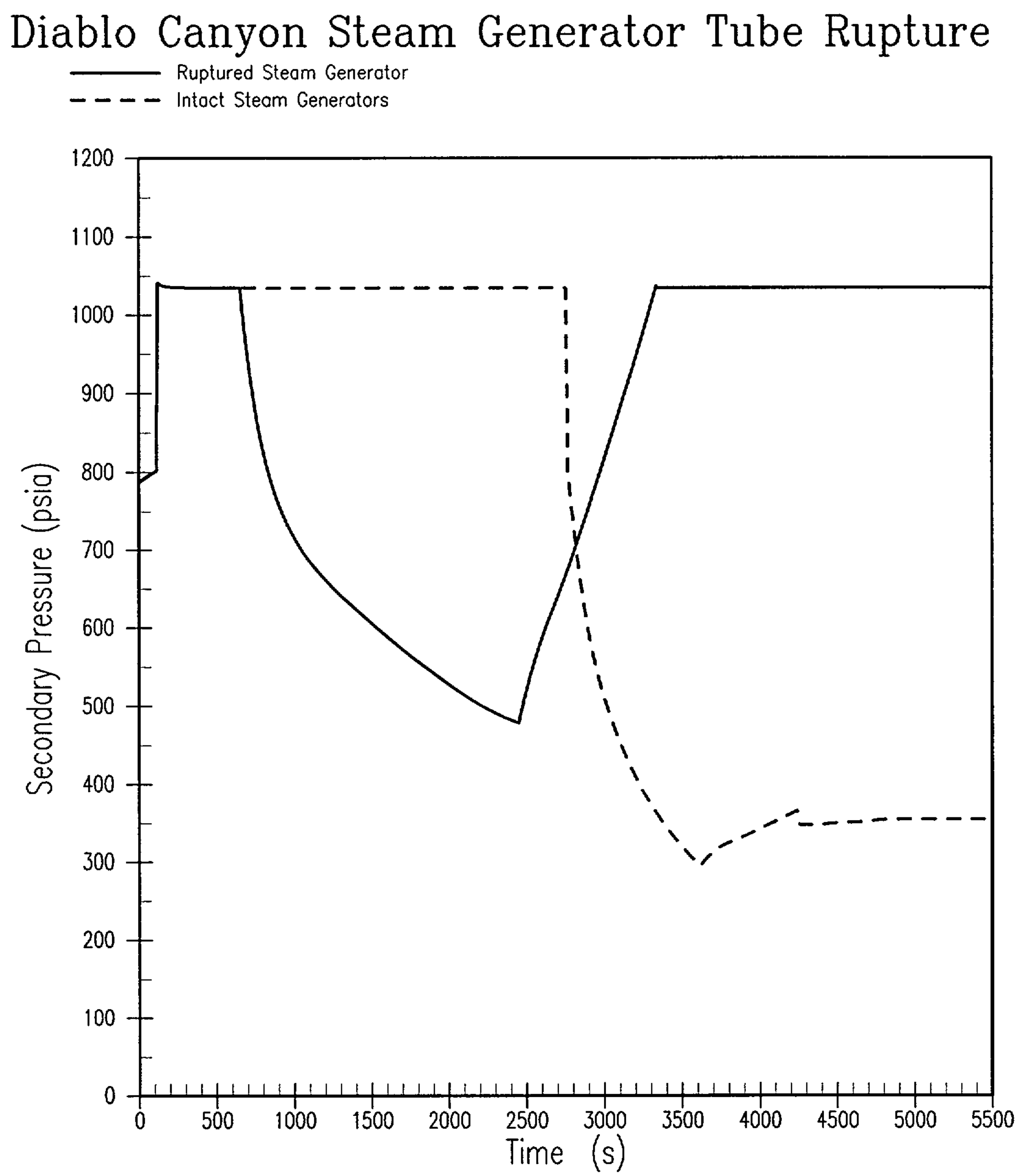

Figure 15.4.3-3 Secondary Pressure

Replace with Figure 15.4.3-4

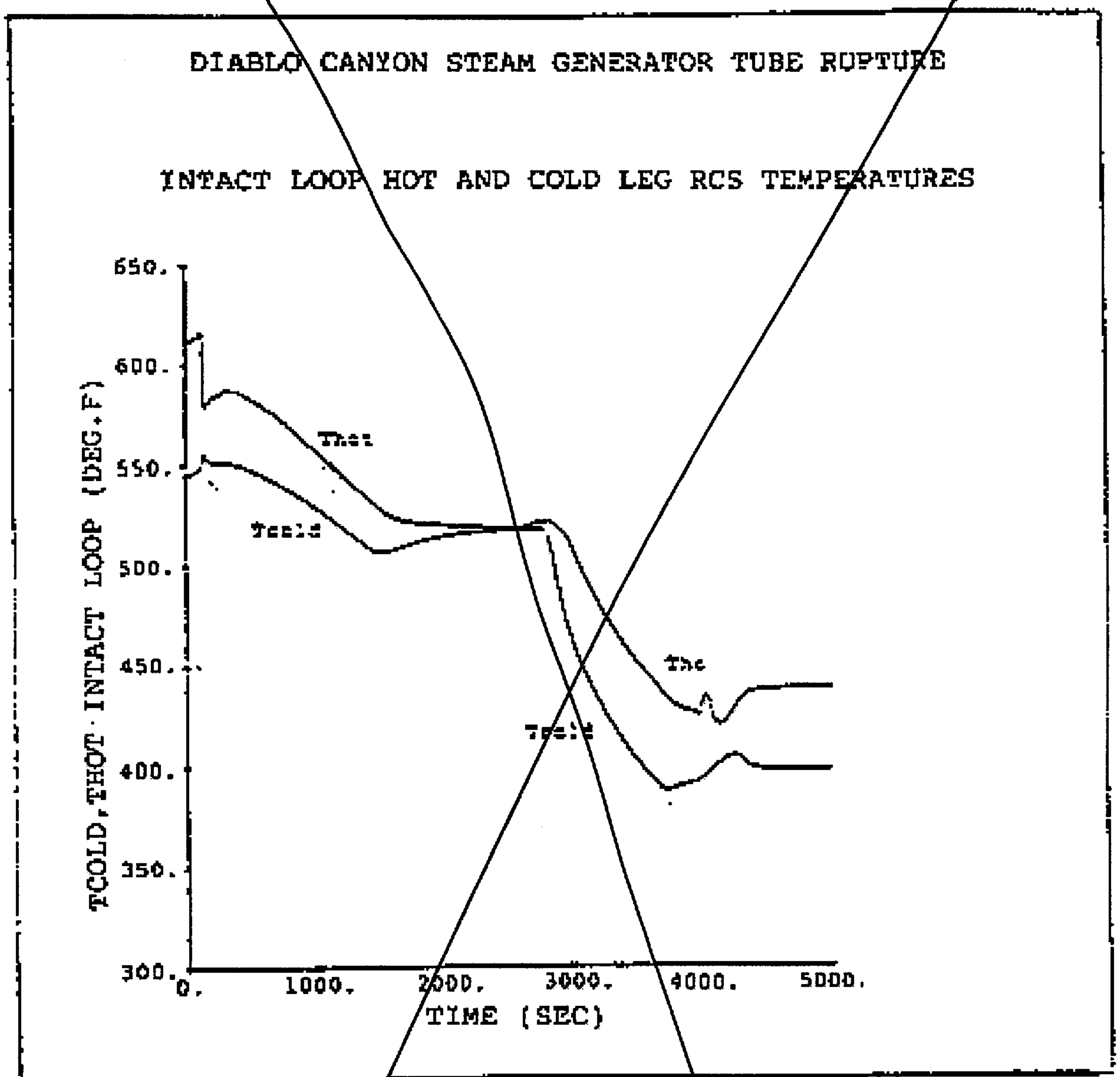

Figure 15.4-102 Intact Loop Hot and Cold Leg RCS Temperature

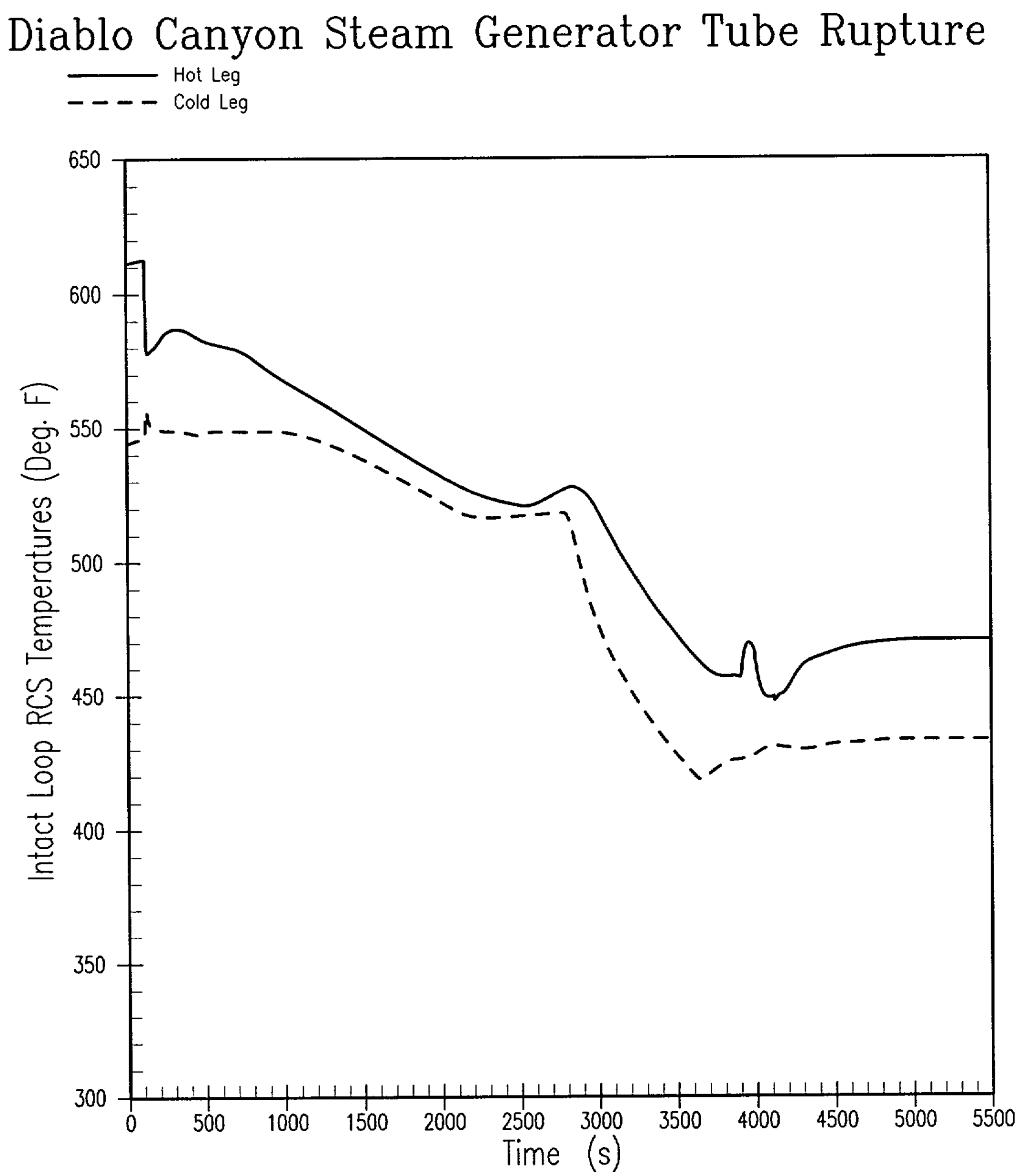

Figure 15.4.3-4 Intact Loop Hot and Cold Leg RCS Temperature

Replace with Figure 15.4.3-5

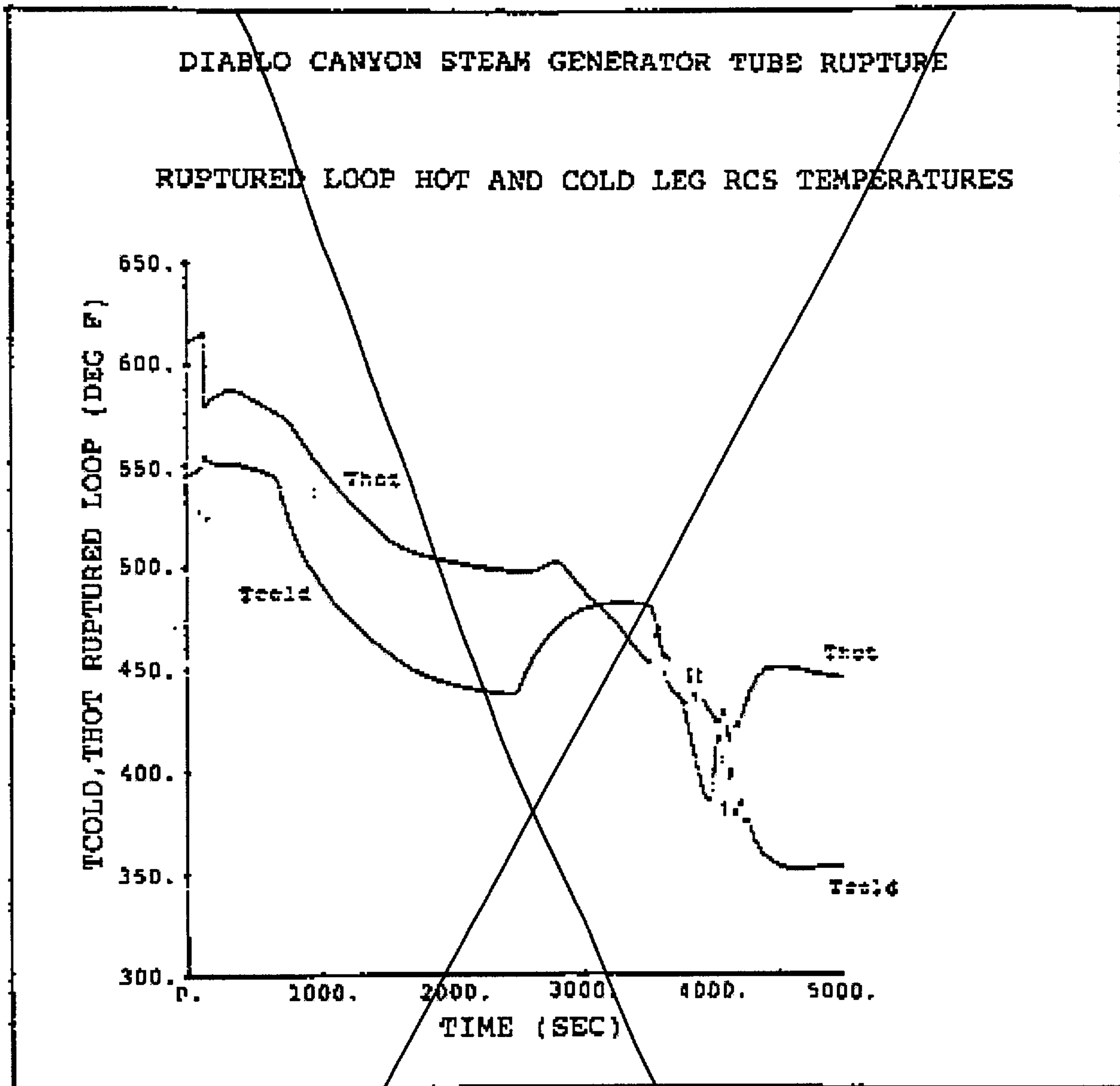

Figure 15.4-103 Ruptured Loop Hot and Cold Leg RCS Temperatures

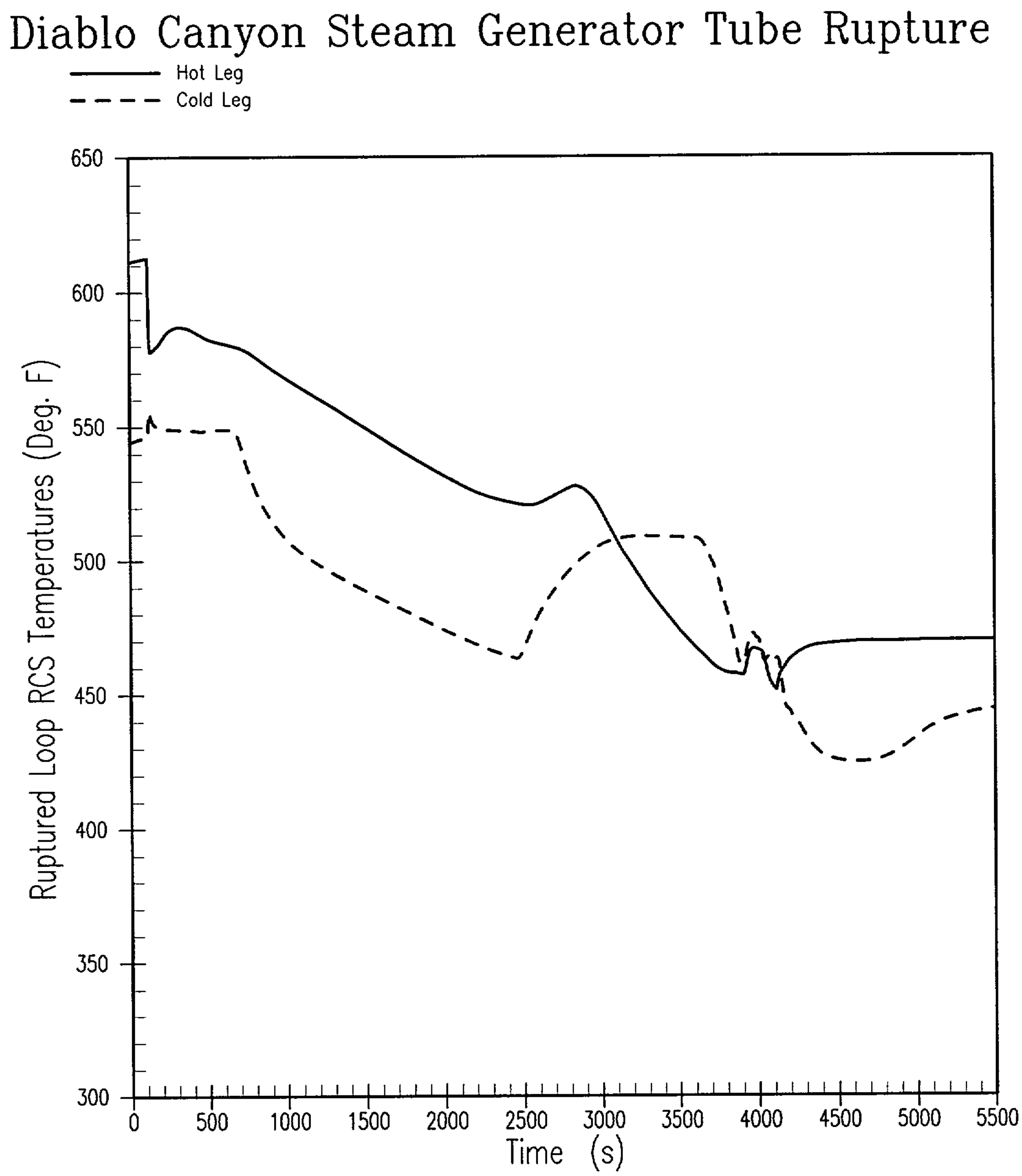

Figure 15.4.3-5 Ruptured Loop Hot and Cold Leg RCS Temperature

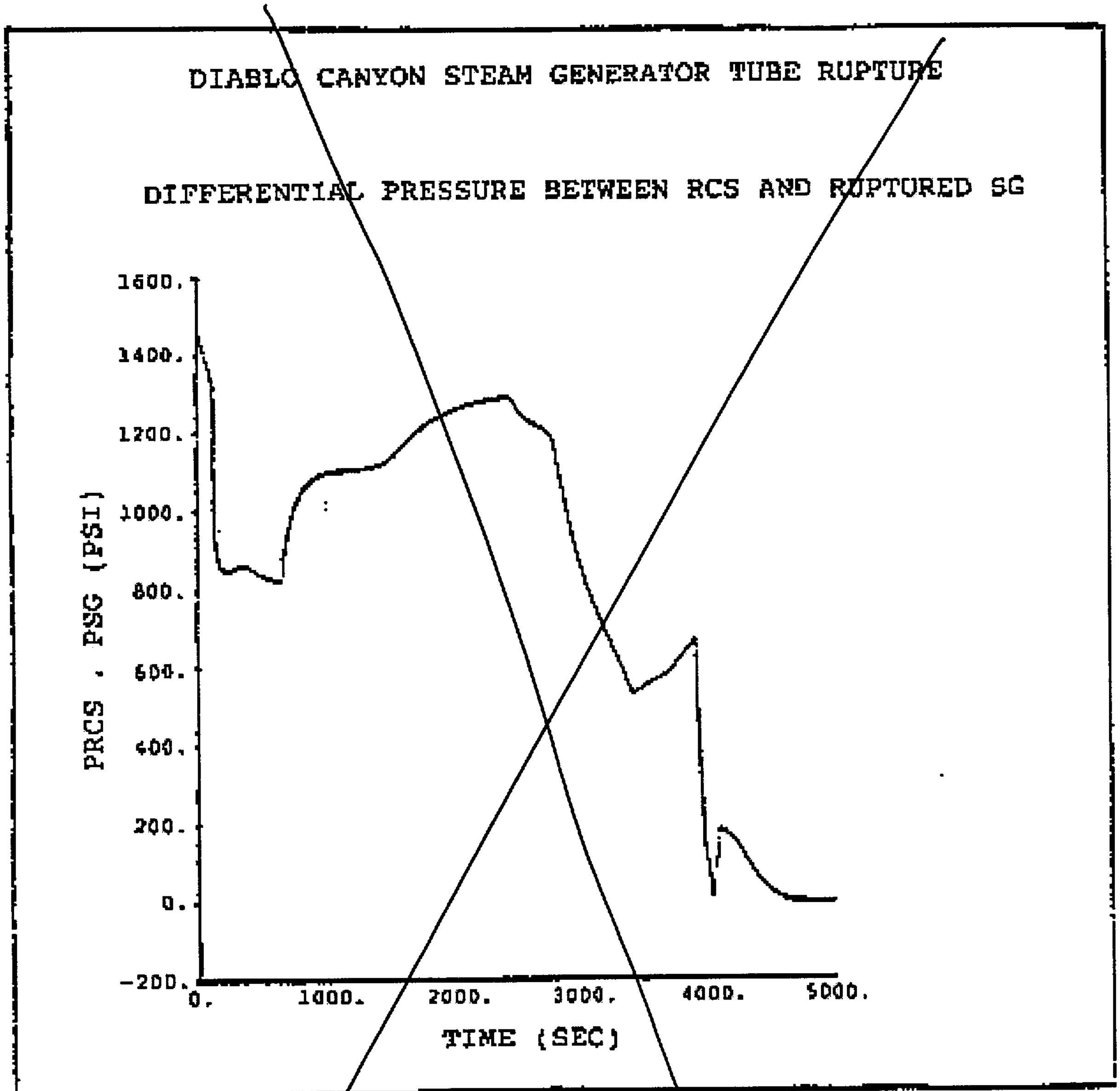

Figure 15.4-104 Differential Pressure Between RCS and Ruptured SG

Replace with Figure 15.4.3-6

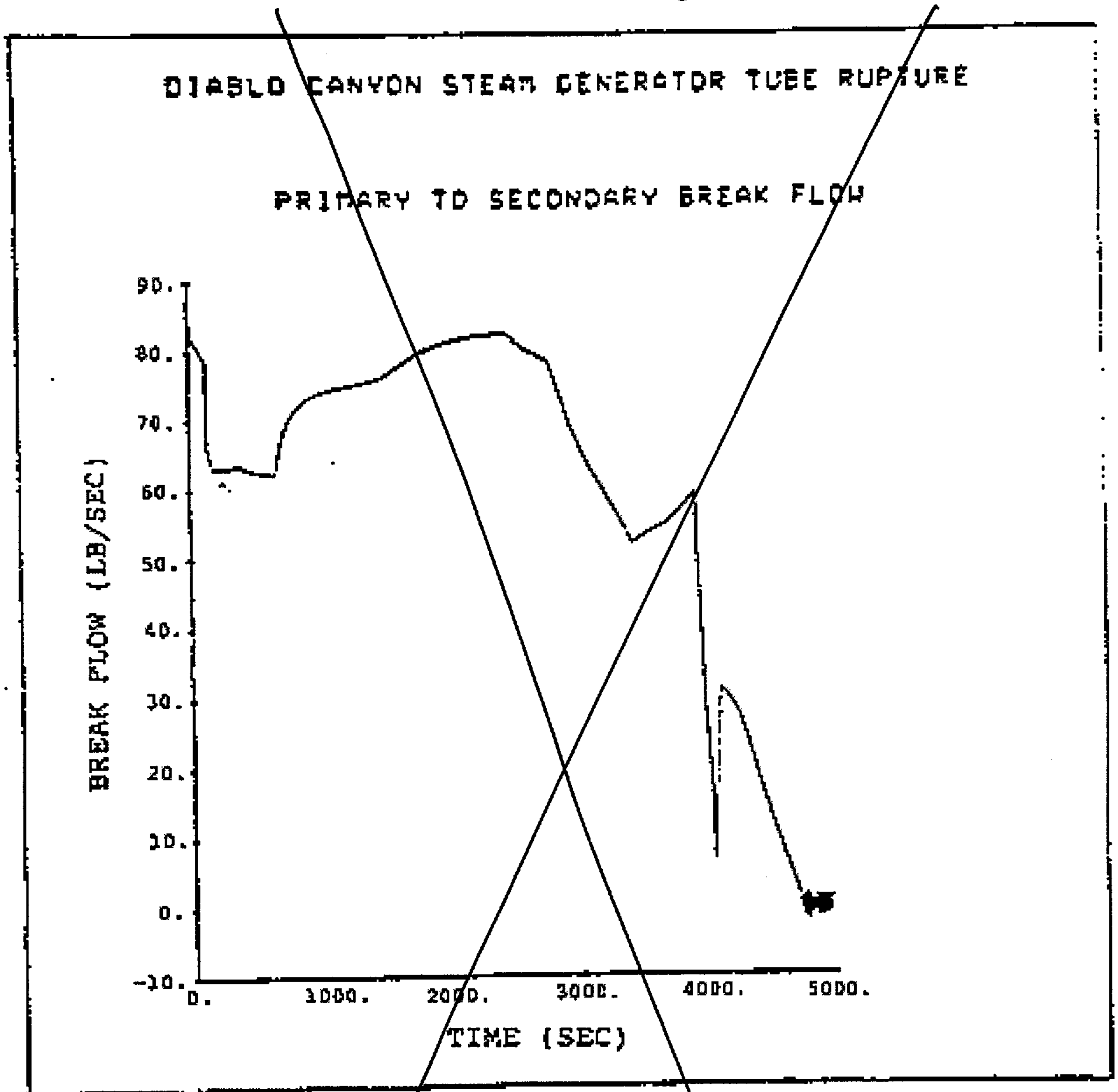

Figure 15.4-105 Primary to Secondary Break Flow Rate

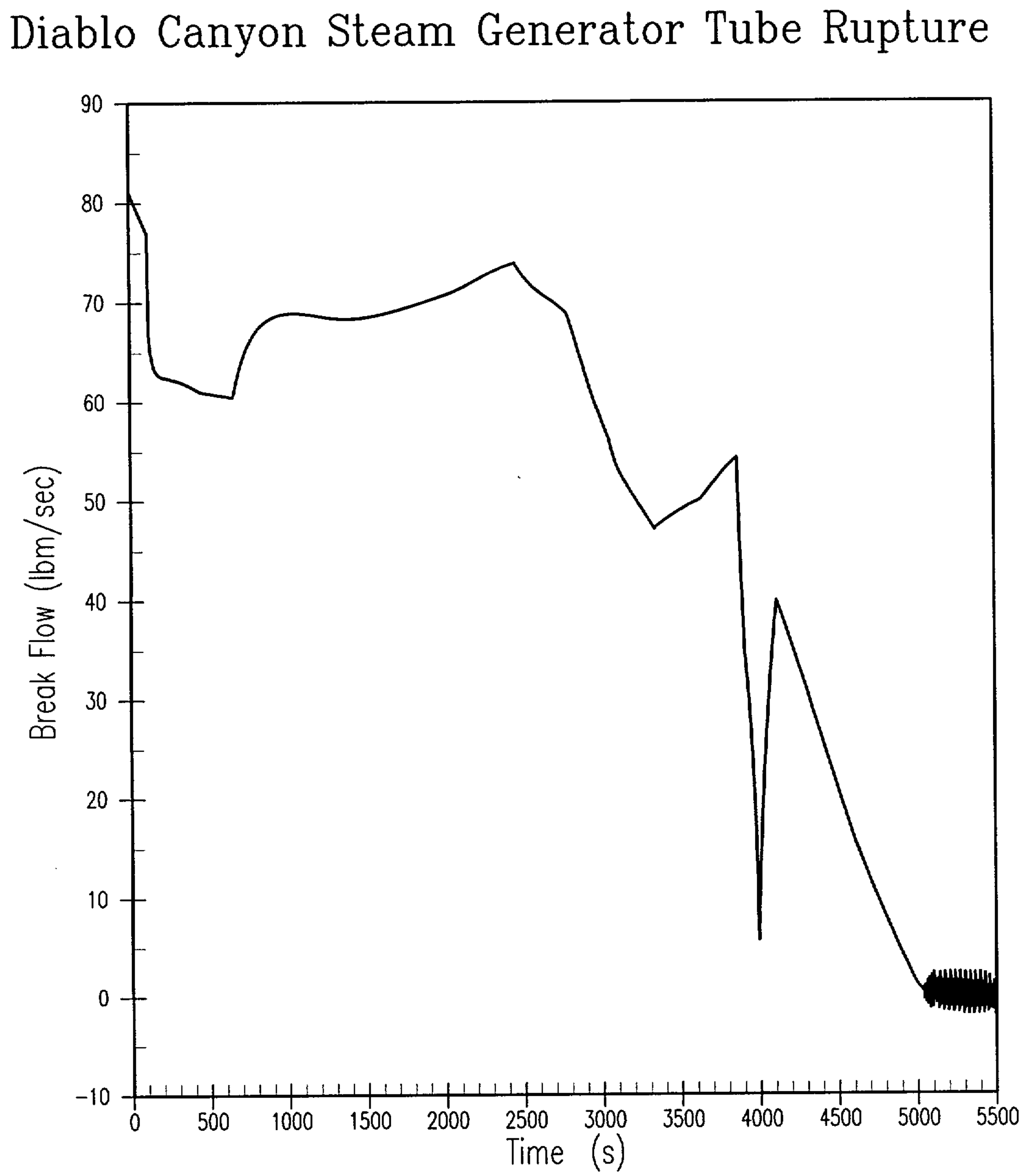

Figure 15.4.3-6 Primary to Secondary Break Flow Rate

Replace with Figure 15.4.3-7

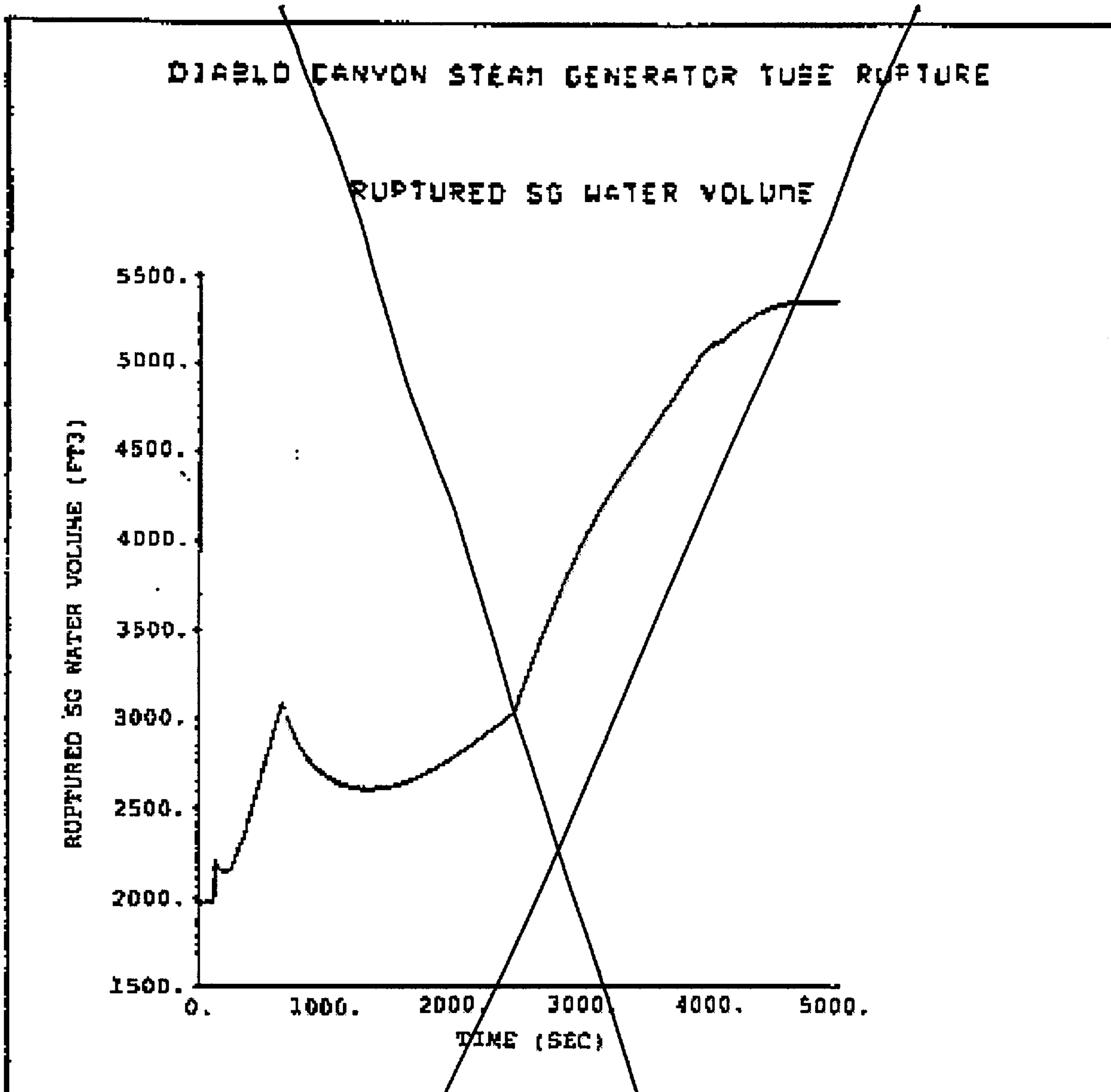

Figure 15.4-106 Ruptured SG Water Volume

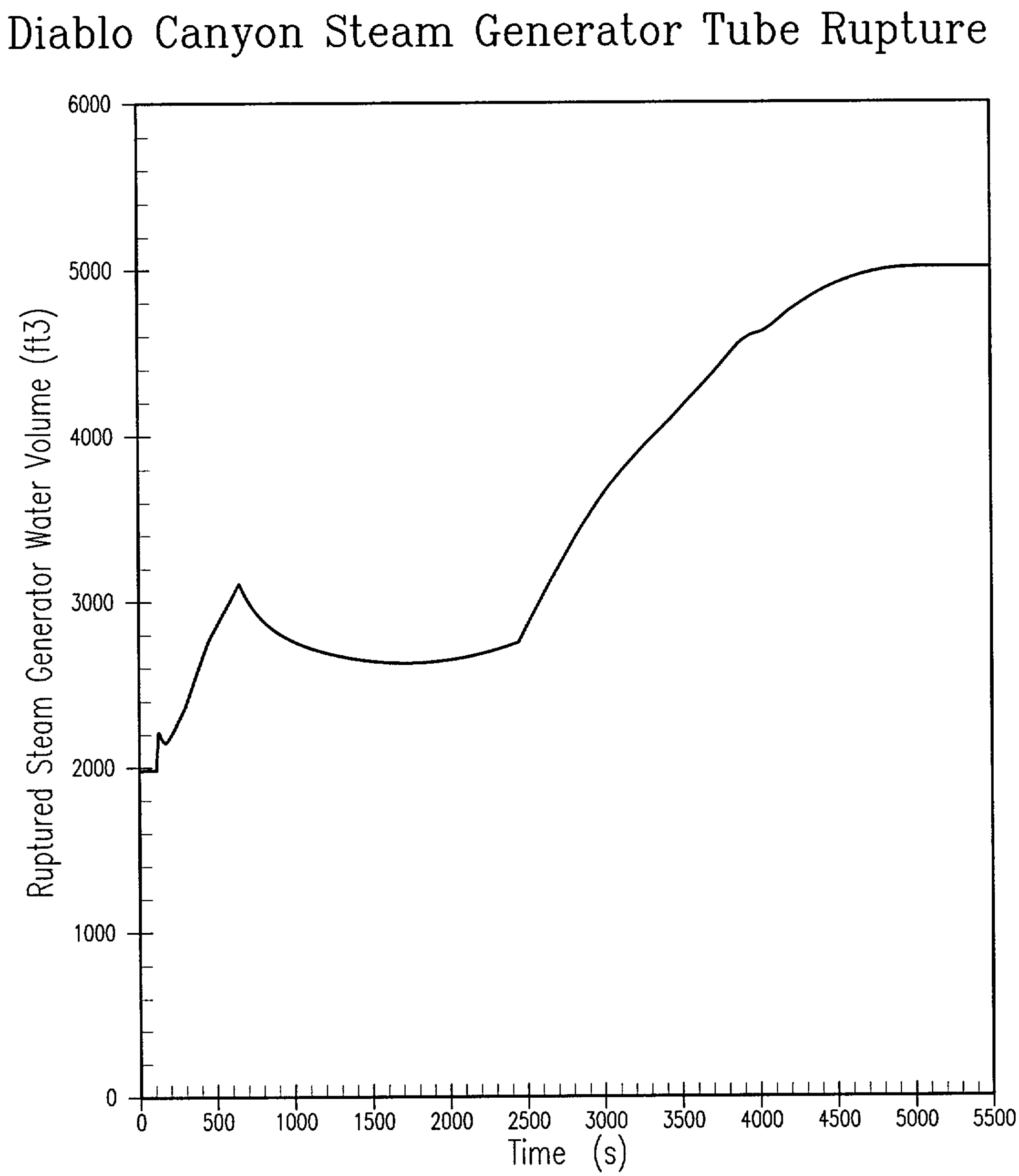

Figure 15.4.3-7 Ruptured SG Water Volume

Replace with Figure 15.4.3-8

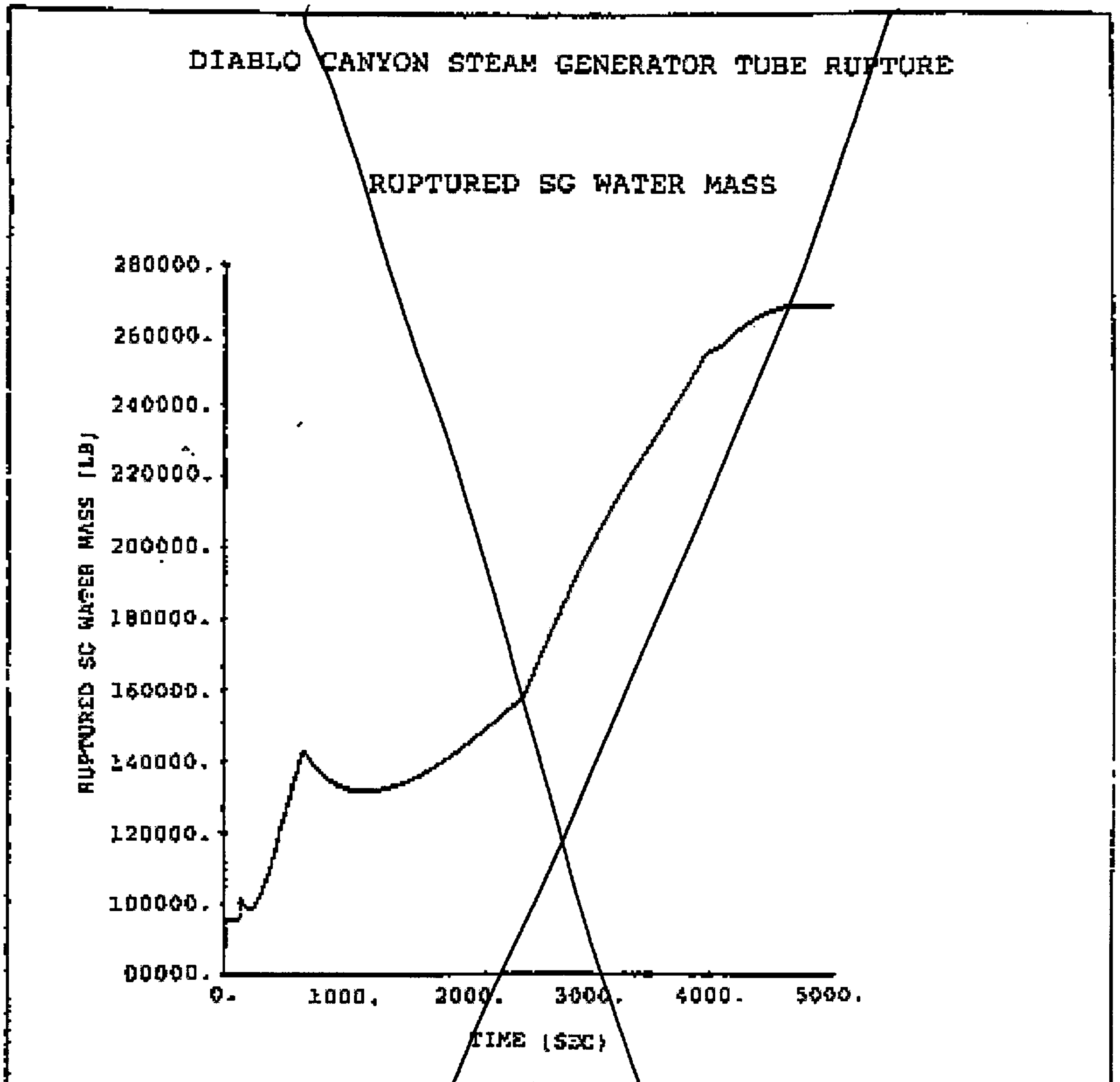

Figure 15.4-107 Ruptured SG Water Mass

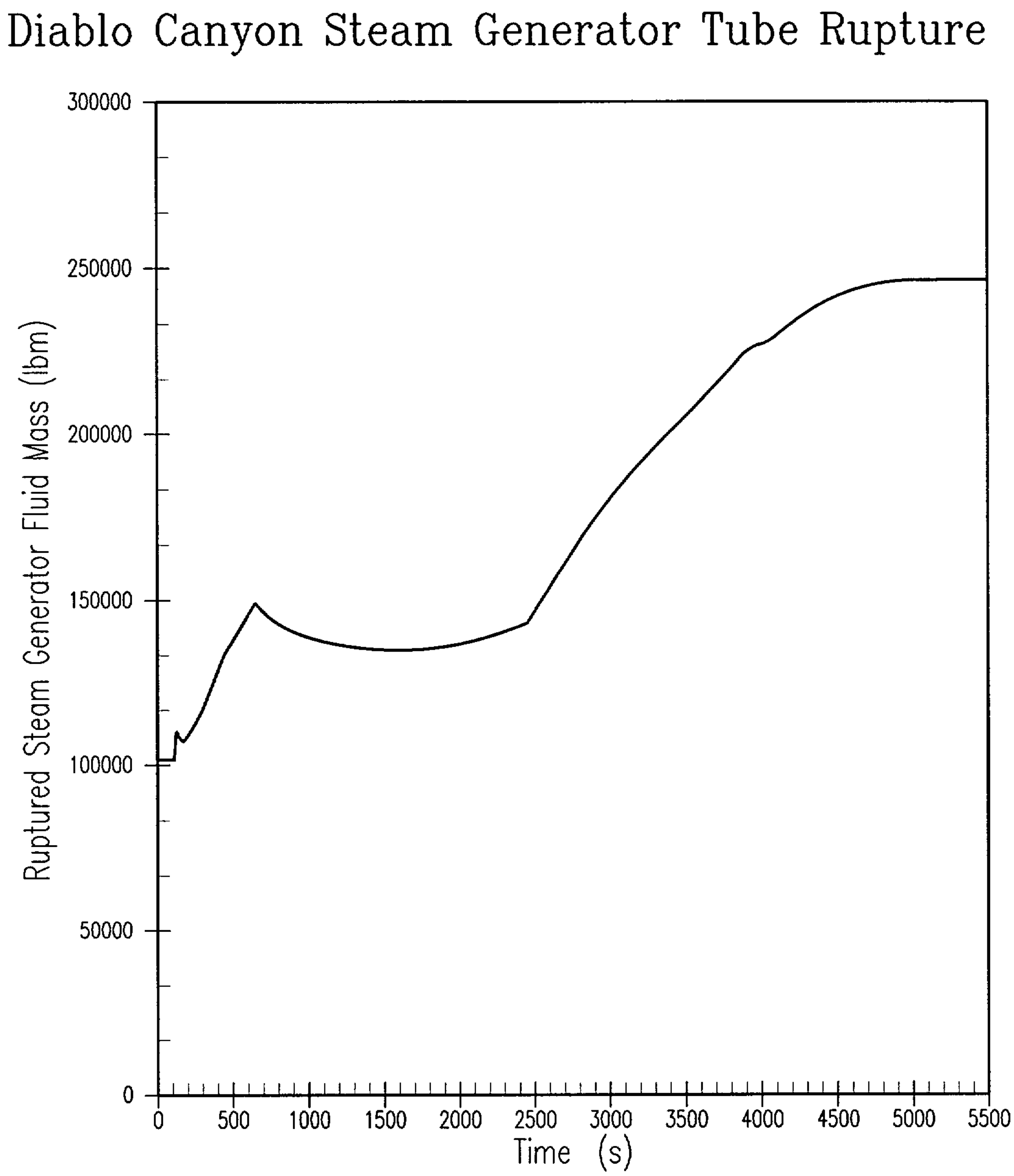

Figure 15.4.3-8 Ruptured SG Water Mass

Replace with Figure 15.4.3-9

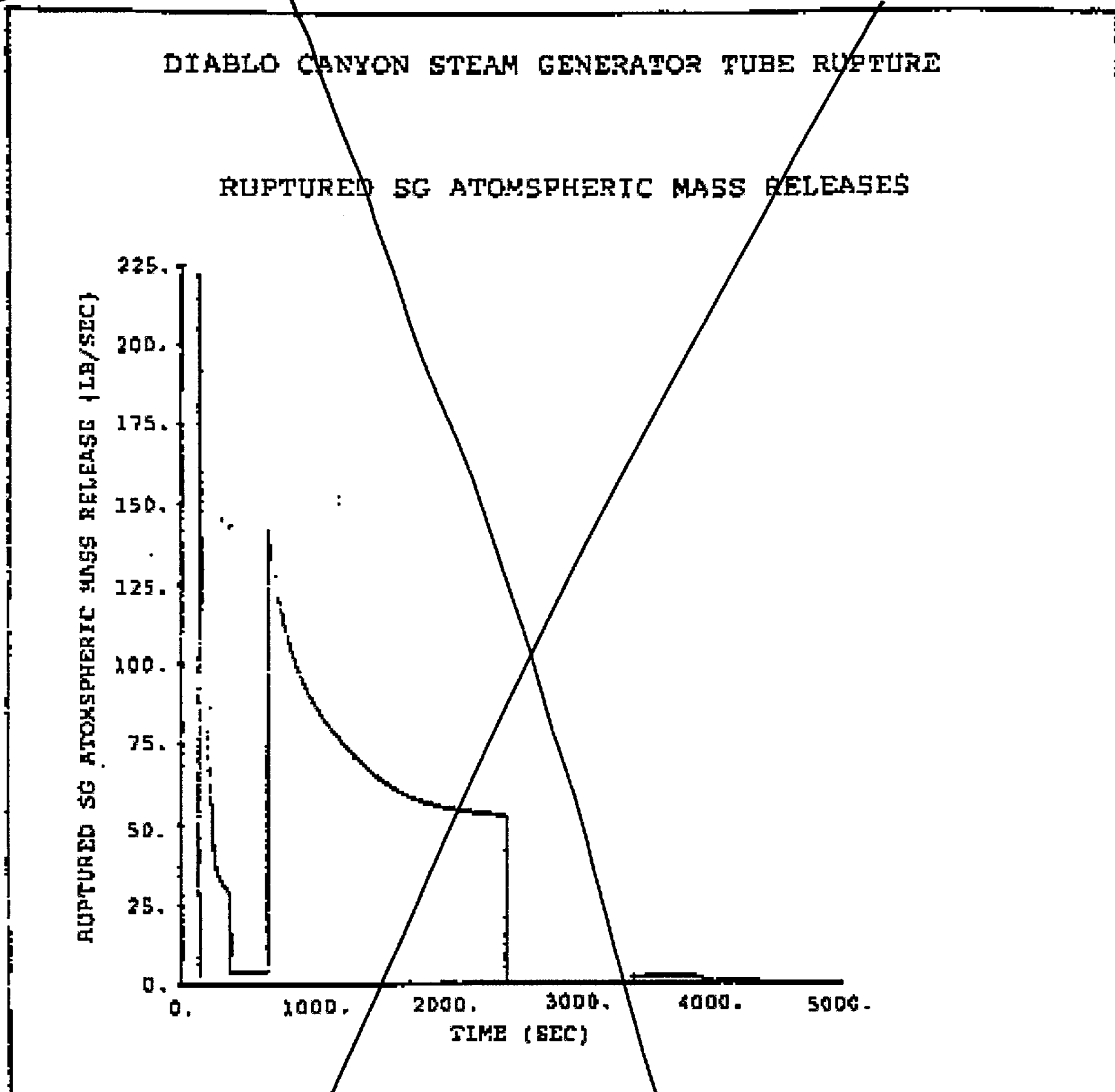

Figure 15.4-108 Ruptured SG Mass Release Rate to the Atomsphere

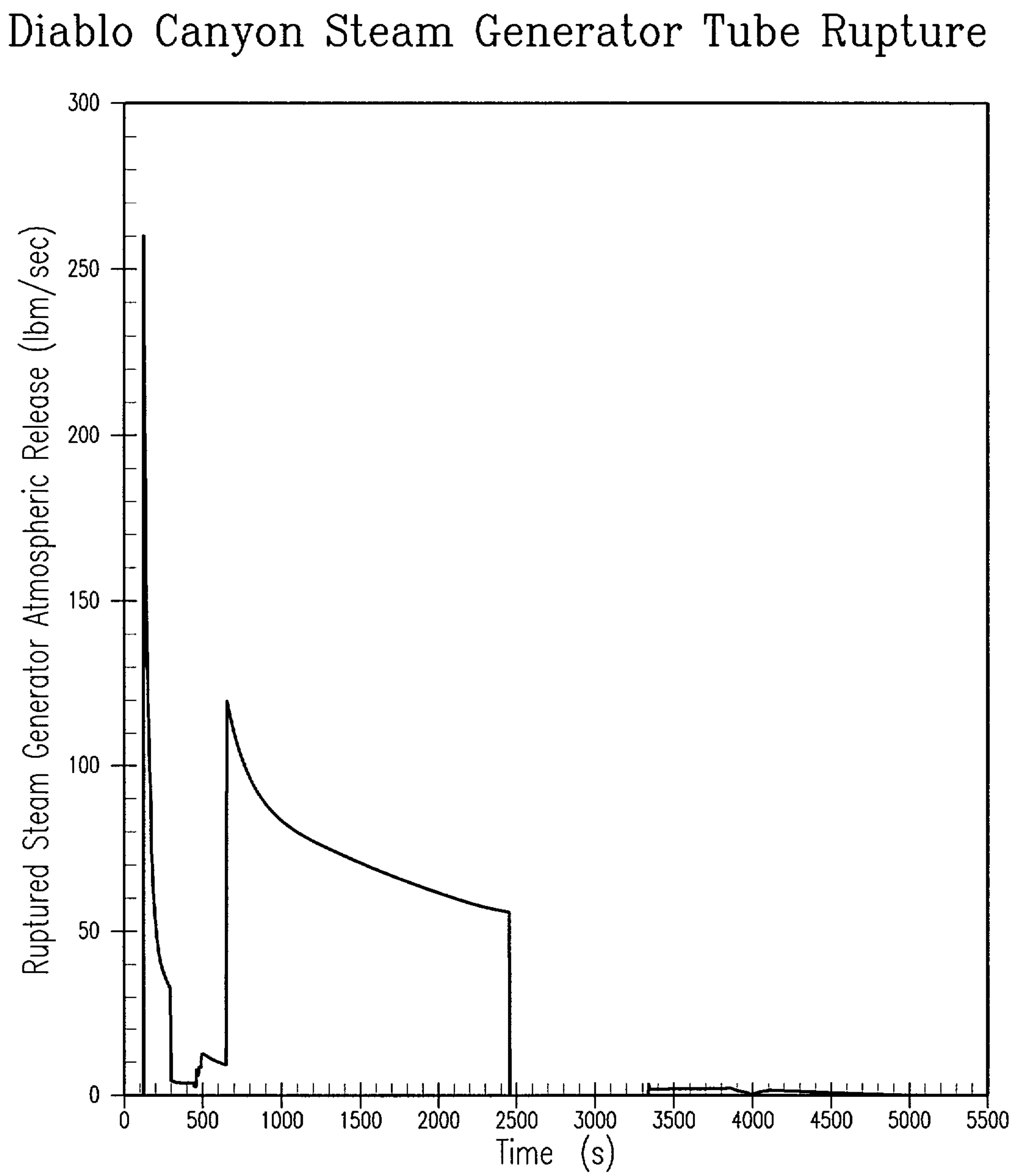

Figure 15.4.3-9 Ruptured SG Mass Release Rate to the Atmosphere

Replace with Figure 15.4.3-10

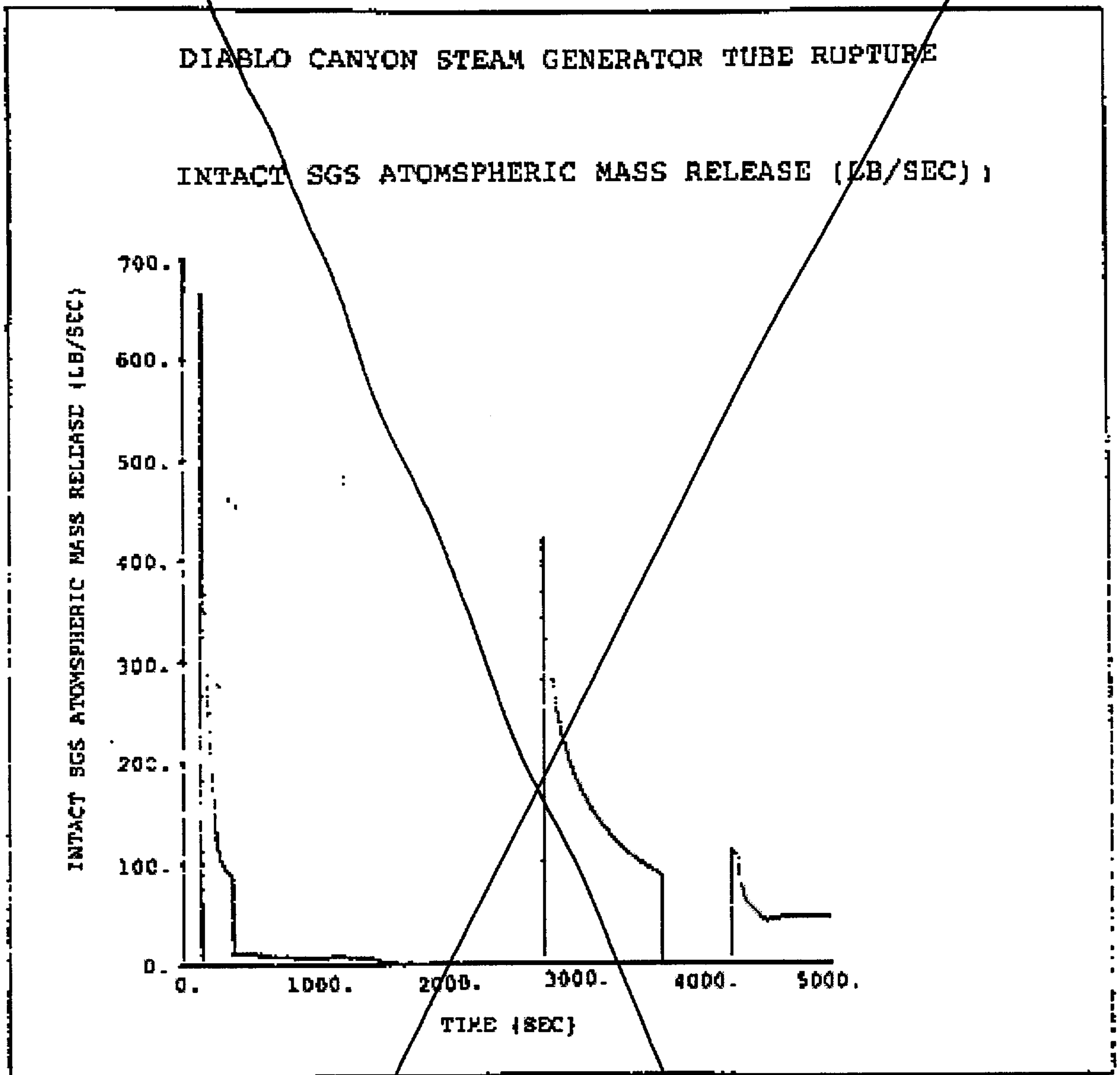

Figure 15.4-109 Intact SGs Mass Release Rate to the Atomsphere

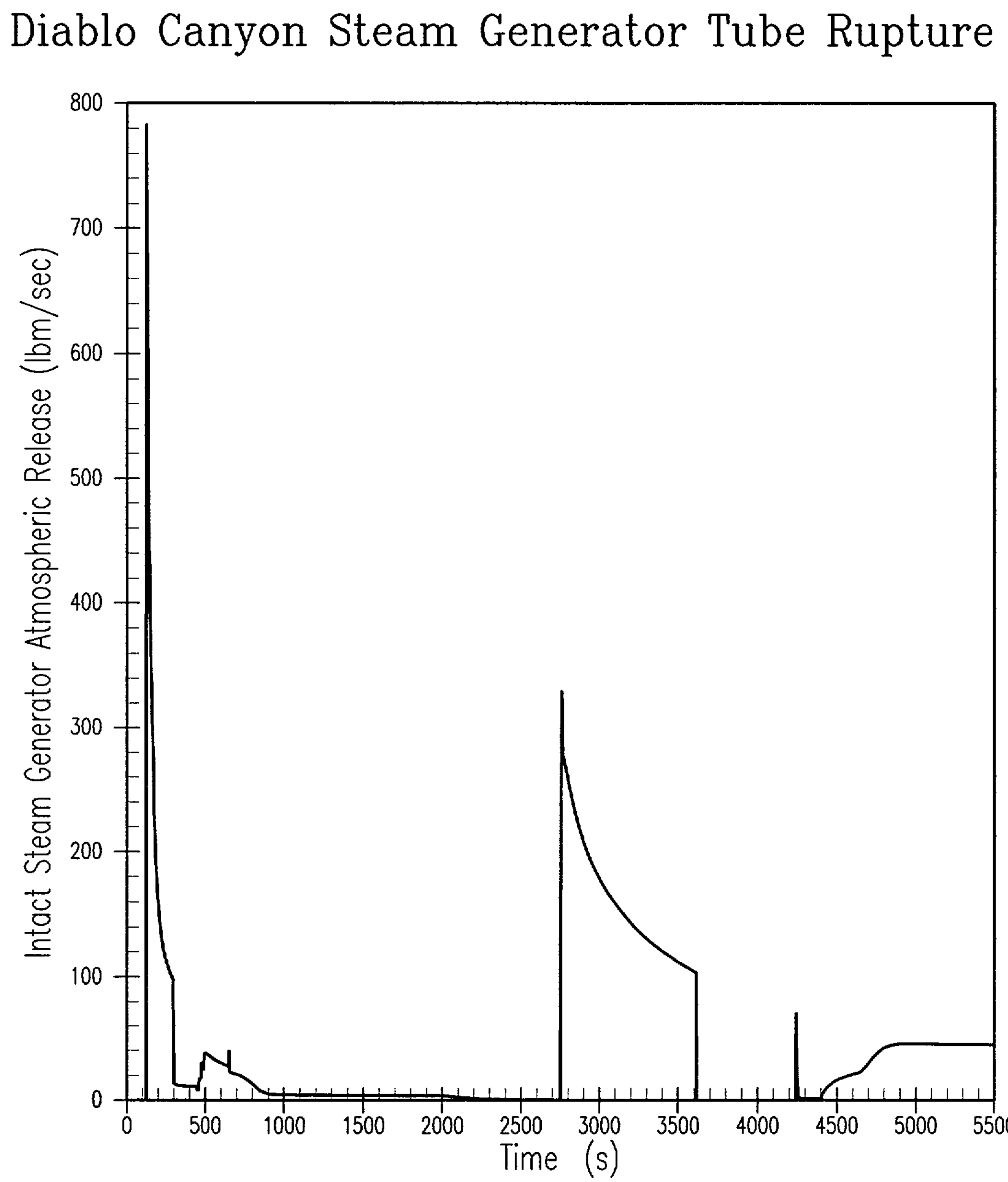

Figure 15.4.3-10 Intact SGs Mass Release Rate to the Atmosphere

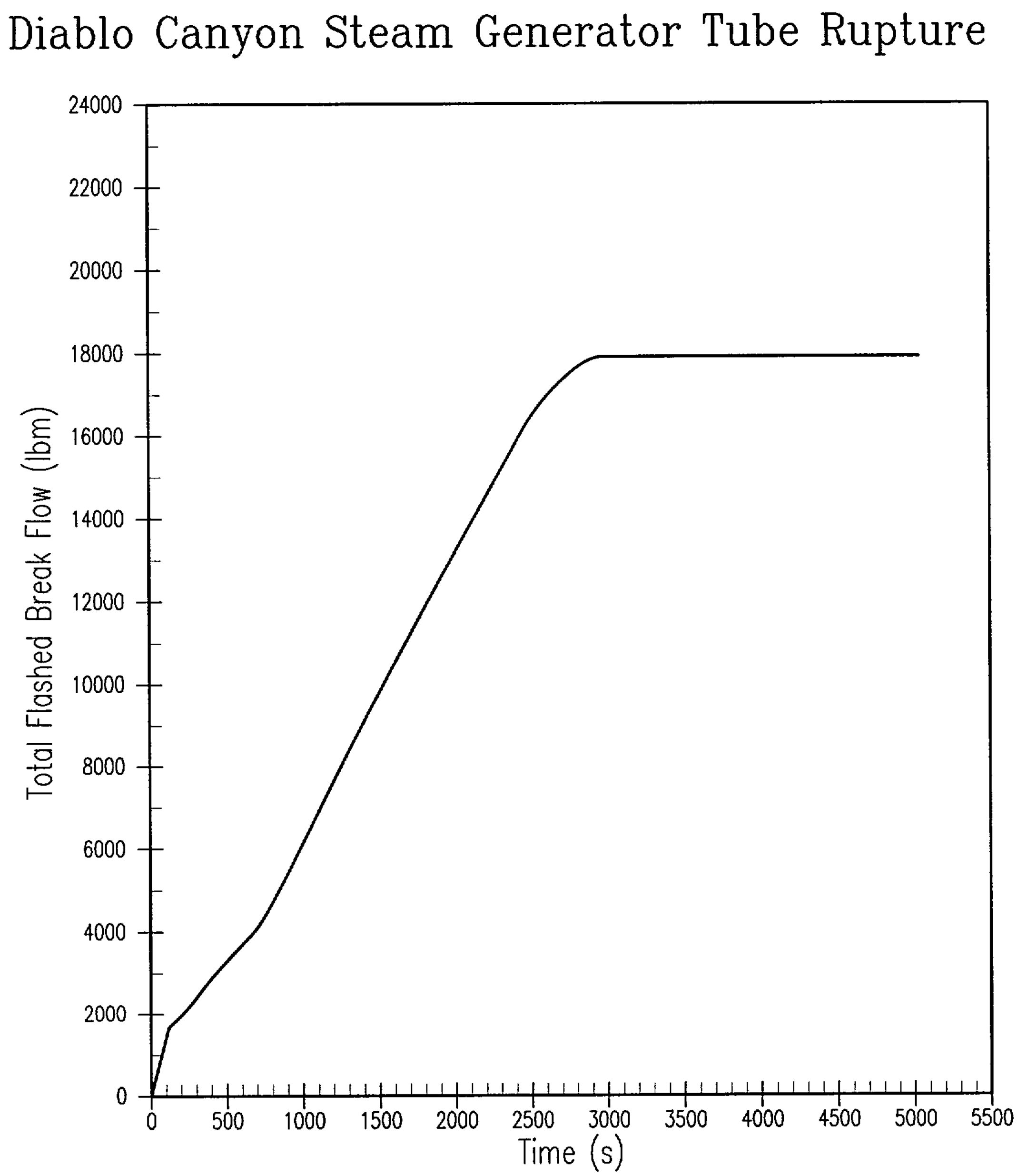

Figure 15.4.3-11 Total Flashed Break Flow

As discussed earlier, thyroid doses which would result from steam line breaks with different sets of assumed conditions can be determined from Figures 15.5-2 through 15.5-5.

It can be concluded from these results that potential doses from major or minor steam line breaks will be well below the guideline levels specified in 10 CFR 100, and that the occurrence of such ruptures would not result in undue risk to the public. A detailed evaluation of potential exposures to control room personnel is made in Section 15.5.17.9, for conditions following a LOCA. The containment shine contribution to control room dose would not be applicable following a steam line break accident. By comparing the activity releases following a steam line break accident, given in Table 15.5-35, with the activity releases calculated for a LOCA, given in Tables 15.5-13 and 15.5-14, it can be concluded that any control room exposures following a steam line break accident will be well below the GDC 19 level.

15.5.18.1 Radiological Assessment for Accident-Induced Leakage to Support Steam Generator Tube Alternate Repair Criteria

For the DBA case described above, it was assumed that the steam generator (SG) tubes will exhibit small leakage during the major steam line rupture at a rate equal to 1 gpm, which exceeds ~~was~~ the operational leakage limit in the DCPP Technical Specifications. However, this leakage rate may be exceeded during the steam line rupture if tube cracks are allowed to remain in service under alternate repair criteria (ARC). If the degraded tubes in the faulted steam generator encounter a differential pressure of sufficient magnitude that causes the cracks in the degraded tubes to expand, there is a potential for primary-to-secondary leakage to increase to a rate that is higher than that during normal operation. This additional leakage is referred to as accident-induced leakage. Implementation of SG tube ARC dictates that a higher performance criteria be established for accident induced leakage. The major steam line rupture analysis must account for accident-induced leakage in the determination of dose consequences. This section provides the updated licensing basis description and radiological consequence analysis for a major steam line rupture analysis using a higher accident-induced leak rate. The NRC approved this analysis in a letter to PG&E dated ~~March 12, 1998, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit No. 1 and Unit No. 2~~." Replace with new NRC SER

The methodology selected for performing the radiological assessment to support use of higher accident-induced leakage follows NRC Standard Review Plan (SRP) 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Revision 2, 1981. Using an accident induced leak rate of 10.5 ~~12.8~~ gpm in the faulted SG, calculations using the LOCADOSE computer program demonstrate that the offsite doses are within 10 percent of 10 CFR 100 limits and control room doses are within GDC 19 limits. As such, 10.5 ~~12.8~~ gpm (at room temperature conditions) represents the SG tube accident leakage performance criteria, in support of SG tube ARC.

The resultant doses from the MSLB event using an accident-induced leak rate of 10.5 ~~12.8~~ gpm are listed below. The limiting case is the accident initiated iodine spike as the thyroid dose at the Exclusion Area Boundary (EAB) is ~~just under~~ at the 30 rem limit. These doses supplement the doses listed in Table 15.5-36 for the DBA and expected cases.

Location	Dose (rem) Thyroid	Beta Skin	Whole Body
Case 1: Accident-Initiated Spike			
EAB (0-2 hr)	29.77	3.37E-2	8.56E-2
LPZ (30 days)	7.29	4.92E-3	1.24E-2
• Dose Limit (10% of 10 CFR 100)	30	25	2.5
Control Room (30 days)	7.49E-1	3.70E-3	2.11E-4
• Dose Limit (GDC 19)	30	5	5
Case 2: Pre-Existing Spike			
EAB (0-2 hr)	74.73	4.15E-2	9.83E-2
LPZ (30 days)	6.45	3.39E-3	7.61E-3
• Dose Limit (10 CFR 100)	300	25	25
Control Room (30 days)	7.78E-1	3.33E-3	1.80E-4
• Dose Limit (GDC 19)	30	5	5

Replace with Insert 15.5.18-1

The input parameters for the dose analysis are summarized below.

(1) The operational (pre-MSLB) primary-to-secondary leak rate was assumed to be 1 gpm to yield a conservatively high isotopic concentration in the secondary system. Use of 1 gpm is more conservative than the ~~previous~~ Technical Specifications operational leak rate limit of 150 gpd per SG.

(2) During the accident, the primary-to-secondary leak rate in the faulted steam generator is assumed at the maximum rate of ~~12.8~~ 10.5 gpm. The primary-to-secondary leak rate in each intact SG was assumed to be at the Technical Specifications operational leak rate limit of 150 gpd; therefore, the total leakage is 450 gpd.

(3) The MSLB occurred in the section of piping between the containment building and the main steam line isolation valves (MSIVs). Prior to control room isolation and pressurization, the control HVAC intake χ/Q is the unfiltered χ/Q taken from the LOCA condition outside containment.

(4) Loss of offsite power is assumed to occur coincident with MSLB accident.

(5) Conservatively, based on the Technical Specifications requirements for the safety injection signal and containment Phase A isolation, the control room will be isolated well within 35 seconds. To add more conservatism in this calculation, the control room is assumed to be isolated in 2 minutes.

(6) All releases were assumed to end after 8 hours, when the plant is placed on the residual heat removal (RHR) system.

(7) For a pre-existing iodine spike, the activity in the reactor coolant is based upon an iodine spike that has raised the reactor coolant concentration to 60 μCi/g of I-131 DEC, based on the Technical Specifications. The secondary coolant activity is 0.1 μCi/g of I-131 DEC, based on the Technical Specifications. Noble gas activity is based on 1 percent failed fuel.

(8) For an accident-initiated (concurrent) iodine spike, the accident initiates an iodine spike in the reactor coolant system (RCS) that increases the iodine release rate from the fuel to a value 500 times greater than the release rate corresponding to an RCS concentration of 1 μCi/g of I-131 DEC. The 1 μCi/g I-131 DEC is based on the Technical Specifications. The iodine activity released to the RCS for the duration of the accident is conservatively assumed to mix instantaneously and uniformly in the RCS. Noble gas activity is based on 1 percent failed fuel. Insert 15.5.18-2

Insert 15.5.18-3

(~~9~~) 10 Following the pipe rupture, auxiliary feedwater to the faulted loop is isolated and the SG is allowed to steam dry. The iodine partition factor for the faulted SG is assumed to be 1.0. Also, the iodine partition factor for the intact SG is conservatively assumed to be 1.0; i.e., no credit is taken for iodine partition.

(~~10~~) 11 All activity in the SGs is released to the atmosphere in accordance with the release rates in Table 15.5-34, with added releases from primary-to-secondary leaks in the faulted loop and intact loops.

Atmospheric steam releases (not including primary-to-secondary leaks):

Ruptured loop	162,784 lb at 45.0 lb/ft^3 (0-2 hr) 0 lb (2-8 hr)
Intact loops	393,464 lb at 45.0 lb/ft^3 (0-2hr) 860,461 lb at 50.0 lb/ft^3 (2-8 hr)

(~~11~~) 12 The source term is based on a composite source term of 3.5 percent and 4.5 percent fuel enrichment. An evaluation has been performed and concluded that the current source term bounds the 5 percent enrichment fuel up to 50,000 MWD/MTU for a 21-month operating cycle.

(~~12~~) 13 Atmospheric Dispersion Factors (sec/m^3)
(Reference Tables 15.5-3 and 15.5-6)

Time	EAB	LPZ	Control Room	
			Pressurized	Infiltration
0-2 hr	5.29E-4	2.20E-5	7.05E-5	1.96E-4
2-8 hr		2.20E-5	7.05E-5	1.96E-4
8-24 hr		4.75E-6	5.38E-5	1.49E-4
24-96 hr		1.54E-6	3.91E-5	1.08E-4
96-720 hr		3.40E-7	2.27E-5	6.29E-5

(~~13~~) 14 Control Room HVAC Flow Rates and Filtration Efficiencies:

Filtered Intake Flow	2100 cfm
Unfiltered Intake Flow	10 cfm
Exhaust Flow	2110 cfm
Filtered Recirculation Flow	2100 cfm

Charcoal Filter Iodine Removal Efficiency

Elemental	95%
Organic	95%
Particulate	95%

(~~14~~) 15 RCS and Secondary Water Volume and Water Mass

RCS water volume	94,000 gallons
RCS water mass	566,000 pounds
Water in SGs	6735.54 ft^3 at 45.0 lb/ft^3 (0-2 hr) and 50.0 lb/ft^3 (2-8 hr)
	Loop 1 1683.88 ft^3
	Loops 2, 3, 4 5051.65 ft^3
Water in Condensers	27243.59 ft^3 at 62.4 lb/ft^3
Water in SGs and Condensers	33979.13 ft^3

15.5.19 ENVIRONMENTAL CONSEQUENCES OF A MAJOR RUPTURE OF A MAIN FEEDWATER PIPE

As reported in Section 15.4.2, a major feedwater line rupture is not expected to cause cladding damage, and thus no release of fission products to the coolant is expected following this accident. If significant radioactivity exists in the secondary system prior to the accident, however, some of this activity will be released to the environment with the feedwater escaping from the pipe rupture. In addition, if an atmospheric steam dump from the unaffected steam generators is necessitated by unavailability of condenser capacity, additional activity will be released. Approximately 1.6 x 10^6 pounds of secondary coolant is the maximum release expected for a full cooldown without any condenser availability.

The radiological consequences of a 1.6 x 10^6 pounds secondary coolant release has been discussed in Section 15.5.18. It can be concluded that potential exposures from major feedwater line ruptures will be well below the guideline levels specified in 10 CFR 100, and that the occurrence of such ruptures would not result in undue risk to the public.

15.5.20 ENVIRONMENTAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE (SGTR)

The SGTR accident is discussed in Section 15.4.3, and the thermal and hydraulic analysis presented in Section 15.4.3.3 provides the basis for the evaluation of radiological consequences discussed in this section.

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.18

INSERT 15.5.18-1

Location	Dose (Rem)		
	Thyroid	Beta Skin	Whole Body
Case 1: Accident-Initiated Spike			
EAB (0-2 hr)	30.0	1.50E-1	9.40E-2
LPZ (30 days)	6.48	1.91E-2	1.18E-2
• Dose Limit (10% of 10 CFR 100)	30.0	2.5	2.5
Control Room (30 days)	6.66E-1	7.09E-3	1.49E-4
• Dose Limit (GDC 19)	30.0	5	5
Case 2: Pre-Existing Spike			
EAB (0-2 hr)	53.05	1.25E-1	7.26E-2
LPZ (30 days)	4.58	9.80E-3	5.56E-3
• Dose Limit (10 CFR 100)	300	25	25
Control Room (30 days)	5.53E-1	6.70E-3	1.27E-4
• Dose Limit (GDC 19)	30	5	5

INSERT 15.5.18-2

To maximize the accident-initiated iodine spiking, a RCS letdown rate of 143 gpm with 100% iodine removal through the filters in the demineralizers is assumed.

INSERT 15.5.18-3

(9) The thyroid dose conversion factors are based on International Commission on Radiological Protection Publication 30 (Reference 21) as documented in Federal Guidance Report (FGR) 11 and FGR 12 (References 41 and 42):

I-131	1.08E+06 (Rem/Ci)
I-132	6.44E+03 (Rem/Ci)
I-133	1.80E+05 (Rem/Ci)
I-134	1.07E+03 (Rem/Ci)
I-135	3.13E+04 (Rem/Ci)

INSERT 15.5.18-4

K.F. Eckerman et al., Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.18

INSERT 15.5.18-5

K.F. Eckerman and J.C. Ryman, External Exposure to Radionuclides in Air, Water, and Soil, Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

15.5.20.1 Offsite Exposures

The evaluation of the radiological consequences of a steam generator tube rupture event assumes that the reactor has been operating at the maximum allowable Technical Specification 1 gpm (Reference 22) limits for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the reactor coolant and in the secondary coolant. Radionuclides from the primary coolant enter the steam generator via the ruptured tube, and are released to the atmosphere through the steam generator PORVs (and safety valves) and via the condenser air ejector exhaust.

and primary to secondary leakage

The quantity of radioactivity released to the environment, due to an SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator, and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated in a manner consistent with the recommendations of Standard Review Plan, Section 15.6.3 (Reference 37).

(1) *Design Basis Analytical Assumptions*

The major assumptions and parameters used in the analysis are itemized in Table 15.5-64.

(2) *Source Term Calculations*

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

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

(i) Accident Initiated Spike - The initial primary coolant iodine concentration is 1 µCi/gm of Dose Equivalent (DE) 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 that the release rate corresponding to the initial primary system iodine concentration. The initial appearance rate can be written as follows:

335 or

$$P_i = A_i \lambda_i \quad (15.5\text{-}15)$$

where:

P_i = Equilibrium appearance rate for iodine nuclide i
A_i = equilibrium RCS inventory of iodine nuclide i corresponding to 1 µCi/gm of DE I-131
λ_i = removal coefficient for iodine nuclide i

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

(b) The initial secondary coolant iodine concentration is 0.1 μCi/gram of DE I-131.

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

(d) The initial noble gas concentrations in the reactor coolant are based upon 1 percent fuel defects. ~~The noble gas inventories, upon which the concentrations are based, were taken from Table 11.1-11.~~

(3) *Radioactivity Transport Model* The analysis conservatively took no

The iodine transport model utilized in this analysis was proposed by Postma and Tam (Reference 34). The model considers break flow flashing, droplet size, bubble scrubbing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. ~~Removal~~ credit ~~is taken~~ for scrubbing of iodine contained in the atomized coolant droplets. ~~when the rupture site is below the secondary water level.~~ The fraction of primary coolant iodine which is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate and the iodine partition coefficient. This analysis conservatively assumes an iodine partition coefficient of 100 between the steam generator liquid and steam phases. ~~when the rupture site is covered. The model takes no scrubbing or mixing credit when the rupture site is above the secondary water level.~~ Droplet removal by the dryers is conservatively assumed to be negligible. The iodine transport model is illustrated in Figure 15.5-19.

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 15.4-14.

(b) The mass of ~~time-dependent fraction of rupture~~ ~~flashed~~ break flow that flashes to steam and is immediately released to the environment is contained in Table 15.4-14 and is presented in Figure ~~15.5-20~~ 15.4.3-11. The break flow flashing fraction was conservatively calculated assuming that 100 percent of the break flow is from the hot leg side of the steam generator, whereas the break flow actually consists of flow 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~~

Insert 15.5.20-1

~~row and the upper anti-vibration bar~~. ~~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 15.5-21. As noted from Figure 15.5-21, the water level in both the ruptured and intact steam generators drops below the top of the tubes after reactor trip, but then begins to increase and recovers the top of the tubes a short time later. The iodine scrubbing efficiency is determined by the method suggested by Postma and Tam (Reference 34). The iodine scrubbing efficiencies are shown in Figure 15.5-22.~~

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

$$A_r = \Sigma_j IA_j \ (1 - eff_j) \qquad (15.5\text{-}16)$$

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

Insert 15.5.20-2

(d) During the time period that the rupture (or leakage) site is uncovered, all of the activity carried by the break (leakage) flow is assumed to be directly released to the environment, i.e., the activity carried by the break (leakage) flow will neither mix with the secondary water nor partition. The rupture site is considered to be covered when the secondary water level is approximately 12 inches over the rupture site (approximately 8 inches over the apex of the tube bundle).

(e) The total primary to secondary leak rate for the three intact steam generators is assumed to be 1.0 gpm. ~~The leak rate is assumed to be 0.70 gpm for the three intact steam generators and 0.30 gpm for the ruptured steam generator.~~ The leakage to the intact steam generators is assumed to persist for the duration of the accident.

Insert 15.5.20-4

(f) The iodine partition coefficient between the liquid and steam of the ruptured and intact steam generators is assumed to be 100 during the time that the rupture (or leakage) site is covered.

(g) 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.

(h) Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 15.5-68. The breathing rates were obtained from NRC Regulatory Guide 1.4 (Reference 35).

Insert 15.5.20-6

(4) *Offsite Dose Calculation*

Offsite thyroid doses are calculated using the equation:

$$D_{Th} = \sum_i \left[(DCF)_i \sum_j \left((IAR)_{ij}\ (BR)_j\ (\chi / Q)_j \right) \right] \qquad (15.5\text{-}17)$$

where:

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

$(BR)_j$ = breathing rate during time interval j in meter^3 second (Table 15.5-68)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j in seconds/meter^3 (Table 15.5-68)

$(DCF)_i$ = thyroid dose conversion factor via inhalation for iodine nuclide i in rem/Ci (Table 15.5-69)

D_{Th} = thyroid dose via inhalation in rem

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

$$D_\gamma = 0.25 \sum_i \left[\bar{E}_\gamma i \sum_j \left((IAR)_{ij}\ (\chi / Q)_j \right) \right] \qquad (15.5\text{-}18)$$

where:

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

$(\chi/Q)_j$ = 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 15.5-70)

D_γ = whole body gamma dose due to immersion in rem

Offsite beta-skin doses are calculated using the equation:

$$D_\beta = 0.23 \sum_i \left[\bar{E}_\beta i \sum_j \left((IAR)_{ij}\ (\chi / Q)_j \right) \right] \qquad (15.5\text{-}19)$$

[a] 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[a]
$(\chi/Q)_j$	=	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 15.5-70)
$D\beta$	=	beta-skin dose due to immersion in rem

(5) *Offsite Dose Results*

Thyroid and whole-body gamma ~~and beta-skin~~ doses at the Exclusion Area Boundary and the outer boundary of the Low Population Zone are presented in Table 15.5-71. ~~All doses are within the allowable guidelines as specified by the Standard Review Plan, Section 15.6.3, and 10 CFR 100.~~ Insert 15.5.20-8

15.5.20.2 Control Room Exposures

Additional analyses were performed to determine the airborne doses to the control room operators from an SGTR. These calculations used the atmospheric releases of radioactivity determined in the analysis discussed in Section 15.5.20.1 and Reference 38. Table 15.5-72 provides the atmospheric releases of iodine activity for accident-initiated and pre-accident iodine spike cases, and Table 15.5-73 provides the atmospheric releases of noble gas activity from an SGTR. Insert 15.5.20-11

The dose calculations used atmospheric dispersion factors for the 0-8 hour period from Table 15.5-6. These χ/Q values, discussed in Section 15.5.4, are conservative for the release locations in a SGTR. Breathing rates for the 0-8 hour period were taking from the onsite design basis case in Table 15.5-7, and control room HVAC parameters were taken from Table 15.5-32 based on the analysis in Section 15.5.17.10.

Table 15.5-74 presents the resulting airborne doses to the control room operators ~~for the duration of the postulated SGTR~~. The resultant doses are well below the guidelines of GDC 19, and are below the corresponding post-LOCA control room exposures presented in Table 15.5-33.

15.5.21 ENVIRONMENTAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

As reported in Section 15.4.4, under adverse circumstances, a locked rotor accident could cause small amounts of fuel cladding failure in the core. If this occurs, some fission products will enter the coolant and will mostly remain in the coolant until cleaned up by the primary coolant demineralizers, or in the case of noble gases, until stripped from the coolant. Following such an incident, there are several possible modes of release of some of this activity to the environment.

 No Changes

The analyses of potential accidents and abnormal operating conditions have been performed using models and assumptions specified in federal regulations and regulatory guides. Conservative methods and assumptions were employed where models or assumptions were not specified by these guidelines, or where specific characteristics of the DCPP units were considered more applicable.

In all accident analyses, the resulting potential radiological exposures to plant personnel, to individual members of the public, and to the general population have been found to be lower than the applicable guidelines and limits specified in 10 CFR 20, 10 CFR 50, and 10 CFR 100. The calculated exposures are summarized in Tables 15.5-58, 15.5-59, and 15.5-60. The results of the accident analyses indicate that the conservation, redundancy, and flexibility incorporated into the plant safety features ensures that these units can be operated without undue risk to the health and safety of the public.

15.5.28 REFERENCES

1. Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plant, N18.2, American Nuclear Society, 1972.

2. Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, US Atomic Energy Commission (AEC), Rev. 1, October 1972.

3. Regulatory Guide 4.2, Preparation of Environmental Reports for Nuclear Power Plants, Directorate of Regulatory Standards, AEC, March 1973.

4. W. K. Burnot, et al, EMERALD (REVISION I) - A Program for the Calculation of Activity Releases and Potential Doses, Pacific Gas and Electric Company, March 1974.

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6. Regulatory Guide Number 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors, AEC, Rev. 1, June 1973.

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14. F. J. Brutschy, et al, Behavior of Iodine in Reactor Water During Plant Shutdown and Startup, General Electric Co. Atomic Power Equipment Department Report, NEDO-10585, August 1972.

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Insert 15.5.20-5

22. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR 82, as amended.

23. Regulatory Guide 1.25, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25) USNRC, March 1972.

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33. ~~Deleted in Revision 12~~. Insert 15.5.20-3

34. A. K. Postma and P. S. Tam, Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident, NUREG-0409, USNRC, January 1978.

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36. M. J. Bell, "ORIGEN - The ORNL Isotope Generation and Depletion Code," ORNL-4628, Oak Ridge National Laboratory, May 1973.

Insert 15.5.20-7

37. Standard Review Plan, Section 15.6.3, Radiological Consequences of Steam Generator Tube Failure (PWR), NUREG-0800, USNRC, July 1981.

38. D. F. Holderbaum, R. N. Lewis, T. A. Miller and K. Rubin, LOFTTR2 Analysis For A Steam Generator Tube Rupture For the Diablo Canyon Power Plant Units 1 and 2, WCAP-11723 (Proprietary)/WCAP-11724 (Nonproprietary), February 1988.

Insert 15.5.20-10

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40. Westinghouse Letter PGE-93-652 dated October 5, 1993, transmitting NSAL-93-016, Revision 1.

41. Deleted in Revision 14 ~~PG&E Calculation N-166 Small Break LOCA Doses, Revision 0, October 31, 1994~~

Insert 15.5.18-4

42. Deleted in Revision 14 ~~PG&E Calculation N-167 Post LOCA Doses, Revision 0, November 4, 1994.~~

Insert 15.5.18-5

43. Deleted in Revision 12.

Insert 15.5.20-9

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.20

INSERT 15.5.20-1

No iodine scrubbing is credited in the analysis and the iodine scrubbing efficiency is assumed to be 0 percent. Thus the location of the tube rupture is not significant for the radiological consequences. However, as discussed in Section 15.4.3.3, in the thermal and hydraulic analysis the tube rupture break flow is calculated conservatively assuming that the break is at the top of the tube sheet.

INSERT 15.5.20-2

The rupture (or leakage) site is assumed to be always covered with secondary water based on Reference 33 which concluded the effect of tube uncovery is essentially negligible for the radiological consequences for the limiting SGTR transient.

INSERT 15.5.20-3

Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue, WCAP-13247, March 1992

INSERT 15.5.20-4

The iodine partition coefficient between the liquid and steam of the ruptured steam generator is assumed to be 100 for non-flashed flow and 1 for flashed flow. The iodine partition coefficient between the liquid and steam of the intact steam generators is assumed to be 100.

INSERT 15.5.20-5

International Commission on Radiological Protection Publication 30, Limits for Intakes of Radionuclides by Workers, 1979.

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.20

INSERT 15.5.20-6

(i) The noble gases in the break flow and primary to secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere. The whole body gamma doses are calculated combining the dose from the released noble gases with the dose from the iodine releases.

(j) For the accident initiated iodine spike case, an iodine spiking factor of 335 obtained from Regulatory Guide 1.183 (Reference 43) is assumed for the exclusion area boundary thyroid dose calculation. For conservatism, an iodine spiking factor of 500 obtained from Standard Review Plan, Section 15.6.3 (Reference 37) is assumed for the accident initiated iodine spike low population zone thyroid dose calculation.

INSERT 15.5.20-7

T. R. England and R. E. Schenter, ENDF-223, "ENDF/B-IV Fission Product Files: Summary of Major Nuclide Data," October 1975.

INSERT 15.5.20-8

All doses are within the limits of 10 CFR 100. All doses are within the allowable guidelines as specified by the Standard Review Plan, Section 15.6.3, except the accident initiated iodine spike exclusion area boundary (0-2 hour) thyroid dose of 30.5 Rem which is 1.5% above the allowable guideline value. The accident initiated iodine spike exclusion area boundary (0-2 hour) thyroid dose is equivalent to the Regulatory Guide 1.183 (Reference 43) methodology total effective dose equivalent (TEDE) of 1.25 Rem and is well below the Regulatory Guide 1.183 accident initiated iodine spike case TEDE limit of 2.5 Rem. The accident initiated iodine spike exclusion area boundary (0-2 hour) thyroid dose of 30.5 Rem is within the 10 CFR 100 dose limit of 300 Rem for the first 2 hours at the exclusion area boundary. Therefore, the accident initiated iodine spike exclusion area boundary (0-2 hour) thyroid dose of 30.5 Rem is acceptable.

INSERT 15.5.20-9

Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, US NRC, July 2000.

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.20

INSERT 15.5.20-10

Steam Generator Tube Rupture (SGTR) Re-analysis Report, Letter from W. R. Rice (Westinghouse) to M. Mayer (PG&E), PGE-01-535, October 26, 2001.

INSERT 15.5.20-11

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake contained in Table 15.5-68 are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 15.5-72. The control room occupancy factors assumed were taken from Table 15.5-32.

Thyroid, whole body gamma, and beta skin doses are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation is continued to account for additional doses due to continued occupancy.

The total primary to secondary leak rate is assumed to be 1.0 gpm. The leakage to the intact steam generators is assumed to persist for the duration of the accident.

The calculations determine the thyroid doses based on a pre-accident iodine spike and based on an accident initiated iodine spike with a spiking factor of 500. Both spike assumptions consider 0.1 μCi/gm D.E. I-131 secondary activity. The whole body doses are calculated combining the dose from the released noble gases with the dose from the iodine releases. The whole body doses are calculated with the limiting iodine releases (either pre-accident spike or accident-initiated iodine spike).

Control room thyroid doses are calculated using the following equation:

$$D_{Th} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right] \qquad (15.5\text{-}19)$$

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.20

where:

D_{Th} = thyroid dose via inhalation (Rem)

DCF_i = thyroid dose conversion factor via inhalation for isotope i (Rem/Ci) (Table 15.5-69)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 15.5-68)

Control room whole body doses are calculated using the following equation:

$$D_{WB} = 0.25 * \left(\frac{1}{GF}\right) * \sum_i E_{\gamma i}\left(\sum_j Conc_{ij}\right) \qquad (15.5\text{-}20)$$

where:

D_{WB} = whole body dose via cloud immersion (Rem)

GF = geometry factor, calculated based on Reference 17, using the equation $GF = \frac{1173}{V^{0.338}}$ where V is the control room volume in ft^3

$E_{\gamma i}$ = average gamma disintegration energy for isotope i (Mev/dis) (Table 15.5-70)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

Control room skin doses are calculated using the following equation:

$$D_\beta = 0.23 * \sum_i E_{\beta i}\left(\sum_j Conc_{ij}\right) \qquad (15.5\text{-}21)$$

where

D_β = whole body dose via cloud immersion (Rem)

$E_{\beta i}$ = average beta disintegration energy for isotope i (Mev/dis) (Table 15.5-70)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)

FSAR UPDATE MARKUP INSERTS FOR FSAR SECTION 15.5.20

INSERT 15.5.20-12

CONTROL ROOM EXPOSURE

Time (hours)	Control Room Filtered Pressurization χ/Q (Sec/m^3)	Control Room Unfiltered Pressurization Zone χ/Q (Sec/m^3)	Control Room Breathing Rate[b] (m^3/Sec)
0-8	7.05×10^{-5}	1.96×10^{-4}	3.47×10^{-4}
8-24	5.38×10^{-5}	1.49×10^{-4}	3.47×10^{-4}
24-96	3.91×10^{-5}	1.08×10^{-4}	3.47×10^{-4}
>96	2.27×10^{-5}	6.29×10^{-5}	3.47×10^{-4}

(b) Regulatory Guide 1.4, Revision 2, June 1974 (Reference 35)

Control Room Parameters Used in Evaluating Radiological Consequences for SGTR Analysis

TABLE 15.5-64

PARAMETERS USED IN EVALUATING
RADIOLOGICAL CONSEQUENCES FOR SGTR ANALYSIS

I. Source Data

A. Core power level, MWt	~~3568~~ 3580
B. Total steam generator tube leakage, prior to accident, gpm	1.0
C. Reactor coolant activity:	
1. Accident initiated spike	The initial RC iodine activities based on 1 µCi/gram of D.E. I-131 are presented in Table 15.5-65. The iodine appearance rates based on an iodine spiking factor of 500 assumed for the accident initiated spike are presented in Table 15.5-66
2. Preaccident spike	Primary coolant iodine activities based on 60 µCi/gram of D.E. I-131 are presented in Table 15.5-65
3. Noble gas activity	The initial RC noble gas activities based on 1% fuel defects are presented in Table 15.5-67
D. Secondary system initial activity	Dose equivalent of 0.1 µCi/gm of I-131, presented in Table 15.5-65
E. Reactor coolant mass, grams	2.16 → ~~2.57~~ x 10^8
F. Initial steam generator mass (each), grams	4.3 x 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 date

A. Ruptured steam generator

1. Rupture flow	See Figure 15.4.3-6 and Table 15.4-14
2. ~~R~~rupture flow ~~flashing fraction~~ Flashed	See Figure ~~15.5-20~~ 15.4.3-11 and Table 15.4-14
3. Iodine scrubbing efficiency	~~See Figure 15.5-22~~ Not Modeled
4. Total steam release, lbs	See Figure 15.4.3-9 and Table 15.4-14
5. Iodine partition coefficient - non-flashed	100 ~~when rupture site is covered~~
- flashed	1.0
~~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~~ 1.0
2. Total steam release, lbs	See Figure 15.4.3-10 and Table 15.4-14
3. Iodine partition coefficient	100 ~~when leakage site is covered~~

C. Condenser

1. Iodine partition coefficient	100

D. Atmospheric dispersion factors	See Table 15.5-68

TABLE 15.5-65

IODINE SPECIFIC ACTIVITIES
IN THE PRIMARY AND SECONDARY COOLANT[a] - SGTR ANALYSIS

	Specific Activity (μCi/gm)		
	Primary Coolant		Secondary Coolant
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
I-131	~~0.69~~ 0.793	~~41.4~~ 47.58	~~0.069~~ 0.0793
I-132	~~0.25~~ 0.204	~~15.0~~ 12.24	~~0.025~~ 0.0204
I-133	~~0.94~~ 1.113	~~56.4~~ 66.78	~~0.094~~ 0.1113
I-134	~~0.13~~ 0.139	~~7.8~~ 8.34	~~0.013~~ 0.0139
I-135	~~0.52~~ 0.589	~~31.2~~ 35.34	~~0.052~~ 0.0589

(a) Based on 1, 60 and .01 μCi/gm of Dose Equivalent I-131 consistent with the DCPP Technical Specifications (Reference 22).

TABLE 15.5-66

IODINE SPIKE APPEARANCE RATES(a) - SGTR ANALYSIS
(CURIES/SECOND)

I-131	I-132	I-133	I-134	I-135
~~1.54~~ 3.69	~~3.21~~ 2.97	~~3.10~~ 6.26	~~3.82~~ 4.27	~~3.04~~ 4.71

(a) The accident initiated spike appearance rate is 500 times the equilibrium appearance rate. The equilibrium appearance rate is calculated based on a letdown flow of 120 gpm with perfect cleanup, a letdown flow uncertainty of 12 gpm, 10 gpm identified reactor coolant system leakage, and 1 gpm unidentified leakage from the reactor coolant system.

TABLE 15.5-67

(a) NOBLE GAS SPECIFIC ACTIVITIES IN THE REACTOR COOLANT BASED ON 1% FUEL DEFECTS - SGTR ANALYSIS

Nuclide	Specific Activity (μCi/gm)
Xe-131m	2.523
Xe-133m	~~3.11~~ 3.911
Xe-133	~~269.5~~ 256.3
Xe-135m	~~0.38~~ 0.449
Xe-135	~~5.42~~ 8.663
Xe-138	~~0.52~~ 0.568
Kr-85m	~~1.8~~ 2.141
Kr-85	~~5.77~~ 6.209
Kr-87	~~0.98~~ 1.232
Kr-88	~~3.07~~ 3.907

(a) Based on a two year fuel cycle at a core power of 3580 MWt, a 75 gpm reactor coolant system letdown flow rate, and a 90% demineralizer iodine removal efficiency.

TABLE 15.5-68

ATMOSPHERIC DISPERSION FACTORS AND BREATHING RATES - SGTR ANALYSIS

OFFSITE EXPOSURE

Time (hours)	Exclusion Area Boundary χ/Q (Sec/m^3)	Low Population Zone χ/Q (Sec/m^3)	Breathing Rate[a] (m^3/Sec)
0-2	5.29×10^{-4}	2.2×10^{-5}	3.47×10^{-4}
2-8	-	2.2×10^{-5}	3.47×10^{-4}

(a) Regulatory Guide 1.4, Revision 2, June 1974 (Reference 35)

Insert 5.5.20-12

TABLE 15.5-69

THYROID DOSE CONVERSION FACTORS[a] - SGTR ANALYSIS
(Rem/Curie)

Nuclide	
I-131	~~1.49~~ 1.07 x 10^6
I-132	~~1.43 x 10^4~~ 6.29 x 10^3
I-133	~~2.69~~ 1.81 x 10^5
I-134	~~3.73~~ 1.07 x 10^3
I-135	~~5.60~~ 3.14 x 10^4

(a) ~~Regulatory Guide 1.109, October 1977~~ (Reference 21)

International Commission on Radiological Protection Publication 30, 1979.

TABLE 15.5-70

AVERAGE GAMMA AND BETA ENERGY FOR NOBLE GASES[(a)] - SGTR ANALYSIS
(MeV/dis)

Nuclide	$\bar{E}_\gamma$	$\bar{E}_\beta$
I-131	0.38	0.19
I-132	2.2	0.52
I-133	0.6	0.42
I-134	2.6	0.69
I-135	1.4	0.43
Xe-131m	0.0029	0.16
Xe-133m	0.02	0.21~~2~~
Xe-133	0.03	0.15~~3~~
Xe-135m	0.43	0.099
Xe-135	0.25 ~~0.246~~	0.32~~5~~
Xe-138	1.2	0.66
Kr-85m	0.16 ~~0.156~~	0.25~~3~~
Kr-85	0.0023	0.25~~1~~
Kr-87	0.79~~3~~	1.3~~3~~
Kr-88	2.2~~1~~	0.25 ~~0.248~~

(a) ENDF-223, October 1975 ~~ORNL-4628, May 1975~~ (Reference 36)

TABLE 15.5-71

OFFSITE RADIATION DOSES FROM SGTR ACCIDENT

	Dose (Rem)	
	Calculated Value	Allowable Guideline Value (Reference 37)
1. Accident Initiated Iodine Spike		
Exclusion Area Boundary (0-2 hr.) Thyroid Dose	30.5[a] ~~28.8~~	30
Low Population Zone (0-8 hr.) Thyroid Dose	2.1 ~~0.3~~	30
2. Pre-Accident Iodine Spike		
Exclusion Area Boundary (0-2 hr.) Thyroid Dose	74 ~~192.4~~	300
Low Population Zone (0-8 hr.) Thyroid Dose	3.2 ~~8.0~~	300
3. Whole-Body Gamma ~~and Beta-Skin~~ Dose		
Exclusion Area Boundary (0-2 hr.) Whole Body Gamma Dose	0.39 ~~0.23~~	2.5 ~~(a)~~
~~Beta-Skin Dose~~	~~0.58~~	~~2.5(a)~~
Low Population Zone (0-8 hr.) Whole-Body Gamma Dose	0.02 ~~0.01~~	2.5 ~~(a)~~
~~Beta-Skin Dose~~	~~0.02~~	~~2.5(a)~~

~~(a) Assumed to apply to the sum of the whole-body gamma and beta-skin doses.~~

(a) This value is equivalent to the Regulatory Guide 1.183 methodology total effective dose equivalent of 1.25 Rem and is well below the Regulatory Guide 1.183 accident initiated iodine spike case total effective dose equivalent limit of 2.5 Rem.

TABLE 15.5-72

ATMOSPHERIC RELEASES OF IODINE ACTIVITY FROM SGTR ACCIDENT

Replace with revised Table 15.5-72

Accident Initiated Iodine Spike		
Nuclide	Curies Released 0-2 hours	Curies Released 2-8 hours
I-131	7.3×10^{1}	1.2×10^{0}
I-132	1.3×10^{2}	8.0×10^{-1}
I-133	1.4×10^{2}	2.1×10^{0}
I-134	1.4×10^{2}	2.0×10^{-1}
I-135	1.3×10^{2}	1.5×10^{0}

Pre-Accident Iodine Spike		
Nuclide	Curies Released 0-2 hours	Curies Released 2-8 hours
I-131	5.5×10^{2}	2.3×10^{0}
I-132	1.9×10^{2}	2.0×10^{-1}
I-133	7.4×10^{2}	2.7×10^{0}
I-134	8.6×10^{1}	2.0×10^{-2}
I-135	4.1×10^{2}	1.1×10^{-1}

TABLE 15.5-72

CONTROL ROOM PARAMETERS USED IN EVALUATING RADIOLOGICAL CONSEQUENCES FOR SGTR ANALYSIS

Control Room Isolation Signal Generated	Time of Safety Injection Signal
Delay in Control Room Isolation After Isolation Signal is Generated	35 Seconds
Control Room Volume	170,000 ft^3
Control Room Unfiltered In-Leakage	10 cfm
Control Room Unfiltered Inflow	
Normal Mode	4200 cfm
Emergency Mode	0 cfm
Control Room Filtered Inflow	
Normal Mode	0 cfm
Emergency Mode	2100 cfm
Control Room Filtered Recirculation	
Normal Mode	0 cfm
Emergency Mode	2100 cfm
Control Room Filter Efficiency	95%

Remove Table

TABLE 15.5-73

ATMOSPHERIC RELEASES OF NOBLE GAS ACTIVITY FROM SGTR ACCIDENT

Nuclide	Curies Released 0-2 hours	Curies Released 2-8 hours
Kr-85m	1.7×10^{2}	3.0×10^{-1}
Kr-85	6.0×10^{2}	2.5×10^{0}
Kr-87	7.8×10^{1}	--
Kr-88	2.8×10^{2}	4.0×10^{-1}
Xe-133m	3.2×10^{2}	1.2×10^{0}
Xe-133	2.8×10^{4}	1.1×10^{2}
Xe-135m	1.4×10^{1}	--
Xe-135	5.4×10^{2}	1.6×10^{0}
Xe-138	2.1×10^{1}	--

TABLE 15.5-74

CONTROL ROOM RADIATION DOSES FROM AIRBORNE ACTIVITY IN SGTR ACCIDENT

	Accident Initiated Iodine Spike, rem	Pre-Accident Iodine Spike, rem	GDC 19 Guideline, rem
Thyroid (0-30 days)	1.4 ~~0.24~~	2.3 ~~1.59~~	30
Whole Body (0-30 days)	2.7×10^{-4} 0.029	2.7×10^{-4} 0.030	5
Beta Skin (0-30 days)	0.02 ~~0.027~~	0.02 ~~0.027~~	30

FIGURES (Continued)

Remove Figure

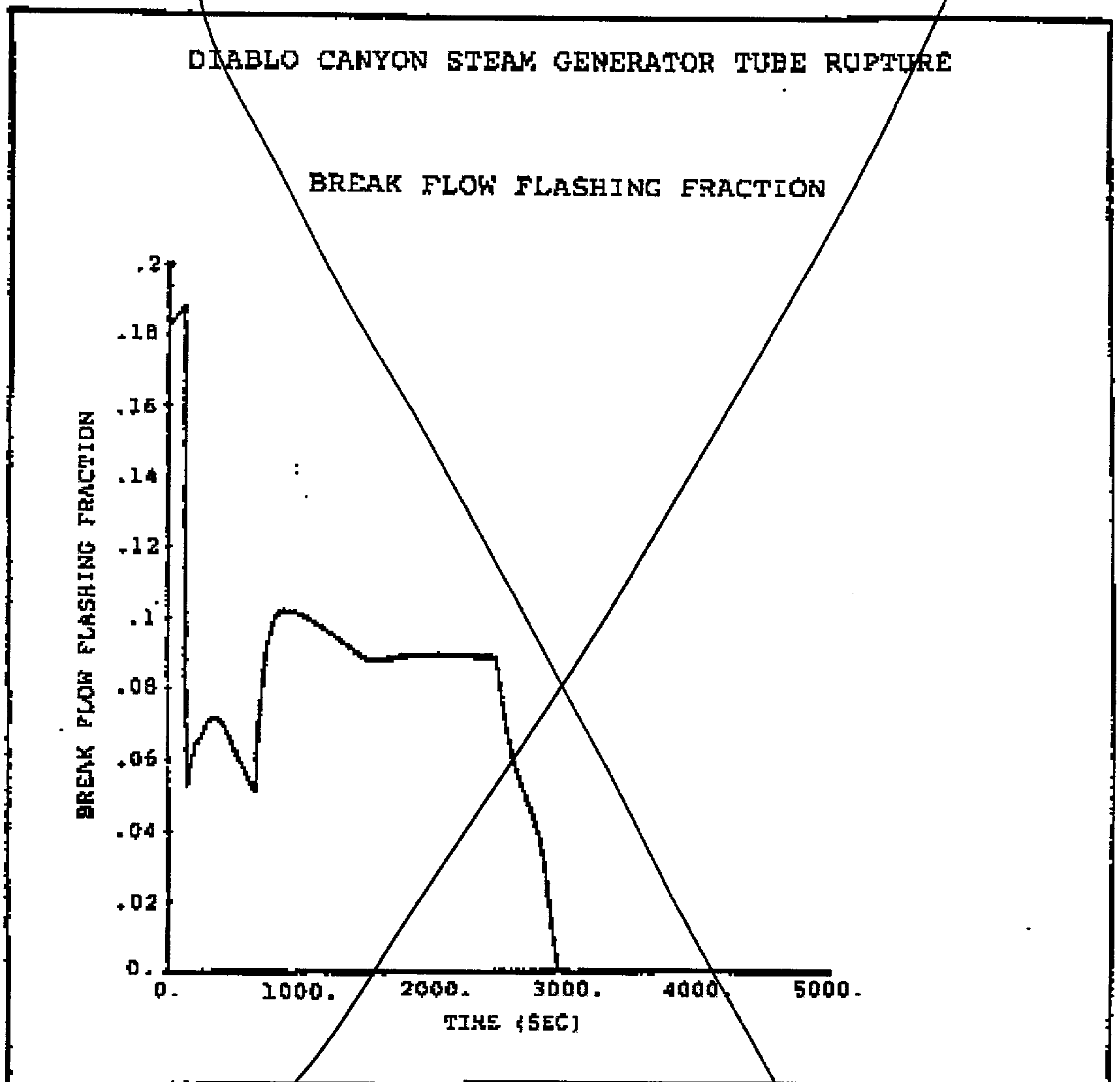

Figure 15.5-20 Break Flow Flashing Fraction - SGTR Analysis

Remove Figure

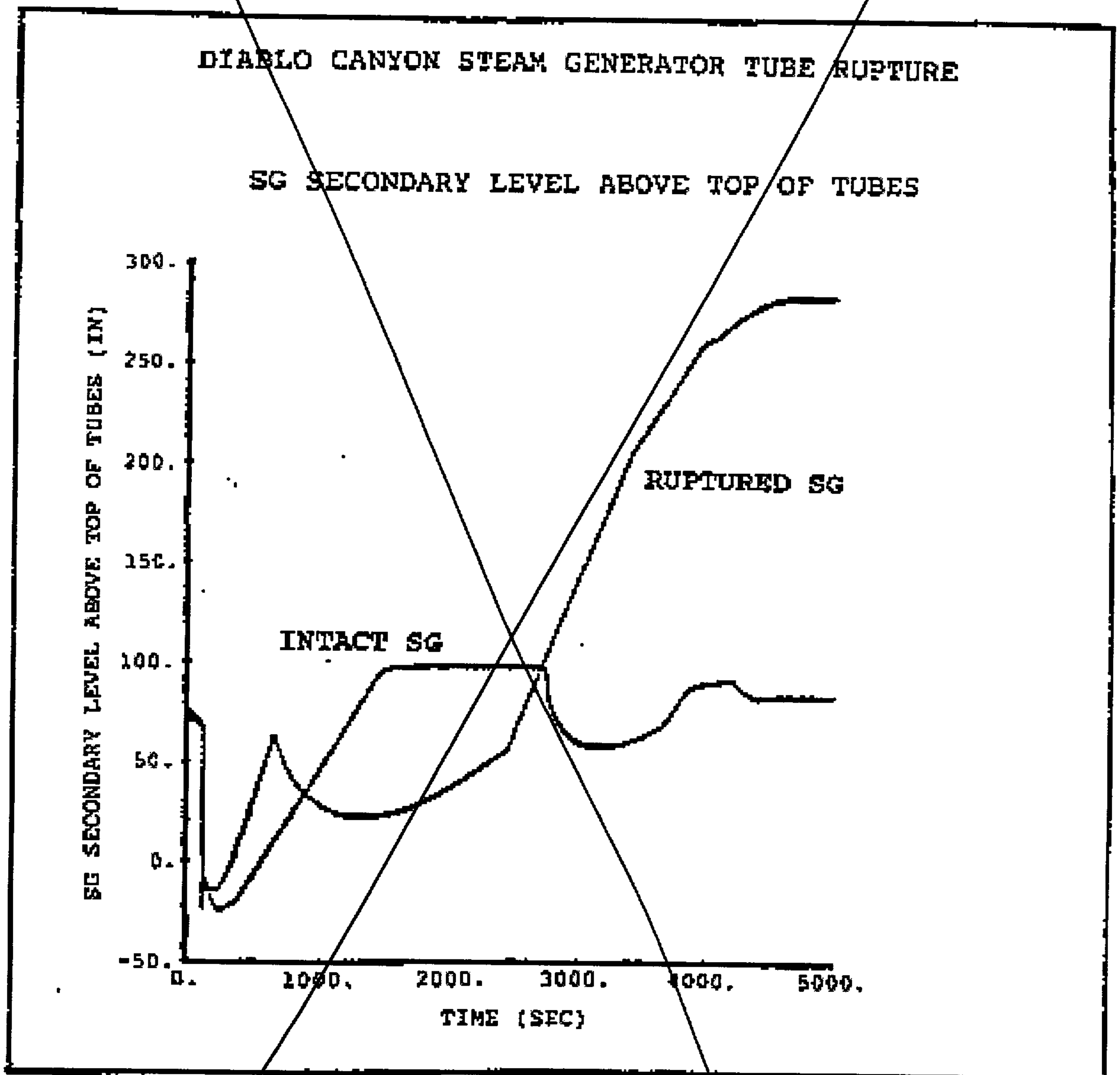

Figure 15.5-21 SG Water Level Above Top of Tubes - SGTR Analysis

Remove Figure

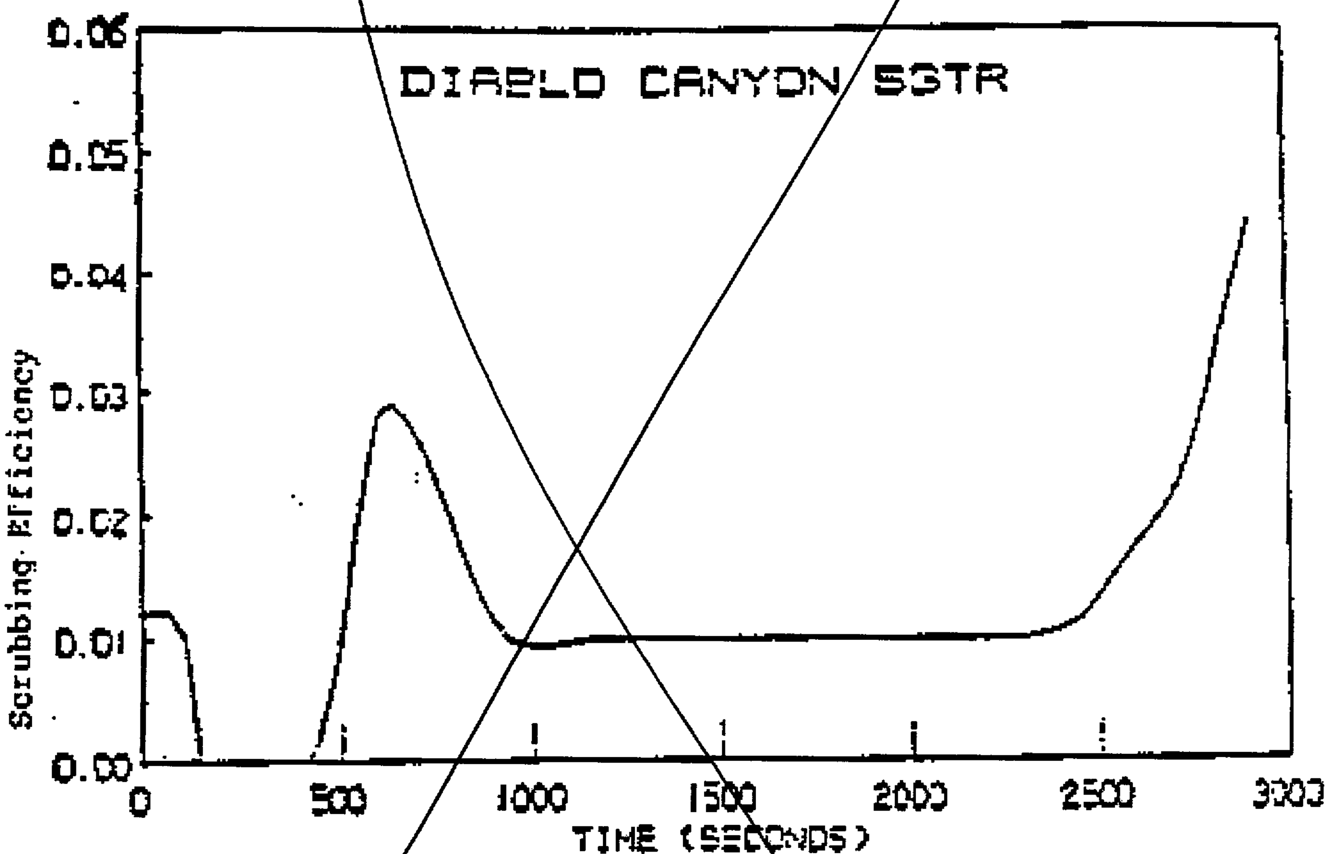

Figure 15.5-22 Iodine Scrubbing Efficiency - SGTR Analysis