ATTACHMENT 5a

Braidwood and Byron Stations

Steam Generator Tube Rupture

Analysis Report

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I INTRODUCTION

The analysis of a design basis steam generator tube rupture (SGTR) event has been performed for Byron and Braidwood Units 1 and 2 to demonstrate that the potential consequences are acceptable. In order to accommodate differences between the plant configurations and intended operating conditions at the different units, one set of calculations was performed for Byron Unit 1 and Braidwood Unit 1 and a second set of calculations was performed for Byron Unit 2 and Braidwood Unit 2. In this report, these are referred to as the Unit 1 and Unit 2 analyses.

The analyses support operation at a core power up to 3658.3 MWt. Transient modeling and input were selected such that the analyses, performed modeling a core power of 3586.6 MWt with a 2% power uncertainty, bound operation with zero uncertainty and core power of 3658.3 MWt. The analyses for Unit 1 support a full power average temperature (T_{avg}) operating range from 588.0°F to 580.0°F as well as a main feedwater temperature range from 449.2°F to 433°F, with up to 5% of the steam generator (SG) tubes plugged. The Unit 2 analyses support a full power T_{avg} operating range from 588.0°F to 575.0°F, with a T_{avg} coastdown to 573.5°F, as well as a main feedwater temperature range from 449.2°F to 435°F, with up to 10% of the SG tubes plugged. The analyses conservatively bound a reactor coolant system (RCS) loop-to-loop flow asymmetry of 7%.

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive primary coolant to the secondary side of the ruptured SG and subsequent release of radioactivity to the atmosphere. Therefore, analyses must be performed to assure that the radiological consequences resulting from an SGTR are within the allowable guidelines. One of the major concerns for an SGTR is the possibility of SG overfill since this could potentially result in a significant increase in the radiological consequences. Therefore, analyses were performed to demonstrate margin to SG overfill, assuming the limiting single failure relative to overfill. The analyses confirmed that SG overfill does not occur. Thermal and hydraulic analyses were also performed to determine the input for use in calculating the radiological consequences, assuming the limiting single failure relative to radiological consequences without SG overfill.

The SGTR transient analyses were performed using the LOFTTR2 computer program following the methodology developed in WCAP-10698-P-A and its Supplement 1 (References 1 and 2). Modifications were made consistent with WCAP-16948-P (Reference 4) to address NSAL-07-11 (Reference 3) which identified a potential non-conservative assumption regarding the direction of conservatism for decay heat in the WCAP-10698-P-A (Reference 1) methodology for evaluating margin to overfill.

The plant response to the event was modeled using the LOFTTR2 computer code with conservative assumptions of break size and location, condenser availability and initial secondary water mass. The analyses include the simulation of the operator actions for recovery from an SGTR based on the Byron and Braidwood unit-specific Emergency Operating Procedures (EOPs), which are based on the Westinghouse Owners Group Emergency Response Guidelines.

The LOFTTR2 analyses were performed for the time period from the SGTR until the primary and secondary pressures equalized (break flow termination). In the margin to overfill analyses presented in Section II, the water volume in the secondary side of the ruptured SG was calculated as a function of time to demonstrate that overfill did not occur. The thermal and hydraulic analyses to develop input to the radiological consequences analyses are presented in Section III. In these analyses, the primary to secondary break flow and the steam releases to the atmosphere from the ruptured and intact SGs were calculated for use in determining the activity released to the atmosphere. The mass releases were calculated with the LOFTTR2 computer code from the initiation of the event until break flow termination. For the time period from break flow termination until all releases are terminated, steam releases from the intact and ruptured SGs were determined from a mass and energy balance. The mass transfer information was used to calculate the radiological consequences at the exclusion area boundary and low population zone and to the operators in the control room. The radiological consequences analyses are presented in Section IV.

I.1 Plant Specific Input Required when Referencing WCAP-10698

Section D of Enclosure 1 of the Staff's SER required additional plant specific input for each utility referencing WCAP-10698. The five items are addressed in the following sections.

I.1.A Operator Response Time

The first item requests a confirmation that simulator and training programs are in place which provide the assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment, and to demonstrate that the operator action times assumed in the analysis are realistic.

Steam Generator Tube Rupture (SGTR) is one of the many abnormal and accident scenarios that are covered during the licensed operator initial and requalification program. Currently, this program is conducted on the simulators located at Training Centers on site at Byron and Braidwood Stations.

Demonstration runs have been performed at Braidwood station for the margin to overfill scenario. The results are presented in Table I-1. The time to close the MSIV does not impact the margin to overfill analysis so long as it has been completed before the cooldown is initiated and does not impact the cooldown initiation time at 21 minutes after event initiation. As demonstrated in Table I-1, the operator action times assumed in the analysis are conservative relative to actual operator performance. Since the plant configuration, emergency operating procedures and training programs for Byron and Braidwood stations are similar, comparable operator response times are expected for Byron Station. As stated above, SGTR scenario is part of the licensed operator initial and requalification program at Byron and Braidwood stations, demonstration runs will continue to be conducted in accordance with the requirements of the program.

I.1.B Site Specific Offsite Dose Calculation

In order to accommodate differences between the plant configurations and intended operating conditions at the different units, one set of calculations was performed for Byron Unit 1 and Braidwood Unit 1 and a second set of calculations was performed for Byron Unit 2 and Braidwood Unit 2. The thermal and hydraulic analyses performed to determine the input for the radiological consequences analyses for a design basis SGTR event are discussed in Section III. The evaluation of the radiological consequences of an SGTR is discussed in Section IV.

I.1.C Structural Analysis of Main Steam Lines

An evaluation to determine impact on the Byron and Braidwood Unit 1 and 2 main steam lines due to the event of filling the main steam lines with water as a result of a SGTR was performed. The main steam lines and associated supports are structurally adequate in the event of water-filled condition as a result of SGTR.

I.1.D Steam Generator Tube Rupture Equipment List

An evaluation to determine systems, components and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s) for Byron and Braidwood Stations was performed. The SGTR margin to overfill analysis ends when the Reactor Coolant System (RCS) pressure is reduced to where it matches the ruptured SG pressure; terminating the break flow into the secondary side of the SG. This occurs after operator action is taken to terminate Safety Injection (SI) flow. For this evaluation the equipment used in the EOPs through SI termination was considered as being credited for accident mitigation.

The NRC acceptance letter of WCAP-10698-P-A (Reference 18) requests in part from each utility referencing WCAP-10698 to provide a list of systems, components and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s) and to include the following:

1. Whether each system and component specified is safety grade.
2. For primary and secondary PORVs and control valves specify the valve motive power and state whether the motive power and valve controls are safety grade.
3. For non-safety grade systems and components state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that non-safety grade components can be utilized for the design basis event.
4. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible.
5. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

Table I-2 identifies the systems, components and instrumentation which are credited for accident mitigation and provides the above information. While steam generator sampling is one of the methods used for identification of a ruptured SG, it is not the only means of identification. As discussed in Section II.2.D, operator actions are not dependent on awaiting sample results.

I.1.E Single Failure Determination

The following single failures were analyzed by Westinghouse within WCAP 10698-P-A for the generic reference plant. An evaluation of the same failures as applicable to Braidwood and Byron Station was performed:

1. Auxiliary Feedwater (AF) Flow Control Valve
2. Steam Generator Power Operated Relief Valve
3. Main Steam Isolation Valve (MSIV) for Ruptured Steam Generator
4. Steam Generator Safety Valve
5. Isolation Valve for Steam Supply Line to Turbine Driven AF Pump
6. Feedwater Flow Control Valve (FWCV) Failure
7. Safety Injection (SI) Reset Device
8. Emergency Diesel Generators
9. Pressurizer PORVs and Block Valves
10. SI Pump Switches
11. Boron Injection Tank (BIT) Isolation Valves

For item 4, the steam generator safety valves are designed to ASME Code Section III standards for steam relief. These valves are not assumed to fail. Item 5 does not apply because the design of Braidwood and Byron station does not include a turbine driven AF pump. For item 11, Braidwood and Byron do not have BIT isolation valves. The equivalent isolation valves to complete SI termination and direct reactor coolant makeup flow through the normal charging line are the High Head Safety Injection valves, 1/2SI8801A/B.

With procedure guidance and the planned modifications discussed in Section II.2.F, items 1, 7, 8, 9, 10, and 11 are determined to be non-limiting.

Sensitivity studies are performed for the following failures to determine the limiting single failure (see Section II.2.E):

- Failure of one intact SG PORV
- Failure of ruptured SG MSIV
- Failure of ruptured SG FWCV

The failure of one intact SG PORV was determined to be the limiting single failure.

Table I-1: Observed Operator Response Time Summary

Data Set	Time to isolate AFW flow to the ruptured SG from start of event	Time to close the ruptured SG MSIV from start of event	Time to initiate cooldown from ruptured SG isolation	Time to initiate depressurization from end of cooldown	Time to terminate SI from end of depressurization
	(min:sec)				
Analysis Input*	9	18	3**	4	3
Crew 1a	4:57	9:12	4:02	1:34	1:29
Crew 4a	6:14	10:11	2:26	1:36	2:07
Crew 4b	5:18	9:40	3:01	2:05	2:03
Crew S	6:10	10:47	3:31	1:28	1:49
Average	5:26	9:33	3:11	1:28	1:32

* Observed operator times are from actual observed crews on the Braidwood simulator. Since these actions are control room based similar times would be expected at Byron. The four crews on the Table represent the most limiting times as follows:

Crew 1a: Longest time to initiate cooldown from ruptured SG isolation.

Crew 4a: Longest time to isolate AFW and terminate SI.

Crew 4b: Longest time to initiate depressurization from end of cooldown.

Crew S: Longest time to initiate cooldown from event initiation.

In addition to the above 4 observed crew times the average time presented on the Table represents the average time for all 10 of the observed crews times.

** An operator response time slightly greater than the 3 minutes assumed in the MTO analysis to initiate a cooldown does not invalidate the outcome of the analysis as long as total time form the start of the event to the time to initiate the cooldown does not exceed the total time of 21 minutes.

Table I-2:Steam Generator Tube Rupture Equipment List

Component	SR/ NSR	Valve Motive Power(1)	Comments
Steam Generators (SG)			
SG Power Operated Relief Valves (PORVs)	SR	HOV	
SG PORV Isolation Valves	SR	Manual	Backup to the SG PORVs
SG Blowdown Isolation Valves	SR	AOV	
SG Blowdown Sample Isolation Valves	SR	AOV	
SG Narrow Range Level	SR (3/SG) NSR (1/SG)		
SG Pressure	SR		
Auxiliary Feedwater System (AF)			
Motor Driven AF Pump	SR		
Diesel Driven AF Pump	SR		
AF Isolation Valves	SR	MOV	
AF Flow Control Loop and Valves	SR	AOV	Backup to the AF Isolation Valves (Safety related air accumulators will be installed. (2))
Feed Water Flow Indication	SR		
AF Flow Control	SR		
Main Steam System (MS)			
Main Steam Isolation Valves (MSIVs)	SR	HOV	
MSIV Bypass Valves	SR	AOV	
Steam Dump Valves	NSR	AOV	Backup to the MSIVs and MSIV Bypass Valves
Main Feedwater Pump Turbine High Pressure Stop Valves	NSR	HOV	Backup to the MSIVs and MSIV Bypass Valves
MS Reheater Start-Up Purge Control Valves	NSR	AOV	Backup to the MSIVs and MSIV Bypass Valves
MS Reheater Shutoff Valves	NSR	MOV	Backup to the MSIVs and MSIV Bypass Valves
Gland Steam Isolation and Bypass Valves	NSR	MOV	Backup to the MSIVs and MSIV Bypass Valves

Table I-2: Steam Generator Tube Rupture Equipment List *(cont.)*

Component	SR/ NSR	Valve Motive Power[(1)]	Comments
MS Header Pressure Controller	NSR		The Design Basis case uses the intact SG PORVs for the RCS cooldown. The Steam Dump Valves are not available in the Design Basis case due to the LOOP.
Steam Dump Mode Selector Switch	NSR		The Design Basis case uses the intact SG PORVs for the RCS cooldown. The Steam Dump Valves are not available in the Design Basis case due to the LOOP.
Reactor Coolant			
Pressurizer PORVs	SR	AOV	
Pressurizer PORV Isolation Valves	SR	MOV	
Pressurizer Spray	SR		
Pressurizer Spray Valves	SR	AOV	
Auxiliary Spray	SR		
Auxiliary Spray Valve	SR	AOV	
Reactor Coolant Pump Seal Injection Isolation Valves	SR	MOV	
Pressurizer Pressure	NSR		Four independent channels of pressure indication are available and other indirect indication of pressurizer pressure is also available.
Pressurizer Water Level	SR		
RCS Pressure	SR		
Core Exit Thermocouples	SR		
Alarms Indicating Leakage from Pressurizer PORVs	NSR		Not part of the design basis case.

Table I-2: Steam Generator Tube Rupture Equipment List *(cont.)*

Component	SR/ NSR	Valve Motive Power(1)	Comments
ECCS			
Centrifugal Charging Pumps	SR		
Safety Injection Pumps	SR		
High Head Safety Injection (Charging Pumps) to RCS Cold Leg Injection Isolation Valves	SR	MOV	A manual isolation valve will be added in series with each High Head Safety Injection valve (1/2SI8801A/B).(2)
Emergency Core Cooling System Pump Hand Switches (Main Control Room)	SR		
Charging Pumps Mini-Flow Isolation Valves Reset Switches	SR		
High Head SI Flow	NSR		Indication is used to determine if RCPs should be stopped. The RCPs are stopped in the Design basis case because of the Loss of Offsite Power (LOOP) event.
SI Pump Discharge Flow	NSR		Indication is used to determine if RCPs should be stopped. The RCPs are stopped in the Design basis case because of the Loss of Offsite Power (LOOP) event.
ESF			
AC Power/Emergency Diesel Generators	SR		
Containment Isolation Valves	SR	AOV	
Charging Line Containment Isolation Valves	SR	MOV	
SI Reset	SR		
Containment Isolation Phase A/B Reset	SR		
SI Sump Isolation Valves Reset Switches	SR		
Steamline SI Reset/Block Switches	SR		

Table I-2: Steam Generator Tube Rupture Equipment List *(cont.)*

Component	SR/ NSR	Valve Motive Power(1)	Comments
Radiation Monitors			
Main Steam Line Radiation Monitor	SR		
Steam Jet Air Ejector/Gland Steam Exhaust Gas Radiation	NSR		Safety related backup provided by the Main Steam Line Radiation Monitors.
SG Blowdown Liquid Radiation	NSR		Safety Related backup provided by the Main Steam Line Radiation Monitors.
Miscellaneous			
Station Air Compressors	NSR		This equipment is part of the recovery step to establish instrument air to the Containment Building.
Iconic Display	NSR		Used to check RCS subcooling. Operators can manually check RCS subcooling margin using safety related RCS pressure and core exit thermocouple instrumentation readings.

Notes:

(1) AOV – Air Operated Valve
HOV – Hydraulically Operated Valve
MOV – Motor Operated Valve

(2) These modifications are discussed in Section II.2.F.

II MARGIN TO STEAM GENERATOR OVERFILL ANALYSES

II.1 Introduction

Analyses were performed to determine the margin to SG overfill for a design basis SGTR event for the Byron and Braidwood units. The analyses were performed using the LOFTTR2 program and the methodology developed in Reference 1, with modifications to address NSAL-07-11 (Reference 3) consistent with WCAP-16948-P (Reference 4), and using plant-specific parameters. This section includes the methods, assumptions and input used to analyze the margin to overfill for the SGTR event as well as the sequence of events for the recovery and the calculated results.

II.2 Input Parameters and Assumptions

The margin to overfill analyses modeled the plant operating at the lower end of the T_{avg} range since a lower operating temperature results in a higher mass flow rate through the broken tube and less steam released from the ruptured SG. The analyses assumed that the plant was operating with the feedwater temperature at the low end of the temperature range since this results in a higher mass of water in the SG at the start of the event. The maximum SG tube plugging (SGTP) was modeled. Maximum tube plugging reduces heat transfer to the ruptured SG, which reduces the mass released by steaming, which in turn reduces margin to overfill. The reduced heat transfer also prolongs the cooldown period, leading to delayed break flow termination. Unit specific sensitivity runs were made to confirm the conservative nature of these plant operating assumptions. The analyses conservatively bound a RCS flow asymmetry of 7%.

II.2.A Design Basis Accident

The accident modeled was a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold leg) side of the SG. The location of the break on the cold side of the SG results in higher primary to secondary break flow than a break on the hot side of the SG. It was also assumed that a loss of offsite power (LOOP) 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.

II.2.B Conservative Assumptions

Plant responses until break flow termination were calculated using the LOFTTR2 computer code. The conservative conditions and assumptions which were used in Reference 1 were also used in the analyses to determine margin to SG overfill for Byron and Braidwood with the exception of the following differences.

1. RCS Flow Asymmetry

   NSAL-00-008 (Reference 5) identified the effect of an RCS loop flow asymmetry on an SGTR as indirect since, although the transient is initiated at full flow, the SGTR is an "asymmetric" event and the flow asymmetry can result in different initial conditions for the loops. With the SGs connected by the steam header, the initial secondary pressure is expected to be similar for all SGs, so the flow asymmetry would have little impact on the calculated break flow. However, with different loop flow rates the SGs would be removing different amounts of energy from the primary system, resulting in different initial secondary fluid mass calculations for the different loops.

The flow asymmetry was modeled in the analyses. From Reference 5 the maximum loop-to-loop flow asymmetry for a "newer" plant is 7% of nominal loop flow. The corresponding loop-to-loop power (SG heat flux) asymmetry was determined and used in steady state secondary side mass calculations to determine the conservative initial secondary side fluid mass.

2. SG Secondary Mass

   A higher initial secondary water mass in the ruptured SG was determined by Reference 1 to be conservative for overfill. The calculation of the initial secondary side water mass conservatively modeled an RCS flow asymmetry of 7%, as discussed above. This was applied in addition to the conservative increase in initial mass applied based on Reference 1 to model the increase in mass that would result from a turbine runback to a lower power, and the consideration of mass uncertainties. The runback is assumed to start at event initiation and continue until the time that the reactor trip setpoint is reached. The power resulting from the turbine runback is determined using an average runback rate of 10% power per minute, with the total power reduction limited to 30% (i.e., 3 minutes of runback).

3. Decay Heat and NSAL-07-11

   NSAL-07-11 (Reference 3) identifies a potential non-conservative assumption regarding the direction of conservatism for decay heat in the WCAP-10698-P-A (Reference 1) methodology for evaluating margin to overfill. For the margin to overfill analyses, higher decay heat yields a benefit by increasing steam releases from the ruptured SG, but results in a penalty from a longer cooldown and a conservatively delayed break flow termination. Conversely, lower decay heat yields a penalty by reducing steam releases from the ruptured SG, but results in a benefit from a shorter cooldown and earlier break flow termination. Similar impacts were identified in WCAP-16948-P (Reference 4) for the AFW and safety injection (SI) flow enthalpies. The relative importance of these competing effects is plant-specific, and plant-specific analyses are required to determine the conservative assumption. Plant-specific sensitivities performed for Byron and Braidwood Units 1 and 2 showed the following to be conservative with respect to margin to overfill for the limiting cases:

   - 1979-2σ American Nuclear Society (ANS) decay heat was conservative compared to the 1971+20% ANS decay heat model specified in WCAP-10698-P-A (Reference 1). For these analyses, the 1979 ANS decay heat model minus 2σ uncertainty was used.
   - Minimum AFW enthalpy was conservative compared to the maximum AFW enthalpy specified by WCAP-10698-P-A (Reference 1). For these analyses, the minimum AFW enthalpy of 0.03 Btu/lbm was modeled.
   - Maximum SI enthalpy was conservative consistent with WCAP-10698-P-A (Reference 1). For these analyses, the maximum SI enthalpy of 80 Btu/lbm was modeled.

The analyses also incorporated the conclusion of WCAP-16948-P (Reference 4) that beginning of life (BOL) minimum reactivity feedback coefficients are conservative for margin to overfill analyses.

II.2.C Plant Input

The following significant plant-specific input was used in the analyses.

1. SG Dimensions

   The SGTR flow is a function of the SG tube inside diameter. The SG tube inside diameters modeled in the analyses are:

   - Unit 1: 0.608 inches.
   - Unit 2: 0.664 inches.

   The available secondary side volume was used to calculate the margin to overfill. The available secondary side volumes modeled in the analyses are:

   - Unit 1: 5122 ft^3
   - Unit 2: 5955 ft^3

2. SG Power-Operated Relief Valve (PORV)

   It was assumed that a loss of offsite power occurs at reactor trip for the SGTR analyses, and thus the SG PORVs open to limit the secondary pressure. The PORV pressure setpoint is 1129.7 psia. The PORVs are relied upon to cool the RCS. A low value for the capacity of the PORVs was modeled since this results in a slower cooldown. The capacity modeled was not based solely on the nominal size of the valves, and addressed the concern identified in TB-07-6 (Reference 6). The PORV capacities modeled in the analyses are:

   - Unit 1: 188 lbm/sec/valve @ 1190 psia (with planned modified valve trim)
   - Unit 2: 114.32 lbm/sec/valve @ 1190 psia.

3. Pressurizer PORV Capacity

   It was assumed that a loss of offsite power occurs at reactor trip for the SGTR analyses, and thus the pressurizer PORV was relied upon to depressurize the RCS. The capacity of 210,000 lbm/hr at 2350 psia was used in the analyses for both units.

4. Auxiliary Feedwater

   It was assumed that the maximum AFW flow was delivered to the SGs following reactor trip and loss of offsite power with no delay. A minimal purge volume (1 ft^3) was modeled to delay delivery of cold AFW to the SGs and maximize steam release. The following AFW flows are modeled in the analyses:

   - Unit 1: 180 gpm/SG for the first 40 seconds, 360 gpm/SG after 40 seconds.
   - Unit 2: 263 gpm/SG for the first 40 seconds, 450.22 gpm/SG after 40 seconds.

Unit 1 AFW flow was limited to the throttled flow value based on the installation of safety related air accumulators (as discussed in Section II.2.f). Since the Unit 2 SGs have sufficient MTO, the Unit 2 AFW flow values were conservatively assumed to fail open even though the throttled flow could have been assumed in the analysis based on the installation of safety related air accumulators.

Flow to the ruptured SG continued at this rate until it was isolated by the operators. Flow to the intact SGs was throttled to maintain the level below 50% on the narrow range span (NRS).

5. Safety Injection Flows

The maximum SI flow was assumed to be initiated at the low pressurizer pressure setpoint of 1884 psia. The flow rates, assuming a density of 62.4 lbm/ft^3, are presented in Table II-1. These were used in the analyses for both units.

II.2.D Operator Action Times

In the event of an SGTR, the operator is required to take actions to stabilize the plant and terminate the primary to secondary break flow. The operator actions for SGTR recovery are provided in the EOPs, and major actions were explicitly modeled in these analyses. The operator actions modeled include isolation of the ruptured SG, cooldown of the RCS, depressurization of the RCS to restore inventory and termination of SI to stop primary to secondary break flow. These operator actions are described below. Note that critical operator response times are periodically validated as part of the Licensed Operator Requalification Program.

1. Identify the ruptured SG

High secondary side activity, as indicated by the main steamline radiation monitor (or other secondary monitors) or high SG sample activity typically will provide the first indication of an SGTR event. The ruptured SG can be identified by a mismatch between steam and feedwater flow, high activity in a SG water sample, or a high radiation indication on the corresponding main steamline radiation monitor. For an SGTR that results in a reactor trip at high power as assumed in these analyses, the SG water level as indicated on the narrow range will decrease significantly for all of the SGs. The AFW flow will begin to refill the SGs, distributing approximately equal flow to each of the SGs. Since primary to secondary break flow adds additional inventory to the ruptured SG, the water level will increase more rapidly in that SG. This response, as displayed by the SG water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured SG.

2. Isolate the ruptured SG

Once the ruptured SG has been identified, recovery actions begin by isolating AFW flow to the ruptured SG and closing the main steam isolation valve (MSIV) on the ruptured SG steamline. In addition to minimizing radiological releases, this also reduces the possibility of filling the ruptured SG by minimizing the accumulation of AFW and enabling the operator to establish a pressure differential between the ruptured and intact SGs as a necessary step toward terminating primary to secondary break flow. In the plant-specific EOPs for SGTR, the operator is directed to verify the level in the ruptured SG is greater than a specified level on the NRS prior to isolating AFW. The required level is 10% NRS for Unit 1 and 14% NRS for Unit 2. (Adverse environment values are not used since the design basis SGTR does not result in an adverse containment environment.) To model the isolation time using the methodology in Reference 1 for the margin to overfill analyses, it was assumed that AFW flow to the ruptured SG was isolated when the level in the SG reached half-way between the plant-specific required level and 50% NRS, or at 9 minutes from event initiation, whichever was longer. The ruptured SG MSIV was assumed to be closed at 18 minutes from event initiation.

3. Cooldown the RCS using the intact SGs

   After isolation of the ruptured SG MSIV, the RCS is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured SG pressure by dumping steam from only the intact SGs. This ensures adequate subcooling in the RCS after depressurization to the ruptured SG pressure in subsequent actions. The analyses assumed that 3 minutes elapses from the time the MSIV was closed until the cooldown was initiated. The tables of required core exit temperatures for cooldown termination (without adverse environment) in the plant-specific EOPs for SGTR were used in the analyses to obtain the target temperature that corresponds to the ruptured SG pressure. Since offsite power was assumed to be lost at reactor trip for these analyses, the normal steam dump system to the condenser could not be used to perform this cooldown and the cooldown was performed by dumping steam via the PORVs on the intact SGs. When the cooldown target temperature was reached the cooldown was terminated. The PORVs on the intact SGs were then used as necessary to maintain that temperature.

4. Depressurize the RCS to restore inventory

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

   The analyses assumed that 4 minutes elapsed from the time the cooldown was terminated until the depressurization was initiated. The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps (RCPs) are running. Since offsite power was assumed to be lost at the time of reactor trip, the RCPs were not running and thus normal pressurizer spray was not available. Therefore, the depressurization was modeled using a pressurizer PORV.

   The RCS depressurization is continued until any of the following conditions in the plant-specific EOPs for SGTR (using setpoints without adverse environment) are satisfied: RCS pressure is less than the ruptured SG pressure and pressurizer level is greater than 12%, or pressurizer level is greater than 68%, or RCS subcooling is less than required to address the subcooling uncertainty. (Byron and Braidwood plant-specific EOPs for SGTR have different pressurizer level setpoints for this step. The lower setpoint was used in the analyses since it potentially results in a higher RCS pressure at the end of the depressurization.)

5. Terminate SI to stop primary to secondary break flow

   The previous actions will have established adequate RCS subcooling, a secondary side heat sink, and sufficient RCS inventory to ensure that the SI flow is no longer needed. When these actions have been completed, the SI flow must be stopped to terminate primary to secondary break flow. The analyses assumed that 3 minutes elapsed from the time the depressurization was terminated until SI could be stopped. SI can be stopped provided the following conditions in the plant-specific EOPs for SGTR (using setpoints without adverse environment) are satisfied: RCS pressure is stable or rising, pressurizer level is greater than 12%, RCS subcooling is greater than required to address the subcooling uncertainty, and a secondary heat sink is confirmed.

   The analyses do not model specific actions after SI termination leading to break flow termination, consistent with the Reference 1 method. The primary to secondary break flow continues after the SI flow is stopped until the RCS and ruptured SG pressures equalize.

It is noted that the total time required to complete the recovery operations consists of both operator action time and system, or plant, response time. For instance, the time for each of the major recovery operations (i.e., RCS cooldown) is primarily due to the time required for the system response, whereas the operator action time is reflected by the time required for the operator to perform the intermediate action steps.

The operator action times to isolate AFW flow to the ruptured SG, to isolate the MSIV on the ruptured SG, to initiate RCS cooldown, to initiate RCS depressurization, and to terminate SI were developed for the design basis analyses. Exelon has determined the corresponding operator action times to perform these operations for Byron and Braidwood (see Table I-1). The operator actions and the corresponding operator action times used for the analyses are summarized in Table II-2.

II.2.E Single Failure Considerations

An evaluation was performed to determine the limiting single failure with respect to margin to SG overfill for an SGTR, consistent with WCAP 10698-P-A, as noted in Section I.1.E above. To aid in identification of the limiting single failure sensitivity runs were performed considering the following failures:

1. Failure of intact SG PORV

   This scenario considered the failure of a PORV to open on one of the intact SGs when the operator performed the RCS cooldown. Since offsite power was assumed to be lost at reactor trip for the SGTR analyses, the SG PORVs were relied upon to cool the RCS. Failure of a PORV on an intact SG to open on demand reduced the steam release capability provided by the PORVs, since only two intact SG PORVs are available for the cooldown. This increased the time required for the cooldown, resulting in increased break flow. The assumption that a minimum of two SG PORVs are available with adequate relieving capacity to conduct the required RCS cooldown is validated by the modifications described in Section II.2.F, Items 2 and 3, below. It is assumed that the PORV on the ruptured SG is isolated. In addition, since the failure of an intact SG PORV scenario assumes a loss of offsite power with an associated loss of Instrument Air (IA), the modification described in Section II.2.F, Item 1, assures that AFW flow control is maintained throughout the event.

2. Failure of ruptured SG MSIV

   This scenario considered the failure of the MSIV on the ruptured SG to close when the operators isolated the ruptured SG. In this case the operators were required to isolate the MSIVs on the intact SGs prior to initiating the cooldown. This was assumed to delay initiation of the cooldown by an additional 2 minutes (i.e., 5 minutes elapsed from the attempt to close the ruptured MSIV as part of ruptured SG isolation until the cooldown is initiated). This delay resulted in increased break flow. The cooldown was then performed using all three of the PORVs on the intact SGs.

3. Failure of ruptured SG feedwater control valve (FWCV)

   This scenario considered the failure of the FWCV on the ruptured SG to automatically reduce main feedwater flow to offset the addition of break flow to the ruptured steam generator. Thus, the mass in the ruptured SG increased in relation to the intact SGs prior to reactor trip. While this additional mass would be expected to provide early identification and isolation of AFW flow to the ruptured SG following reactor trip, no reduction in the operator action time for AFW isolation was credited. The initial secondary SG water mass was not increased to account for the impact of turbine runback. This modeling is consistent with the FWCV failure presented in WCAP-10698-P-A (Reference 1). The cooldown was performed using all three of the PORVs on the intact SGs.

The intact SG PORV failure was determined to be the limiting single failure. The penalty from the delay in cooldown initiation that resulted from the MSIV failure was offset by the faster cooldown obtained by use of the PORVs on three intact SGs. The impact of continued steam leakage of 5 lbm/sec for 30 minutes from the ruptured SG via unisolated flow paths was investigated for the MSIV failure case and it was determined to result in increased margin to overfill since the mass released was greater than the additional break flow that resulted from the small reduction in the ruptured SG pressure. Compared to the intact SG PORV failure case, there was a benefit in the FWCV failure case since the initial ruptured SG mass penalty for turbine runback was not included in the FWCV failure scenario. The FWCV failure case also received a benefit of a shorter cooldown due to the availability of the third intact SG PORV for the cooldown.

In the determination of the limiting failure the margin to overfill at the time of SI termination for the different sensitivity runs was compared. The operator responses after SI termination are not dependent on the specific failure scenario and the time for the operators to take positive control for termination of break flow would be consistent between the different single failure scenarios.

II.2.F Modifications to Support MTO Single Failure Considerations

Byron and Braidwood Stations will be implementing plant modifications to support the Steam Generator Margin to Overfill Reanalysis assumptions. The four modifications are as follows:

- Install safety related air accumulator tanks to support AFW flow control
- Increase the capacity of the SG Power Operated Relief Valves (PORV's) (on Unit 1 only)
- Install Uninterruptible Power Supplies (UPS) on 2 of the 4 SG PORVs
- Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

Below is a brief description of each modification:

1. Install Safety Related Air Accumulator Tanks to Support Auxiliary Feedwater (AF) Flow Control

   This modification will install two instrument air accumulator tanks (one per train on each unit) to provide a safety related air supply for the Auxiliary Feedwater Flow Control Valves. The tanks will be capable of providing 30 minutes of air supply to the Auxiliary Feedwater Flow Control Valves (AF005's). The tanks will be safety related. In addition, the modification will install two check valves in series for each tank to separate the non safety portion of the instrument air system from the safety related air accumulator tanks and tubing. A relief valve will be installed on each tank to provide overpressure protection. The electronic controls associated with the flow control loop have been verified to be safety related. This modification is planned to be installed in accordance with 10 CFR 50.59.

2. Increase the Capacity of the Steam Generator Power Operated Relief Valves (PORVs) on Unit 1 Only

   Byron and Braidwood will be replacing the SG PORV valve trim to increase the capacity of the valve from approximately 420,000 lbs/hr to 736,000 lbs/hr. By increasing the valve capacity the operators can cool down and depressurize the Reactor Coolant System (RCS) more rapidly which will equalize the pressure between the RCS and the secondary side of the steam generator which terminates the flow from the RCS to the secondary. This reduces the inflow to the secondary and therefore increases the margin to overfill. NRC approval of this modification, as it is included in the SGTR reanalysis, is required prior to installation as this change results in more than a minimal increase in the accident dose as defined in NEI 96-01, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, dated November 2000.

3. Install Uninterruptible Power Supplies (UPS) on two of the four Steam Generator (SG) Power Operated Relief Valves

   The modification to install uninterruptible power supplies to the SG PORVs is prompted by the resolution of Unresolved Items (URIs) from the 2009 Component Design Bases Inspection (CDBI) at Byron Station (URI 05000454/2009007-03; URI 05000455/2009007-03). The URIs involved a concern with respect to the single failure assumptions used in Byron Station's analysis for a Steam Generator Tube Rupture (SGTR) event. The NRC documented their position regarding these URIs in Reference 14. The NRC verified that this same SGTR-related concern was also applicable to Braidwood Station as documented in Reference 16. Byron Station responded to the NRC in Reference 15; and Braidwood Station responded to the NRC in Reference 17. In these letters, both Byron Station and Braidwood Station committed to installing the UPS modification to resolve the single failure concern. This modification places the SGTR analysis in compliance with NRC regulations and preserves the assumptions in the SGTR analysis.

   The modification will install two UPS units on each Unit at Byron and Braidwood Stations. At Byron and Braidwood Stations, there are four SG PORV's; two PORVs power from electrical Division 1, and the other two PORVs powered from electrical Division 2. One UPS will be installed on each electrical division's power supply to one of the SG PORVs it supplies. Currently, a potential single failure of the Division 1 (or 2) 480VAC Unit Substation exists that would disable the power supplies to two SG PORV's. Following the implementation of this modification, the normal power feed will continue to be a Motor Control Center (MCC) powered from a 480VAC Unit Substation; however, on a loss of power to the 480VAC Unit Substation, the UPS would provide a backup power supply to one of the two PORVs from that division. By installing one UPS to a PORV powered from each electrical division, a single failure of a 480VAC Unit Substation would only adversely impact one SG PORV. The other division would still have power supplied to both SG PORV's. Therefore, at least two SG PORV's will be available to support cool down and depressurization of the RCS during a SGTR event (this also assumes that the ruptured SG PORV is isolated and unavailable for cooldown). This modification is planned be installed in accordance with 10 CFR 50.59.

4. Install a manual isolation valve upstream of each High Head Safety Injection valve (1/2SI8801A/B)

   As mentioned in Section I.1.E, Byron and Braidwood do not have BIT isolation valves, but the two parallel High Head Safety Injection motor operated valves (1/2SI8801A/B) provide the equivalent function. These valves provide the means to isolate the high head safety injection flow during the SI termination phase. If a single failure of one of these valves occurs, it would be necessary to secure the charging pumps in order to depressurize the reactor coolant system (RCS) to stop the break flow into the ruptured steam generator and to prevent overfilling the RCS. If the charging pumps were secured, injection flow to the reactor coolant pumps seals ceases. While reactor coolant pump seal design provides for loss of seal injection, it is not a desirable condition.

   To address the potential single failure of an SI8801A/B valve to close, a modification will be implemented which installs a manual valve upstream of each SI8801A/B valve to provide for isolation of high head safety injection without the need to stop all charging pumps. During normal plant conditions these manual valves will be locked in the open position. Upon failure of an SI8801A/B valve to close when demanded, an operator will be dispatched to locally close the upstream manual isolation valve. As described above, this manual action is not replacing an automatic function but is simply an equivalent manual action for locally isolating high head safety injection flow into the RCS. This modification is planned to be installed in accordance with 10 CFR 50.59.

II.3 Description of Analyses

The LOFTTR2 analysis for the limiting Unit 1 and Unit 2 margin to overfill cases are described below. For both units the limiting case with respect to margin to SG overfill considered operation at the minimum operating temperature (580.0°F for Unit 1 and 575.0°F for Unit 2), with the minimum main feedwater temperature (433.0°F for Unit 1 and 435.0°F for Unit 2), the maximum SGTP level (5% for Unit 1 and 10% for Unit 2), and the failure of a PORV on an intact SG to open when the operator performed the RCS cooldown. The sequences of events for these transients are presented in Table II-3.

Although the lower T_{avg} is conservative for margin to overfill, this does not mean that a T_{avg} coastdown will be more limiting. Since the Overtemperature Delta-T (OTΔT) reactor trip nominal T_{avg} and nominal delta-T values in the setpoint equation are not reset to reflect the lower operating T_{avg}, the reactor trip will be delayed. The benefits of delaying reactor trip versus the disadvantages of operating at a lower T_{avg} were incorporated in a case considered in the analysis. The Unit 2 case modeling the T_{avg} coastdown to 573.5°F showed the same margin to overfill as the case considering the Unit 2 minimum operating temperature of 575.0°F. It was concluded that the low T_{avg} case would be reported as the limiting case but that future analyses would continue to examine the impact of T_{avg} coastdown.

Following the tube rupture, water flowed from the primary into the secondary side of the ruptured SG since the primary pressure is greater than the SG pressure. In response to this loss of coolant, pressurizer level decreased as shown in Figure II-1 (Unit 1) and Figure II-7 (Unit 2). The RCS pressure also decreased as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2) as the steam bubble in the pressurizer expanded. As the RCS pressure decreased due to the continued primary to secondary break flow automatic reactor trip occurred on an Overtemperature-ΔT (OTΔT) trip signal.

After reactor trip, core power rapidly decreased to decay heat levels. The turbine stop valves closed and steam flow to the turbine was terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remained 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 caused the secondary side pressure to increase rapidly after reactor trip, as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2), until the SG PORVs (and safety valves if their setpoints are reached) lifted to dissipate the energy, as shown in Figure II-6 (Unit 1) and Figure II-12 (Unit 2). As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated following reactor trip.

The RCS pressure and pressurizer level continued to decrease after reactor trip as energy transfer to the secondary system shrank the primary coolant and the tube rupture break flow continued to deplete primary inventory. The decrease in RCS inventory resulted in a low pressurizer pressure SI signal. The SI flow increased the RCS inventory and the RCS pressure trended toward the equilibrium value where the SI flow rate would equal the break flow rate.

AFW flow to the ruptured SG was assumed to be isolated 9 minutes after the start of the event, and the ruptured SG MSIV was assumed to be closed at 18 minutes. The ruptured SG level was well above the level required for identification and isolation by these times.

After isolation of the ruptured SG, a 3-minute operator action time was assumed prior to initiating the cooldown. Due to the assumed failure of one of the intact SG PORVs to open only two intact SGs were credited for the cooldown. It was therefore assumed that the PORVs on two intact SGs were opened for the RCS cooldown at 21 minutes after the start of the event, as shown in Figure II-6 (Unit 1) and Figure II-12 (Unit 2). The cooldown was continued until the cooldown termination temperature obtained from EOPs was reached. When this condition was satisfied the operator closed the PORVs to terminate the cooldown. This cooldown ensured that there would be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured SG pressure. The reduction in the intact SG pressure required to accomplish the cooldown is shown in Figure II-1 (Unit 1) and Figure II-8 (Unit 2). The pressurizer level and RCS pressure also decreased during this cooldown process due to shrinkage of the RCS as shown in Figure II-1 and Figure II-2 (Unit 1) and Figure II-7 and Figure II-8 (Unit 2).

The PORVs on the intact SGs which were used for the cooldown also automatically opened as necessary to maintain the prescribed RCS temperature to ensure that subcooling was maintained. When the PORVs were opened, the increased energy transfer from the RCS to the secondary system also aided in the depressurization of the RCS to the ruptured SG pressure after the SI flow was terminated.

After termination of the cooldown, a 4-minute operator action time was assumed prior to the RCS depressurization. In these analyses, the RCS depressurization was terminated when the RCS pressure was reduced to less than the ruptured SG pressure and the pressurizer level was above the required value, since there was adequate subcooling margin and the high pressurizer level setpoint was not reached. The RCS depressurization is shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2). The depressurization reduced the break flow as shown in Figure II-3 (Unit 1) and Figure II-9 (Unit 2) and increased SI flow to refill the pressurizer, as shown in Figure II-1 (Unit 1) and Figure II-7 (Unit 2).

After termination of the depressurization, a 3-minute operator action time was assumed prior to SI termination. The SI flow was terminated at this time since the requirements for SI termination were satisfied. (RCS subcooling was greater than the required allowance for subcooling uncertainty, minimum AFW flow was available or at least one intact SG level was in the narrow range, the RCS pressure was stable or increasing, and the pressurizer level was greater than the required value.) After SI termination the RCS pressure began to decrease as shown in Figure II-2 (Unit 1) and Figure II-8 (Unit 2).

II.4 Acceptance Criteria

The analyses were performed to demonstrate that the secondary side of the ruptured SG did not completely fill with water. The available secondary side volume of a single SG is 5122 ft^3 for Unit 1 and 5955 ft^3 for Unit 2. Margin to overfill is demonstrated provided the transient calculated SG secondary side water volume is less than these values. No credit is taken for the volume of the nozzle or any steam piping.

II.5 Results

The primary to secondary break flow rate throughout the recovery operations is presented in Figure II-3 (Unit 1) and Figure II-9 (Unit 2). The ruptured SG fluid mass is shown in Figure II-4 (Unit 1) and Figure II-10 (Unit 2). The water volume in the ruptured SG is presented as a function of time in Figure II-5 (Unit 1) and Figure II-11 (Unit 2). The peak ruptured SG water volume for Unit 1 is 5028 ft^3 resulting in 94 ft^3 of margin to overfill. The peak ruptured SG water volume for Unit 2 is 5685 ft^3 resulting in 270 ft^3 of margin to overfill. Therefore, it is concluded that overfill of the ruptured SG will not occur for a design basis SGTR for Byron and Braidwood Units 1 and 2.

II.6 Conclusions

It is concluded that overfill of the ruptured SG will not occur for a design basis SGTR for Byron and Braidwood Units 1 and 2.

Table II-1 Safety Injection Flows for Design Basis SGTR Analyses

Pressure (psia)	Total Injection Flow Rate (gpm)
600	1328.1
900	1190.9
1000	1140.7
1100	1087.9
1200	1030.9
1300	968.5
1400	898.7
1500	816.2
1600	687.7
1700	548.6
1800	524.9
1900	500.5
2000	472.6
2100	440.6
2200	407.1
2300	371.9
2400	333.8
2500	281.2
2600	221
2700	119.3
2739	20.7

Table II-2 Operator Action Times For Design Basis SGTR Analyses*

Action	Time
Operator action time to isolate AFW flow to ruptured SG	Margin to Overfill: Later of 9 minutes from event initiation or the time for the level to reach 30% NRS for Unit 1 / 32% NRS for Unit 2. Input to Dose: Time for the level to reach 10% NRS for Unit 1 / 14% NRS for Unit 2, but not earlier than 5 minutes from event initiation**.
Operator action time to isolate MSIV on ruptured SG	18 minutes from event initiation.
Operator action time to initiate cooldown	Margin to Overfill: 3 minutes from time of steamline isolation*** Input to Dose: 3 minutes from time of failed open PORV isolation***
Cooldown	Calculated by LOFTTR2
Operator action time to initiate depressurization	4 minutes from end of cooldown
Depressurization	Calculated by LOFTTR2
Operator action time to terminate SI following depressurization	Maximum of 3 minutes from end of depressurization or time to satisfy termination criteria
Pressure equalization	Calculated by LOFTTR2

* A simple validation of the operator action times modeled in the analyses requires that the operators perform each of the actions within the time listed in the table above. However, evaluations can be performed to show that acceptable results continue to be obtained if some of these action times are traded off. For example, performing the earlier actions (e.g. isolating AFW) in a shorter time than analyzed allows for the possibility that the later times (e.g. starting the depressurization) may be increased over the values used in the analyses and still demonstrate acceptable results.

** The assumption of a minimum of 5 minutes from event initiation until AFW isolation used in the input to dose analyses is not a critical operator action time and does not impose a requirement on the operators.

*** The assumption of a minimum of 3 minutes from the time of steamline isolation to the initiation of a cooldown is not a critical operator action time and does not impose a requirement on the operators as long as the cooldown is initiated in 21 minutes.

Table II-3 Sequence of Events for Limiting Margin to Overfill Analyses

Event	Unit 1 Time (seconds)	Unit 2 Time (seconds)
Steam Generator Tube Rupture	0	0
Reactor Trip (OTΔT) and LOOP	200.4	139.3
AFW Initiated	201	140
SI Actuated	474	317
AFW Flow to Ruptured SG Isolated	540	540
Ruptured SG MSIV Closed	1080	1080
RCS Cooldown Initiated	1260	1260
RCS Cooldown Terminated	1790	1958
RCS Depressurization Initiated	2032	2200
RCS Depressurization Terminated	2130	2302
SI Terminated	2311	2482
Break Flow Terminated	3458	3258

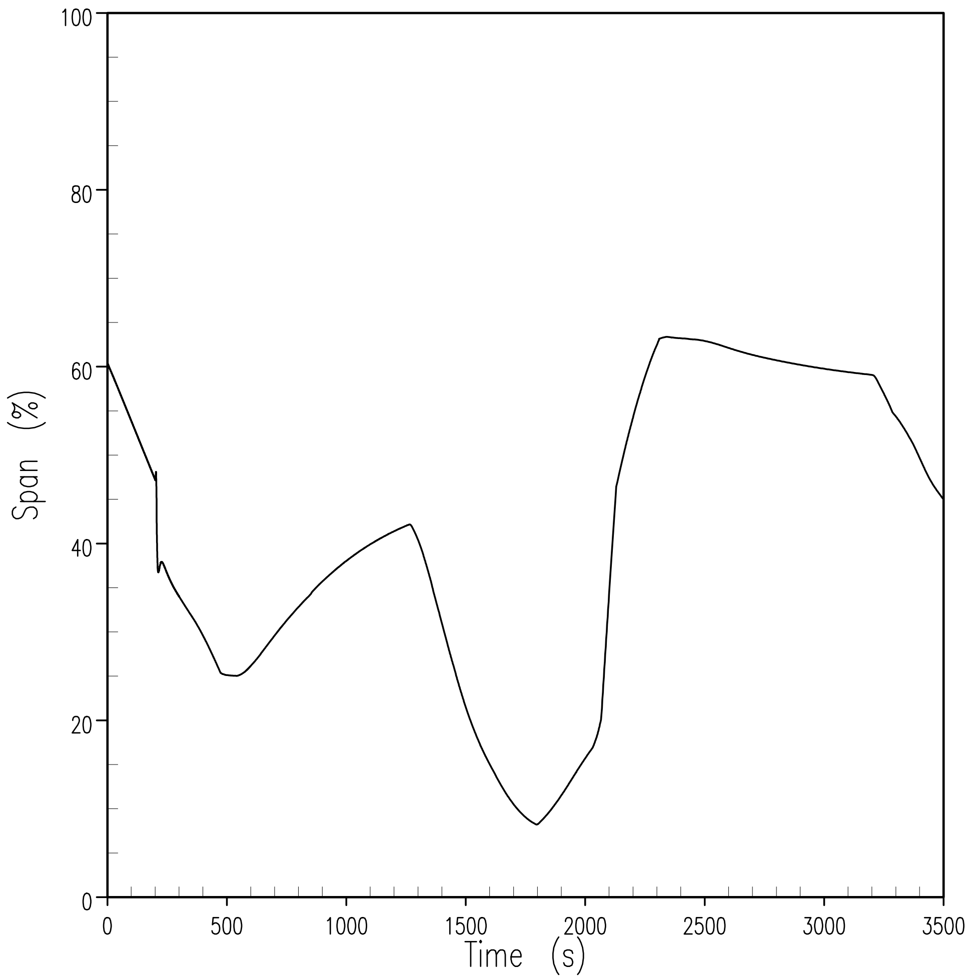

**Figure II-1 Pressurizer Level –
Unit 1 Margin to Overfill Analysis**

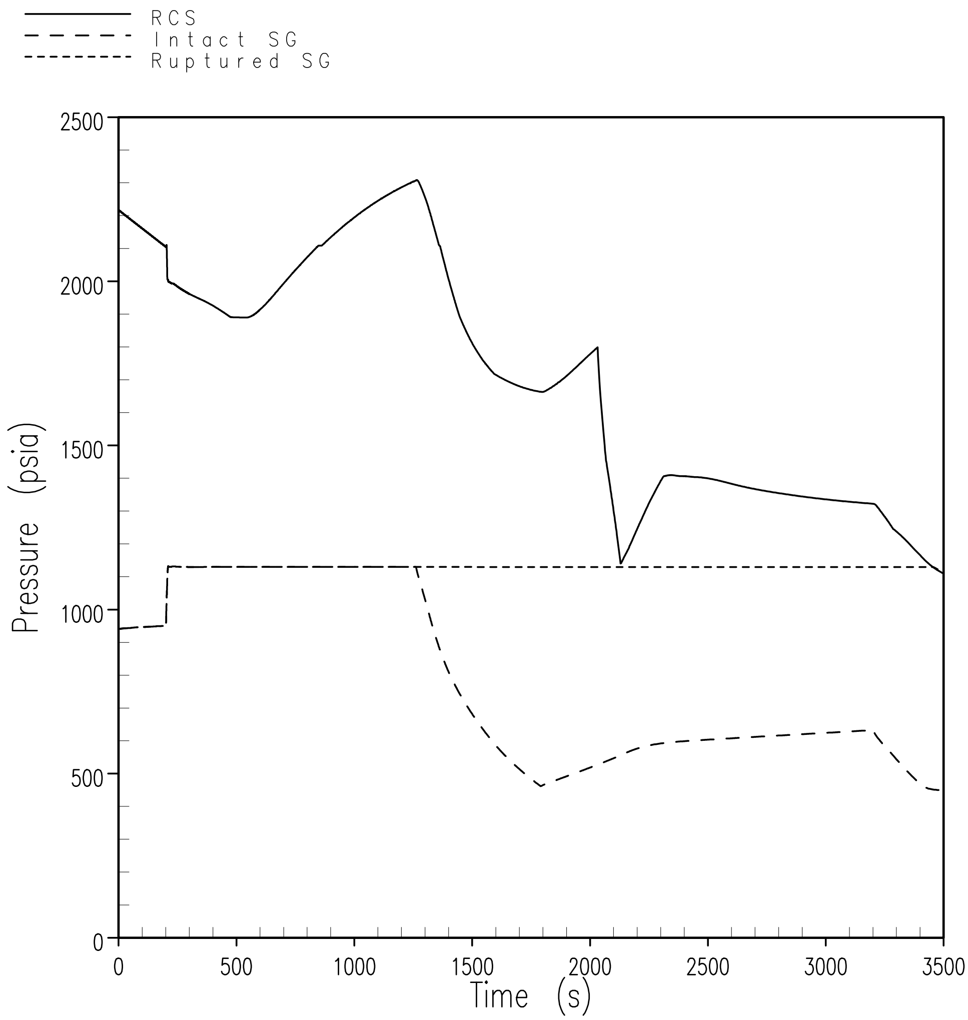

Figure II-2 RCS and Secondary Pressures – Unit 1 Margin to Overfill Analysis

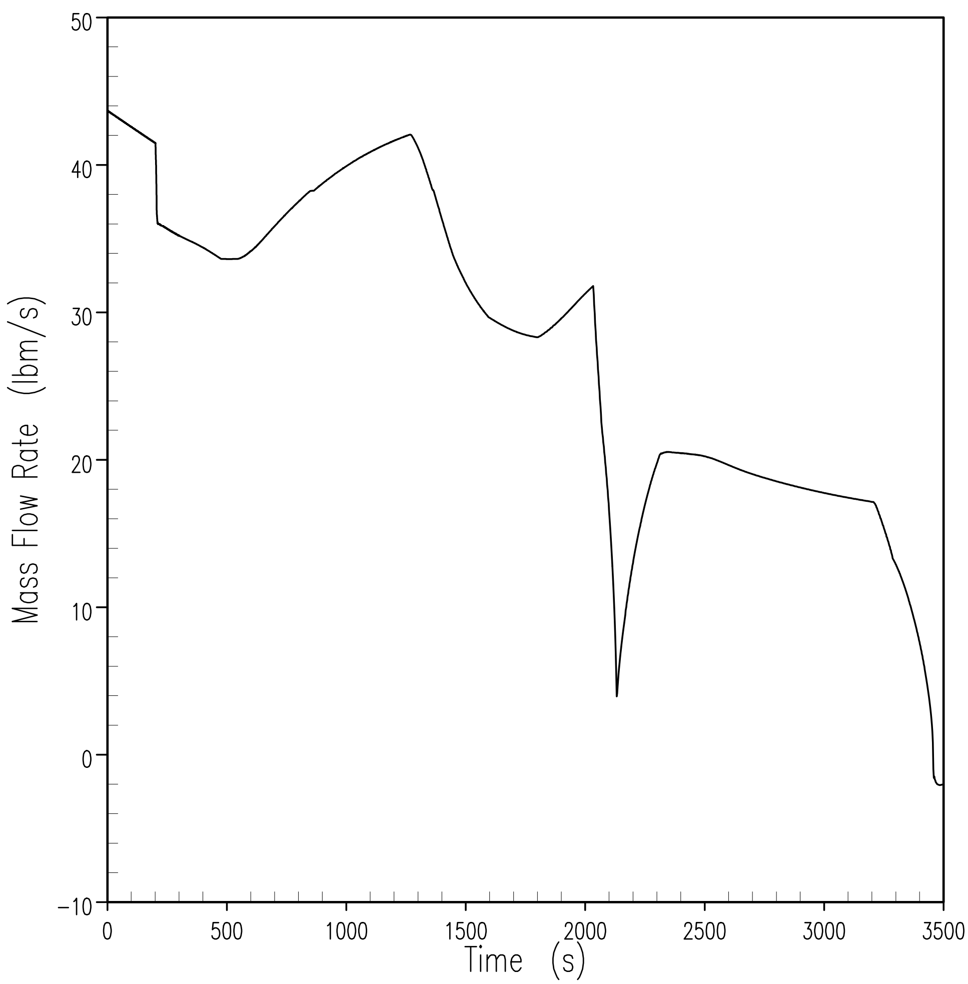

**Figure II-3 Primary to Secondary Break Flow –
Unit 1 Margin to Overfill Analysis**

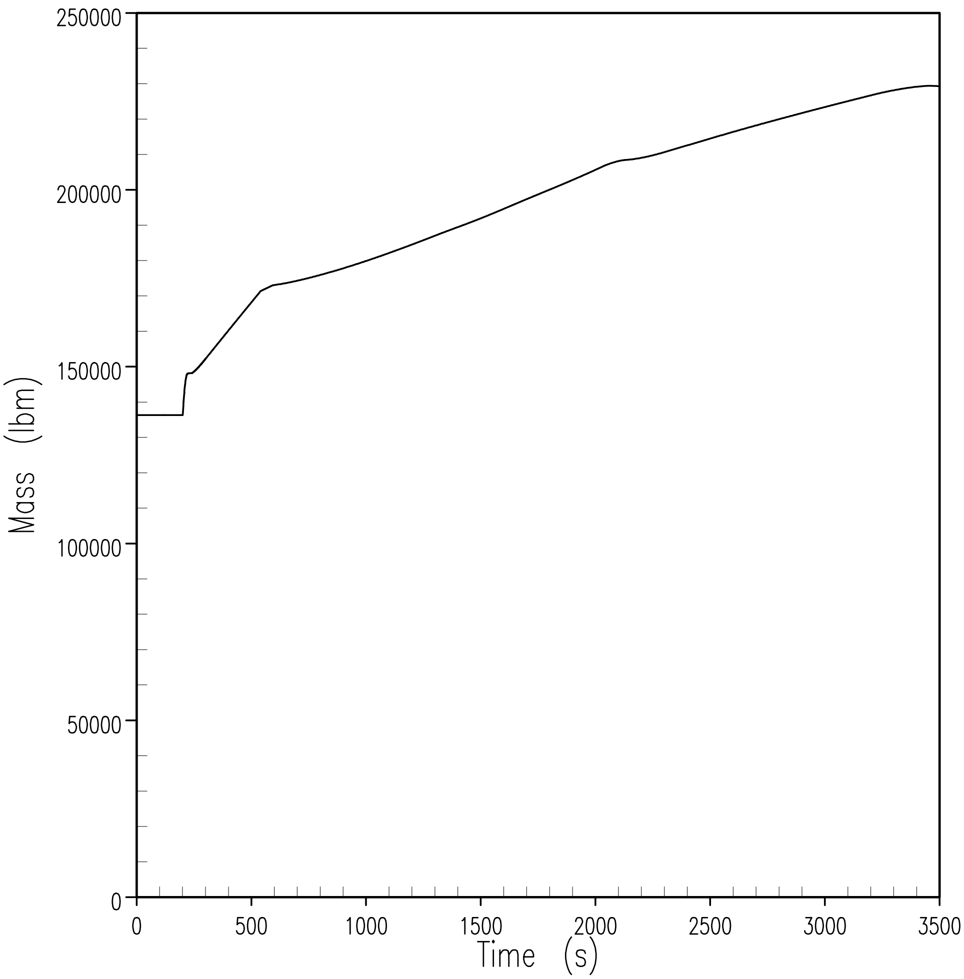

**Figure II-4 Ruptured SG Fluid Mass –
Unit 1 Margin to Overfill Analysis**

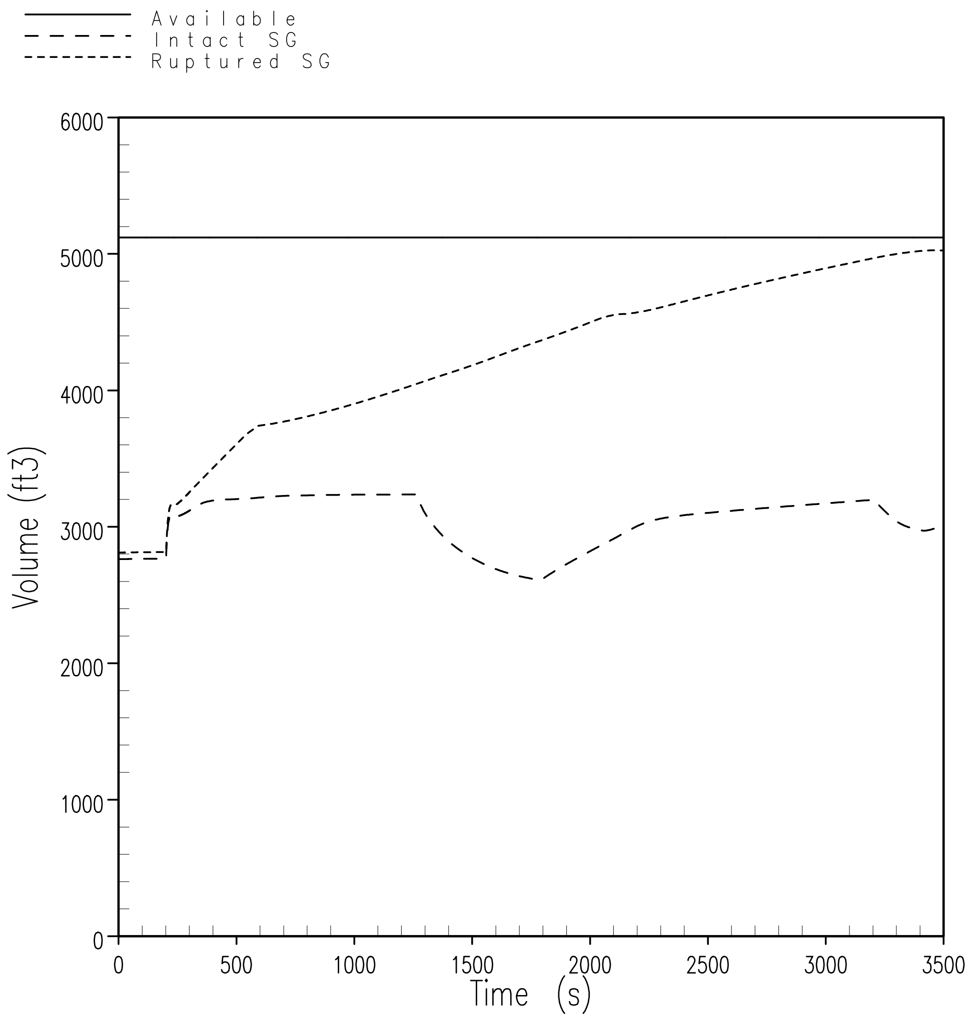

**Figure II-5 SG Water Volumes –
Unit 1 Margin to Overfill Analysis**

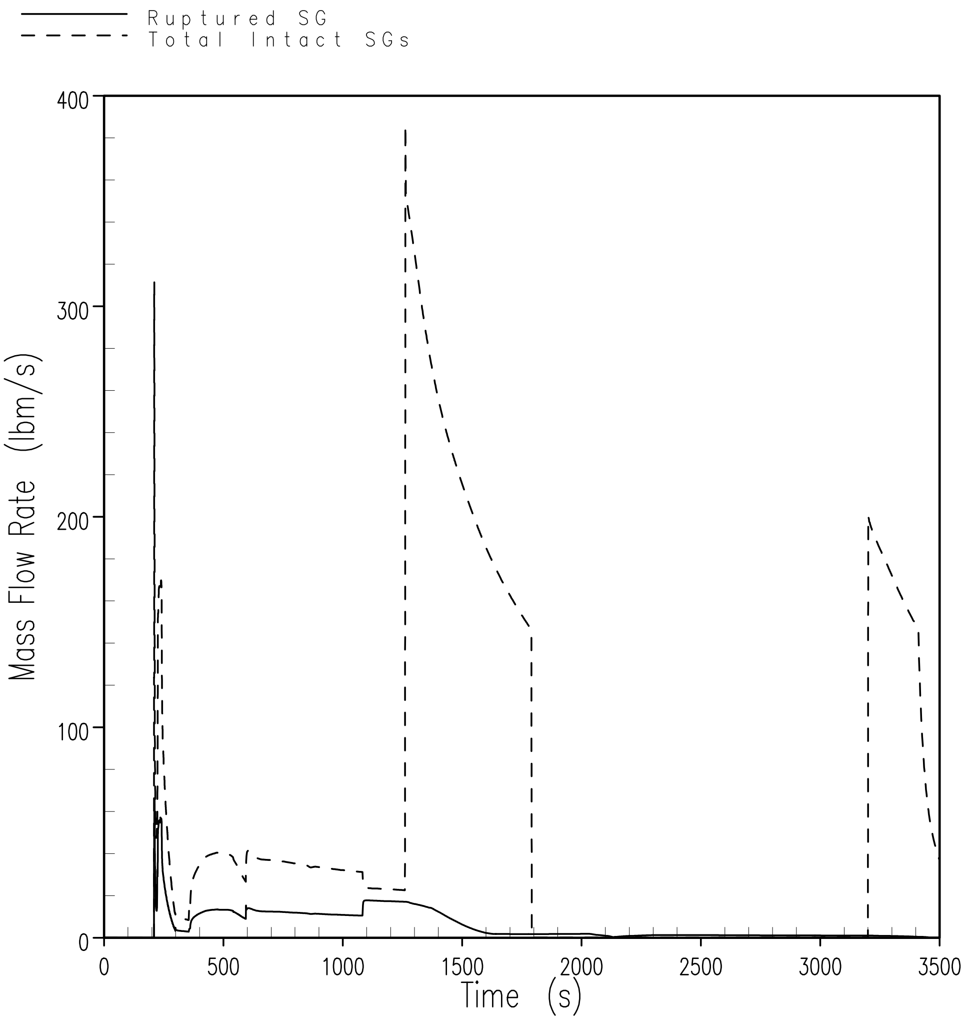

**Figure II-6 SG Steam Releases –
Unit 1 Margin to Overfill Analysis**

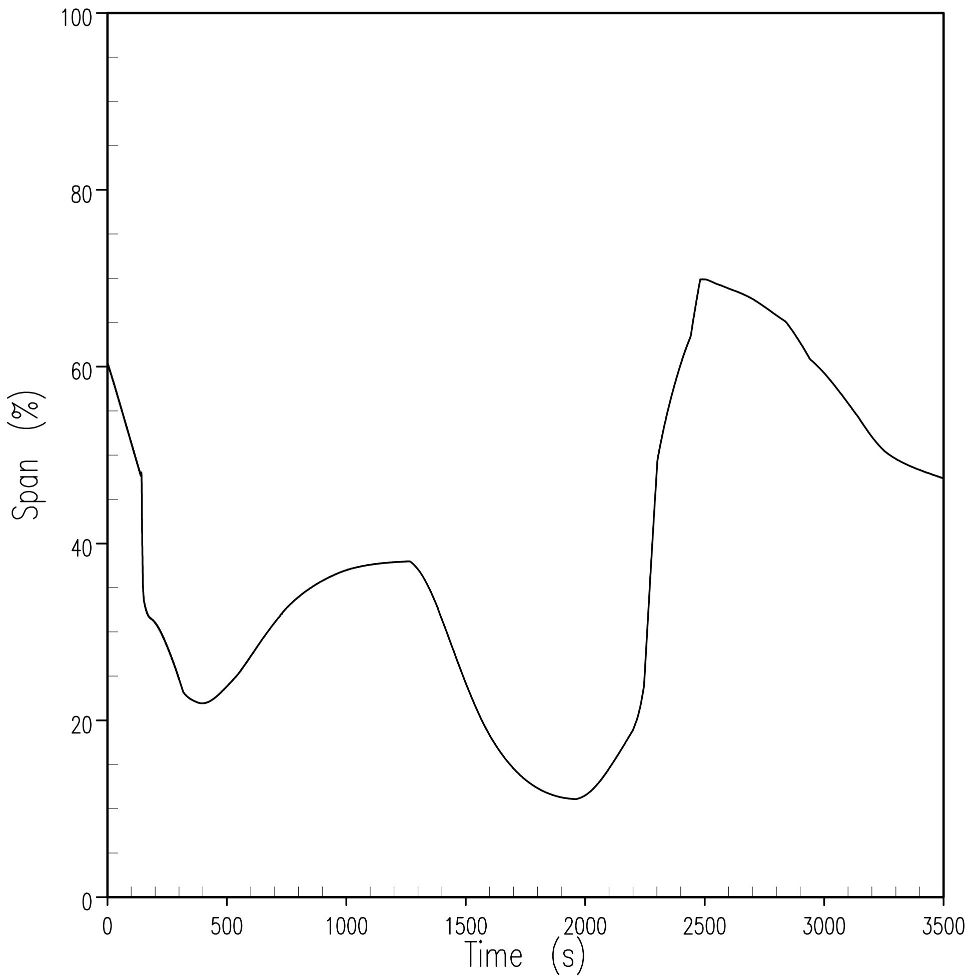

Figure II-7 Pressurizer Level –
Unit 2 Margin to Overfill Analysis

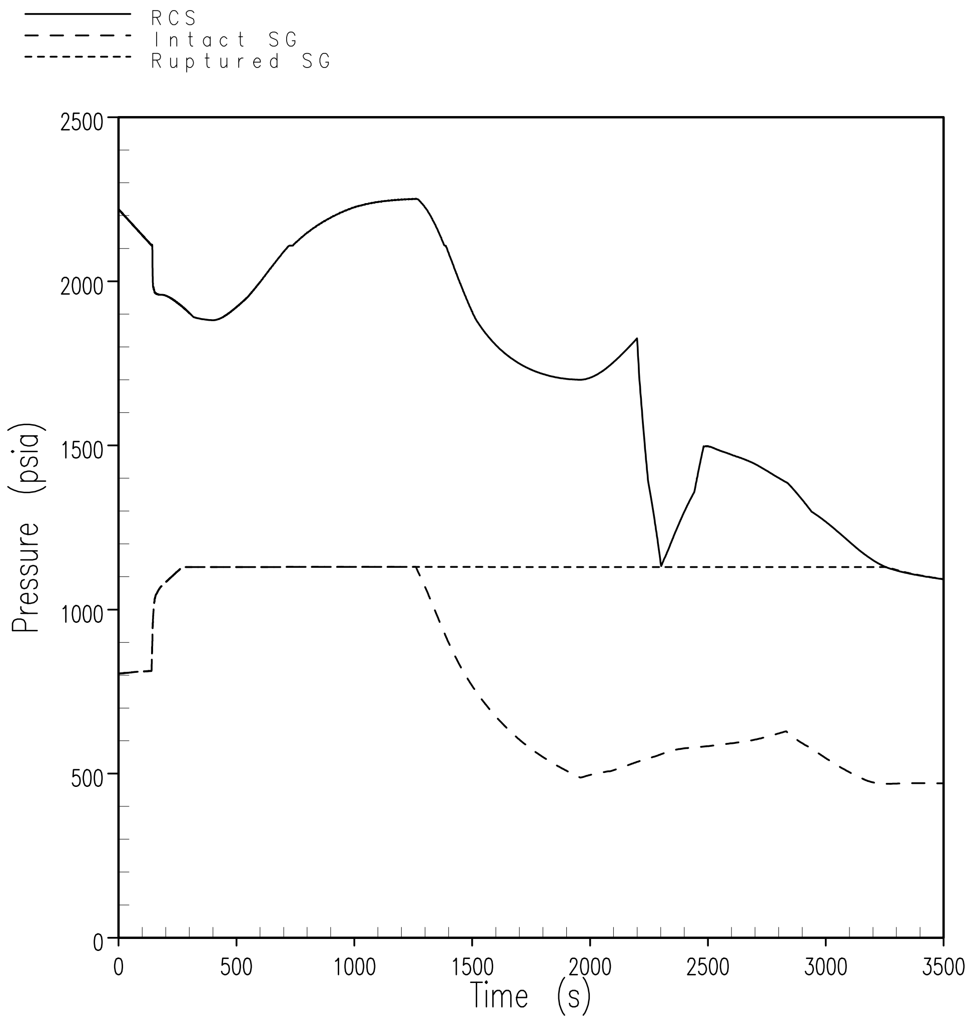

Figure II-8 RCS and Secondary Pressures – Unit 2 Margin to Overfill Analysis

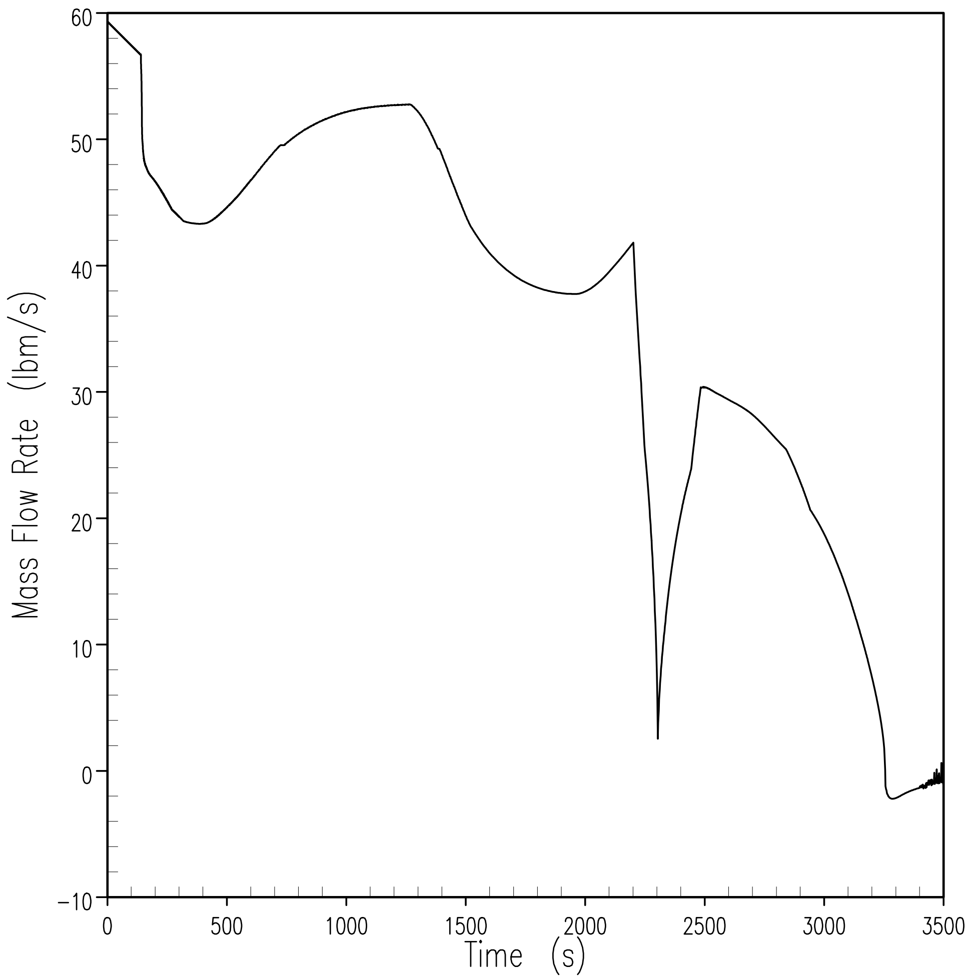

**Figure II-9 Primary to Secondary Break Flow –
Unit 2 Margin to Overfill Analysis**

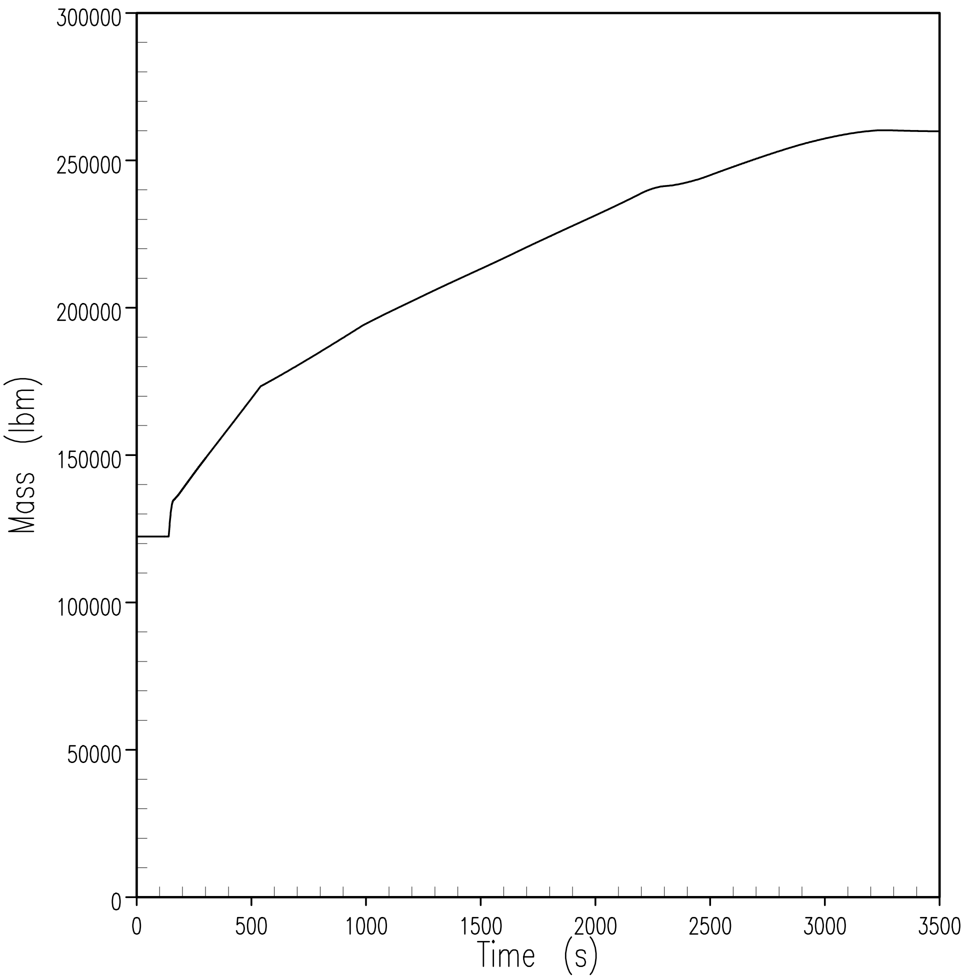

**Figure II-10 Ruptured SG Fluid Mass –
Unit 2 Margin to Overfill Analysis**

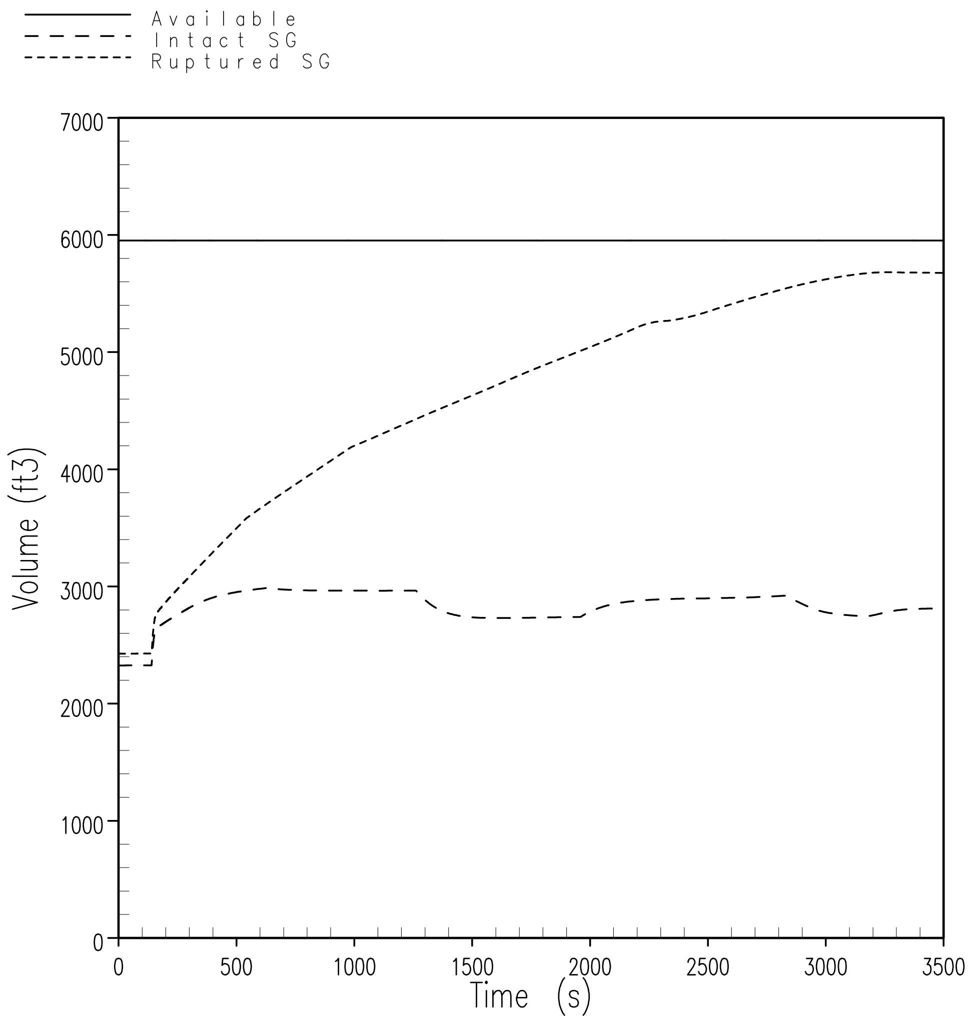

**Figure II-11 SG Water Volumes –
Unit 2 Margin to Overfill Analysis**

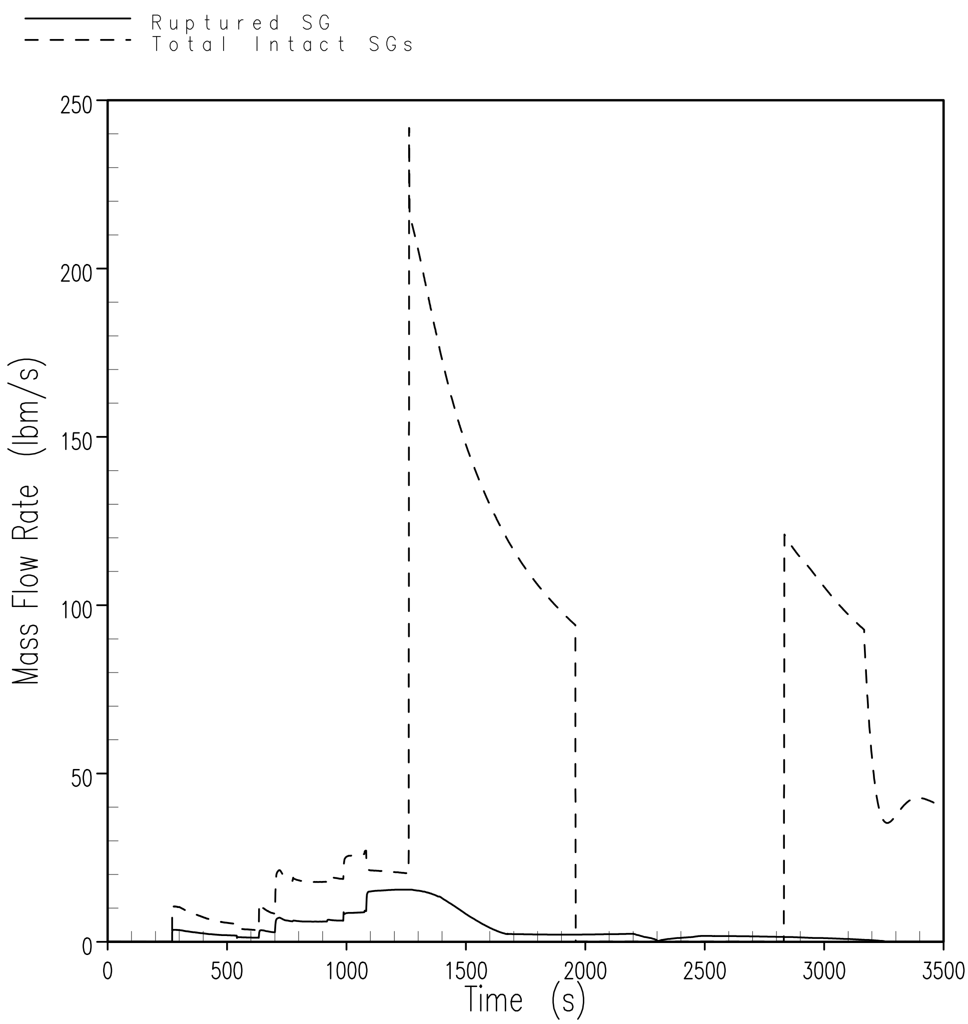

Figure II-12 SG Steam Releases –
Unit 2 Margin to Overfill Analysis

III THERMAL AND HYDRAULIC ANALYSES FOR RADIOLOGICAL CONSEQUENCES

III.1 Introduction

Thermal and hydraulic analyses were performed to determine the input for the radiological consequences analyses for a design basis SGTR event for the Byron and Braidwood units. The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in References 1 and 2, and using the plant-specific parameters. This section includes the methods, assumptions and input used to analyze the SGTR event, as well as the sequence of events for the recovery and the calculated results.

III.2 Input Parameters and Assumptions

The thermal-hydraulic analyses, which determined the mass releases for the radiological consequences analyses, modeled the plant operating at both the high and the low end of the T_{avg} range. SG tube plugging at 0% and 5% for Unit 1, and 0% and 10% for Unit 2, were also studied.

III.2.A Design Basis Accident

The design basis accident modeled was a double-ended break of one SG tube located at the top of the tube sheet on the outlet (cold-leg) side of the SG. The location of the break on the cold side of the SG results in higher primary to secondary break flow than a break on the hot side of the SG, as determined by Reference 1. However, the break flow flashing fraction was conservatively calculated assuming that all of the break flow comes from the hot-leg side of the SG. The combination of these conservative assumptions results in a very conservative calculation of the radiological consequences. It was also assumed that loss of offsite power occurred 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. Due to the assumed loss of offsite power, the condenser was not available for steam releases once the reactor was tripped. Consequently, after reactor trip, steam was released to the atmosphere through the SG PORVs.

III.2.B Single Failure Consideration

Based on the information in Reference 2, the most limiting single failure with respect to radiological consequences is a failed open PORV on the ruptured SG. Failure of this PORV causes an uncontrolled depressurization of the SG, which increases primary to secondary break flow and the steam release to the atmosphere. The lower secondary pressure also results in a higher break flow flashing fraction. Pressure in the ruptured SG will remain below that in the primary system until the failed PORV can be isolated, and recovery actions completed.

III.2.C Conservative Assumptions

This section includes a discussion of the methods and assumptions used to analyze the SGTR event and to calculate the mass released the sequence of events during the recovery operations, and the calculated results.

Most of the assumptions used for the margin to overfill analyses are also conservative for the radiological consequences analyses. The major differences in the assumptions that were used for the LOFTTR2 analyses for radiological consequences compared to those used in the margin to overfill analyses are discussed below.

1. SG Secondary Mass

   Plant-specific sensitivity studies determined that a maximum initial secondary water mass resulted in increased steam releases and flashed break flow. Therefore, the transient calculation considered the effects of turbine runback and RCS flow asymmetry in the same manner as the margin to overfill analyses. However, a lower secondary mass is conservative for the dose analyses. Additional cases were analyzed with minimum initial secondary water mass (without the turbine runback mass increase and with the maximum main feedwater temperature) and the results conservatively incorporated in the dose analyses. A lower secondary mass also results in a lower ruptured SG pressure when the ruptured SG is failed open and this was considered in the confirmation that the pressure did not fall below 320 psig noted in the operator action time discussion below.

2. Decay Heat and NSAL-07-11

   As noted in NSAL-07-11 (Reference 3) SGTR thermal and hydraulic analyses for input to radiological consequences analyses have no competing effects with respect to decay heat. Higher decay results in increased steam releases from the ruptured SG and a longer cooldown, leading to a later break flow termination. These effects are conservative for the SGTR radiological consequences calculation, and thus, lower decay heat was not considered. Similarly, the maximum AFW and SI enthalpies were used. The following changes were made to the related assumptions used in the margin to overfill analyses:

   - The 1971+20% ANS decay heat model specified by WCAP-10698-P-A (Reference 1) was used for these analyses.
   - Maximum AFW enthalpy is conservative consistent with WCAP-10698-P-A (Reference 1). For these analyses, the maximum AFW enthalpy of 91.12 Btu/lbm was modeled.

3. Flashing Fraction

   When calculating the fraction of break flow that flashes to steam, 100% of the break flow was assumed to come from the hot-leg side of the break. Since the tube rupture flow actually consists of flow from the hot-leg and cold-leg sides of the SG, the temperature of the combined flow will be less than the hot-leg temperature and the flashing fraction would be correspondingly lower. Thus, this assumption is conservative.

III.2.D Plant Input

The significant plant-specific input is the same as modeled in the margin to overfill analyses except for the changes listed below.

1. SG Power-Operated Relief Valve (PORV)

   The PORV on the ruptured SG was assumed to fail open therefore the maximum capacity of the PORVs was modeled. The PORV capacities modeled in the analyses are:

   - Unit 1: 212.4 lbm/sec/valve @ 1190 psia,
   - Unit 2: 133.89 lbm/sec/valve @ 1200 psia.

2. Auxiliary Feedwater

   It was assumed that the minimum AFW flow (302 gpm/SG) was delivered to the SGs following reactor trip and loss of offsite power with a maximum delay (63 seconds). The maximum purge volume (160 ft^3) was modeled to delay delivery of cold AFW to the SGs and maximize steam release. Flow to the ruptured SG continued at this rate until it was isolated by the operators. Flow to the intact SGs was throttled to maintain the level below 50% NRS. The same input was used for Unit 1 and Unit 2.

III.2.E Operator Action Times

The major operator actions required for the recovery from an SGTR are discussed in Section II.2.D, and the operator action times used for the analyses are presented in Table II-2. With the exception of the time to isolate AFW flow to the ruptured SG the operator action times assumed for the margin to overfill analyses were also used for the radiological consequences analyses. Earlier AFW isolation results in higher releases, so it was assumed that AFW flow to the ruptured SG was isolated when level in the SG reached the plant-specific required level (but not before 5 minutes). The assumption of a minimum of 5 minutes from event initiation until AFW isolation used in the input to dose analyses is not a critical operator action time and does not impose a requirement on the operators. While isolation of AFW earlier than 5 minutes would have some impact on the exact calculated mass release data, the data is rounded and margin added in developing the input to the dose analysis (as noted in Section IV.2.A) which ensures that the reported doses are conservative.

For the radiological consequences analyses, the PORV on the ruptured SG was assumed to fail open at the time the ruptured SG is isolated. 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. An operator can locally close the block valve for the PORV on the ruptured SG within 30 minutes after the failure. Thus, it was assumed that the ruptured SG PORV was isolated at 30 minutes after the valve is assumed to fail open. After the ruptured SG PORV was isolated, an additional delay time of 3 minutes (Table II-2) was assumed before initiation of the RCS cooldown. The cooldown was performed using the PORVs on all three of the intact SGs. The cooldown target temperature was selected based on the ruptured SG pressure.

III.2.F Mass Release Calculations

The mass releases were determined for use in evaluating the offsite and control room radiological consequences of the SGTR using the methodology of Reference 2. The steam releases from the ruptured and intact SGs, and primary to secondary break flow into the ruptured SG and the associated flashing fraction, were determined for the period from accident initiation until 2 hours after the accident and from 2 to 8 hours after the accident.

In the LOFTTR2 analyses, the SGTR recovery actions in the EOPs were simulated until the termination of primary to secondary break flow. After the primary to secondary break flow is terminated, the operators will continue the SGTR recovery actions. The plant is then cooled and depressurized to cold shutdown conditions. In accordance with the methodology in Reference 2 it was assumed that the operators perform the post-SGTR cooldown using steam release to the atmosphere. This method results in a conservative evaluation of the long-term releases for use in the radiological consequences analyses compared to the other cooldown methods in the EOPs.

The high level actions for the post-SGTR cooldown method using steam releases are discussed below.

1. Prepare for cooldown to cold shutdown

   The initial steps to prepare for cooldown to cold shutdown will be continued if they have not already been completed. A few additional steps are also performed prior to initiating cooldown. These include isolating the cold leg SI accumulators to prevent unnecessary injection, energizing pressurizer heaters as necessary to saturate the pressurizer water and to provide for better pressure control, and assuring shutdown margin in the event of a potential boron dilution due to in-leakage from the ruptured SG.

2. Cooldown RCS to Residual Heat Removal (RHR) system temperature

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

3. Depressurize RCS to RHR system pressure

   When the cooldown to RHR system temperature is completed, the pressure in the ruptured SG is decreased by releasing steam from the ruptured SG. It was assumed that the ruptured SG is depressurized by releasing steam via the PORV. As the ruptured SG pressure is reduced, the RCS pressure is maintained equal to the pressure in the ruptured SG in order to prevent excessive in-leakage of secondary side water or additional primary to secondary break flow. Although normal pressurizer spray is the preferred means of RCS pressure control, auxiliary spray or a pressurizer PORV can be used to control RCS pressure if pressurizer spray is not available.

4. Cooldown to cold shutdown

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

III.3 Description of Analyses

The LOFTTR2 results for the limiting Unit 1 and Unit 2 input to dose analyses are described below. For both units the limiting case with respect to the input to dose considered operation at the maximum operating temperature (588.0°F for both units), with the minimum main feedwater temperature (433.0°F for Unit 1 and 435.0°F for Unit 2), the minimum steam generator tube plugging (SGTP) level (0% for both units), and the failure of the PORV on the ruptured SG in the full open position when the operator closes the MSIV. The sequences of events for these transients are presented in Table III-1.

Following the tube rupture, water flowed from the primary into the secondary side of the ruptured SG since the primary pressure was greater than the SG pressure. In response to this loss of coolant, pressurizer level decreased as shown in Figure III-1 (Unit 1) and Figure III-10 (Unit 2). The RCS pressure also decreased as shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2) as the steam bubble in the pressurizer expanded. As the RCS pressure decreased due to the continued primary to secondary break flow, an automatic reactor trip occurred on an OTΔT trip signal.

After reactor trip, core power rapidly decreased to decay heat levels. The turbine stop valves closed and steam flow to the turbine was terminated. The steam dump system is designed to actuate following reactor trip to limit the increase in secondary pressure, but the steam dump valves remained 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 caused the secondary side pressure to increase rapidly after reactor trip, as shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2) until the SG PORVs (and safety valves if their setpoints were reached) lift to dissipate the energy, as shown in Figure III-4 (Unit 1) and Figure III-13 (Unit 2). As a result of the assumed loss of offsite power, main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated following reactor trip.

The RCS pressure and pressurizer level continued to decrease after reactor trip as energy transfer to the secondary system shrunk the RCS and the tube rupture break flow continued to deplete primary inventory. The decrease in RCS inventory resulted in a low pressurizer pressure SI signal. The SI flow increased the RCS inventory and the RCS pressure trended toward the equilibrium value where the SI flow rate would equal the break flow rate.

AFW flow to the ruptured SG was isolated 5 minutes after the start of the event, and the ruptured SG MSIV was closed at 18 minutes. The ruptured SG level was above the level required for identification and isolation by these times.

The ruptured SG PORV was assumed to fail open when the MSIV was closed at 18 minutes. The failure caused the ruptured SG to depressurize rapidly, which resulted in an increase in primary to secondary break flow. The depressurization of the ruptured SG increased the break flow and energy transfer from primary to secondary, which resulted in RCS pressure and temperature decreasing more rapidly than in the margin to overfill analyses. The ruptured SG depressurization caused a cooldown in the intact SGs loops. The operators identified that the ruptured SG PORV had failed open and closed the associated block valve 30 minutes after the failure. At 48 minutes the ruptured SG PORV block valve was closed and the ruptured SG pressure began to increase as shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2). The ruptured SG pressure was confirmed to be above 320 psig at all times in the transient. This was also confirmed for transients run with less limiting mass transfer results, but greater ruptured SG pressure reductions. The lowest ruptured SG pressure for all cases analyzed was greater than 377 psia for Unit 1 and 440 psia for Unit 2.

After the block valve for the ruptured SG PORV was closed, a 3-minute operator action time was assumed prior to initiating the cooldown. The PORVs on all three of the intact SGs were opened at 51 minutes for the RCS cooldown as shown in Figure III-4 (Unit 1) and Figure III-13 (Unit 2). The depressurization of the ruptured SG due to the failed open PORV affected the RCS cooldown target temperature. The target temperature was determined based upon the pressure in the ruptured SG at the time the cooldown was initiated. The cooldown was continued until the cooldown termination temperature obtained from EOPs was reached. When this condition was satisfied the operator closed the PORVs to terminate the cooldown. The cooldown ensured that there would be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured SG pressure. The reduction in the intact SG pressure required to accomplish the cooldown is shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2). The pressurizer level and RCS pressure also decreased during this cooldown process due to shrinkage of the RCS as shown in Figure III-1 (Unit 1) and Figure III-10 (Unit 2).

The PORVs on the intact SGs also automatically opened to maintain the prescribed RCS temperature to ensure that subcooling was maintained. When the PORVs were opened, the increased energy transfer from the RCS to the secondary system also aided in the depressurization of the RCS to the ruptured SG pressure after the SI flow was terminated.

After termination of the cooldown, a 4-minute operator action time was assumed prior to the RCS depressurization. In these analyses, the RCS depressurization was terminated when the RCS pressure was reduced to less than the ruptured SG pressure and the pressurizer level was above the required value, since there was adequate subcooling margin and the high pressurizer level setpoint was not reached. The RCS depressurization is shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2). The depressurization reduced the break flow as shown in Figure III-3 (Unit 1) and Figure III-12 (Unit 2) and increased SI flow to refill the pressurizer, as shown in Figure III-1 (Unit 1) and Figure III-10 (Unit 2).

After termination of the depressurization, a 3-minute operator action time was assumed prior to SI termination. The SI flow was terminated at this time since the requirements for SI termination were satisfied. (RCS subcooling was greater than the required allowance for subcooling uncertainty, minimum AFW flow was available or at least one intact SG level was in the narrow range, the RCS pressure was stable or increasing, and the pressurizer level was greater than the required value.) After SI termination the RCS pressure began to decrease as shown in Figure III-2 (Unit 1) and Figure III-11 (Unit 2).

III.3.A Calculation of Mass Releases

The operator actions for the SGTR recovery up to the termination of primary to secondary break flow were simulated in the LOFTTR2 analyses. Thus, the steam releases from the ruptured and intact SGs, along with the break flow into the ruptured SG, were determined from the LOFTTR2 results for the period from the initiation of the accident until the break flow was terminated.

Following the termination of break flow, it was assumed that the RCS and intact SG conditions were maintained stable until the cooldown to cold shutdown was initiated. The PORVs for the intact SGs were then assumed to be used to start to cool down the RCS to the RHR system operating temperature of 350°F, at the maximum allowable cooldown rate of 100°F/hr. The RCS and the intact SG temperatures at 2 hours were determined using the RCS and intact SG parameters at the time of break flow termination and the RCS cooldown rate. The steam releases from the intact SGs for the period from break flow termination until 2 hours were determined from a mass and energy balance using the calculated RCS and intact SG conditions. Since the ruptured SG is isolated, no change in the ruptured SG conditions was assumed to occur until subsequent depressurization.

The RCS cooldown was assumed to continue after 2 hours until the RHR system operating temperature of 350°F was 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 operating pressure (using a bounding value of 300 psia) via steam release from the ruptured SG PORV, since this maximizes the steam release from the ruptured SG to the atmosphere which is conservative for the evaluation of the radiological consequences. The RCS pressure was also assumed to be reduced concurrently as the ruptured SG is depressurized. It was assumed that the RCS cooldown and depressurization to RHR operating conditions were completed within 8 hours after the accident. The steam releases from 2 to 8 hours were determined for the intact SGs from a mass and energy balance using the RCS and intact 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 break flow termination and saturated conditions at the RHR operating pressure.

After 8 hours, it was assumed that further plant cooldown to cold shutdown as well as long-term cooling was provided by the RHR system. Therefore, the steam releases to the atmosphere were terminated at 8 hours.

III.4 Acceptance Criteria

The analyses were performed to calculate the mass transfer data for input to the radiological consequences analyses. As such, no acceptance criteria are defined. The results of the analyses were used as input to the radiological consequences analyses presented in Section IV.

III.5 Results

III.5.A Results of LOFTTR2 Analyses

The primary to secondary break flow rate throughout the recovery operations is presented in Figure III-3 (Unit 1) and Figure III-12 (Unit 2). The break flow flashing fraction was calculated using the ruptured hot leg loop temperature presented in Figure III-5 (Unit 1) and Figure III-14 (Unit 2). The flashing fraction is presented in Figure III-6 (Unit 1) and Figure III-15 (Unit 2), the integrated flashed break flow is presented in Figure III-7 (Unit 1) and Figure III-16 (Unit 2). The ruptured SG PORV steam release rate is presented in Figure III-4 (Unit 1) and Figure III-13 (Unit 2), along with the total intact SG PORV steam release rate. The ruptured steam water volume is shown in Figure III-8 (Unit 1) and Figure III-17 (Unit 2). The water volume in the ruptured SG when the break flow is terminated is significantly less than the available SG volume. The ruptured SG fluid mass is shown in Figure III-9 (Unit 1) and Figure III-18 (Unit 2).

III.5.B Mass Release Results

The mass release calculations were performed using the methodology discussed in Section III.3.A. For the time period from initiation of the accident until break flow termination, the releases were determined from the LOFTTR2 results for the time prior to reactor trip and following reactor trip. Since the condenser was in service until reactor trip, any radioactivity released to the atmosphere prior to reactor trip would be through the condenser vacuum exhaust. After reactor trip, the releases to the atmosphere were assumed to be via the SG PORVs.

The transfer and release data are presented in Table III-2 and Table III-3 (Unit 1) and Table III-4 and Table III-5 (Unit 2).

RCS and SG mass data was obtained from the LOFTTR2 analyses for use in the dose analyses. The data is presented Table III-6 for both units. The initial RCS masses are conservatively low values accounting for the maximum SGTP (5% for Unit 1 and 10% for Unit 2) and maximum RCS operating T_{avg} (588.0°F for both units). The initial SG secondary masses are conservatively low values accounting for the minimum SGTP (0%) and minimum RCS operating T_{avg} (580.0°F for Unit 1 and 575.0°F for Unit 2). Although the RCS and intact SG masses dropped early in the transient, the minimum values were only slightly below the initial values and the mass was significantly above the initial mass for the remainder of the transient. (Although Figure III-8 (Unit 1) and Figure III-17 (Unit 2) show that the intact SG volume dropped below the initial value for the limiting case, and the mass would follow this trend, the minimum mass for this case was still greater than the initial mass provided in Table III-6 for a case modeling minimum initial SG mass.) The Unit 1 ruptured SG mass dropped significantly from the initial value, therefore the minimum transient value is also provided. This minimum transient mass was obtained from a case modeling minimum initial SG mass, and is significantly lower than that shown in Figure III-9 which presents the limiting transient relative to the releases. The Unit 2 ruptured SG mass did not drop significantly from the initial value so the minimum transient value is not provided for use in the dose analyses. (Although Figure III-18 shows that the Unit 2 ruptured SG mass dropped below the initial value for the limiting case, the minimum mass for this case was greater than the initial mass provided in Table III-6 from a case modeling minimum initial SG mass.)

III.6 Conclusions

The analyses performed to calculate the mass transfer data for input to the radiological consequences analyses were completed and the data was tabulated for the limiting cases. The results of the analyses were used as input to the radiological consequences analyses presented in Section IV.

Table III-1 Sequence of Events for Limiting Input to Radiological Consequences Analyses		
	Unit 1	**Unit 2**
Event	**Time (seconds)**	**Time (seconds)**
Steam Generator Tube Rupture	0	0
Reactor Trip (OTΔT) and LOOP	217	160
AFW Actuated	280	223
AFW Flow to Ruptured SG Isolated	300	300
SI Actuated	387	305
Ruptured SG MSIV Closed	1080	1080
Ruptured SG PORV Fails Open	1082	1082
Ruptured SG PORV Block Valve Closed	2882	2882
RCS Cooldown Initiated	3062	3062
Break Flow Flashing Terminated	3304	3390
RCS Cooldown Terminated	4284	4922
RCS Depressurization Initiated	4526	5164
RCS Depressurization Terminated	4618	5250
SI Terminated	4798	5430
Break Flow Terminated	5478	6234
Two hours from SG Tube Rupture	7200	7200
Time RHR System Takes Over Cooling	28800	28800

Table III-2 Unit 1 Break Flow and Flashed Break Flow			
Start of Period (sec)	**End of Period (sec)**	**Total Break Flow during Period (lbm)**	**Total Flashed Break Flow during Period (lbm)**
0	217	8832	1346
217	1082	31822	2090
1082	1982	43335	5173
1982	2882	47081	5390
2882	3062	9347	677
3062	3304	11053	278
3304	4526	44734	
4526	5478	16215	
5478	7200		
7200	28800		

Table III-3 Unit 1 Intact and Ruptured SG Steam Flow to Atmosphere			
Start of Period (sec)	**End of Period (sec)**	**Total Intact SGs Steam Flow to Atmosphere during Period (lbm)**	**Total Ruptured SG Steam Flow to Atmosphere during Period (lbm)**
0	217	742000*	248900*
217	1082	91020	43810
1082	1982	15180	96690
1982	2882	240	68300
2882	3062	0	0
3062	3304	93660	0
3304	4526	147900	0
4526	5478	38100	400
5478	7200	138700	0
7200	28800	945800	30100

* Pre-trip steam releases are through the condenser.

Table III-4 Unit 2
Break Flow and Flashed Break Flow

Start of Period (sec)	End of Period (sec)	Total Break Flow during Period (lbm)	Total Flashed Break Flow during Period (lbm)
0	160	8565	1391
160	1082	42671	2829
1082	1982	51578	5290
1982	2882	55269	5701
2882	3062	10998	838
3062	3390	17588	554
3390	5164	76740	
5164	6234	23706	
6234	7200		
7200	28800		

Table III-5 Unit 2
Intact and Ruptured SG Steam Flow to Atmosphere

Start of Period (sec)	End of Period (sec)	Total Intact SGs Steam Flow to Atmosphere during Period (lbm)	Total Ruptured SG Steam Flow to Atmosphere during Period (lbm)
0	160	544500*	183000*
160	1082	85050	43240
1082	1982	19080	73660
1982	2882	4530	54800
2882	3062	240	0
3062	3390	84990	0
3390	5164	153810	1800
5164	6234	47700	1000
6234	7200	104900	0
7200	28800	947000	50400

* Pre-trip steam releases are through the condenser.

Table III-6
RCS and SG Mass Data

	Unit 1	Unit 2
Event	**Liquid Mass (lbm)**	**Liquid Mass (lbm)**
Initial RCS	5.48E5	4.88E5
Initial Ruptured SG	1.03E5	6.89E4
Minimum Ruptured SG	4.15E4	>6.89E4
Initial Intact SGs (total)	3.10E5	2.06E5

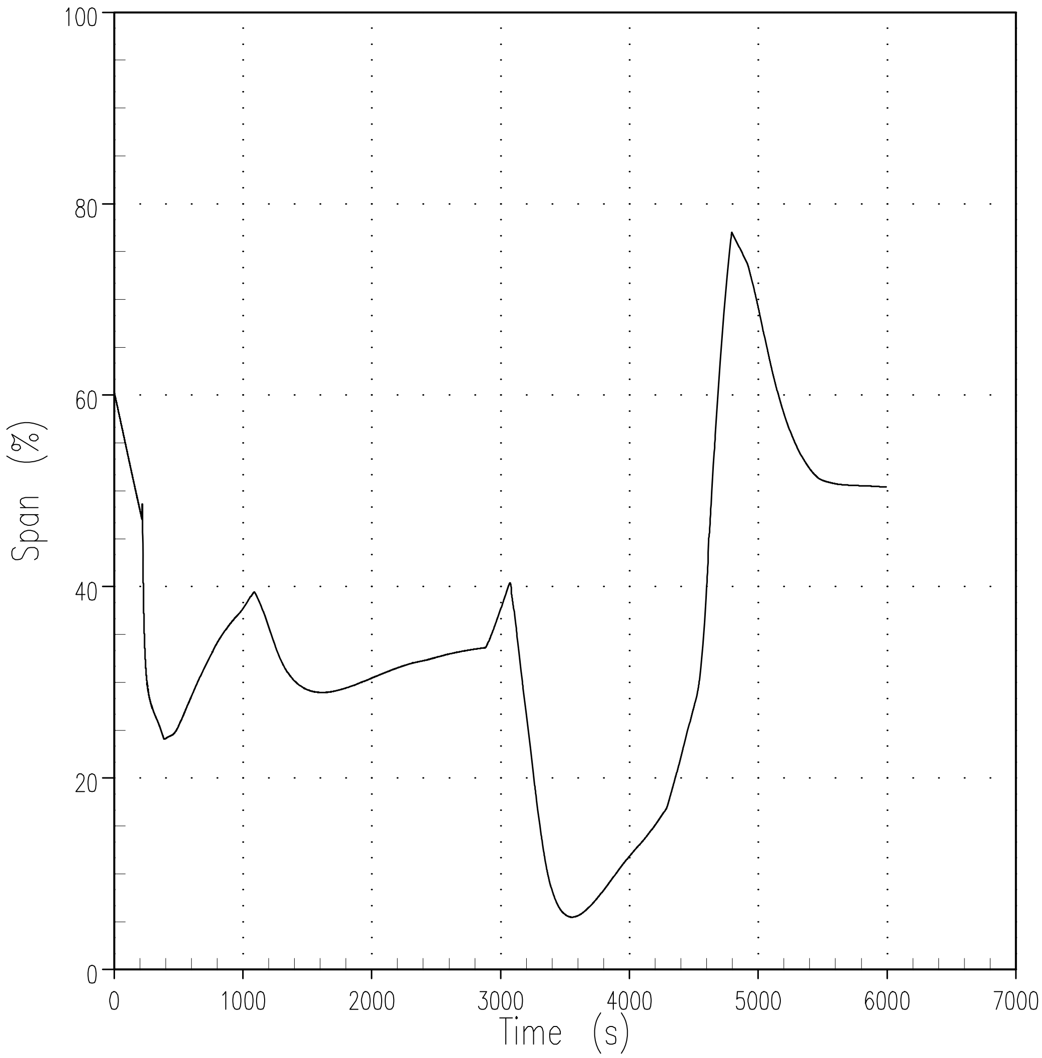

Figure III-1 Pressurizer Level –
Unit 1 Input to Radiological Consequences Analysis

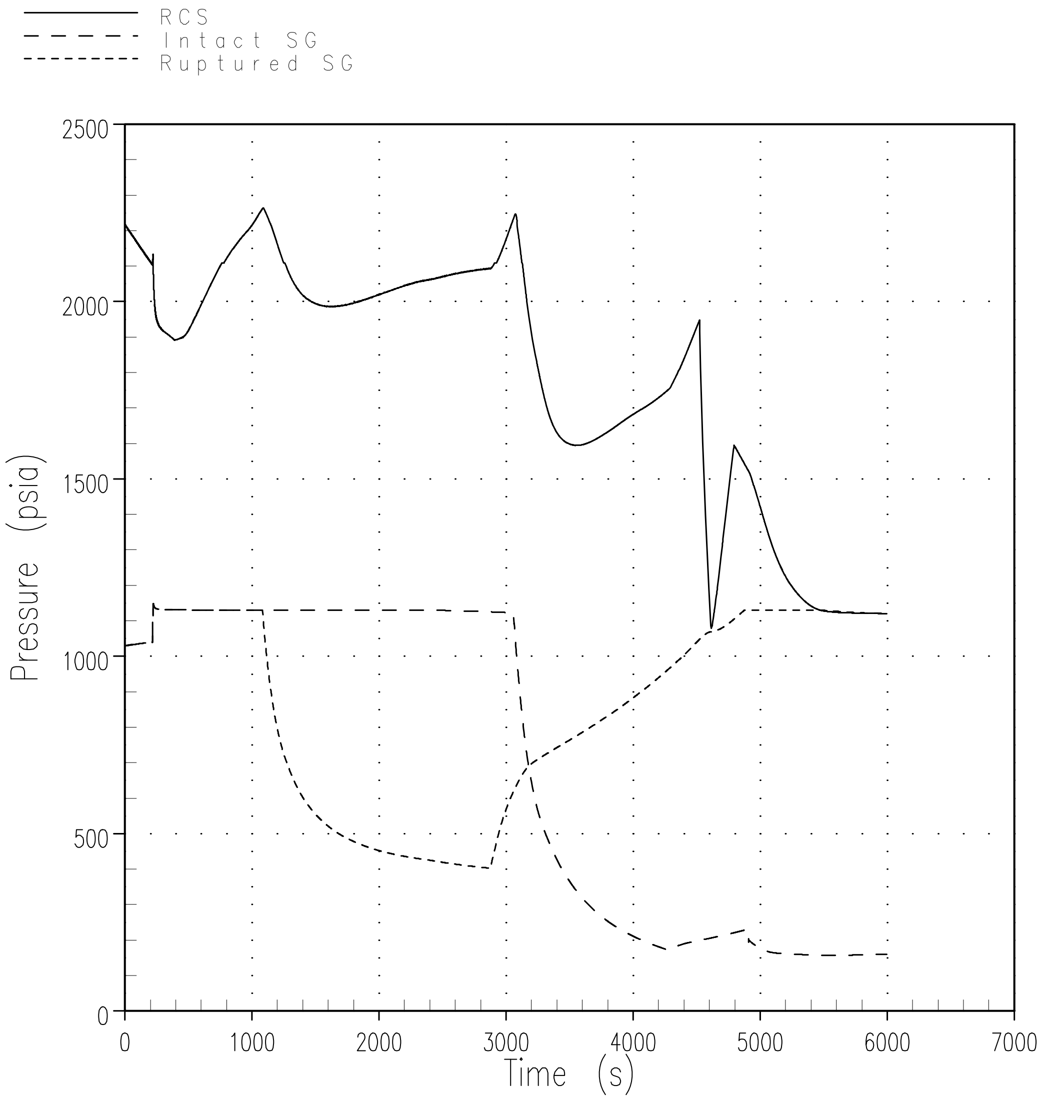

**Figure III-2 RCS and Secondary Pressure –
Unit 1 Input to Radiological Consequences Analysis**

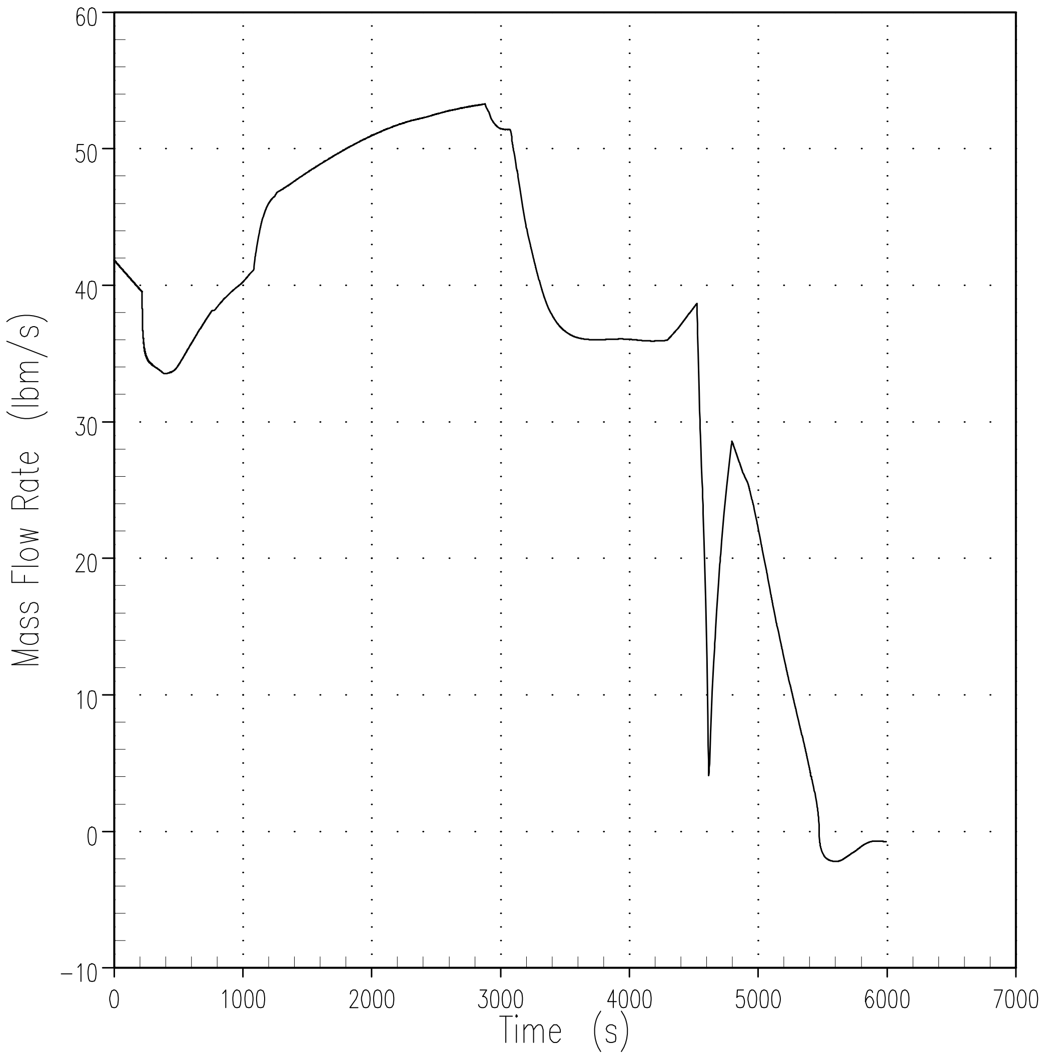

**Figure III-3 Primary to Secondary Break Flow –
Unit 1 Input to Radiological Consequences Analysis**

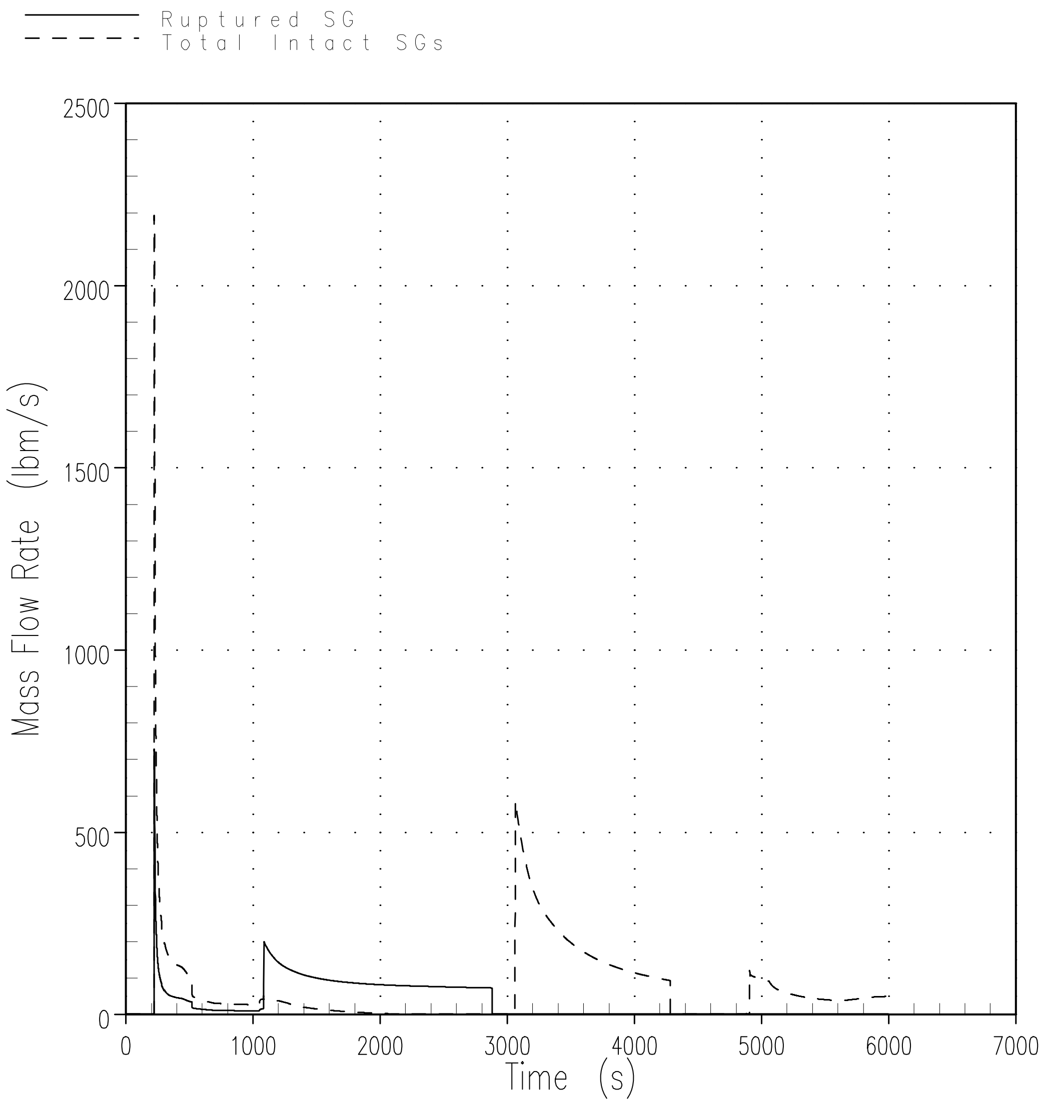

**Figure III-4 SG Mass Release Rate to the Atmosphere –
Unit 1 Input to Radiological Consequences Analysis**

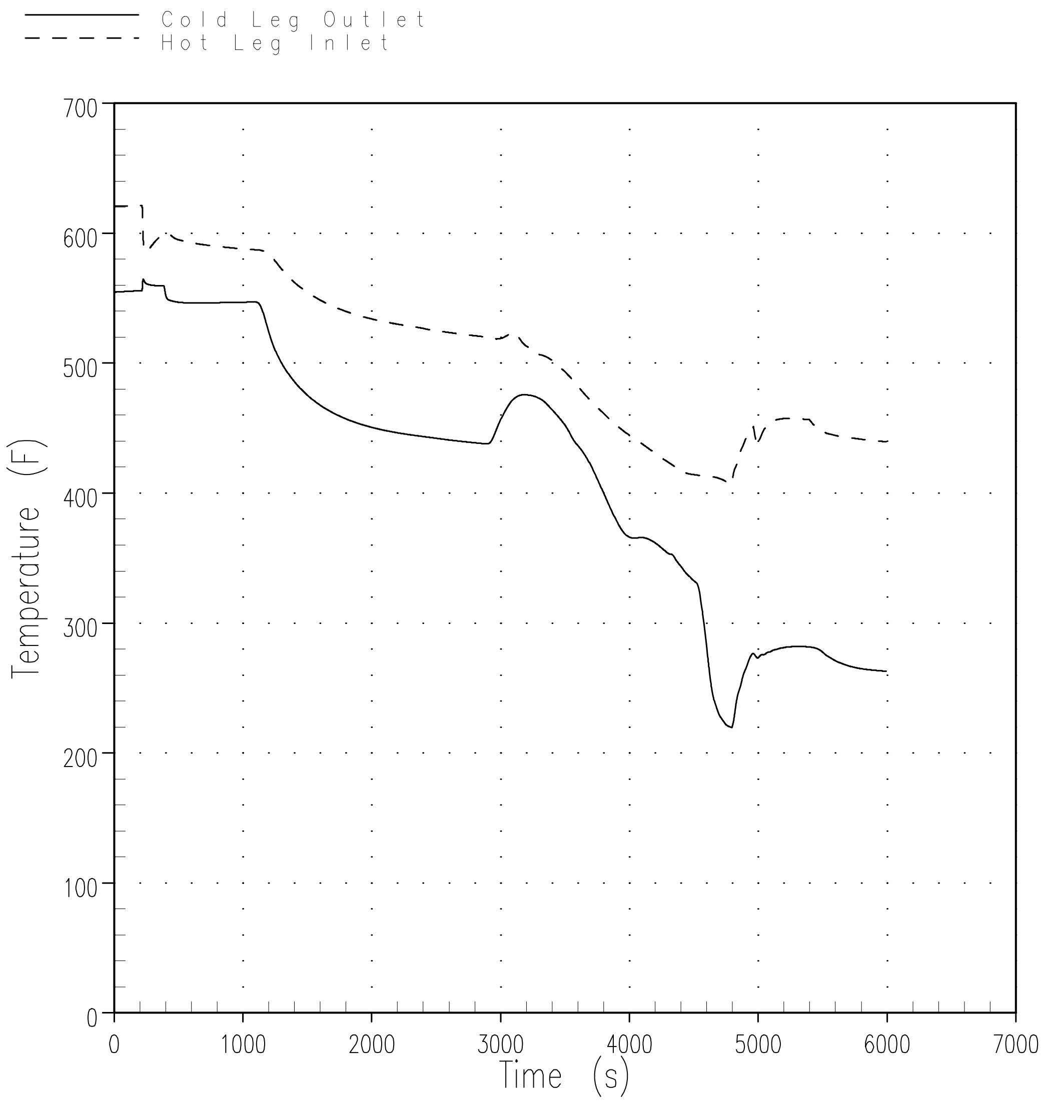

Figure III-5 Ruptured Loop Hot & Cold Leg Temperatures – Unit 1 Input to Radiological Consequences Analysis

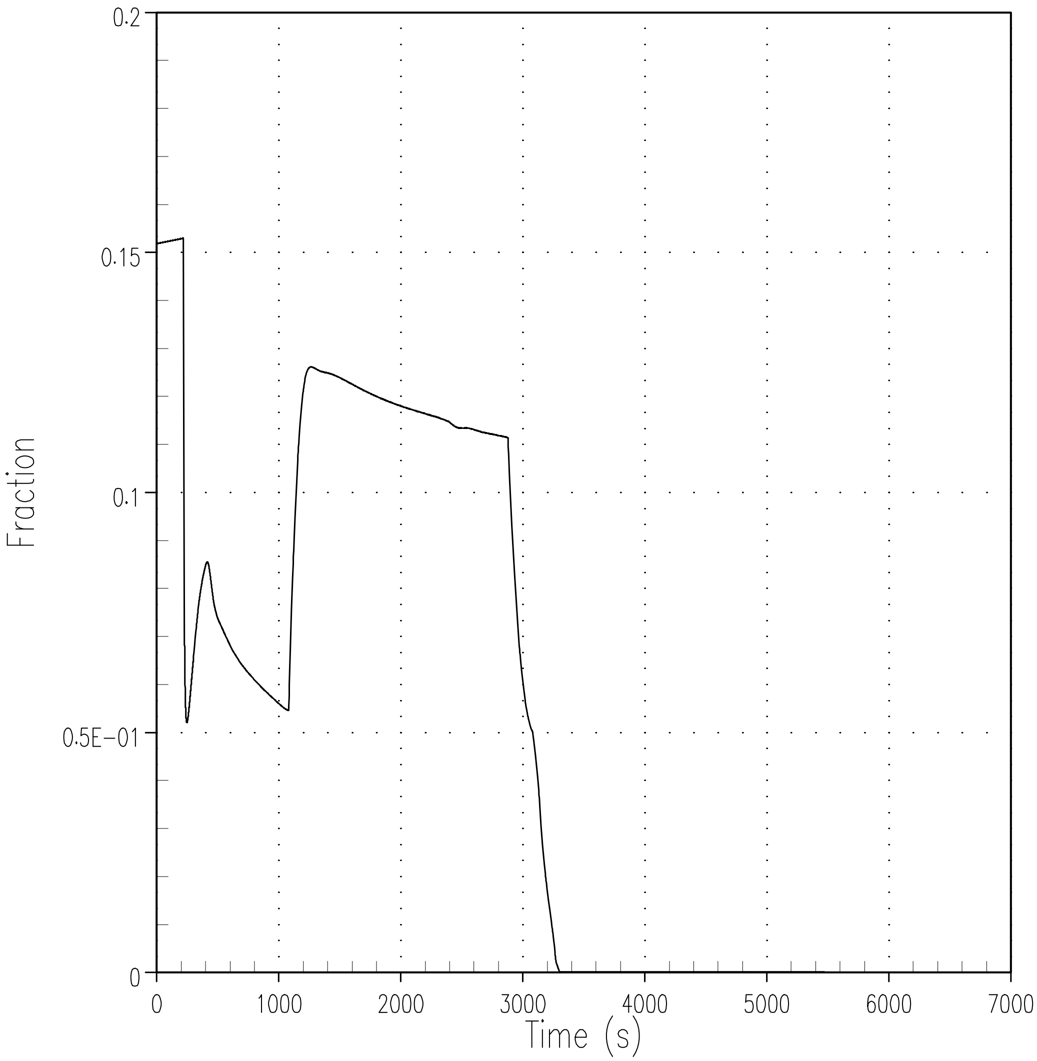

**Figure III-6 Break Flow Flashing Fraction –
Unit 1 Input to Radiological Consequences Analysis**

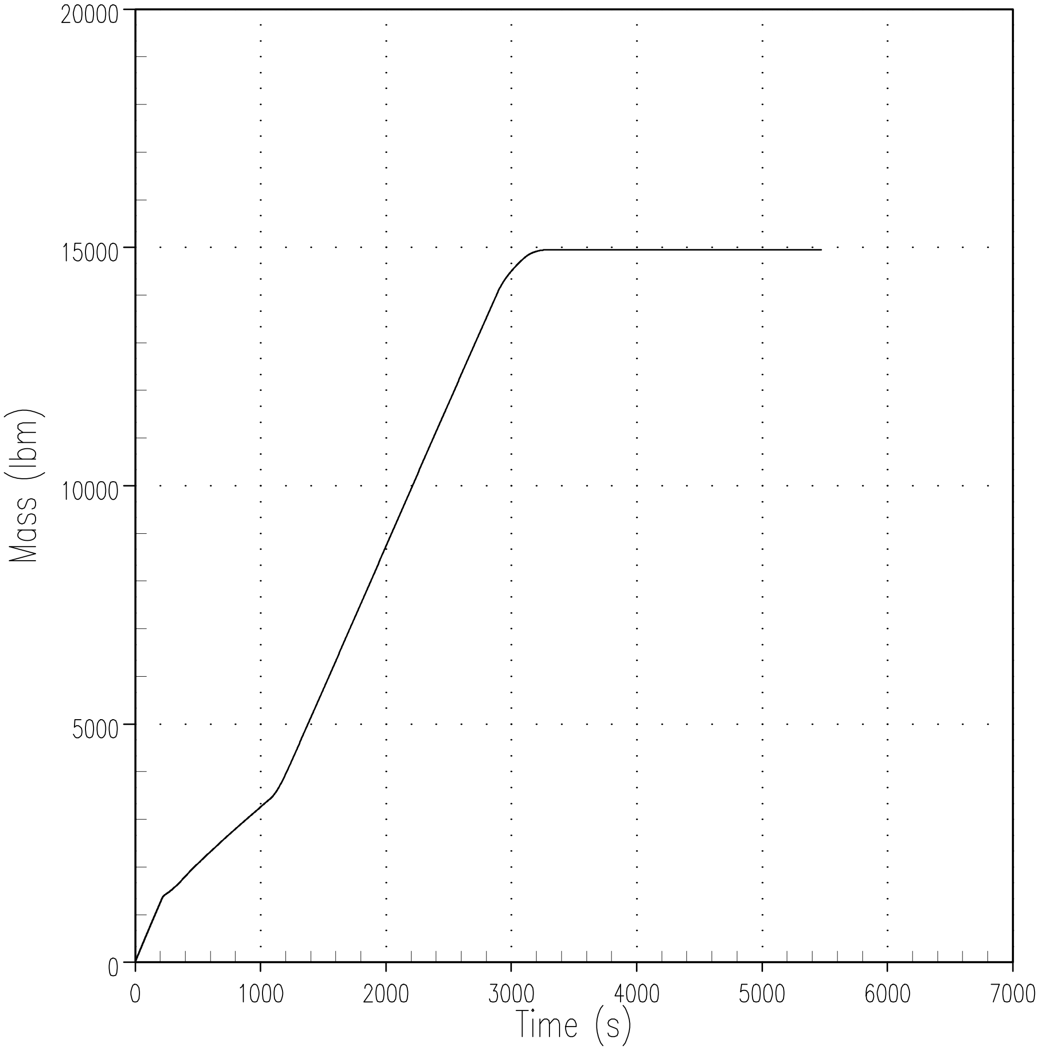

**Figure III-7 Total Flashed Break Flow –
Unit 1 Input to Radiological Consequences Analysis**

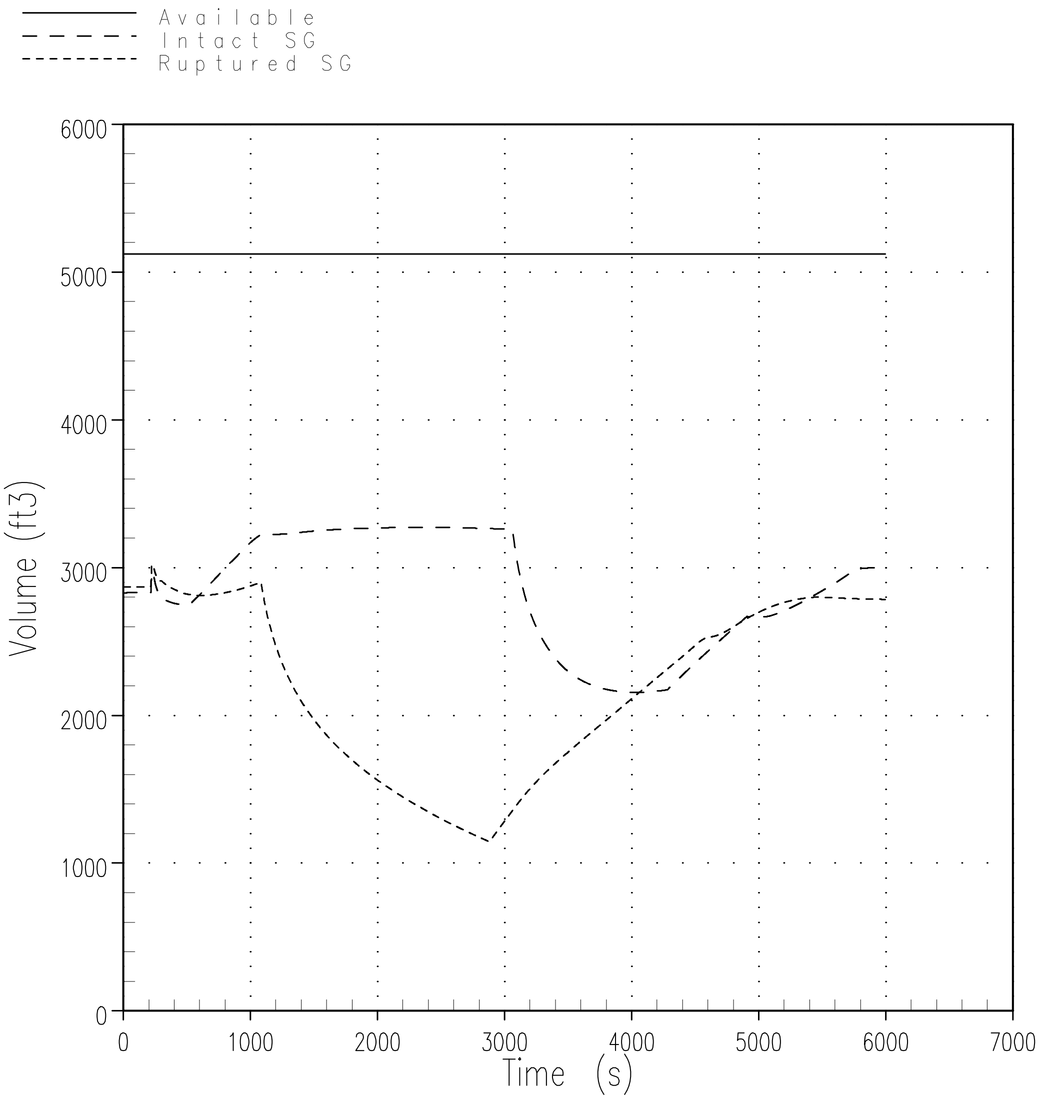

Figure III-8 SG Water Volumes –
Unit 1 Input to Radiological Consequences Analysis

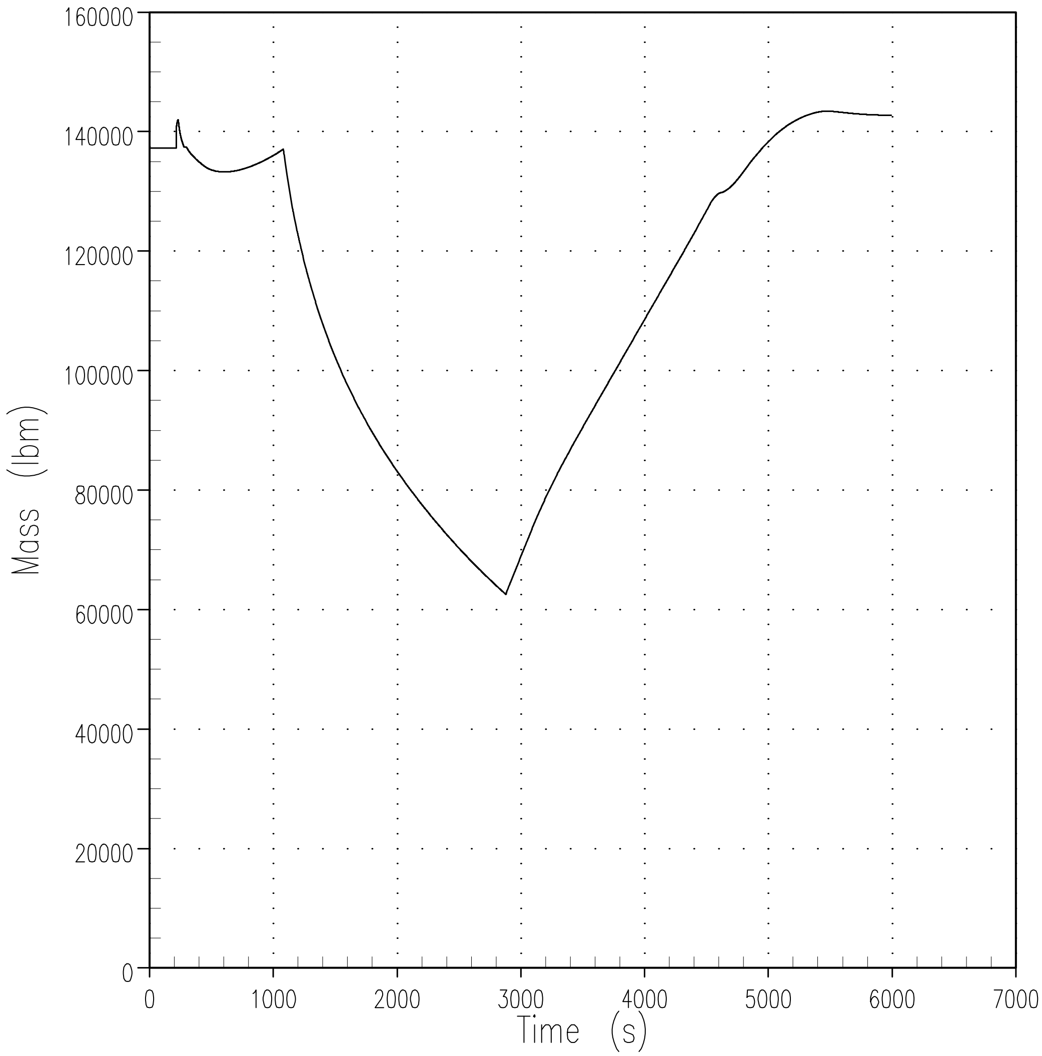

**Figure III-9 Ruptured SG Fluid Mass –
Unit 1 Input to Radiological Consequences Analysis**

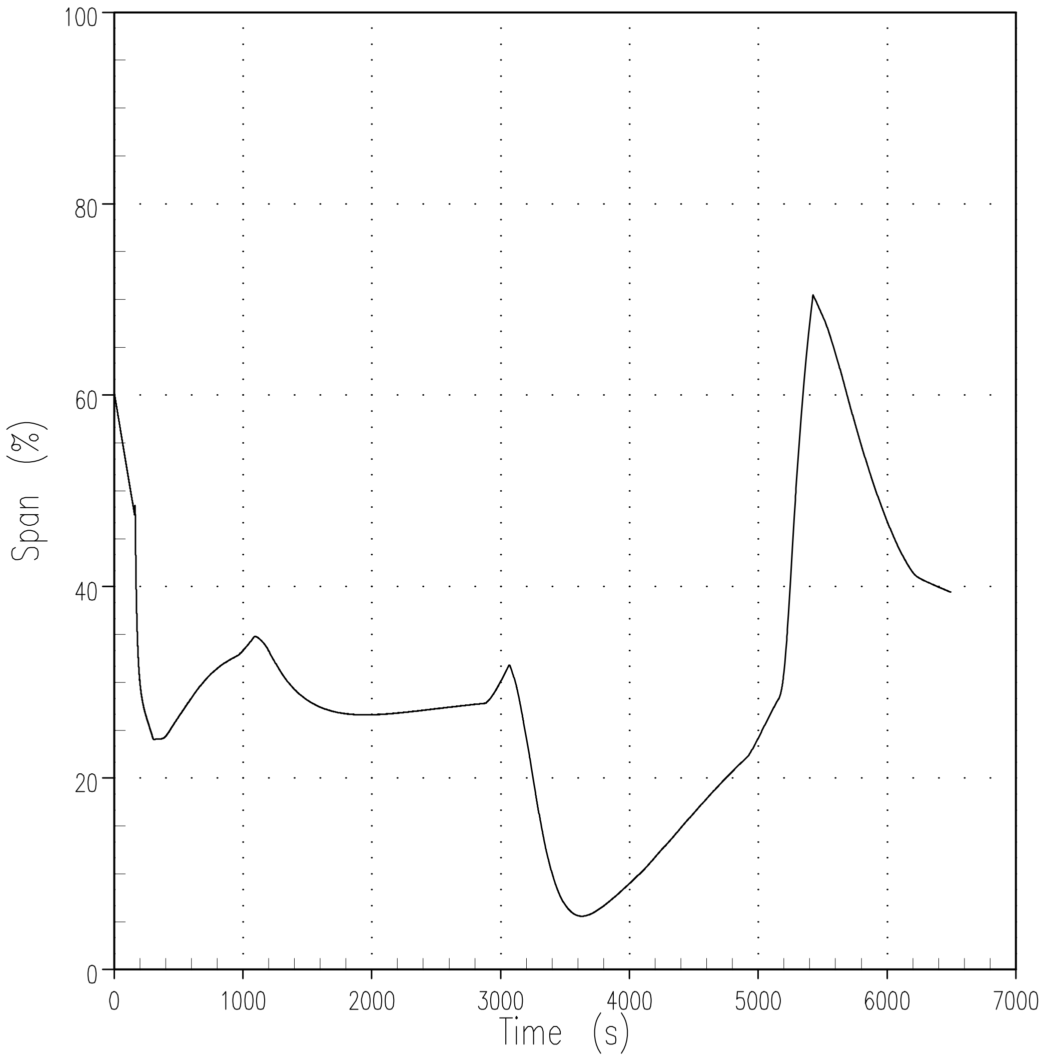

Figure III-10 Pressurizer Level –
Unit 2 Input to Radiological Consequences Analysis

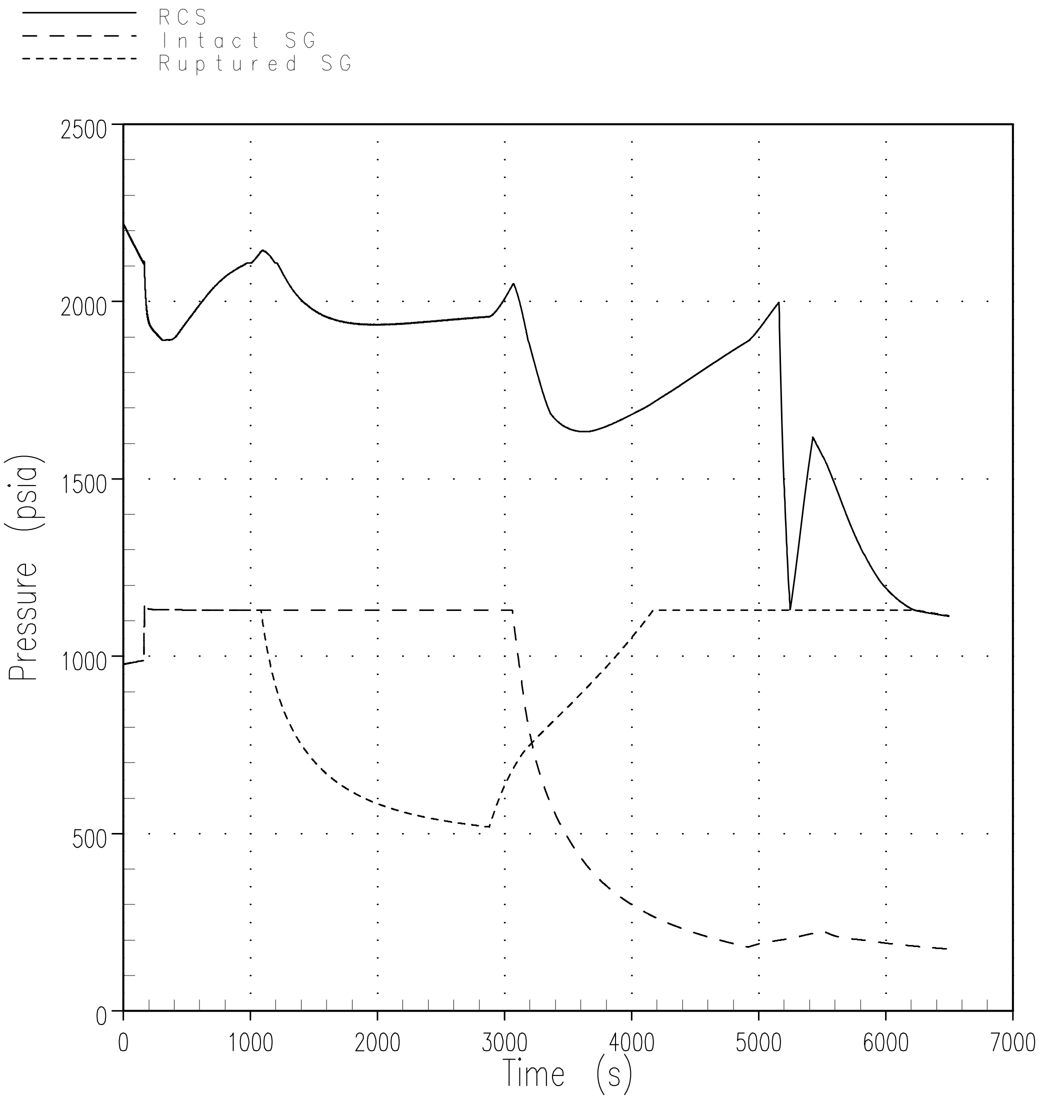

**Figure III-11 RCS and Secondary Pressure –
Unit 2 Input to Radiological Consequences Analysis**

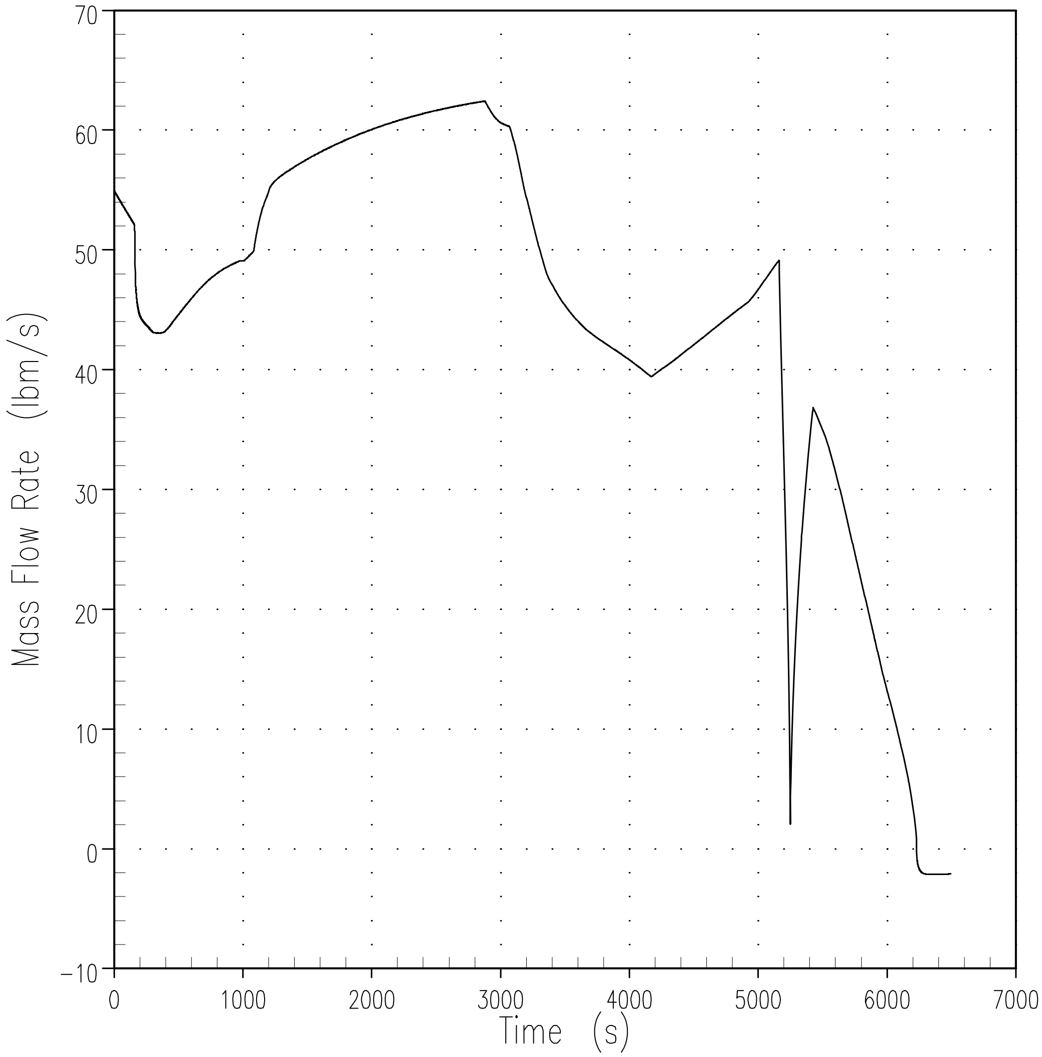

Figure III-12 Primary to Secondary Break Flow – Unit 2 Input to Radiological Consequences Analysis

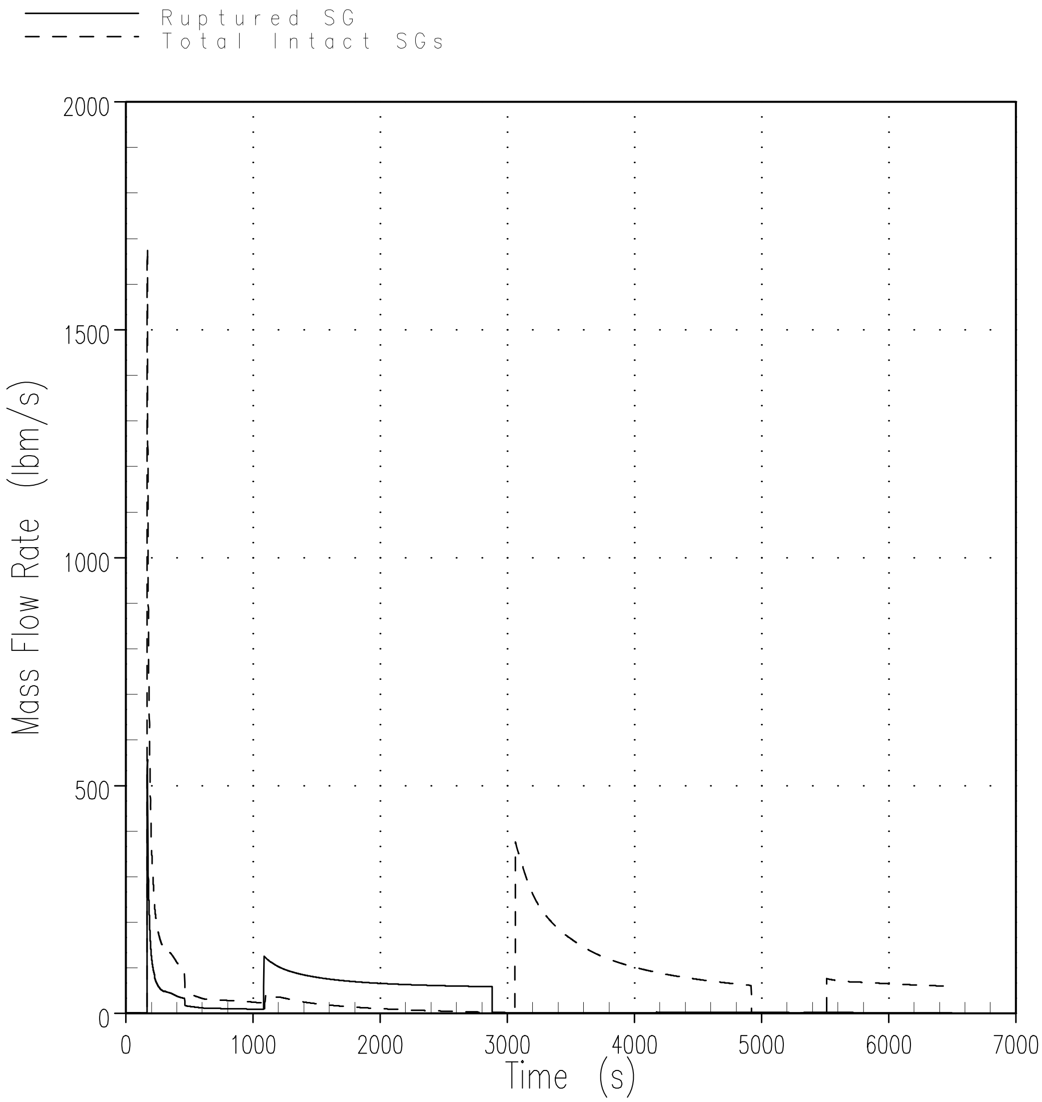

**Figure III-13 SG Mass Release Rate to the Atmosphere –
Unit 2 Input to Radiological Consequences Analysis**

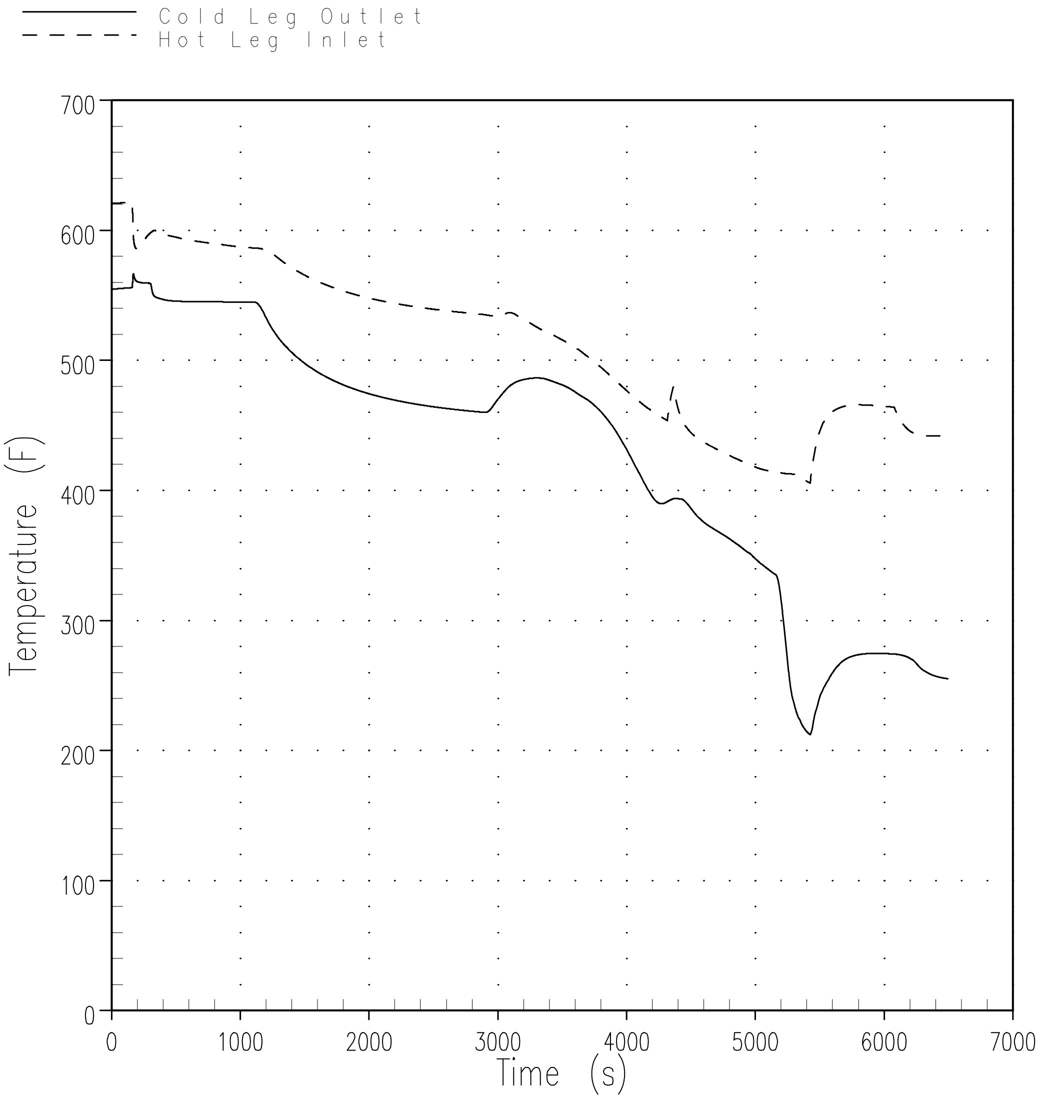

Figure III-14 Ruptured Loop Hot & Cold Leg Temperatures – Unit 2 Input to Radiological Consequences Analysis

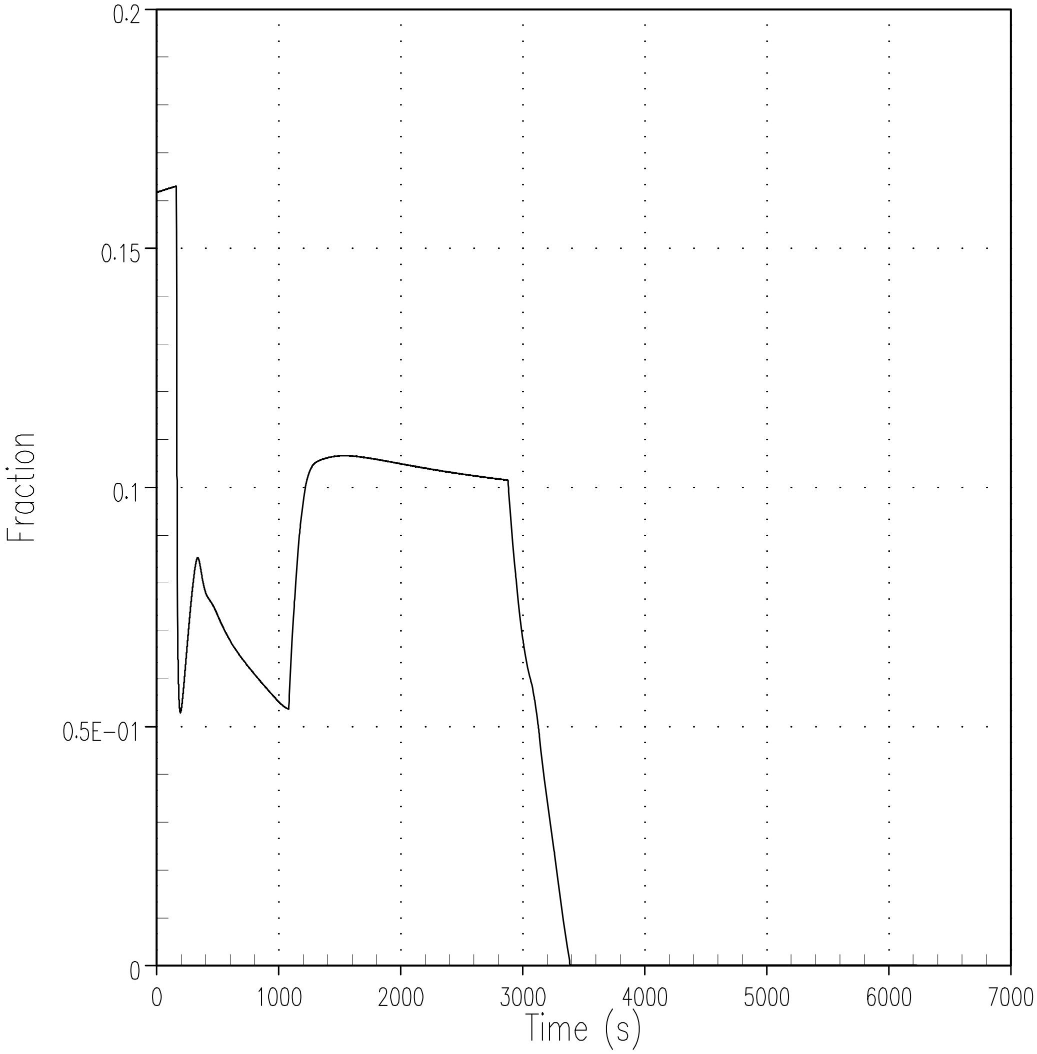

**Figure III-15 Break Flow Flashing Fraction –
Unit 2 Input to Radiological Consequences Analysis**

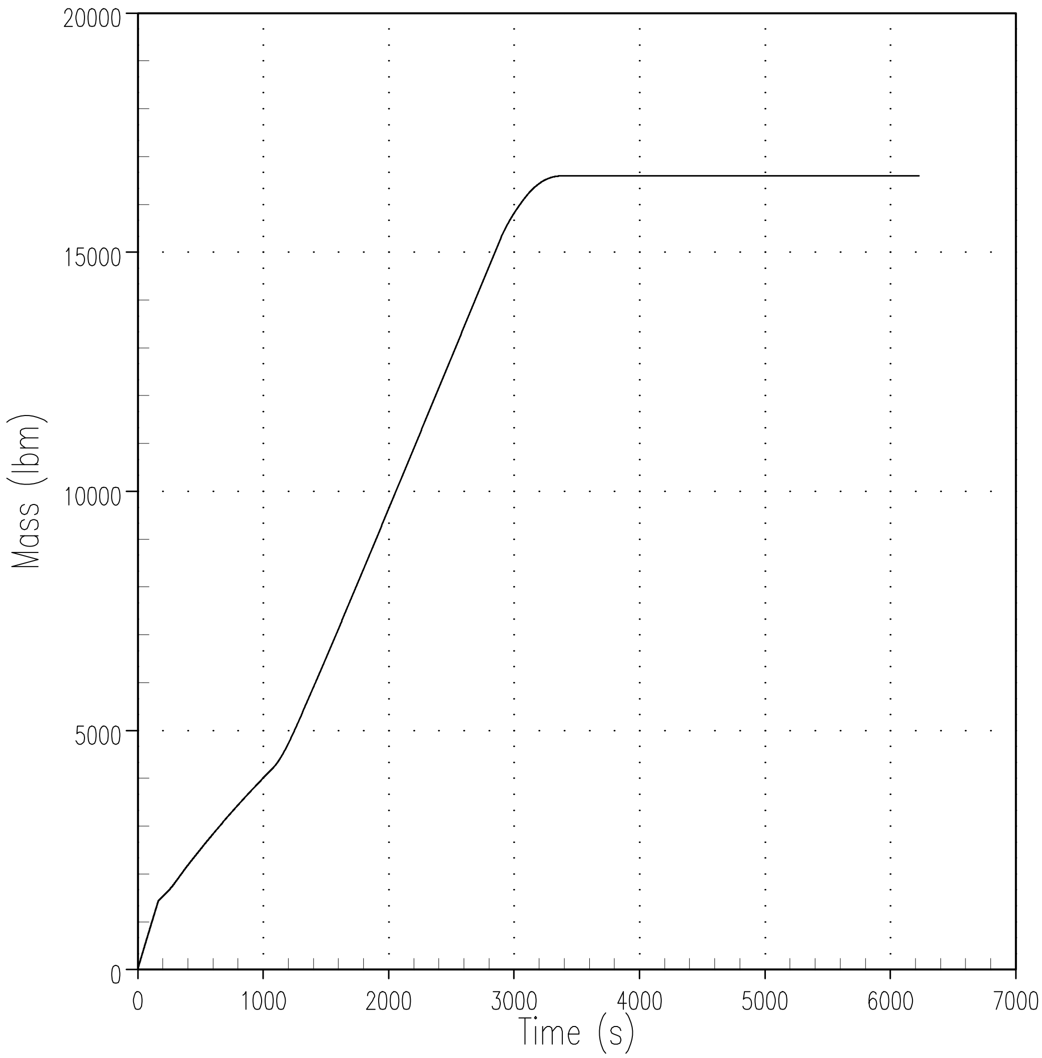

**Figure III-16 Total Flashed Break Flow –
Unit 2 Input to Radiological Consequences Analysis**

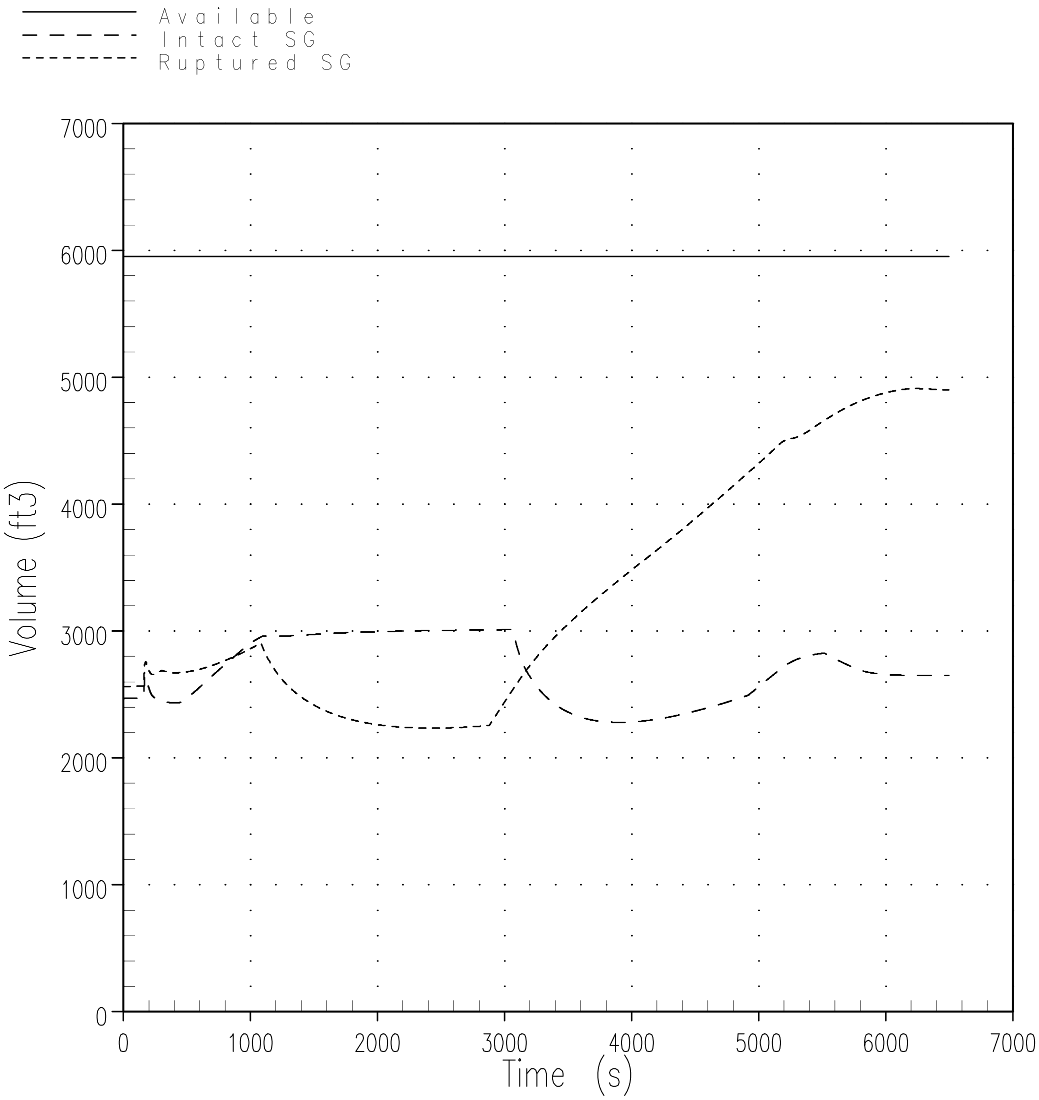

**Figure III-17 SG Water Volumes –
Unit 2 Input to Radiological Consequences Analysis**

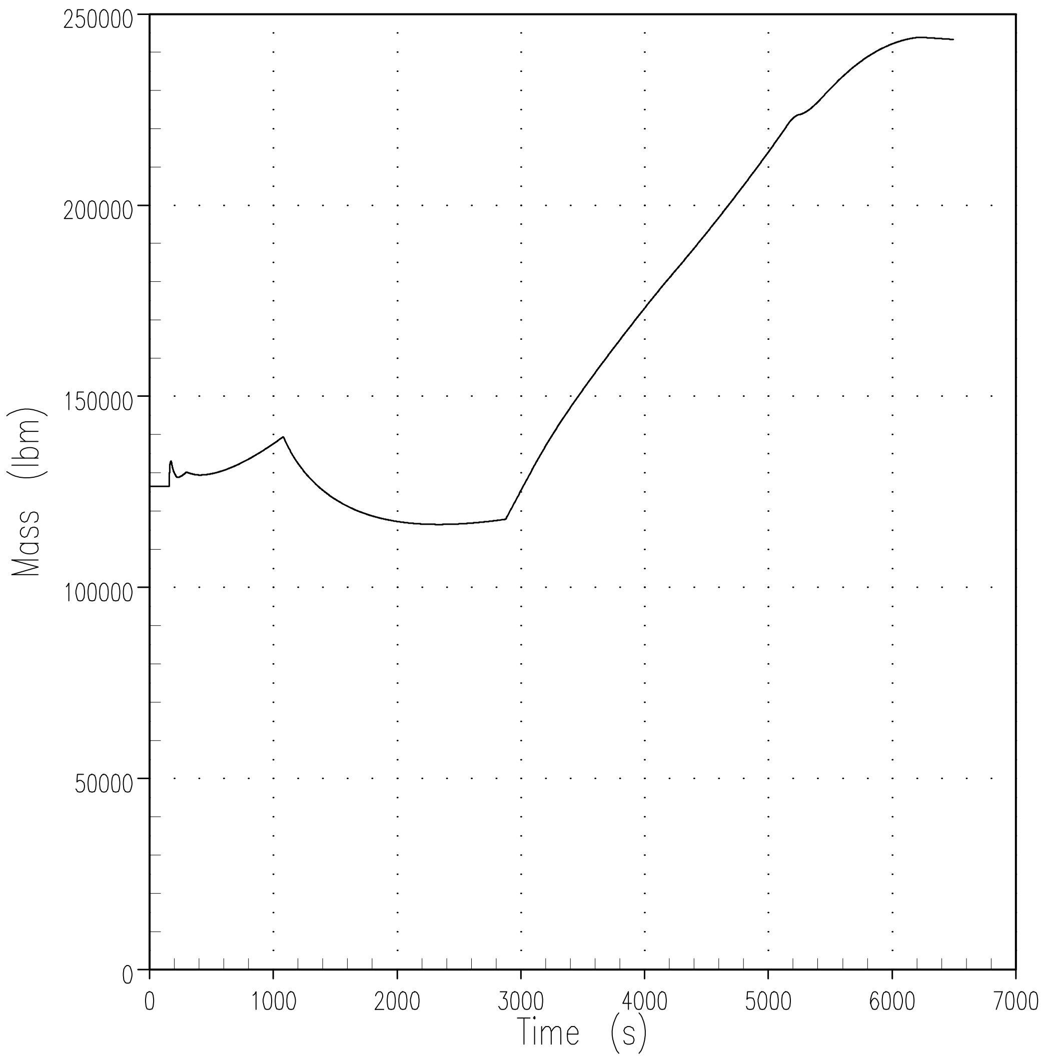

**Figure III-18 Ruptured SG Fluid Mass –
Unit 2 Input to Radiological Consequences Analysis**

IV RADIOLOGICAL CONSEQUENCES ANALYSES

IV.1 Introduction

The evaluation of the radiological consequences of an SGTR assumes that the reactor has been operating at the Technical Specification limits for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radionuclides in the secondary coolant. During an SGTR, activity from the primary coolant enters the secondary coolant via the ruptured tube. Primary to secondary leakage is released to the atmosphere through the SG safety valves or PORVs and the condenser air ejector exhaust.

The amount of radioactivity released to the environment due to an SGTR depends upon primary and secondary coolant activities, iodine spiking effects, primary to secondary break flow, break flow flashing, 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 SG and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively modeled for a design basis double-ended rupture of a single SG tube.

Section III of this report presents the mass releases for the SGTR event assuming failure of the PORV on the ruptured SG. The resulting offsite and control room doses were calculated as presented in this section. The SGTR radiological consequences analyses were performed following the guidance provided in Regulatory Guide (RG) 1.183 (Reference 7) and using the RADTRAD code, Version 3.03 (Reference 8). The calculations determined the doses based on a pre-accident iodine spike and on an accident initiated iodine spike.

The following sections discuss the methods and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results.

IV.2 Input Parameters and Assumptions

Major assumptions and parameters are discussed in this section and summarized in Tables IV-1 and IV-1a.

IV.2.A Mass Transfer Assumptions

Break flow, flashed break flow, and steam releases from the intact and ruptured SGs were modeled using data from the thermal and hydraulic analyses in Section III of this report. The dose analyses conservatively added 10% to the mass transfer data presented in Section III. This margin was added to allow flexibility in addressing minor changes in the transient calculation without requiring a reanalysis of the doses. The 10% increase in mass transfers results in approximately a 10% increase in the calculated doses. In addition, approximately 5% margin was added to the total calculated doses to arrive at the doses reported. This additional margin was included to allow flexibility in addressing minor changes in the dose analysis input without requiring a reanalysis or changes in the reported dose.

The RCS and intact SG masses were modeled at the initial values listed in Table III-6. For Unit 1 the ruptured SG mass was modeled as the average of the initial and minimum values listed in Table III-6, but for Unit 2 the ruptured SG mass was modeled at the initial value listed in Table III-6. The difference in approach for the two analyses was explained in Section III.5.B.

The primary to secondary leak rate to the three intact SGs was assumed to be 0.218 gpm/SG. The leakage to the intact SGs was assumed to persist for the duration of the accident. Cold conditions were assumed in selecting the density of 62.4 lbm/ft^3 to model the mass flow rate of 5.46 lbm/min for this leakage.

IV.2.B Source Term Assumptions

The radionuclide concentrations in the primary coolant and secondary coolant prior to and following the SGTR, were based on the following:

1. There is no fuel damage as a result of the postulated SGTR. The iodine concentrations in the RCS were based upon either a pre-accident iodine spike or an accident-initiated iodine spike as outlined in RG 1.183 (Reference 7).
    a. Pre-Accident Spike - A reactor transient occurred prior to the SGTR and raised the primary coolant iodine concentration to 60 μCi/gm Dose Equivalent (DE) I-131 which is the Technical Specification limit for iodine concentration excursions beyond equilibrium conditions.
    b. Accident-Initiated Spike - The primary coolant iodine concentration was initially at the equilibrium operation Technical Specification limit of 1.0 μCi/gm DE I-131. Coincident with the SGTR, an iodine spike was initiated. This spike increased the iodine release rate from the fuel to the coolant to a value 335 times greater than the release rate corresponding to the initial RCS iodine concentration. The spike was assumed to continue until 8 hours after the start of the event.
2. The initial secondary coolant iodine concentration was 0.1 μCi/gm DE I-131. This is the Technical Specification limit.
3. The chemical form of iodine released from the SGs to the environment was 97% elemental and 3% organic consistent with the guidance of RG 1.183 (Reference 7).
4. The initial concentration of noble gases in the RCS was at the Technical Specification limit of 603 μCi/gm DE Xe-133.
5. No noble gases were present in the secondary coolant at the start of the event since retention of noble gases in water is negligible.

The iodine and noble gas concentration data and the equilibrium iodine appearance rates are presented in Table IV-2.

IV.2.C Additional Assumptions for Dose Calculations

Offsite power was assumed to be lost at reactor trip. This assumption was used in the thermal-hydraulic analyses (Section III) to maximize break flow and steam release through the ruptured SG PORV. Prior to reactor trip, a condenser iodine partition coefficient of 100 was assumed. This condenser partition coefficient is supported by Table 1 of WCAP-8159 (Reference 9) which identifies a total iodine separation factor of 10,000 for the condenser vent. With a partition coefficient of 100 associated with the steam releases from the SG (discussed below), this leaves a factor of 100 associated with the condenser. After reactor trip and loss of offsite power, flow to the condenser was isolated.

The iodine transport model used in these analyses accounted for break flow flashing, steaming, and partitioning. The model assumed that a fraction of the iodine carried by the break flow became airborne immediately due to flashing. All of the iodine in the flashed break flow was assumed to be transferred instantly out of the SG. The fraction of iodine in the break flow that was not assumed to become airborne immediately mixed with the secondary coolant and was assumed to become airborne at a rate proportional to the steaming rate. The water/steam iodine partition coefficient of 100 from RG 1.183 (Reference 7) was used.

All noble gases in the break flow and primary to secondary leakage were assumed to be transferred instantly out of the SG to the atmosphere.

Decay of radioactivity was credited in the fuel, the RCS and the SGs prior to release. No credit was taken for the radioactive decay during release and transport or for cloud depletion by ground deposition during transport after release to the environment. Decay of activity in the control room (CR) was credited.

Atmospheric dispersion factors (χ/Q) for the exclusion area boundary (EAB), the low population zone (LPZ), and the CR that were used to model the spread of the released activity from the release point to the receptor are presented in Table IV-3.

The breathing rate of 3.5E-4 m^3/sec was applied for all dose locations for the assumed 8-hour release consistent with the guidance of RG 1.183 (Reference 7).

Total effective dose equivalent (TEDE) doses were calculated at the EAB, the LPZ and in the CR. TEDE is the sum of the committed effective dose equivalent (CEDE) dose from inhalation and effective dose equivalent (EDE) dose from external exposure. The inhalation dose conversion factors (DCFs) from Reference 10 and external exposure DCFs from Reference 11 were used in determining the dose resulting from the released activity. Values used are presented in table IV-4. The EAB doses were determined for the limiting two-hour time interval which is the first two hours following the SGTR since break flow was terminated within the first two hours. The LPZ doses were determined for the 8-hour duration of releases defined by the RHR cut in time used in the thermal and hydraulic analyses in Section III.

The CR ventilation system was assumed to be placed in the emergency mode of operation 30 minutes after the initiation of the SGTR. Although all releases were terminated when the RHR system was put in service, the calculation of CR doses was continued for 30 days to account for additional doses due to continued occupancy of the CR. The inflow (filtered and unfiltered) to the CR and the CR filtered recirculation flow were used in the calculation of the CR doses. CR parameters used in the analyses are presented in Table IV-5.

IV.3 Acceptance Criteria

The offsite dose limits for the two iodine spike scenarios are specified in RG 1.183 (Reference 7) and Standard Review Plan (SRP) 15.0.1 (Reference 12). The doses at the EAB and the LPZ for an SGTR with an assumed pre-accident iodine spike must meet the 10 CFR 50.67 limit of 25 rem TEDE, while the

EAB and LPZ doses for an SGTR with an assumed accident-initiated iodine spike must meet the limit of 2.5 rem TEDE specified in RG 1.183. The EAB doses are calculated for the limiting two hours. The LPZ doses are calculated up to the time releases are terminated, which is the RHR cut in time of 8 hours used in the thermal and hydraulic analyses in Section III.

The CR dose limit of 5 rem TEDE is specified in SRP 6.4 (Reference 13) based on 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. The CR doses are calculated for 30 days.

IV.4 Results

The doses for both units for all locations are presented in Table IV-6 compared to the limits. Note that the doses calculated for Unit 2 are higher than those calculated for Unit 1.

IV.5 Conclusions

The doses at the EAB, LPZ, and in the CR resulting from an SGTR are within the applicable limits.

Table IV-1 Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
Reactor Coolant System Iodine	
Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gm DE I-131. These are 60 times the values given in Table IV-2 which are based on 1.0 μCi/gm DE I-131.
Accident-Initiated Spike	Initial primary coolant iodine activities based on 1.0 μCi/gm DE I-131. The iodine appearance rates for the accident-initiated spike are 335 times the equilibrium appearance rates which are given in Table IV-2. The spike continues until 8 hours from the start of the event.
Noble Gas Activity	Primary coolant noble gas activities based on 603 μCi/gm DE Xe-133. Values are given in Table IV-2. No noble gases are contained in the secondary coolant.
Secondary Coolant System Iodine	Initial secondary coolant iodine activity based on 0.1 μCi/gm DE I-131. Values are given in Table IV-2.
Iodine Chemical Fractions	97% elemental, 3% organic, no particulates
RCS Mass	Constant at initial values given in Table III-6.
Intact SGs Mass	Constant at initial values given in Table III-6.
Ruptured SG Mass	Unit 1: Constant at average of initial and minimum values given in Table III-6. Unit 2: Constant at initial value given in Table III-6.
Transient Timing	Times are given in Table III-1.
Ruptured SG Transient Release Data	
Rupture Flow*	Unit 1: Table III-2 and Figure III-3 Unit 2: Table III-4 and Figure III-12
Flashed Rupture Flow*	Unit 1: Table III-2 and Figure III-7 Unit 2: Table III-4 and Figure III-16
Steam Releases*	Unit 1: Table III-3 and Figure III-4 Unit 2: Table III-5 and Figure III-13

* The dose analyses conservatively added 10% to the mass transfer data presented in these tables and figures. This margin was added to allow flexibility in addressing minor impacts on the transient calculation without requiring a reanalysis.

Table IV-1 *(continued)* Summary of Parameters Used in Evaluating the Radiological Consequences of a Steam Generator Tube Rupture	
Intact SGs Transient Release Data	
Primary to Secondary Leakage	0.218 gpm/SG (5.46 lbm/min)
Steam Releases*	Unit 1: Table III-3 and Figure III-4 Unit 2: Table III-5 and Figure III-13
Iodine Partition Coefficients	
Condenser	100 (Applied until reactor trip and loss of offsite power)
Steam Release from SGs	100
Flashed Break Flow Release from Ruptured SG	1.0
Atmospheric Dispersion Factors	Values are given in Table IV-3.
Breathing Rate	3.5E-4 m^3/sec for all time intervals at all dose locations.
Control Room Modeling	See Table IV-5.

* The dose analyses conservatively added 10% to the mass transfer data presented in these tables and figures. This margin was added to allow flexibility in addressing minor impacts on the transient calculation without requiring a reanalysis.

Table IV-1a Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter	Current AOR Input Values	Reanalysis Input Values	Comment
Dose Model			
Regulatory guidance	RG 1.183 Rev 0	RG 1.183 Rev 0	No change
Nuclide Parameters			
Nobles gases Kr-85m	1.80 uCi/g	1.80 uCi/g	No change
Kr-85	7.11 uCi/gram	7.11 uCi/gram	No change
Kr-87	1.15 uCi/gram	1.15 uCi/gram	No change
Kr-88	3.35 uCi/gram	3.35 uCi/gram	No change
Xe-131m	N/A	3.31 uCi/gram	Not originally modeled because not defined in standard 60 nuclides in RADTRAD Users manual, Table 1.4.3.2-3, published December 1997. This new proposed analysis models this value.
Xe-133m	N/A	3.65 uCi/gram	Not originally modeled because not defined in standard 60 nuclides in RADTRAD Users manual, Table 1.4.3.2-3, published December 1997. This new proposed analysis models this value.
Xe-133	251.0 uCi/gram	251.0 uCi/gram	No change
Xe-135m	N/A	0.49 uCi/gram	Not originally modeled because not defined in standard 60 nuclides in RADTRAD Users manual, Table 1.4.3.2-3, published December 1997. This new proposed analysis models this value.

Table IV-1a *(continued)* Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter	Current AOR Input Values	Reanalysis Input Values	Comment
Xe-135	7.72 uCi/gram	7.72 uCi/gram	No change
Xe-138	N/A	0.66 uCi/gram	Not originally modeled because not defined in standard 60 nuclides in RADTRAD Users manual, Table 1.4.3.2-3, published December 1997. This new analysis models this value.
Iodine I-131	0.742 uCi/gram	0.742 uCi/gram	No change
I-132	0.979 uCi/gram	0.979 uCi/gram	No change
I-133	1.35 uCi/gram	1.35 uCi/gram	No change
I-134	0.243 uCi/gram	0.243 uCi/gram	No change
I-135	0.842 uCi/gram	0.842 uCi/gram	No change
Offsite X/Q			All offsite X/Q values were updated for finer wind speed categories per RG 1.23 Revision 1. This was also a commitment per RS-06-019.
Exclusion Area Boundary			
0-2 hours	5.36E-04 sec/m^3	6.18E-04 sec/m^3	
Low Population Zone			
0-2 hours	9.32E-05 sec/m^3	1.10E-04 sec/m^3	
2-8 hours	4.50E-05 sec/m^3	5.13E-05 sec/m^3	
8-24 hours	3.12E-05 sec/m^3	3.51E-05 sec/m^3	
24-96 hours	1.41 E-05 sec/m^3	1.53E-05 sec/m^3	
96-720 hours	4.54E-06 sec/m^3	4.68E-06 sec/m^3	

Table IV-1a *(continued)* Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter Current	AOR Input Values	Reanalysis Input Values	Comment
Activity Related Data			
Maximum nominal reactor coolant system (RCS) iodine activity (uCi/gram Dose Equivalent (DE) I-131)	1.0 uCi/gram	1.0 uCi/gram	No change
Maximum RCS iodine spike activity	60 uCi/gram Pre-accident	60 uCi/gram Pre-accident	No change, per RG 1.183
Accident initiated Iodine spike factor	335 times greater than release rate equilibrium values	335 times greater than release rate equilibrium values	
Maximum nominal secondary iodine activity (uCi/gram DE I-131)	0.1 uCi/gram	0.1 uCi/gram	No change
Design fuel defect level for noble gas activity	1%	1%	No change
Equilibrium Iodine Appearance (Release) Rate			No change
I-131	0.416 Ci/min	0.416 Ci/min	No change
I-132	1.754 Ci/min	1.754 Ci/min	No change
I-133	0.924 Ci/min	0.923 Ci/min	Slight change due to rounding
I-134	0.926 Ci/min	0.926 Ci/min	No change
I-135	0.826 Ci/min	0.826 Ci/min	No change
Accident initiated iodine spike	8 hours	8 hours	No change

Table IV-1a *(continued)* Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter Current	AOR Input Values	Reanalysis Input Values	Comment
Thermal Hydraulic Transient Data			
Time to reach RHR cut-in conditions	8 hours	8 hours	No change
Maximum primary-to-secondary leakage to intact SGs (gpm/SG)	0.218	0.218	No change
Condenser iodine partition factor	0.01	0.01	No change
Sequence of events, break-flow, flashed break flow, and steam releases	See UFSAR Tables 15.6-6, 6a and 6b	See Tables III-1, 2, 3, 4, 5 and 6	The sequence of events and subsequent release rates were changed from the existing model as discussed in Section III.
Control Room Modeling			
Control room volume	200,000 ft^3	200,000 ft^3	No change
Normal mode unfiltered makeup flow (cfm)	6424	6424	No change
Normal mode filtered makeup flow (cfm)	0	0	No change
Normal mode filtered recirculation flow (cfm)	0	0	No change
Emergency mode unfiltered makeup flow (cfm)	0	0	No change
Emergency mode filtered makeup flow (cfm)	8575	8575	No change
Emergency mode filtered recirculation flow (cfm)	39,150	39,150	No change

Table IV-1a *(continued)* Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter Current	AOR Input Values	Reanalysis Input Values	Comment
Control room unfiltered inleakage (cfm)	500	500	The initial NRC approved analysis assumed 1000 cfm was changed to 500 cfm under a separate LAR. The corrections resulted in a decrease in SGTR reported dose and thus the calculation was not submitted as part of that amendment but updated and implemented under 50.59. The approved LAR for this change was "Braidwood Station, Units 1 and 2, and Byron Station Unit Nos. 1 and 2- Issuance of Amendments Re: Revised Application of Alternative Source Term" dated February 5th, 2009.
Control room filter efficiencies (%)			
Elemental iodine	95	95	No change
Organic iodine	95	95	No change
Particulates (Aerosol)	99	99	No change
Control room recirculation filter efficiencies			
Elemental iodine	90	90	No change
Organic iodine	90	90	No change

Table IV-1a *(continued)* Summary of Comparison of AST Parameters Used in Steam Generator Tube Rupture Dose Analysis

Parameter Current	AOR Input Values	Reanalysis Input Values	Comment
Particulates (Areosol)	0	0	The NRC approved analysis assumed 80% efficiency which was incorrect and correct under a separate LAR. The corrections resulted in a decrease in SGTR reported dose and thus the calculation was not submitted as part of that amendment but updated and implemented under 50.59. The approved LAR for this change was "Braidwood Station, Units 1 and 2, and Byron Station Unit Nos. 1 and 2- Issuance of Amendments Re: Revised Application of Alternative Source Term" dated February 5th, 2009.
Time control room heating, ventilation, and air conditioning (HVAC) emergency mode is initiated after start of design basis accident	0.5 hours	0.5 hours	No change
Control room atmospheric dispersion factors			
0-0.5 hours	1.77E-03 sec/m^3	1.77E-03 sec/m^3	No change
0.5-2 hours	8.14E-04 sec/m^3	8.14E-04 sec/m^3	No change
2-8 hours	6.98E-04 sec/m^3	6.98E-04 sec/m^3	No change
8-24 hours	3.12E-04 sec/m^3	3.12E-04 sec/m^3	No change
24-96 hours	1.95E-04 sec/m^3	1.95E-04 sec/m^3	No change
96-720 hours	1.67E-04 sec/m^3	1.67E-04 sec/m^3	No change

Table IV-2 Specific Activities in the Primary Coolant and Associated Iodine Appearance Rates and Specific Activities in the Secondary Coolant

Nuclide	RCS Concentration Based on 1.0 µCi/gm DE I-131 (µCi/gm)	Equilibrium Appearance Rate (Ci/min)	RCS Concentration Based on 603 µCi/gm DE Xe-133 (µCi/gm)	SG Concentration Based on 0.1 µCi/gm DE I-131 (µCi/gm)
I-131	0.742	0.416	-	0.0742
I-132	0.979	1.754	-	0.0979
I-133	1.35	0.923	-	0.135
I-134	0.243	0.926	-	0.0243
I-135	0.842	0.826	-	0.0842
Kr-85m	-	-	1.80	-
Kr-85	-	-	7.11	-
Kr-87	-	-	1.15	-
Kr-88	-	-	3.35	-
Xe-131m	-	-	3.31	-
Xe-133m	-	-	3.65	-
Xe-133	-	-	251.0	-
Xe-135m	-	-	0.49	-
Xe-135	-	-	7.72	-
Xe-138	-	-	0.66	-

Table IV-3 Atmospheric Dispersion Factors

Time Period	Exclusion Area Boundary (sec/m^3)	Low Population Zone (sec/m^3)	Control Room (sec/m^3)
0 to 0.5 hours	6.18E-4	1.10E-4	1.77E-3
0.5 to 2 hours	6.18E-4	1.10E-4	8.14E-4
2 to 8 hours	6.18E-4	5.13E-5	6.98E-4

Table IV-4 Dose Conversion Factors

Nuclide	Reference 11 EDE Dose Conversion Factor - Cloudshine ($Sv\text{-}m^3/Bq\text{-}sec$)	Reference 10 CEDE Dose Conversion Factor - Inhaled (Sv/Bq)
I-131	1.82E-14	8.89E-09
I-132	1.12E-13	1.03E-10
I-133	2.94E-14	1.58E-09
I-134	1.30E-13	3.55E-11
I-135	7.98E-14	3.32E-10
Kr-85m	7.48E-15	-
Kr-85	1.19E-16	-
Kr-87	4.12E-14	-
Kr-88	1.02E-13	-
Xe-131m	3.89E-16	-
Xe-133m	1.37E-15	-
Xe-133	1.56E-15	-
Xe-135m	2.04E-14	-
Xe-135	1.19E-14	-
Xe-138	5.77E-14	-

Table IV-5 Control Room Modeling	
Transition from Normal Mode Ventilation, to Emergency Mode	The CR ventilation emergency mode is initiated at 0.5 hours after the start of the accident.*
CR Volume	200,000 ft^3
CR Unfiltered In-Leakage	500 cfm
CR Unfiltered Makeup Flow	
Normal Mode	6424 cfm
Emergency Mode	0 cfm
CR Filtered Makeup Flow	
Normal Mode	0 cfm
Emergency Mode	8575 cfm
CR Filtered Recirculation Flow	
Normal Mode	0 cfm
Emergency Mode	39,150 cfm
CR Filter Efficiency	
Filtered Makeup Flow	95% for elemental and organic iodines
Filtered Recirculation	90% for elemental and organic iodines
CR χ/Q	Values are given in Table IV-3
CR Occupancy Factors	1.0 for the first day 0.6 from 1 to 4 days 0.4 after 4 days
* This time could be shortened since the analyses already consider a single failure of the ruptured SG PORV being stuck in the full open position in the transient presented in Section III. The conservative time has been retained to be consistent with the original analyses.	

Table IV-6 SGTR Radiological Consequences Analyses Results

Scenario	Location	Current Analysis U1/U2 TEDE Dose (rem)	MUR Unit 1 TEDE Dose (rem)*	MUR Unit 2 TEDE Dose (rem)*	TEDE Dose Limit (rem)
Pre-Accident Iodine Spike	Exclusion Area Boundary (0 to 2 hours)	0.721	3.5	3.7	25
	Low Population Zone (0 to 8 hours)	0.165	0.63	0.69	25
	Control Room (0 to 30 days)	0.717	1.7	2.0	5
Accident-Initiated Iodine Spike	Exclusion Area Boundary (0 to 2 hours)	0.327	1.8	2.1	2.5
	Low Population Zone (0 to 8 hours)	0.077	0.33	0.41	2.5
	Control Room (0 to 30 days)	0.183	0.46	0.56	5

* Approximately 5% margin was added to the total calculated dose to arrive at the dose reported. This additional margin is included to allow flexibility in addressing minor impacts on the dose analyses without requiring a reanalysis or changes in the reported dose.

V OVERALL CONCLUSIONS

It was concluded in Section II that SG overfill would not occur for a design basis SGTR. The radiological consequences of an SGTR were evaluated using the thermal hydraulic results from the analyses presented in Section III. The resulting doses at the EAB, LPZ, and in the CR are within the applicable limits, as presented in Section IV.

V.1 SGTR Margin to Steam Generator Overfill Analysis

The updated steam generator tube rupture (SGTR) accident is discussed in Section II of this attachment. The accident analyses demonstrate that SG overfill does not occur. The analyses were performed using the LOFTTR2 program and the methodology developed in Reference 1, with modifications to address NSAL-07-11 (Reference 3) consistent with WCAP-16948-P (Reference 4), and using plant-specific parameters. The MTO analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt. Therefore, the analyzed RTP power is bounding for the MUR power uprate.

V.2 SGTR Thermal and Hydraulic Analysis for Radiological Consequences

The thermal and hydraulic analyses were performed using the LOFTTR2 program and the methodology developed in References 1 and 2, and using the plant-specific parameters. From these predictions, the RCS and SG water masses, the ruptured SG break flow, the fraction of this break flow that flashes directly to steam, and the steam releases from the ruptured and intact SGs through the MSSVs and PORVs are calculated for input to the dose analyses. The thermal-hydraulic analyses assumed a core power of 3658.3 MWt, or 102% of 3586.6 MWt to generate this data. Therefore, the analyzed RTP power is bounding for the MUR power uprate.

V.3 SGTR Radiological Consequences Analysis

The steam generator tube rupture radiological analyses are based upon the alternative source term (AST) as defined in Regulatory Guide (RG) 1.183, with acceptance criteria as specified in RG 1.183 for offsite doses and in 10 CFR 50.67 for the control room. The analyses involve the transfer of activity from the primary to the secondary side of the SGs and then to the environment. The RCS iodine and noble gas source terms are scaled to the Technical Specification Dose Equivalent (DE) Iodine-131 and Xenon-133 limits in the primary coolant, which removes the power dependence from the analysis. The various parameters from the thermal-hydraulic analyses are consistent with a core power of 3658.3 MWt, or 102% of 3586.6 MWt. The resulting doses at the EAB, LPZ, and in the CR remain within the applicable limits; therefore, the results of the SGTR radiological analyses are acceptable under MUR power uprate conditions.

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