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10 CFR 50.90

U S Nuclear Regulatory Commission
ATTN: Document Control Desk,
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Response to Requests for Additional Information (RAI) Associated with Adoption of the Alternative Source Term (AST) Methodology (TAC NOS. ME2609 and ME2610)

References:
1. NSPM Letter to US NRC, "License Amendment Request (LAR) to Adopt the Alternative Source Term Methodology," dated October 27, 2009 (ADAMS Accession No. ML093160583).
2. US NRC Letter to NSPM, "Prairie Island Nuclear Generating Plant, Units 1 and 2 - Request for Additional Information (RAI) Associated with Adoption of the Alternative Source Term (AST) Methodology (TAC NOS. ME2609 and ME2610)," dated May 12, 2011 (ADAMS Accession No. ML103540433).

In Reference 1, the Northern States Power Company, a Minnesota corporation doing business as Xcel Energy (hereafter "NSPM") , requested an amendment to the Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP). The proposed amendment requested adoption of the Alternative Source Term (AST) methodology, in addition to TS changes supported by AST design basis accident radiological consequence analyses.

During a January 26, 2011 teleconference, NSPM discussed delayed implementation of the AST LAR. It was noted that the delayed implementation could be documented as a License Condition. Enclosure 1 provides a proposed License Condition to address delayed implementation of the AST LAR. NSPM will implement the License Condition within 30 days following issuance of the AST License Amendment (LA), and implement the balance of the LA, in accordance with the terms of the License Condition.

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In Reference 2, the Nuclear Regulatory Commission (NRC) Staff requested additional information to support their review of Reference 1. Enclosure 2 to this letter provides the responses to the Staff RAIs, specifically, responses to RAIs from the Reactor Systems Branch.

NSPM submits this supplement in accordance with the provisions of 10 CFR 50.90.

The supplemental information provided in this letter does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the October 27, 2009 submittal, as supplemented by letters dated April 29, 2010 (ADAMS Accession No. ML101200083), May 25, 2010 (ADAMS Accession No. ML101460064), June 23, 2010 (ADAMS Accession No. ML101760017), August 12, 2010 (ADAMS Accession No. ML102300295), and December 17, 2010 (ADAMS Accession No. ML103510322).

In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Gregory Myers, P.E., at 651-267-7263.

<u>Summary of Commitments</u>

This letter contains no new commitments or revisions to existing commitments.

I declare under penalty of perjury that the forgoing is true and correct.
Executed on JUN 22 2011

Mark A. Schimmel

Mark A. Schimmel
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, PINGP, USNRC
Resident Inspector, PINGP, USNRC
State of Minnesota

ENCLOSURE 1

License Additional Conditions (Retyped)

Unit 1 B-3
Unit 2 B-3

4 pages follow

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Amendment Number	Additional Conditions	Implementation Date
158	The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 158 shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.	October 31, 2002
158	The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table LR, "Less Restrictive Changes – Relocated Details," and Table R, "Relocated Specifications," attached to the NRC staff's safety evaluation dated July 26, 2002. Those requirements shall be relocated to the appropriate documents no later than October 31, 2002.	October 31, 2002
	The Alternative Source Term (AST) License Amendments ___/___ will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Amendment Number	Additional Conditions	Implementation Date
	NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Amendment Number	Additional Conditions	Implementation Date
149	The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 149 shall be as follows: For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval, which begins on the date of implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to the implementation of this amendment.	October 31, 2002
149	The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table LR, "Less Restrictive Changes – Relocated Details," and Table R, "Relocated Specifications," attached to the NRC staff's safety evaluation dated July 26, 2002. Those requirements shall be relocated to the appropriate documents no later than October 31, 2002.	October 31, 2002
	The Alternative Source Term License Amendments ___/___ will be implemented after installation of the Unit 2 Replacement Steam Generators (RSGs).	Within 90 days after completion of the outage in which the Unit 2 RSGs are installed

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Amendment Number	Additional Conditions	Implementation Date
	NSPM will provide the NRC written notification when Unit 2 RSG installation is complete and AST License Amendment implementation has commenced.	Within 30 days after completion of the outage in which the Unit 2 RSGs are installed

Enclosure 2

Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI)

In order for the NRC staff to continue its review, the following additional information is needed:

A. <u>RAI Related to the MTO Analysis</u>

A.1 If the licensee chooses to continue to use its simulator for the MTO analysis, it should provide additional information related to the computer codes and RCS physical models in the simulator for the NRC staff to review and approve. The additional information provided should include: a discussion of the methodology; computation device manuals; user's manuals and guidelines; scaling reports; assessment reports and uncertainty assessment reports as described in the applicable sections of Regulatory Guide 1.203, "Transient and Accident Analysis Methods."

The information should show that: the constituent equations representing the RCS thermal-hydraulics are correct and complete; the correlations for the heat transfer and flow rate determination are adequately supported by the applicable test data; the nodal scheme appropriately models the RCS; the mathematical methods provide stable solutions; the time step used for the mathematical solution does not result in divergent conditions; the system responses of the RCS for both with and without a loss of AC power are validated by comparing with the applicable integrated and separated effects test data; and the MTO analyses show that the assumptions and the plant conditions used result in a maximum response time for the AST application.

<u>Response</u>

As described in the response to question 2, below, the Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter, "NSPM"), has decided to perform a Steam Generator Tube Rupture (SGTR) margin-to-overfill (MTO) Analysis aligning as closely as possible to the NRC approved methodology described in Westinghouse topical report WCAP-10698-P-A. Thus, NSPM is not using the simulator for the MTO analysis, with the exception of substantiating operator action times in support of the analysis described below.

A.2 *Alternatively, the licensee may perform an SGTR MTO analysis for PINGP at current licensed thermal power conditions. The analysis should align as closely as possible to an NRC-approved methodology described in a Westinghouse topical report, WCAP-10698-P-A. However, since the licensee has stated that a limiting single failure is not in the PINGP licensing basis, this exception to the WCAP-10698-P-A methodology will be acceptable. The requested analytical results should include sequences of the event with specification of operator actions and the associated times credited in the analysis, and the response of key plant parameters versus time.*

Response

NSPM has performed sensitivity analyses to determine the limiting margin-to-overfill (MTO) scenarios at 1683 MWt, which is the current licensed reactor core power level of 1677 MWt, plus calorimetric uncertainties. The analyses followed the methodology in WCAP-10698-P-A, with the exception of the assumption of a single failure.

The analyses were performed using the LOFTTR2 thermal hydraulic model consistent with the methodology in WCAP-10698-P-A.

The results indicate a margin-to-overfill of 186 ft^3 in the ruptured steam generator for the limiting scenario. The limiting scenario models 0% steam generator tube plugging (SGTP), low decay heat, maximum safety injection (SI) enthalpy and minimum auxiliary feedwater (AFW) enthalpy. No water is transferred into the steam lines.

The analyses were performed utilizing the configuration of the replacement steam generators (Framatome ANP 56/19). As discussed in a 1/26/11 teleconference between NRC and NSPM, AST implementation will be delayed until after implementation of the Unit 2 replacement steam generator (RSG) is complete. See the Enclosure 1 for the associated License Condition.

The sequence of events for the limiting scenario analysis is presented in Table 1. Figures 1 through 8 provide the time-dependant values of the following parameters for the limiting MTO scenario:

- Reactor Coolant System (RCS) and Secondary Pressures (Intact and Ruptured Steam Generators)
- Primary-to-Secondary Break flow rate
- Steam Generator (SG) Water Volumes (Intact and Ruptured Steam Generators)
- Pressurizer Level
- Intact Steam Generator Inlet and Outlet Temperatures
- Ruptured Steam Generator Inlet and Outlet Temperatures
- Steam Generator Steam releases
- Steam Generator Narrow Range Level (Ruptured Steam Generator)

Table 1: Sequence of Events

Event	Time (sec)
Tube Rupture	0
Reactor Trip	49
Auxiliary Feedwater (AFW) Initiation	50
Safety Injection (SI) Actuation	119
Ruptured SG AFW Isolation	251
Close Main Steam Isolation Valve (MSIV)	1130
Initiate Cooldown with Intact SGs	1190
Establish Charging Flow	1192
Terminate Cooldown	1626
Initiate Depressurization	1866
Terminate Depressurization	1962
Stop SI Flow	2082
Balance Charging and Letdown Flows	2982
Break Flow < 0	3212

Figure 1: RCS and Secondary Pressures

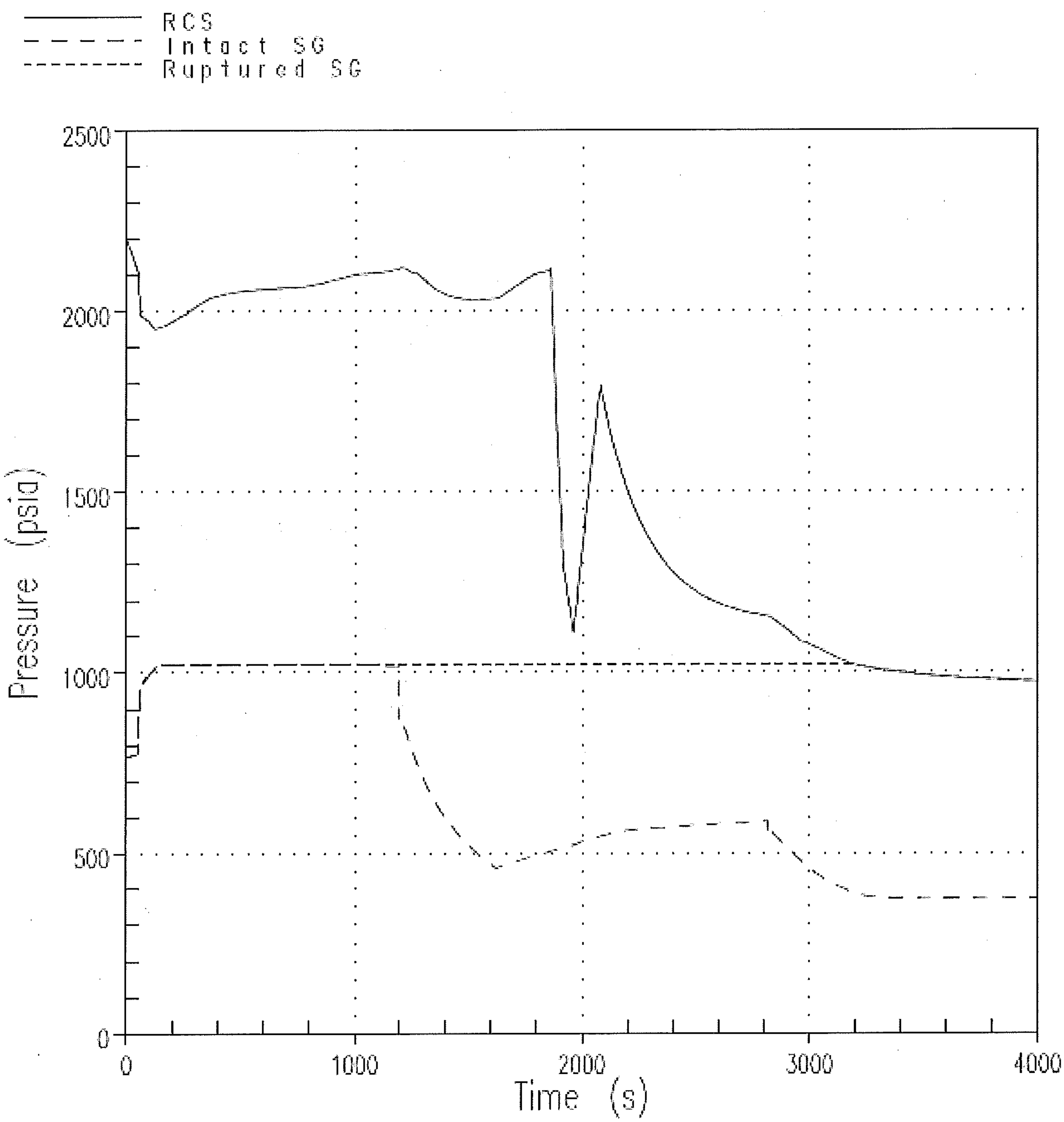


Figure 2: Primary-to-Secondary Break Flow Rate

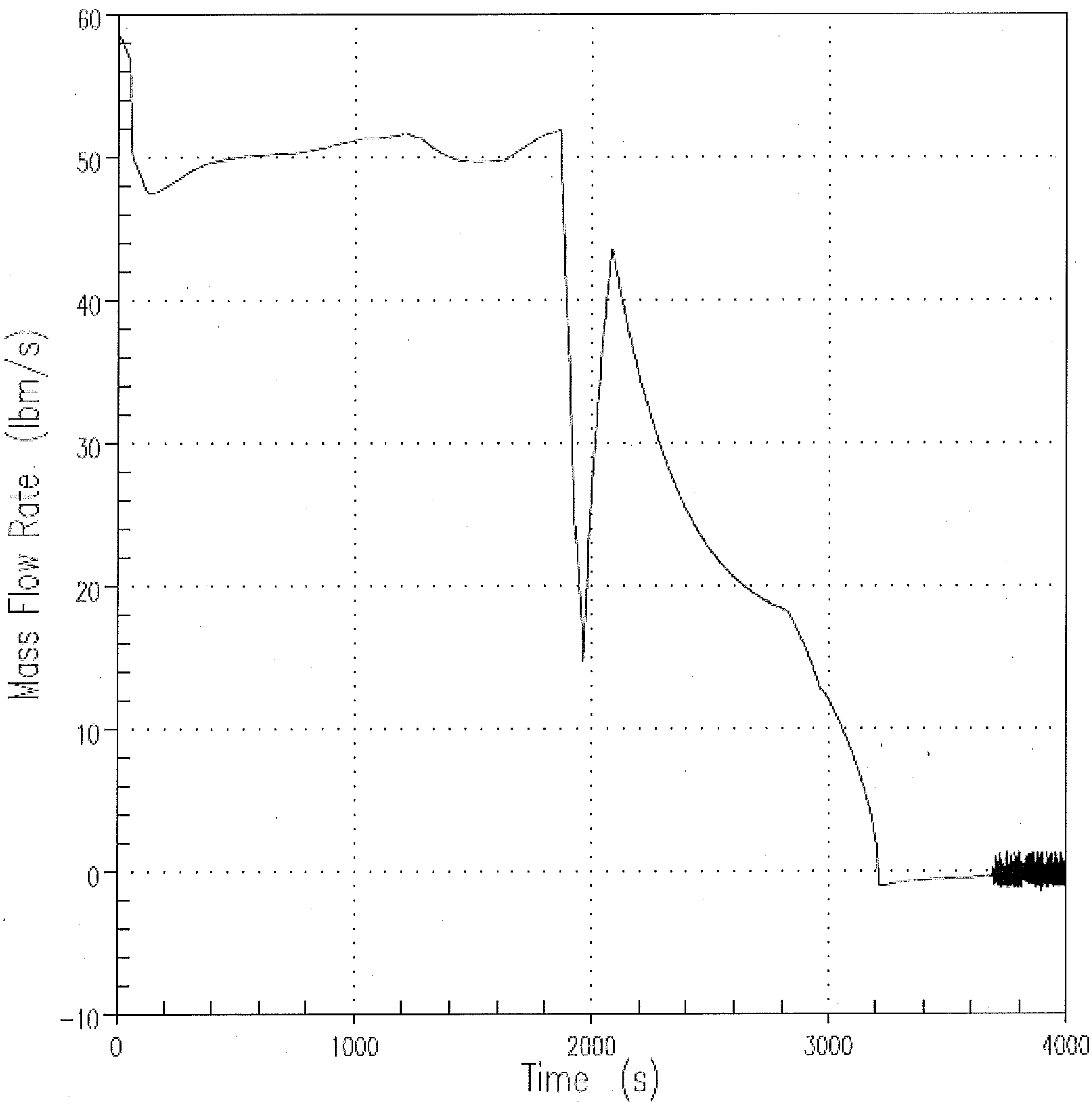


Figure 3: Steam Generator Water Volumes

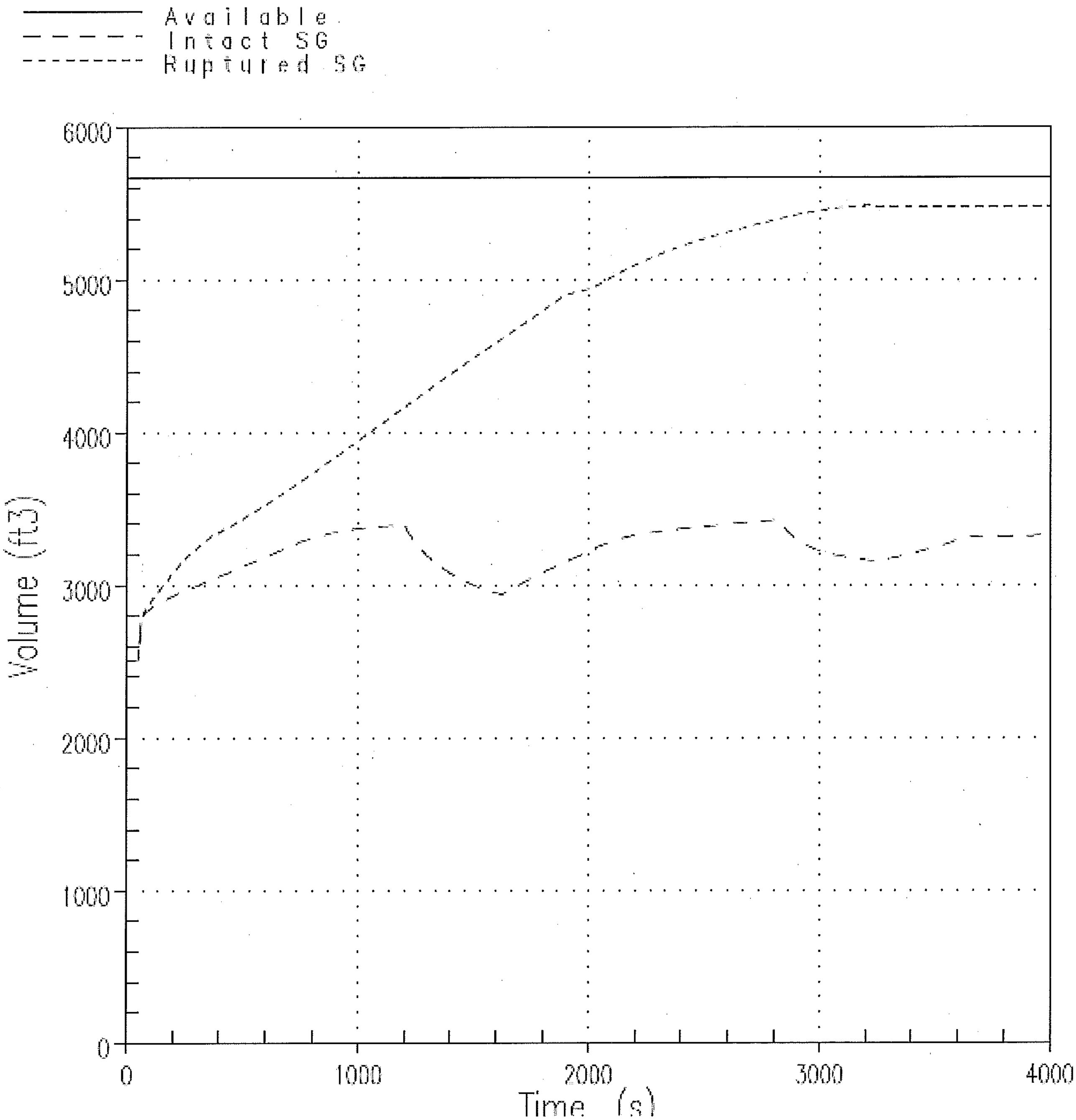


Figure 4: Pressurizer Level

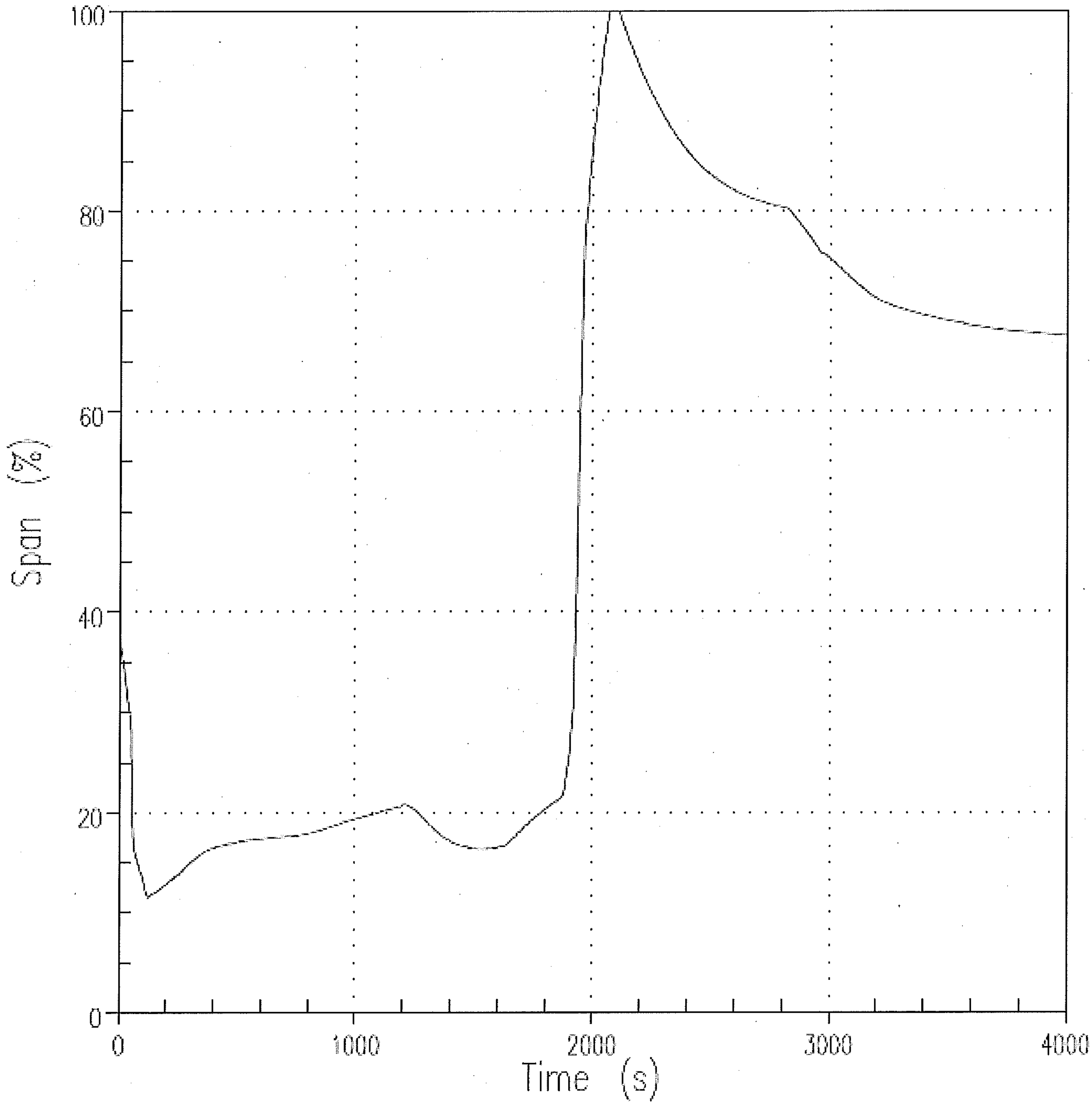


Figure 5: Intact SG Inlet and Outlet Temperatures

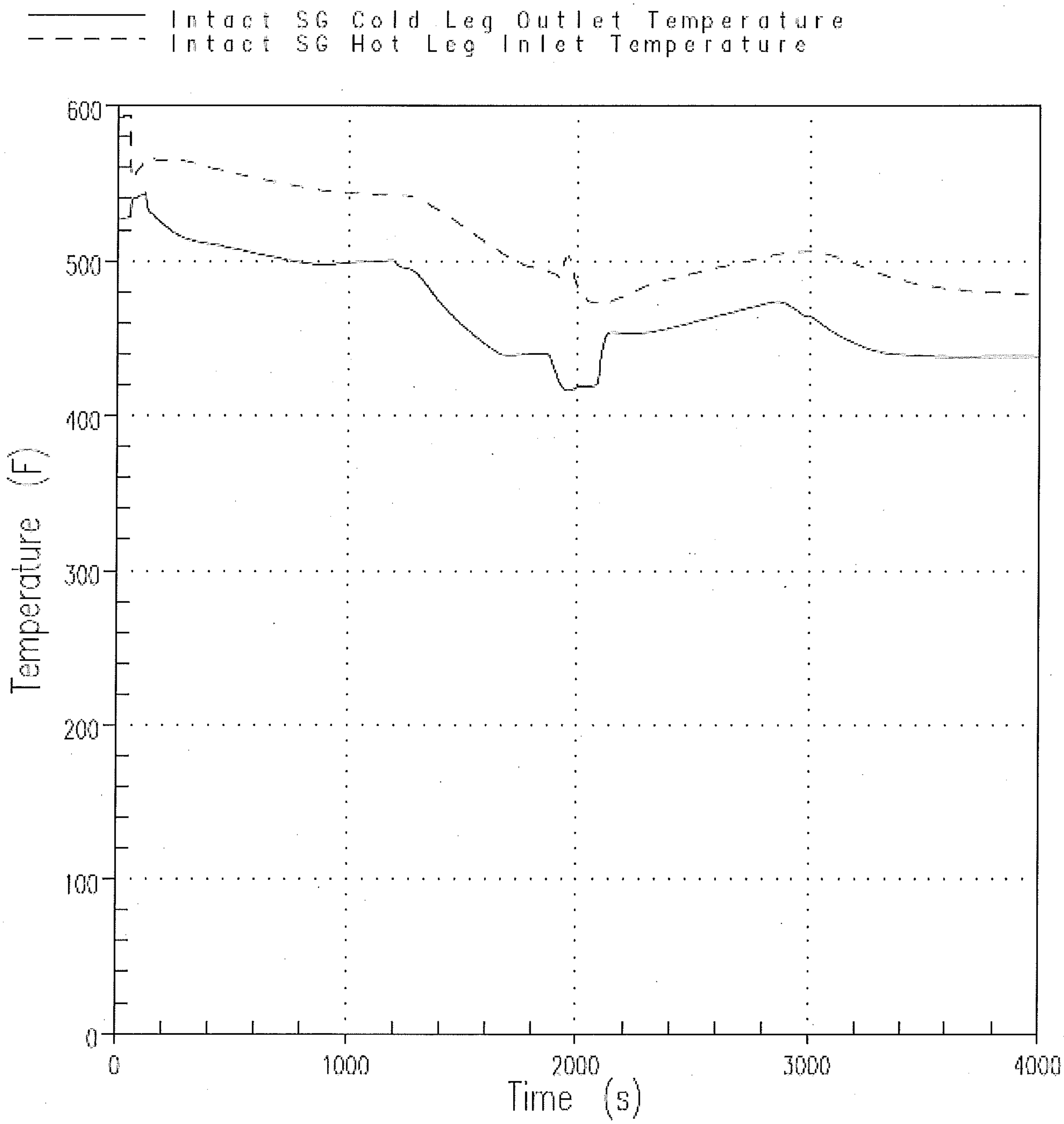


Figure 6: Ruptured SG Inlet and Outlet Temperatures

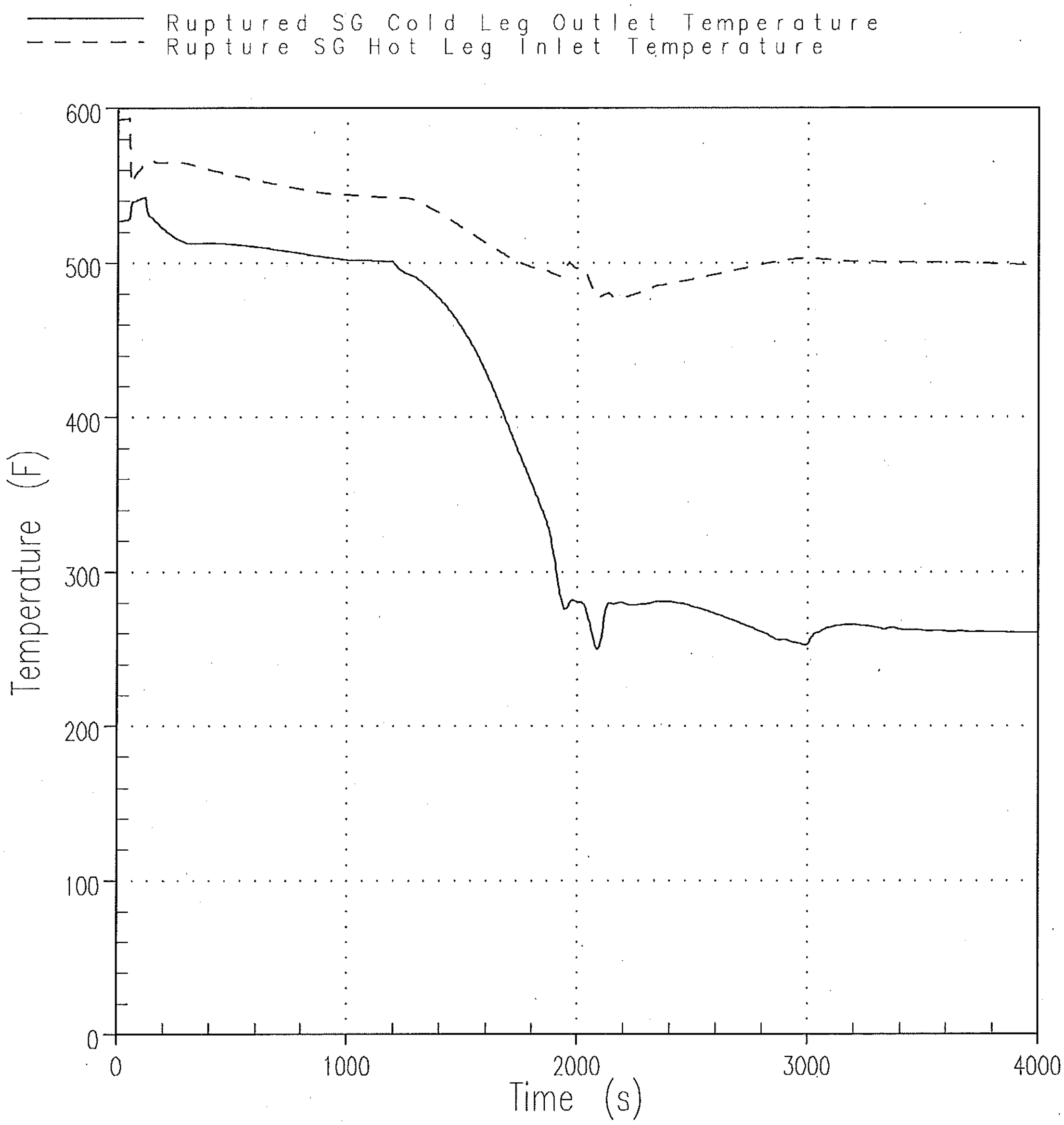


Figure 7: SG Steam Releases

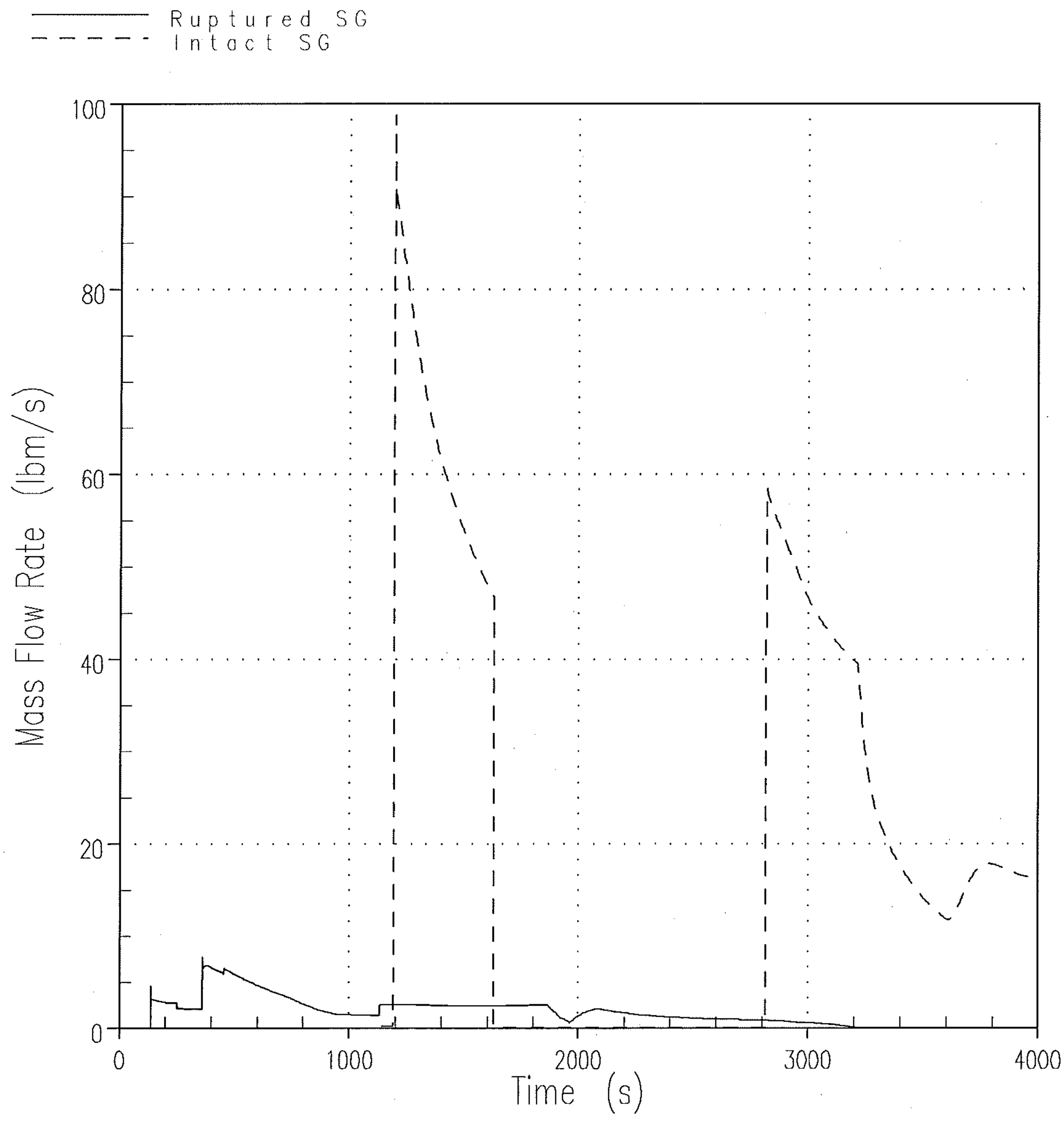
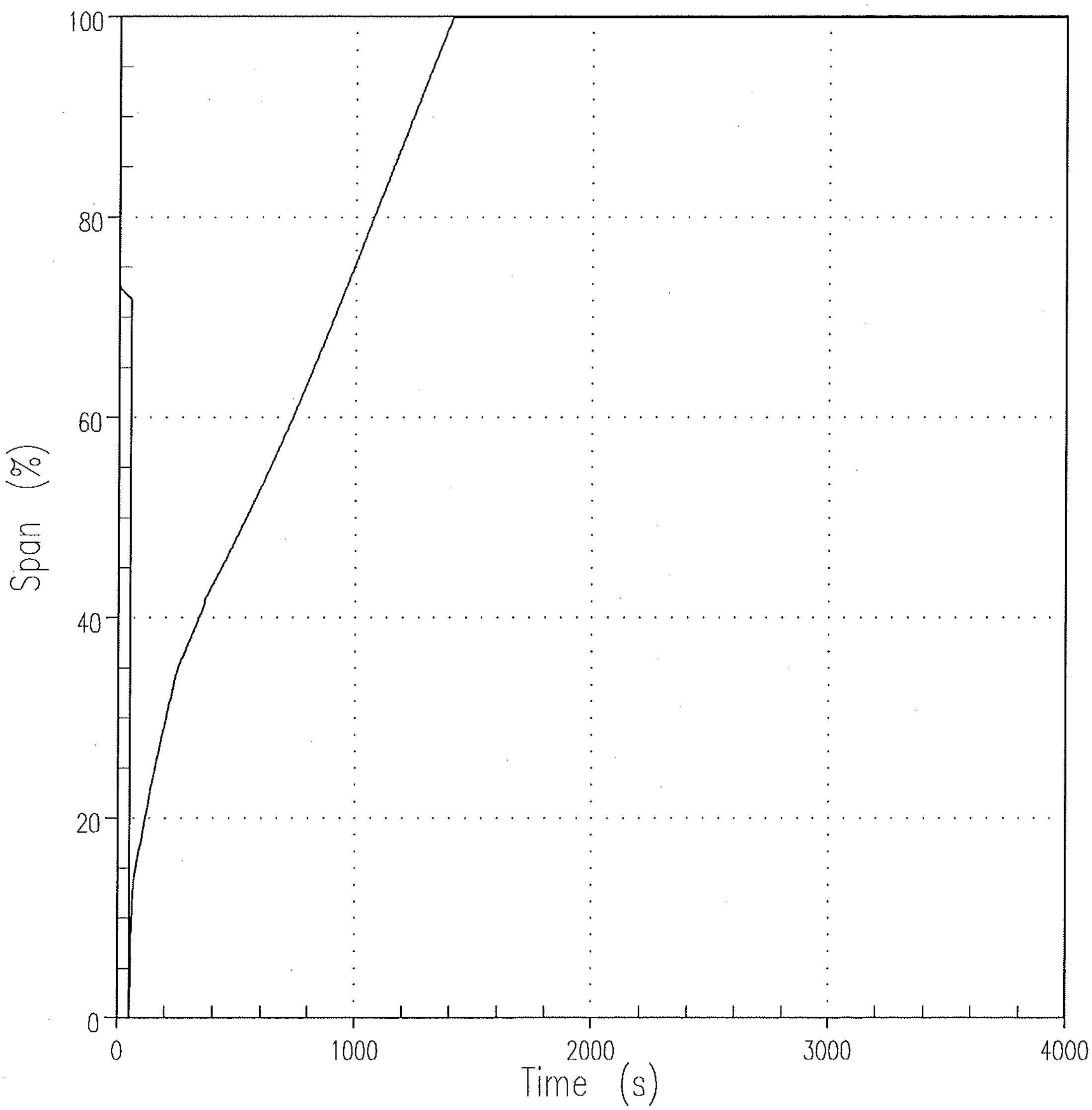


Figure 8: Ruptured SG Narrow Range Level



In addition to providing the analytical results, please address the following:

A.2.a Address compliance with the conditions and restrictions specified in the NRC safety evaluation reports approving the WCAP-10698-P-A methodology.

Response

NRC safety evaluation report approving the WCAP-10698-P-A methodology, dated March 30. 1987, Enclosure 1, Section (D) identifies plant specific inputs that are required to support a margin to overfill analysis that references WCAP-10698-P-A (for clarification, the plant specific inputs identified in the NRC SER for WCAP-10698-P-A are identified in italics).

(1) Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required 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.

Compliance

A NSPM fleet administrative procedure establishes the process to capture analysis-credited operator actions, such as SGTR analysis operator actions, and documents and validates the actual timing of operator actions. The Prairie Island Nuclear Generating Plant (PINGP) Operations Manager has overall responsibility for the operator action time validation process.

NSPM procedures require consideration of all the time critical operator actions necessary to accomplish the nuclear safety functions for each design basis event. The safety analyses document the time critical operator actions and their associated instrumentation and controls required for design basis events. Any equipment required to perform time critical operator actions is identified and assured that it is available.

Each time critical operator action (TCOA) is validated on a periodic basis or as needed in response to plant procedure changes, crew human performance methodology changes, and plant modifications that affect completion time.

Simulator validation of time critical operator actions is performed unless:

- Simulator validation is impractical due to modeling constraints;
- Time critical operator actions are performed outside the control room;
- Simulator validation is combined with walkthrough validation since the event involves control room and local operator action;
- For changes that do not warrant simulator validation due to nature or scope

The specific time critical tasks are controlled in a station specific procedure maintained by station Operations. The procedure identifies the licensing and design basis requirements for each time critical operator action, the time requirement and the time validation and training requirements.

The response to RAI A.2.f provides confirmation that operator actions credited in the analysis are consistent with procedures and action times are conservative, resulting in a minimum margin to overfill.

(2) *A site specific SGTR radiation offsite consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP Section 15.6.3, as supplemented by the guidance in Reference (1).*

Compliance

As described below in the response to RAI A.2.g, a SGTR radiological consequence analysis was provided by Reference 1, and supplemented by Reference 3. As described below in the response to RAIs B.1 and B.2, a supplemental thermal hydraulic analysis demonstrates that the thermal hydraulic mass transfers resulting from realistic modeling of operator actions following a SGTR event are bounded by the mass transfers modeled in the previously submitted SGTR radiological consequence analysis. As described above, the MTO analysis predicts that the ruptured steam generator will not be overfilled. Thus, liquid will not be released through the SG PORV or Safety Valves. Therefore, the previously submitted SGTR radiological consequence analysis is bounding.

(3) *An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.*

Compliance

As described below in the response to RAI A.2.c the analysis results determine that there will be no liquid release or water filled conditions in the main steam line. Therefore, an evaluation of the structural adequacy of the main steam lines and associated supports under water filled conditions is not necessary.

(4) A list of systems, components, and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s). Specify whether each system and component specified is safety grade. 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. 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. 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. 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.

Compliance

The PINGP SGTR Emergency Operating Procedure (EOP), identifies the following systems, components, and instruments for accident mitigation. Note that the EOP identifies multiple means and equipment available to the operators to perform required mitigation functions. Therefore, not all of the equipment in Table 2 below is required for any postulated SGTR event.

Table 2
Systems, Components, and Instruments Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
CV-31098, 31099 CV-31116, 31117	Main Steam Isolation Valves (MSIVs)	Safety Related SR	N/A	
MV-32045, 32047 MV-32048, 32050	MSIV Bypass Valves	SR	N/A	
CV-31084, 31089 CV-31102, 31107	SG Power Operated Relief Valves (PORVs)	SR	N/A	The motive power to open the SG PORV is supplied from the Instrument Air (IA) system; which is reliable but not safety related. Redundant IA compressors are powered from diesel-backed safeguards electrical buses. The air compressors are automatically loaded on to the emergency diesel generators in response to a loss of offsite power (LOOP). This is further described below in the response to RAI A.2.d.
1HC-468, 478 2HC-468, 478	SG PORV Control Board Controllers	Non-Safety Related NSR	N	The power supply to the SG PORV control board controllers is safety related. The power supply to the controller will be available during a LOOP.

Table 2
Systems, Components, and Instruments
Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
44032, 44033 44532, 44533	SG PORV Control Board Indicating Lights	NSR	N	The power supply to the SG PORV control board valve position indicating lights is safety related. The power supply to the valve position indicating lights will be available during a LOOP.
MV-32016, 32017 MV-32019, 32020	Steam Supply from Ruptured SG to Turbine Driven Auxiliary Feedwater (AFW) Pump	SR	N/A	
MV-32040, 32046 MV-32044, 32051 MV-32043, 32049 MV-32058, 32059	SG Blowdown Isolation Valve	SR	N/A	
MV-32238, 32381 MV-32239, 32382 MV-32246, 32383 MV-32247, 32384	AFW Isolation Valve to Ruptured SG	SR	N/A	
145-201, 145-331 245-201, 245-331	Auxiliary Feedwater Pumps	SR	N/A	

Table 2
Systems, Components, and Instruments Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
CV-31231, 31232 CV-31233, 31234	Pressurizer PORV	SR	N	The motive power to open the Pressurizer PORVs is IA, which is reliable but not safety-related. Redundant IA compressors are powered from diesel-backed safeguards electrical buses. The air compressors are automatically loaded on to the emergency diesel generators in response to a loss of offsite power (LOOP). In addition, a Seismic Category I passive air accumulator is provided inside of containment as a back-up air supply for the Pressurizer PORVs.
CS-46259, 46260 CS-49576, 49577	Pressurizer PORV Control Switches	NSR	N/A	The power supply to the Pressurizer PORV control switches is safety related. The power supply to the control switches will be available during a LOOP
CV-31329 CV-31421	Pressurizer Auxiliary Spray Valves	SR	N/A	
MV-32195, 32196 MV-32197, 32198	Pressurizer Block Valves	SR	N/A	

Table 2
Systems, Components, and Instruments
Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
CS-46241, 46242 CS-49555, 49556	Pressurizer Heaters	NSR	N	The power supply to the pressurizer heaters is from a safety related power source.
CS-46295, 49579	Pressurizer Auxiliary Spray Valve Control Switches	NSR	N	The power supply to the Pressurizer Auxiliary Spray valve control switches is safety related. The power supply to the control switches will be available during a LOOP
145-071, 145-072 245-071, 245-072	Safety Injection (SI) Pumps	SR	N/A	
145-041, 145-042, 145-043, 245-041, 245-042, 245-043	Chemical and Volume Control System (CVCS) Charging Pumps	SR	N/A	
Various	CVCS Letdown System	NSR	N	The analysis credits operation of the letdown or excess letdown system for balancing CVCS charging flow following securing of the SI Pumps (with appropriate time delay). Note: excess letdown provides a backup method to normal letdown.
CS-46182, 46183 CS-49546, 49547	Safety Injection Reset Circuitry	SR	N/A	

Table 2
Systems, Components, and Instruments
Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
CS-46083, 46084 CS-49663, 49664	Containment Isolation Reset Circuitry	SR	N/A	
121, 122 & 123	Instrument Air Compressors	NSR	N	Redundant IA compressors are powered from diesel-backed safeguards electrical buses. The air compressors are automatically loaded on to the emergency diesel generators in response to a loss of offsite power (LOOP).
CS-46154, 46155 CS-46590, 46591	Control Switches for providing Instrument Air to Containment	SR	N/A	
1LI-461, 462, 463 1LI-471, 472, 473 2LI-461, 462, 463 2LI-471, 472, 473	SG Water Level Indication – Narrow Range	NSR	N	Instrument Loops from sensing line through transmitter are safety related. The indication is on the non safety related side of the current to pneumatic transmitter (I/I). The power supply is safety related and will be available during a LOOP. Three indication channels for each Steam Generator are available in the Control Room.

Table 2
Systems, Components, and Instruments Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
TE13234 to 13272 TE13407 to 13445	Core Exit Thermocouples	SR	N/A	
1LI-426, 427, 428 2LI-426, 427, 428	Pressurizer Water Level Indication	Augmented Quality AQ	N/A	Instrument Loops from sensing line through transmitter are safety related. The indicators are augmented quality. The power supply is safety related and will be available during a LOOP. Three indication channels per unit are available in the Control Room.
1PI-709, 710 2PI-709, 710	RCS Pressure Indication	AQ	N/A	The indicators and transmitters are augmented quality. The power supply is safety related and will be available during a LOOP. Two indication channels per unit are available in the Control Room.
1PI-468, 469, 482A, 478, 479, 483A 2PI-468, 469, 482A, 478, 479, 483A	SG Pressure Indication	NSR	N	Instrument Loops from sensing line through transmitter are safety related. The indicators are non safety related. The power supply is safety related and will be available during a LOOP. Three indication channels per Steam Generator are available in the Control Room.

Table 2
Systems, Components, and Instruments
Available for SGTR Mitigation

Equipment / Component ID#	Equipment/ Component Name	Safety Class (SR, AQ, NSR)	If Non-Safety Related, is Safety Grade Backup Available (Y or N)	Discussion
1(2)RM-15	Condenser Air Ejector Radiation Monitor	NSR	N See Note 1	Radiation monitor is non safety related. The radiation monitor is maintained and tested in accordance with the Offsite Dose Calculation Manual. The power supply to the radiation monitor is safety related and will be available during a LOOP.
1(2)RM-19	Steam Generator Blowdown Liquid Radiation Monitor	NSR	N See Note 1	The radiation monitor is maintained and tested in accordance with the Offsite Dose Calculation Manual. The power supply to the radiation monitor is safety related and will be available during a LOOP.
1(2)RM-51 1(2)RM-52	Main Steam Line Radiation Monitor	NSR	N/A See Note 1	Radiation monitor is non safety related. The radiation monitor is maintained and tested in accordance with the Offsite Dose Calculation Manual. The power supply to the radiation monitor is non-safety related, backed up by a non-safety related diesel generator and expected to be available during a LOOP.

1. The installed radiation monitors and the SG level transmitters are used in the EOPs to identify the ruptured SG. As directed by the Emergency Operating Procedures, SG water sampling is performed to confirm the identification of the ruptured SG during a SGTR event that does not result in abnormal radiation monitor indication. The high primary to secondary flow rate due to a design basis tube rupture results in SG water level increase that provide relatively quick indication of a ruptured SG. SG chemistry sample analysis time is not a critical aspect of the accident mitigating actions for scenarios that could be challenging with respect to overfill. Therefore, the time duration for sampling and analysis of SG secondary water would not delay the response to this event.

(5) *A survey of plant primary and "balance-of-plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and effect of the difference on the margin to overfill should be provided.*

Compliance

Consistent with the reference plant from WCAP-10698-P-A, PINGP Units 1 and 2 are Westinghouse designed plants utilizing similar features such as reactor trip setpoints, safety injection system, auxiliary feedwater system, SG relief valves, and emergency operating procedures for the SGTR scenario. PINGP plant-specific inputs were used in the analysis and applied consistent with the WCAP-10698-P-A methodology. There are no major design differences between the WCAP-10698-P-A reference plant and the PINGP Units that would affect the methodology application. Also, the exclusion of a single failure from the analysis is acceptable and consistent with the PINGP current licensing basis for the SGTR scenario. In addition, the NRC recently approved application of this methodology for the Point Beach Nuclear Plant (ADAMS Accession Nos. ML110880039 and ML110450159), which is a very similar Westinghouse NSSS two-loop plant, as documented by Reference 4.

A.2.b *List in a table the nominal values with the associated uncertainties, and corresponding values used in the MTO analysis for the major input initial conditions described in WCAP-10698-P-A. Discuss the bases used to select the numerical values of the input parameters and show that the numerical values used are conservative, resulting in a minimum SG MTO during an SGTR event. In addition, provide a basis for the target cooldown temperature used in the analysis.*

Response

NSPM has performed a series of analyses of the limiting margin-to-overfill (MTO) scenarios 1683 MWt, which is the current licensed reactor core power level of 1677 MWt, plus calorimetric uncertainties.. The analyses followed the methodology in WCAP-10698-P-A, with the exception of the assumption of a single failure. A comparison of the WCAP-10698-P-A modeling to that used in the PINGP SGTR MTO analysis is provided in Table 4 below. The nominal values, associated uncertainties and corresponding values used in the MTO analyses for major input initial conditions are provided by Table 3 below.

Table 3

Major Input Initial Conditions

Parameter	Nominal Value	Uncertainty	Value Modeled in MTO Analysis
Core Power	1677 MWt	0.36% or 6 MWt	1683 MWt
Reactor Coolant System Pressure	2250 psia	60 psi	2190 psia
Pressurizer level	33% span	5% span	38% span
Steam Generator Level	44% Narrow Range Span (NRS)	10% NRS	73% NRS[1]
Low Pressurizer Pressure Reactor Trip Setpoint	1915 psia	65 psi	1980 psia
Low Pressurizer Pressure Safety Injection Actuation	1845 psia	103 psi	1948 psia[2]

1. Includes 10% increase in initial steam generator mass plus mass added due to turbine runback.

2. The nominal trip setpoint is 1802.1 psia. The modeled value is conservatively based on the actual plant setting instead of the lower nominal setpoint.

Table 4: Comparison of WCAP-10698-P-A Modeling to the Analysis Assumptions

Parameter	WCAP-10698-P-A Modeling Direction of Conservatism	PINGP SGTR MTO Analysis
Initial Conditions		
Power	Full-power (nominal + uncertainty)	Full-power (nominal + uncertainty)
RCS Pressure	Minimum	Minimum
Pressurizer Water Level	Maximum	Maximum
Steam Generator (1) Secondary Mass	Maximum	Maximum
Break Location	Cold-leg	Cold-leg
Offsite Power Availability		
Offsite Power	LOOP	LOOP
Protection Setpoints and Errors		
Reactor Trip Delay	Minimum	Minimum
Turbine Trip Delay	Minimum	Minimum
SG Relief or Safety Valve setpoint	Minimum	Minimum
Pressurizer pressure trip setpoint	Maximum	Maximum
Pressurizer pressure SI setpoint	Maximum	Maximum
Safeguards Capacity		
SI Flow Rate	Maximum	Maximum
AFW Flow Rate (isolation on SG level)	Minimum	Maximum(2)
AFW System Delay	Minimum	Minimum
AFW Temperature	Maximum	Minimum(2)
Control Systems		
CVCS Operation	Not operating	Operating(3)
Pressurizer Heater Control	Not operating	Not operating
Turbine runback mass penalty	Included	Included
Reactor Coolant Pump Running	Not Operating	Not Operating
Decay Heat		
Decay Heat	Maximum	Minimum (ANS 1979-2σ(2))
Single Failure		
Single Failure	Included	Not Included, consistent with current licensing basis

Table 4: Comparison of WCAP-10698-P-A Modeling to the Analysis Assumptions		
Parameter	**WCAP-10698-P-A Modeling Direction of Conservatism**	**PINGP SGTR MTO Analysis**
Operator Actions		
Operator Response Times	Maximum	Maximum

(1) Consistent with the discussion of power in WCAP-10698-P-A, the initial steam generator mass is more conservatively calculated without inclusion of the initial power uncertainty since it results in a higher mass.

(2) Plant-specific sensitivities for PINGP concluded that it is more conservative to model minimum AFW temperature and decay heat rather than maximized as prescribed by WCAP-10698-P-A for the margin-to-overfill analysis. Also, plant-specific sensitivities concluded it is more conservative to model maximum AFW flow rate rather than minimum as prescribed by WCAP-10698-P-A for the margin-to-overfill analysis.

(3) It is conservative to model charging flow in this analysis since its initiation is modeled by operator actions well after reactor trip. When charging flow is initiated automatically, its impact in delaying reactor trip is the basis for no charging flow being modeled per WCAP-10698-P-A.

The emergency operating procedures provide a table of reactor coolant system cooldown target temperatures corresponding to a range of secondary pressures. The target cooldown temperature was taken from the EOP table and was based on the conservatively modeled steam generator PORV setpoint modeled in the analyses. The steam generator PORV setpoint modeled is 1020 psia, which corresponds to the no-load reactor coolant system average temperature. A target temperature of 505^0F was chosen in accordance with the emergency operating procedures.

A.2.c Ensure that the limiting liquid release pathway and scenario are identified. Include consideration of the steam line equipment water-release failures discussed in WCAP-11002-P (Note that the NRC staff discussed WCAP-11002-P in its evaluation of WCAP-10698-P-A, but did not find that it provided an acceptable method for performing a licensing basis safety analysis). If a liquid release is predicted, provide analyses of the static and dynamic structural effects in the main steam system and of the consequences of passing water through the steam pressure relief valves.

Response

Based on the analyses described above there is no predicted liquid release or water filled conditions in the main steam lines. Therefore, an analysis of the static and dynamic structural effects of the main steam system and the consequences of passing water through the steam pressure relief valves is not necessary.

A.2.d Under the assumed LOOP conditions, address the functionality of each power operated relief valve (PORV). Discuss what, if any, mitigating function the PORV provides, and its capability to perform that function under the assumed LOOP conditions. If the valve's actuation must be manual, provide information to demonstrate that the operator is capable of actuating the valve within the analytically assumed time.

<u>Response</u>

Each SG has one PORV located on the main steam line between the SG and the MSIV. As part of the LOFTTR2 modeling for the SGTR MTO analysis, the PORV on each SG performs the following functions:

- The PORV for the ruptured SG is closed. The position of the PORV during normal power operation is de-energized in the closed position and the fail safe position for the PORV is closed.
- The PORV for the intact SG is used by the operator in the Control Room to cooldown the Reactor Coolant System (RCS).

The PORV controllers are powered from 120 VAC instrumentation buses; which are safety related (battery backed). The PORV position indication lights are powered from 125 VDC buses, which are powered from safety-related batteries. The PORVs require air to operate remotely from the control room. The PORVs receive air from the instrument air (IA) system headers, which supply air to both Unit 1 and 2 PORVs. There are three instrument air compressors. During normal power operation, two compressors are operating to provide compressed air for both units. One compressor is fully loaded and the other compressor is loaded part of the time. The IA compressors are powered from the safety-related 480 VAC buses, which, during a LOOP, are powered from emergency diesel generators (EDGs). During a LOOP the EDGs will automatically restore power to the safeguards buses. The IA compressors are automatically loaded on to the associated EDG at step 5 (30 seconds following the occurrence of the LOOP assuming 10 seconds for the EDG to be up to speed and voltage) with no action required from the control room operators. During the 30 seconds that the IA compressors are not operating prior to being loaded on the EDGs, the IA system air receivers maintain IA system pressure. In addition, the SG PORVs or Pressurizer PORVs are not credited in the analysis during the brief time period following the initiation of the accident that the IA compressors are not operating.

A.2.e One of the key parameters that will affect the results of the SG MTO analysis during an SG tube rupture event is the initial SG water level, which is a function of the initial power level. The MTO analysis to be submitted should consider the effects of initial SG water levels corresponding to power levels that capture 95 percent of the operating time during a fuel cycle. Also, for the range of power levels that envelop 95 percent of operating time, provide trending data for the corresponding SG water levels to show that conservative initial SG water levels (with the inclusion of measurement uncertainties, thus resulting in a smaller margin to SG overfill) have been selected.

Response

PINGP operates at power levels of approximately 100% for more than 95% of the operating time during a fuel cycle. The data shown on the attached Figure 9 for Unit 1 and Figure 10 for Unit 2 represents the time period of July 2008 through May 2011. On Figure 9, the entire time period of early September to late November 2009, the reactor was shutdown.

The 100% power nominal setpoint for steam generator level control is 44% narrow range. Steam Generator narrow range water level data for both Unit 1 and Unit 2 was reviewed for the same time period used in Figures 9 and 10 for displaying power. The data is shown on the attached Figures. For each Unit a Figure is provided for each Steam Generator indicated narrow range water level for the same periods of time that are shown in Figures 9 and 10 for the power levels. As shown in the Steam Generator water level plots, during full power operation, the water level varies from the program level by a small amount. The data shows that this variance is less than +/- 1% indicated level. As shown on Table 3, included as part of the response to RAI A.2.b, above, an assumed initial SG narrow range water level of 73% is used in the MTO analysis. It is noted that the 73% initial SG water level includes accounting for the turbine runback. Thus, actual steam generator indicated level for 100% steady state power operation is bounded by the initial SG water level assumed in the analysis.

Figure 9, Page 1
Unit 1 Power Level and Steam Generator
Narrow Range Water Level

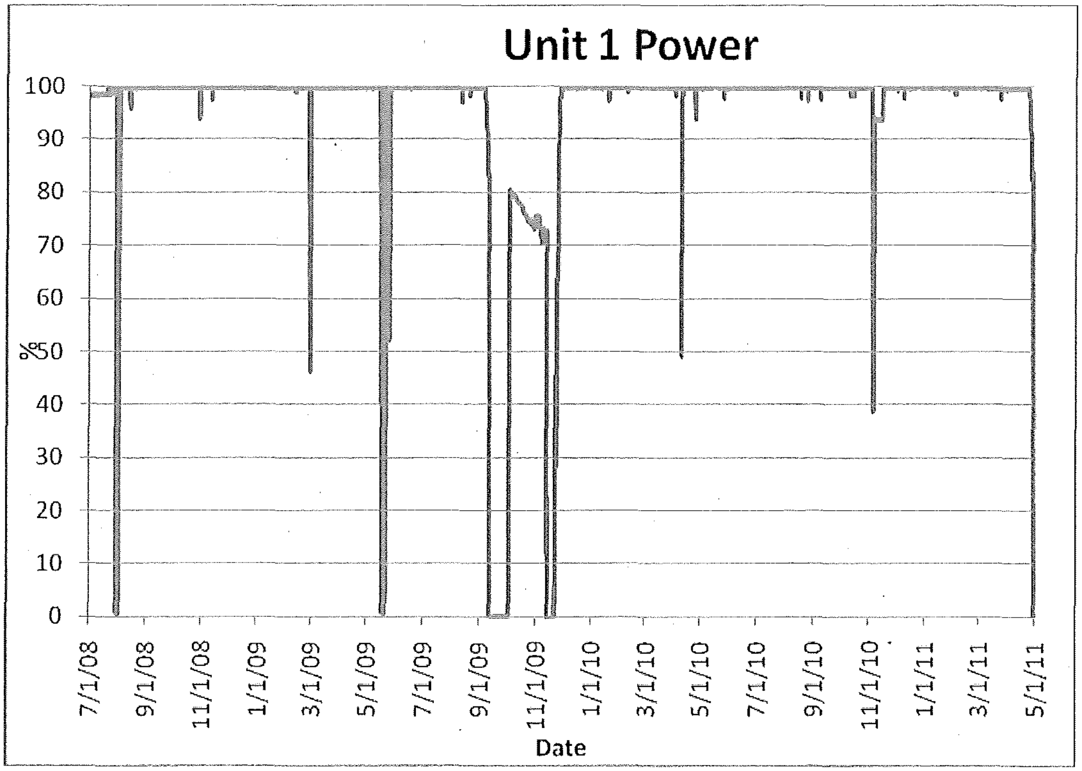
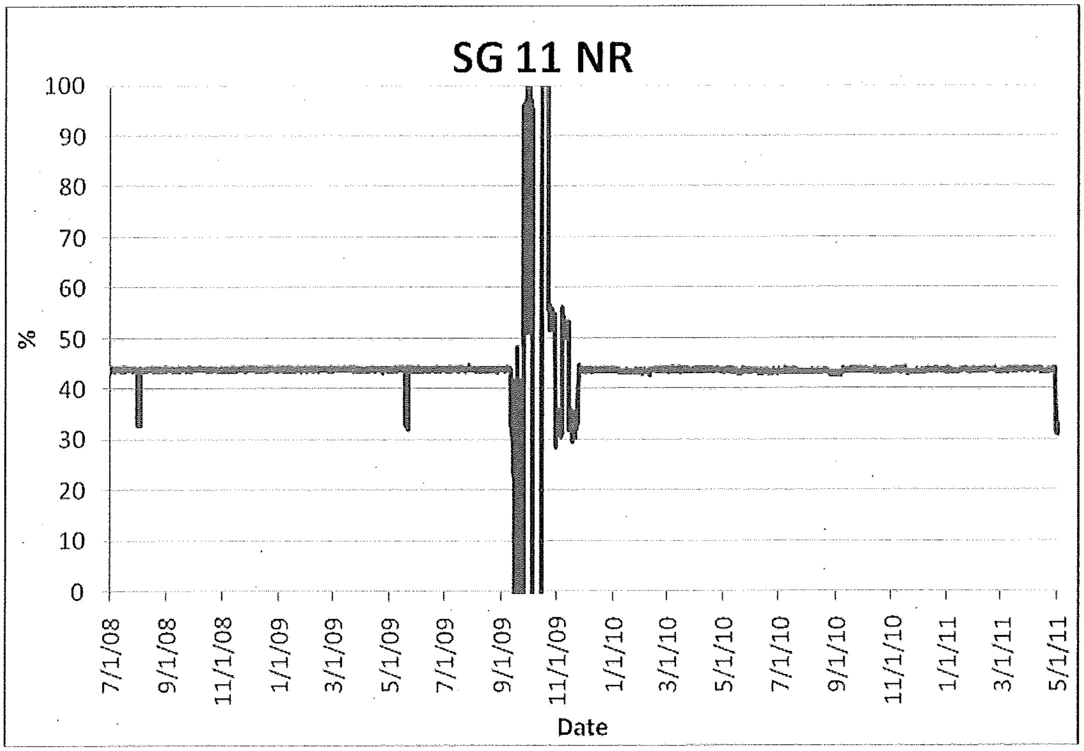


Figure 9, Page 2
Unit 1 Power Level and Steam Generator
Narrow Range Water Level



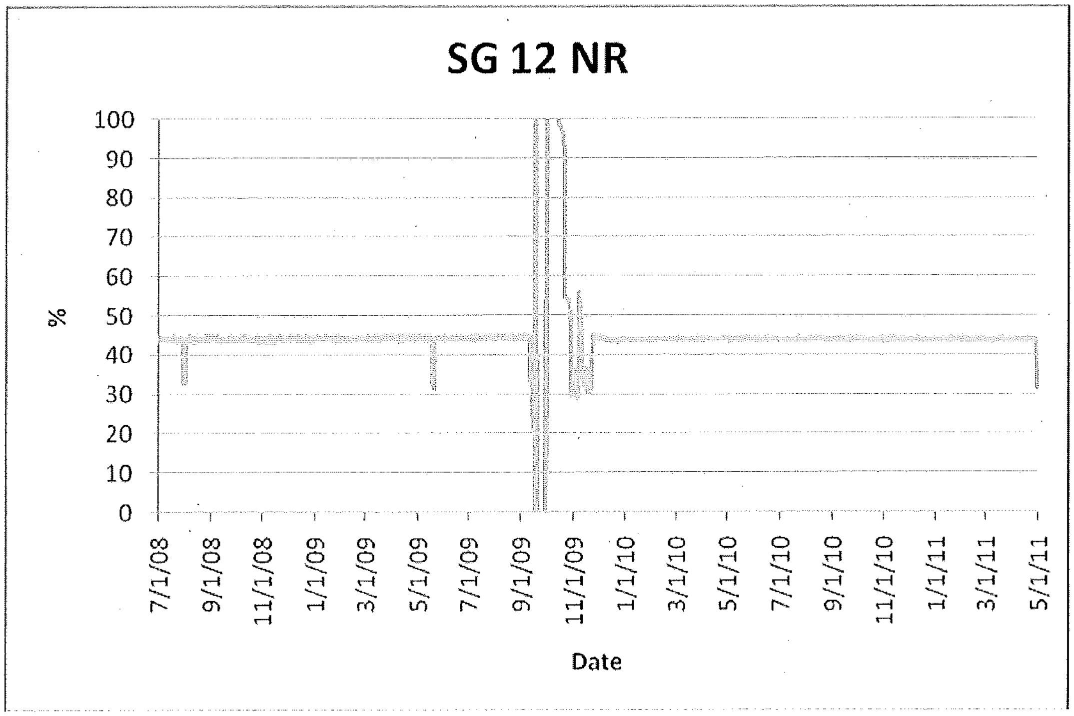
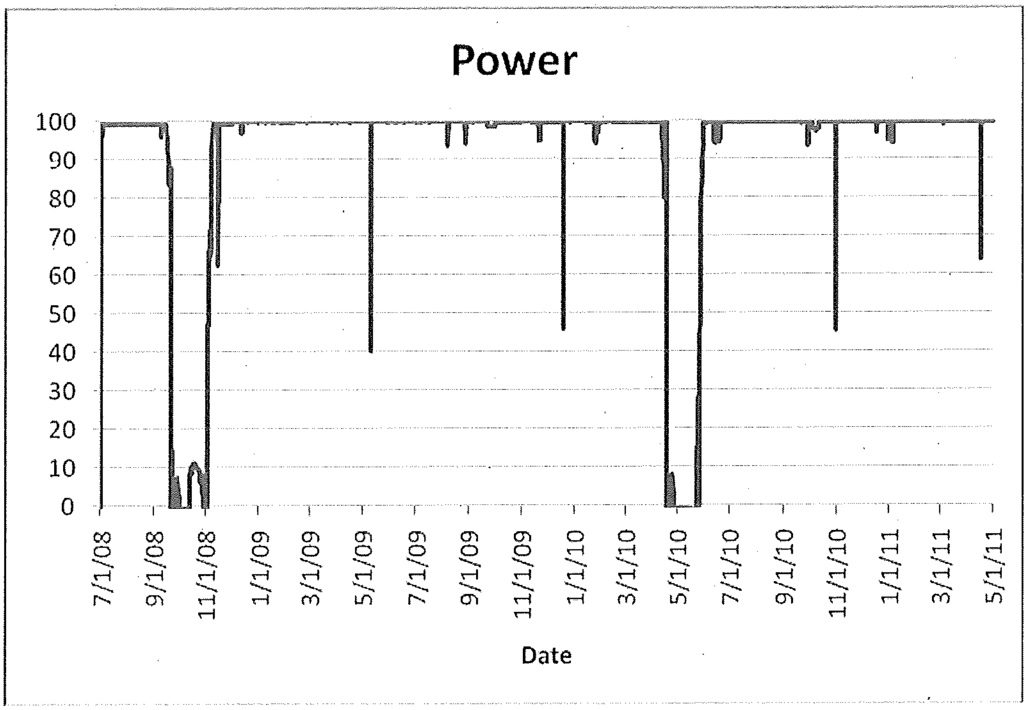


Figure 10, Page 1
Unit 2 Power Level and Steam Generator Narrow Range (NR) Water Level



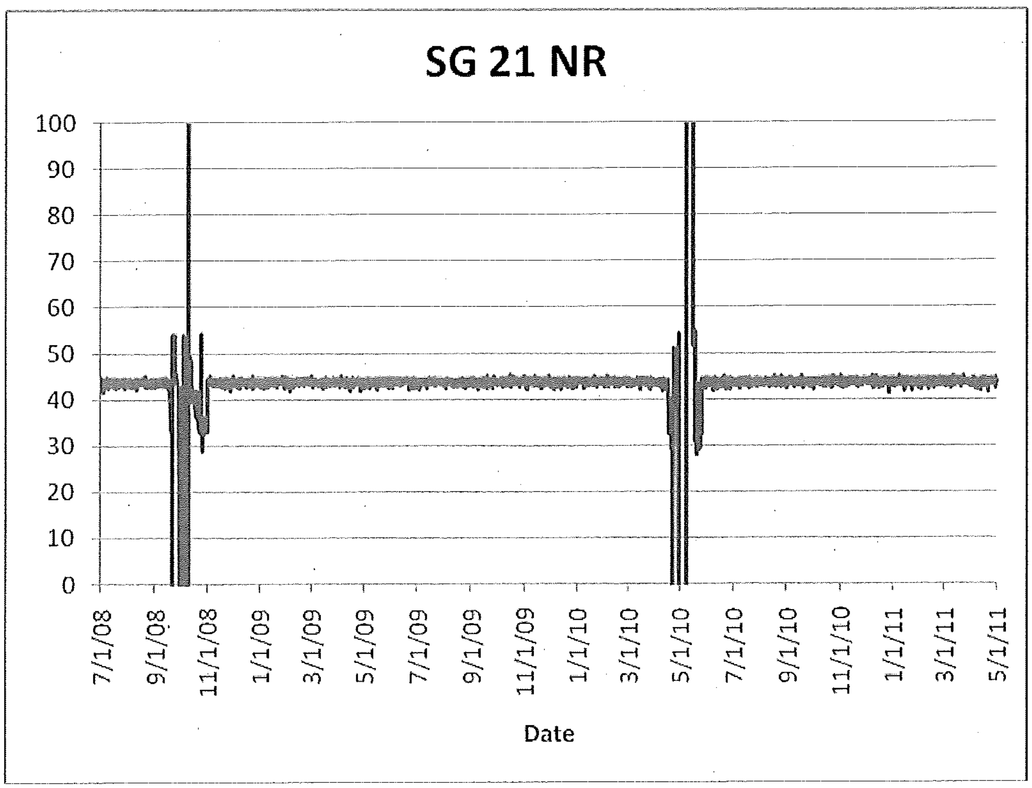
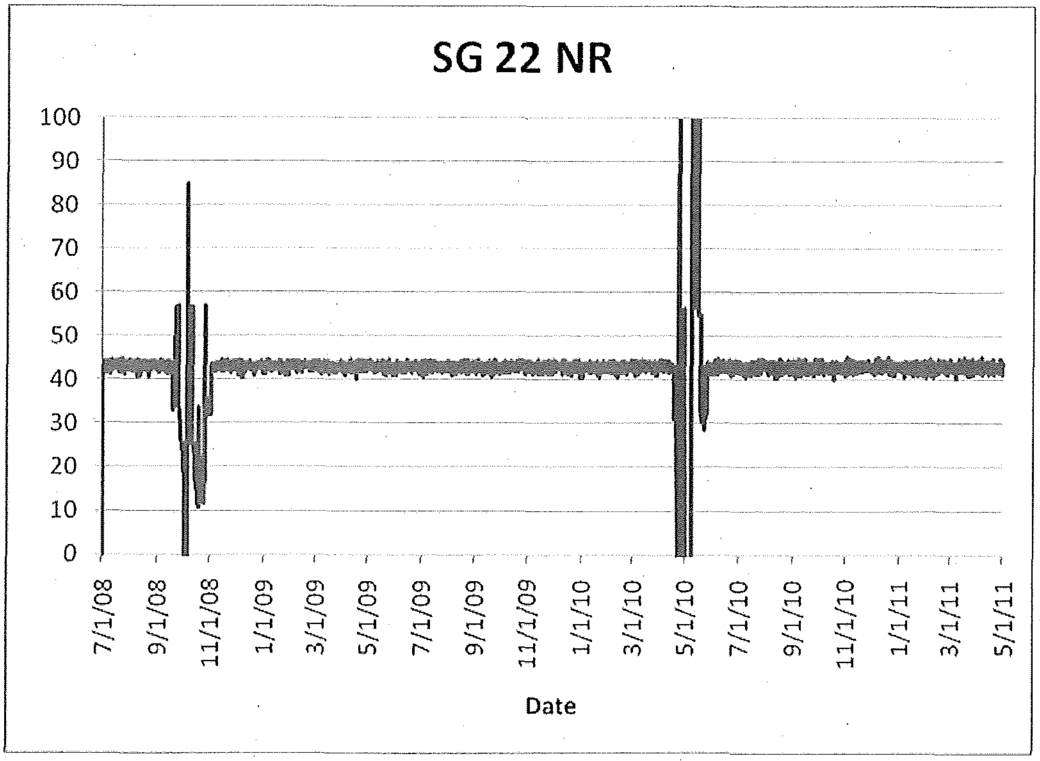


Figure 10, Page 2
Unit 2 Power Level and Steam Generator
Narrow Range (NR) Water Level



A.2.f *Identify operator actions and associated action times credited in the analysis. Where an operator action is credited, confirm that such action is consistent with station procedures and action times are conservative, resulting in a minimum SG MTO.*

Response

The operator actions and associated time frames credited in the MTO analysis are shown in the following table.

Table 5
Operator Actions Credited in Margin to Overfill Analysis

Operator Action	Time for Operator Action
Isolation of AFW to the ruptured SG	Isolated based on SG water level (see below discussion)
Initiation of RCS cooldown time following reactor trip	19 minutes
Initiation of RCS depressurization following termination of RCS cooldown	4 minutes
Secure Safety Injection pumps following termination of RCS depressurization	2 minutes
Balance letdown and charging following securing Safety Injection Pumps.	15 minutes

Isolation of AFW is based on Steam Generator water level. Emergency Operating Procedures direct control room operators to isolate AFW to the ruptured SG when the indicated water level is greater than 5%. To be conservative, the analysis assumes that this action is not performed until the indicated water level reaches 35%. A SG water level of 35% is conservative relative to values observed in the simulator and result in a minimum margin to overfill.

The operator actions credited in the analysis are consistent with the Emergency Operating Procedures (EOPs) for mitigating a SGTR. These same EOPs are used for training and simulator time validation. The operator action times credited in the analysis are conservative and result in a minimum margin to overfill.

A.2.g Update the licensing basis radiological consequence analyses for the AST conditions to reflect radiological consequences of the above-identified limiting release, should they be more severe than the current, proposed, radiological analysis. Since the NRC staff is allowing the single failure exception to the WCAP-10698-P-A methodology, the above requested analysis represents an event that has a significantly higher likelihood of occurrence.

Response

As described above, the MTO analysis predicts that the ruptured steam generator will not be overfilled. Thus, liquid will not be released through the SG PORV or Safety Valves, and the SGTR radiological consequence analysis provided in the Reference 1 LAR, as supplemented by Reference 2 remains bounding. Therefore, there is no need to update the radiological consequence analysis previously provided.

A.2.h *Identify how procedures address the steam generator overfill condition. What parameters do operators monitor to help ensure that overfill does not occur?*

Response

The PINGP procedural guidance for a SGTR event is consistent with Westinghouse Owners' Group (WOG) Emergency Response Guidelines (ERGs). For a SGTR event with loss of offsite power, as the event progresses, water level in the ruptured steam generator could potentially go off scale high on control room indications. Once this condition is reached, however, there is significant volume (approximately 1300 ft^3) available to accommodate break flow into the ruptured steam generator. In order to minimize the potential for overfill, the procedural guidance directs the operator to continue with rapid cooldown and depressurization of the RCS, secure safety injection, and maintain RCS and ruptured steam generator pressures equal until transition to a recovery procedure.

A.2.i For any revised radiological consequence analyses, provide the basis for the assumed flashing fraction, if it is less than 100 percent.

Response

This response is not required as, described above in the response to question 2.g, it was not necessary to update the radiological consequence analyses.

B. RAI Related to the SGTR Mass Release Analysis

B.1 Information on page 116 of the October 27, 2009 LAR indicates that the results of a recent Westinghouse SGTR analysis were used to determine: (1) primary coolant releases to the ruptured SG; (2) steam mass releases from ruptured SG to the environment; and (3) steam mass releases from intact SG to the environment.

Provide a discussion of the Westinghouse SGTR analysis for mass releases determination and verify that the methods used in the analysis are NRC- approved methods, and address compliance with restrictions and conditions specified in the NRC safety evaluation report approving the methods and computer codes. The requested information should also include the plant parameters considered in the analysis, identify the major input initial conditions and the worst single failure used in the analysis, discuss the bases used to select the numerical parameters and demonstrate that the numerical values with consideration of the uncertainties and fluctuations around the nominal values are conservative, resulting in maximum mass releases during an SGTR event. The results to be provided should include sequences of the event with specification of operator actions, associated times credited in the analysis and their bases for acceptance, and the response of key parameters versus time.

Also, address the acceptability of the analysis performed at the extended power uprate (EPU) power level to the AST application, which is based on the current power level.

Response

As indicated in Section 3.7.5.2 (page 114) of Reference 1, the current licensing basis analysis for the SGTR is described in the Updated Safety Analysis Report (USAR) (Section 14.5.4.3) and is consistent with that contained in the original Final Safety Analysis Report (FSAR). This analysis method is the original and current licensing basis for PINGP Units 1 and 2. It also was used as the basis for the SGTR dose analyses in AST applications for a number of other plants. NRC approval of these applications is documented in References 4 through 9. As discussed further below, computer codes are not used for the analysis, specific operator actions are not modeled and a detailed sequence of events is not generated. The PINGP licensing basis does not include consideration of a single failure. The calculation includes conservative consideration of a higher power level corresponding to a planned extended power uprate (EPU) power level which bounds current operation since it results in increased steam releases from the ruptured and intact steam generators (SGs). The calculations will be repeated in detail for the EPU and confirmed to remain bounding.

The SGTR calculations performed for the earlier Westinghouse plants, including PINGP, did not include a computer analysis to determine the plant transient behavior following a SGTR. Rather, a simplified thermal-hydraulic approach was utilized. This simplified thermal-hydraulic analysis assumes that primary-to-secondary break flow continues until 30 minutes from the start of the event and includes conservative assumptions that maximize the primary to secondary break flow and the steam release to the atmosphere for use in calculating the radiological dose consequences of the event. The current licensing basis for PINGP Units 1 and 2 is consistent with plants which received their operating license prior to the R. E. Ginna Nuclear Power Plant SGTR event which prompted development of the WCAP-10698-P-A steam generator tube rupture methodology.

The accident considered is the double-ended rupture of a single steam generator tube. The primary to secondary break flow rate is calculated using the orifice equation and neglecting the frictional losses in the tube. It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the reactor coolant system (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. The analysis assumes that reactor trip and SI actuation occur simultaneously when the pressurizer pressure decreases to the SI actuation setpoint. Loss of offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator safety valves. Immediately following reactor trip and SI actuation it is assumed that the RCS pressure stabilizes at the equilibrium point where the incoming SI flowrate equals the outgoing break flowrate. The equilibrium primary-to-secondary break flow is assumed to persist until 30 minutes after the initiation of the SGTR.

A portion of the break flow will flash directly to steam upon entering the secondary side of the ruptured SG. Although not included in previous PINGP SGTR calculations, the calculations performed to provide input to the radiological consequences analysis for the LAR incorporates a break flow flashing fraction. Since a transient break flow calculation is not performed, a detailed time dependent flashing fraction that incorporates the expected changes in primary side temperatures cannot be calculated. Instead, a conservative calculation of the flashing fraction is performed using the limiting conditions from the break flow calculation. Two time intervals are considered, as in the break flow calculations: pre- and post-reactor trip (SI initiation occurs concurrently with reactor trip). Since the RCS and SG conditions are different before and after the trip, different flashing fractions would be expected. For the flashing fraction calculations it is conservatively assumed that all of the break flow is at the hot leg temperature and that there is no reduction in hot leg temperature despite the reactor trip and subsequent plant cooldown. This is an especially limiting assumption since it maintains a constant flashing fraction from the time of trip until 30 minutes when break flow is terminated, while in an actual SGTR the operators must perform the plant cooldown using the intact SG to assure subcooling at the ruptured SG pressure and this cooldown must be performed at a point in the transient well before break flow termination.

The steam released from the steam generators to the environment after reactor trip until 30 minutes is determined from a mass and energy balance for the primary and secondary systems. The energy which must be dissipated during this period includes the energy generated in the core and the change in the plant sensible heat between the initial and final conditions. The steam released from the ruptured SG is determined by dividing the total steam release by the number of steam generators in the plant. This is a conservative simplification since it assumes that the ruptured SG participates equally in removing the decay heat in the period from reactor trip until break flow termination while the plant emergency operating procedures instruct that only the intact SG should be used to perform the cooldown. The steam release calculation also conservatively neglects energy absorption by the injection of relatively cold safety injection flow directly into the RCS.

After 30 minutes, it is assumed that steam is released only from the intact SG in order to dissipate the core decay heat and to cool the plant down to the residual heat removal (RHR) system operating conditions. (During post-SGTR cooldown, the pressure in the ruptured steam generator is assumed to be decreased by the backfill method in which core decay heat and RCS fluid energy is dissipated by releasing steam from the intact steam generator. This is the preferred approach in the plant emergency operating procedures since it minimizes the radioactivity released to the atmosphere.) A primary and secondary side mass and energy balance is used to calculate the steam release from the intact SG from 0 to 2 hours, from 2 to 8 hours, and from 8 to 14 hours when the RHR is assumed to be in service removing all decay heat.

A summary of the key inputs used in the calculation follows:

- Core power level of 1811 MWt
- Nominal RCS pressure of 2250 psia
- RCS average temperature range of 560.9°F to 574°F
- 0% to 10% steam generator tube plugging (SGTP)

- AREVA Model 56/19 SGs for Unit 1 and Westinghouse Model 51 SGs for Unit 2
- Low pressurizer pressure SI actuation setpoint = 1845 psia
- Lowest steam generator safety valve reseat pressure = 1016 psia (includes 4% main steam safety valve (MSSV) blowdown and 3% tolerance)
- Maximum high head safety injection (HHSI) flow rates are assumed. The injection flow is used to determine the equilibrium RCS pressure, where injection flow equals break flow and the corresponding break flow is modeled from reactor trip until 30 minutes. Transient changes in pressure would reduce the injection flow and result in break flow rates lower than this equilibrium value. These are not considered.
- Decay heat based on the 1971 American Nuclear Society (ANS) decay heat model +20%

Eight distinct cases were considered. The first four cases modeled the Unit 1 SGs at the varying conditions of 0% and 10 % SGTP (and the associated secondary side conditions), and high and low values for the RCS average temperature. The second four cases modeled the Unit 2 SGs at the same varying combinations of tube plugging, and RCS average temperature. These 8 cases were considered individually to determine the primary-to-secondary break flow and steam releases to the atmosphere for the dose analysis between 0 and 30 minutes. The limiting break flow from all of the different calculations along with the limiting steam released to the atmosphere are used in the dose calculation. A single calculation was performed to calculate the long-term steam releases from the intact steam generator for the time intervals 0 to 2 hours, 2 to 8 hours, and 8 to 14 hours. The Unit 2 SG configuration provided the bounding break flow and steam releases.

The break flow flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side. Therefore, the highest flashing will be predicted for the case with the highest primary side temperatures. Similarly, a lower secondary side pressure maximizes the difference in the primary and secondary enthalpies, although a lower pressure would have a higher heat of vaporization that would result in less flashing. The highest possible pre-trip flashing fraction based on the range of operating conditions covered by this analysis is for a case with a hot leg temperature of 606.8°F, RCS pressure of the SI setpoint of 1845 psia and initial secondary pressure of 740 psia. All cases consider the same post-trip RCS equilibrium pressure of 2062 psia and post-trip SG pressure of 1016 psia. The corresponding calculated flashing fractions are 0.18 before trip and 0.12 after trip.

The results of the analysis are reflected in the input listed in Table 3.7-7 (page 116) of Reference 1.

Following approval of the WCAP-10698-P-A methodology, the NRC did not require plants that received their operating licenses prior to the R.E. Ginna Nuclear Power Plant SGTR event to update their analyses to the new methodology. In order to confirm the conservative nature of the licensing basis input to dose mass releases, a supplemental input to dose analysis was performed for PINGP Units 1 and 2 at planned EPU conditions. The supplemental analysis modeled operator responses leading to break flow termination consistent with the PINGP SGTR Emergency Operating Procedure. The supplemental thermal hydraulic evaluation was performed to verify that the radiological consequences analysis input determined by the licensing basis hand calculation input to dose analysis discussed previously are bounding and conservative despite continuation of break flow beyond 30 minutes.

The thermal hydraulic input to dose analysis was performed using the LOFTTR2 computer code and modeling from WCAP-10698-P-A and its supplement. The evaluation includes explicit simulation of operator actions leading to break flow termination based on the PINGP EOPs and simulator studies specific to PINGP Units 1 and 2. The analysis considers the allowable vessel average temperature and SGTP ranges consistent with the hand calculation input to dose analysis, as well as a consistent power level. The analysis models reactor trip on over-temperature delta-temperature and SI actuated on low pressurizer pressure. The analysis does not include consideration of a single failure, assumes nominal plant conditions without consideration of uncertainties, and assumes nominal initial secondary mass consistent with the approach approved in the Point Beach Nuclear Plant safety evaluation report (SER) (Reference 4). Consideration of uncertainties on nominal conditions would have a small impact on the flashed break flow. Following reactor trip and the assumed loss of offsite power, the reactor coolant system temperatures trend towards the no-load temperature, independent of the initial conditions assumed. The initial secondary mass mainly impacts the steam releases. The licensing basis analysis assumes the ruptured SG participates equally in removing the decay heat in the period from reactor trip until break flow termination while the supplemental analysis utilizes the intact SG for the cooldown, consistent with the plant EOPs. Adding conservatism to the initial SG mass would not change the conclusion that the licensing basis analysis is bounding. Conservatisms contained in the supplemental evaluation include consideration of maximum SI flow rate, minimum auxiliary feedwater (AFW) flow rate, maximum AFW initiation delay, and maximum decay heat. Note that the decay heat model includes uncertainty to maximize the releases and time required to cool the RCS.

Another conservatism included in the supplemental analysis is the break flow flashing fraction, which is determined using the hot leg temperature. Since the tube rupture flow calculated using the LOFTTR2 code consists of flow from the hot leg and cold leg sides of the SG, the actual break flow temperature and the flashing fraction is much lower. This approach of maximizing flashing fraction by assuming all flow is at the hot leg temperature is similar to the licensing basis analysis approach; however, the LOFTTR2 code is able to calculate a flashing fraction based on the transient changes in primary and secondary side conditions. This results in reduced flashing fractions than those calculated using the licensing basis approach.

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 PINGP Units 1 and 2 EOPs, and major actions were explicitly modeled in this analysis. The main operator actions modeled leading to break flow termination are identification and isolation of the ruptured steam generator, cooldown and depressurization of the RCS to reduce break flow and restore inventory, and termination of SI flow to stop primary-to-secondary break flow. The operator actions modeled in the analysis are discussed in more detail as follows.

Following the tube rupture, the RCS pressure decreases as shown in Figure 11 due to the primary-to-secondary break flow. In response to this depressurization, the reactor trips on overtemperature-ΔT at approximately 89 seconds. The main feedwater flow was assumed to be terminated and AFW flow was assumed to be automatically initiated with a maximum delay following reactor trip and the coincident LOOP. After reactor trip, core power rapidly decreases to decay heat levels and the RCS depressurization becomes more rapid. The steam dump to condenser system is inoperable due to the assumed LOOP, which results in the secondary pressure rising to the steam generator PORV setpoint as shown in Figure 11. The decreasing pressurizer pressure leads to an automatic SI signal on low pressurizer pressure at approximately 192 seconds. Following SI initiation, the high head safety injection (HHSI) flow begins to restore the reactor coolant inventory and the RCS pressure trends toward the equilibrium value where the SI flow rate equals the break flow rate. After the SGTR and reactor trip, the following operator actions are modeled to mitigate the SGTR event.

1. Isolate auxiliary feedwater flow to the ruptured steam generator

Following reactor trip, auxiliary feed flow to the ruptured steam generator is stopped based on a steam generator level of greater than 35% narrow range. For an SGTR that results in a reactor trip at high power as assumed in this analysis, the steam generator water level as indicated on the narrow range will decrease significantly for both steam generators. The AFW flow will begin to refill the steam generators. Since primary-to-secondary leakage adds additional inventory to the ruptured steam generator, the water level will increase more rapidly in that steam generator. This response, as displayed by the steam generator water level instrumentation, will result in operator isolation of AFW flow to the affected steam generator.

2. Identify the ruptured steam generator

   The SGTR EOP instructs the operators to identify the ruptured steam generator based on a number of criteria, including an unexpected increase in any steam generator narrow range level and high radiation indications. The first actions to isolate AFW to the affected steam generator have already provided indication that it is a ruptured steam generator. Primary-to-secondary leakage will continue to add additional inventory to the ruptured steam generator so the water level will continue to increase even after AFW isolation. This response, as displayed by the steam generator water level instrumentation, provides confirmation of an SGTR event and also identifies the ruptured steam generator.

3. Isolate steam flow from the ruptured steam generator

   Once the ruptured steam generator has been identified, operators continue recovery actions by isolating steam flow from the ruptured steam generator. This enables the operators to establish a pressure differential between the ruptured and intact steam generators as a necessary step toward terminating primary-to-secondary break flow. Isolation of steam flow from the ruptured steam generator was assumed to be completed immediately following AFW termination to provide conservative ruptured steam generator releases.

4. Cooldown the RCS using the intact steam generator

   After isolation of the ruptured steam generator steamline, actions are taken to cool the RCS as rapidly as possible by dumping steam from the intact steam generator. Since offsite power is lost, the RCS is cooled by dumping steam to the atmosphere using the atmospheric dump valves on the intact steam generator. An operator action is assumed to initiate cooldown 19 minutes from reactor trip, at 1230 seconds. The cooldown is continued until the core exit temperature is less than the target temperature, which provides the necessary RCS subcooling at the ruptured steam generator pressure. The reduction in the intact steam generator pressure required to accomplish the cooldown is shown in Figure 11. When the target temperature is reached at 1936 seconds as determined by the LOFTTR2 code, it is assumed that the operator terminates the cooldown and maintains the RCS temperature using the intact steam generator atmospheric dump valves. This cooldown ensures that there will be adequate subcooling in the RCS after the subsequent depressurization of the RCS to the ruptured steam generator pressure.

5. Establish charging flow

   Following initiation of the RCS cooldown, the PINGP procedures instruct the operators to establish charging flow to help maintain RCS inventory lost through the primary-to-secondary break flow. Modeling initiation of charging flow is conservative for a SGTR analysis and is consistent with the PINGP specific EOPs and operator training. Actions to initiate charging flow are performed at 1232 seconds.

6. Depressurize the RCS to reduce break flow and restore reactor coolant inventory

   The RCS is depressurized to reduce the break flow rate. SI flow will tend to increase RCS pressure until break flow matches injection flow. Consequently, HHSI flow must be terminated to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be assured. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after HHSI flow is stopped. If sufficient subcooling is not available, or a high level in the pressurizer is approached, the depressurization is terminated.

   The RCS depressurization is performed using normal pressurizer spray if the reactor coolant pumps are running. Since offsite power is assumed to be lost at the time of reactor trip, the reactor coolant pumps are not running and, thus, normal pressurizer spray is not available. Therefore, the depressurization is modeled using a single pressurizer power operated relief valve.

   After the RCS cooldown is completed, a conservatively bounding 7-minute operator action time is included prior to the RCS depressurization. The RCS depressurization is initiated at 2356 seconds and continued until any of the following conditions are satisfied: RCS pressure is less than the ruptured steam generator pressure and pressurizer level is greater than the allowance of 7% for pressurizer level uncertainty, or pressurizer level is greater than 75%, or RCS subcooling is less than the 20°F allowance for subcooling uncertainty. The LOFTTR2 code determined conditions are met for RCS depressurization termination at 2446 seconds. The RCS depressurization reduces the break flow rate as shown in Figure 12 and increases SI flow to refill the pressurizer.

7. Terminate SI flow

   The previous actions establish adequate RCS subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to ensure that HHSI flow is no longer needed. When these actions have been completed, the HHSI flow must be stopped to prevent re-pressurization of the RCS and to terminate primary-to-secondary leakage. The HHSI flow is terminated at this time if RCS subcooling is greater than the 20°F allowance for subcooling uncertainty, maximum AFW flow is available or the intact steam generator level is within the required range, the RCS pressure is stable or increasing, and the pressurizer level is greater than the 7% allowance for uncertainty. After depressurization is completed, an operator action time of 2 minutes was assumed prior to HHSI flow termination. Since the LOFTTR2 code determined that the above requirements are satisfied, SI flow termination actions were performed at 2566 seconds. After HHSI flow termination, the RCS pressure begins to decrease as shown in Figure 11.

8. Balance charging flow to minimize primary-to-secondary leakage

   Once HHSI has been stopped, charging flow, letdown flow, and pressurizer heaters will then be controlled to prevent re-pressurization of the RCS and re-initiation of leakage into the ruptured steam generator. In the LOFTTR2 modeling, charging flow is terminated in place of modeling the balance of charging and letdown flow. An operator action time of 15 minutes was assumed prior to effective termination of charging flow. Charging flow was effectively terminated at 3466 seconds. No actions are modeled to reduce RCS pressure after termination of SI or charging flow. The break flow gradually reduces the RCS pressure until it equals the secondary pressure and break flow is terminated. Break flow termination occurs at 3830 seconds. The primary-to-secondary break flow rate throughout the recovery operations is presented in Figure 12. 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 actions and corresponding operator action times used for the analysis are summarized in Table 6. These operator response time inputs were obtained from a simulated SGTR event at PINGP.

   Table 8 contains a comparison of the thermal hydraulic mass transfer results for the 30-minute licensing basis hand calculation input to dose analysis to the supplemental thermal hydraulic input to dose analysis. The calculated sequence of events is presented in Table 7. Figures 11 through 16 contain plant transient responses to the tube rupture event including primary and secondary pressure, primary-to-secondary break flow, primary-to-secondary flashing fraction, and secondary steam releases. Conservatism was added to the reported results to cover plant changes that may impact the SGTR analysis, such that a recalculation does not need to be performed for minor changes to the plant. Note that Figures 11 through 16 do not reflect the added conservatism.

The supplemental thermal hydraulic analysis demonstrates that, despite the continuation of break flow beyond the 30-minute assumption used in the licensing basis SGTR analysis, the thermal hydraulic mass transfers resulting from a transient analysis modeling operator actions are bounded by those calculated for the licensing basis analysis. Table 8 shows that the supplemental thermal hydraulic analysis results in an approximate increase of 18% in total tube rupture break flow (pre- and post-trip) and 13% in intact steam generator steam releases (trip to 2 hours), while showing a decrease of approximately 77% in total flashed break flow (pre- and post-trip) and 52% in ruptured steam generator steam releases. The difference in intact steam generator steam releases would not have a significant impact on a dose analysis due to the minimal activity contained in the intact steam generator. Flashed break flow, which shows the largest reduction compared to the licensing basis analysis, would be expected to have the greatest impact on the SGTR radiological dose consequences analysis since it is generally modeled as a direct release from the RCS to the environment with no mitigation in the secondary side of the ruptured steam generator. The increase in total break flow is more than offset by the reduction in actual releases (i.e., flashed break flow and ruptured steam generator steam release). Note from Table 7 that the break flow flashing stops at 1442 seconds, during cooldown using the intact steam generator. As seen in Figure 14, the flashing fraction is reduced over time and this reduction is taken into account for the resulting mass transfers. In contrast, the licensing basis analysis maintains a constant flashing fraction for the entire 30-minute break flow duration.

As such, the 30-minute mass transfer data from the licensing basis hand calculation used in the dose analysis provides conservative results when compared to a transient analysis modeling operator actions, with break flow duration lasting longer than 30 minutes. The evaluation was performed without inclusion of a single failure, assumes nominal conditions without uncertainties, and does not minimize initial secondary mass. The approach is judged to be acceptable, since a comparison of the releases with the licensing basis hand calculation, to those determined by the transient analysis modeling operator actions, provides a credible demonstration that the licensing basis hand calculation is quite conservative. The comparison and results are similar to those shown for the Point Beach Nuclear Plant SER (Reference 4).

Table 6
Operator Action Times for PINGP Supplemental Thermal-Hydraulic Analysis

Action	Time
Operator action time to isolate auxiliary feedwater flow to the ruptured steam generator following reactor trip	AFW isolated on SG level, LOFTTR2-calculated
Operator action time to close main steam isolation valve to isolate steam flow from the ruptured steam generator[1]	Immediately following AFW isolation
Operator action time to initiate cooldown following reactor trip	19 minutes
Operator action time to establish maximum charging flow	Immediately following cooldown initiation
Plant response to complete cooldown	LOFTTR2-calculated
Operator action time to initiate depressurization following completion of cooldown	7 minutes
Plant response to complete depressurization	LOFTTR2-calculated
Operator action time to terminate emergency core cooling system (ECCS) flow following completion of depressurization	2 minutes
Operator action time to balance letdown and charging flow following safety injection termination	15 minutes
Plant response until break flow termination resulting from primary and secondary pressure equalization	LOFTTR2-calculated

[1] *No operator action time was given. A minimum time after AFW isolation is used to provide conservative ruptured steam generator releases to atmosphere.*

Table 7 Sequence of Events for PINGP SGTR Supplemental Thermal Hydraulic Analysis	
Event	**Time (sec)**
SGTR Occurs	0
Reactor Trip and Loss of Offsite Power	89
Initiation of Auxiliary Feedwater	149
Initiation of Safety Injection	192
Isolation of Auxiliary Feedwater Flow to Ruptured Steam Generator	738
Isolation of Main Steam Isolation Valve to Ruptured Steam Generator[2]	740
Initiation of Cooldown with Intact Steam Generators	1230
Initiation of Maximum Charging Flow	1232
Break Flow Flashing Stops	1442
Termination of Cooldown	1936
Initiation of Depressurization	2356
Termination of Depressurization	2446
Termination of Safety Injection	2566
Balance of Charging and Letdown Flow	3466
Termination of Break Flow	3830

[2] A minimum time following AFW isolation is assumed for MSIV isolation to provide conservative ruptured steam generator releases to atmosphere.

Table 8

Comparison of Thermal Hydraulic Results for PINGP SGTR Input to Dose Analysis

	Releases Presented in AST Submittal	**Supplemental Thermal Hydraulic EPU Input to Dose Analysis**
Pre-Trip Break Flow[3]	14,600 lbm	5,500 lbm
Post-Trip Break Flow[3]	125,400 lbm	159,300 lbm
Pre-Trip Flashed Break Flow	2,630 lbm	900 lbm
Post-Trip Flashed Break Flow	15,050 lbm	3,100 lbm
Post-Trip Ruptured Steam Generator Steam Release	80,500 lbm	38,700 lbm
Intact Steam Generator Releases from Trip to 2 hours	237,100 lbm	266,900 lbm[4]
Intact Steam Generator Releases from 2 to 8 hours[5]	569,000 lbm	
Intact Steam Generator Releases from 8 to 14 hours[5]	416,100 lbm	

[3] Includes flashed break flow.

[4] The supplemental analysis only calculates intact SG steam releases from trip until break flow termination. Therefore, the 30 minute to 2 hour intact SG release of 156,600 lbm calculated in the licensing basis analysis for the AST submittal was included in the presented value to provide comparable time period results.

[5] Values calculated in licensing basis are assumed to apply to both analyses.

Figure 11: SGTR Supplemental Input to Dose Evaluation
RCS and Secondary Pressures

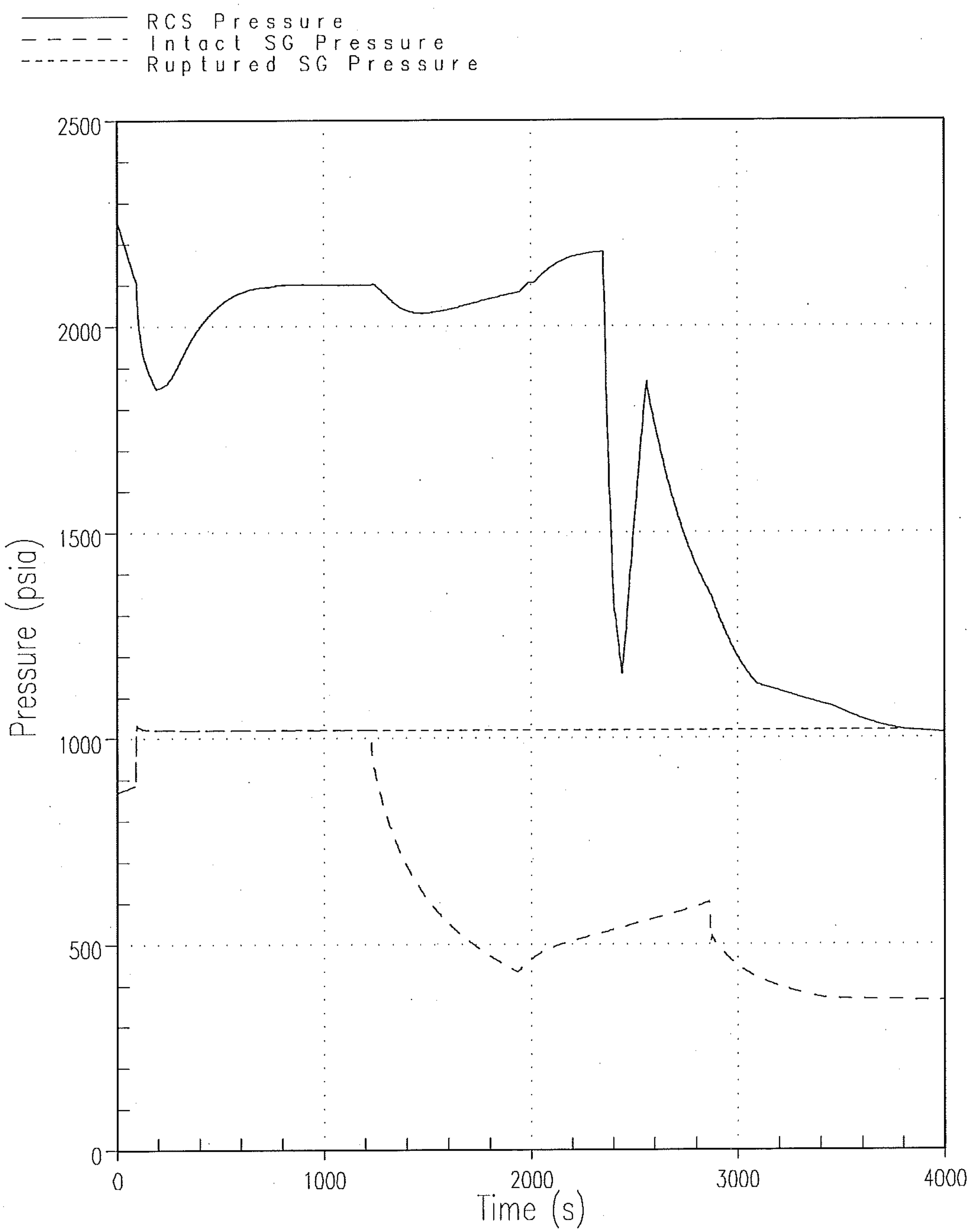


**Figure 12: SGTR Supplemental Input to Dose Evaluation
Primary-to-Secondary Break Flow Rate**

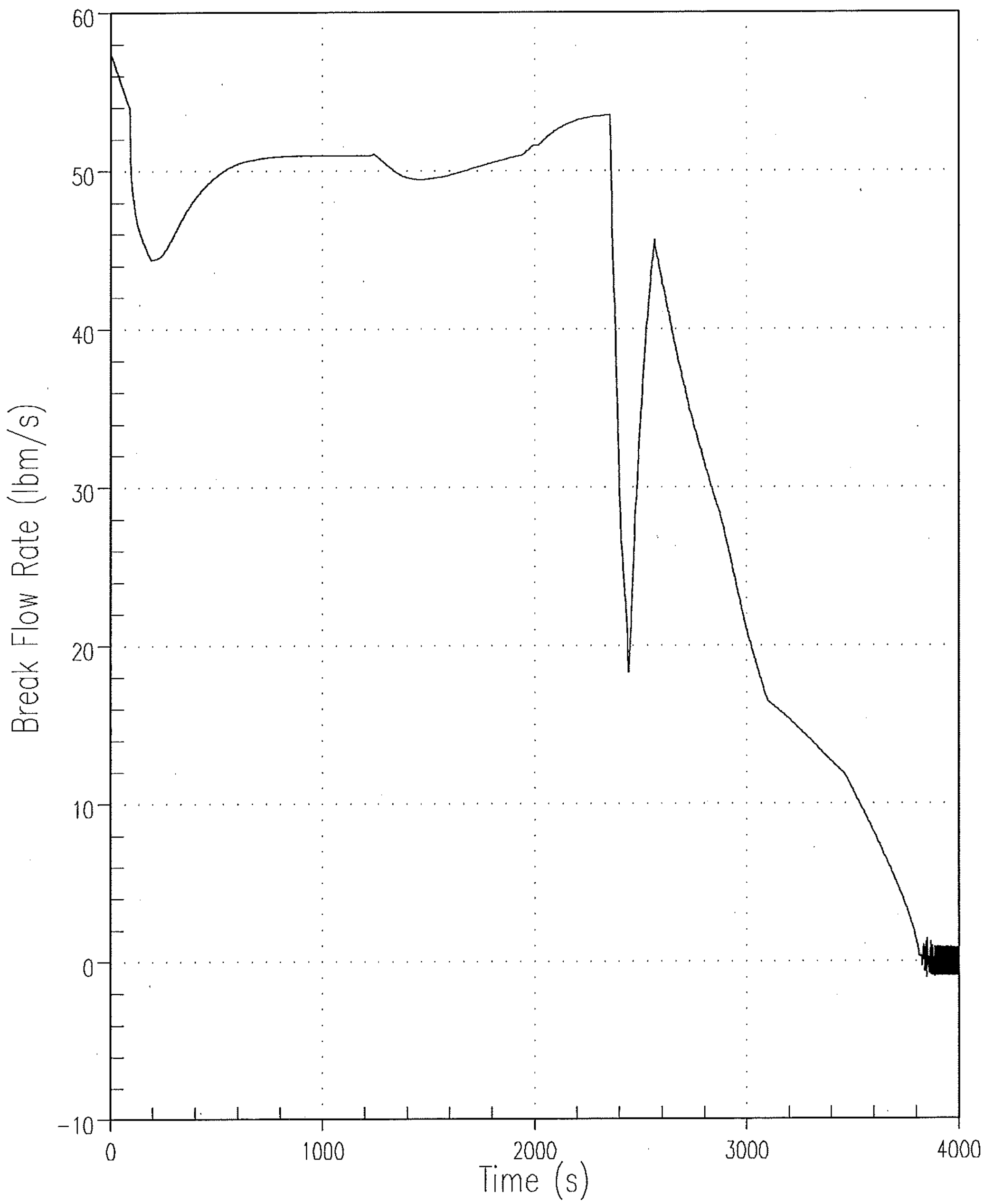


**Figure 13: SGTR Supplemental Input to Dose Evaluation
Integrated Primary-to-Secondary Break Flow**

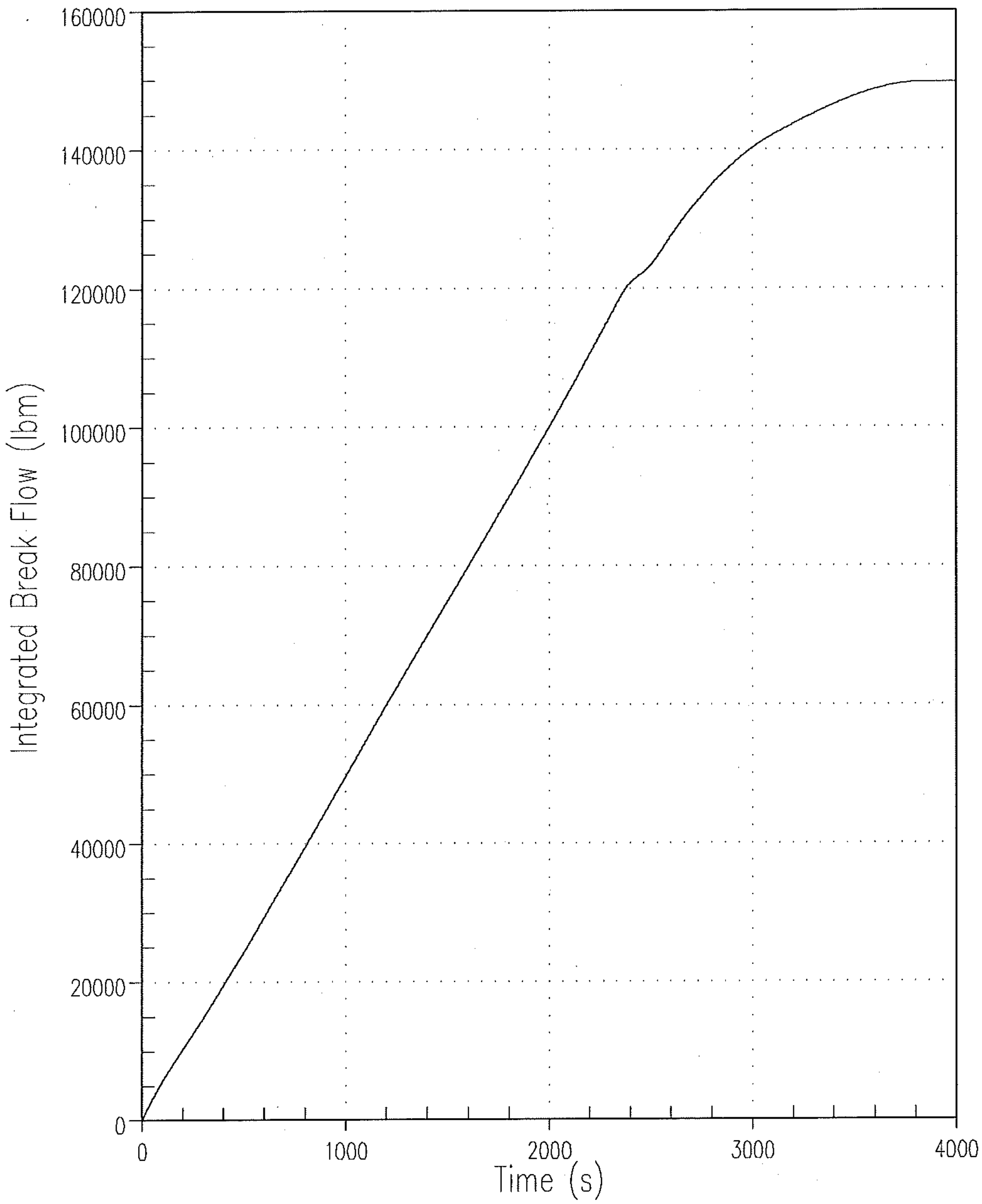


Figure 14: SGTR Supplemental Input to Dose Evaluation Primary-to-Secondary Break Flow Flashing Fraction

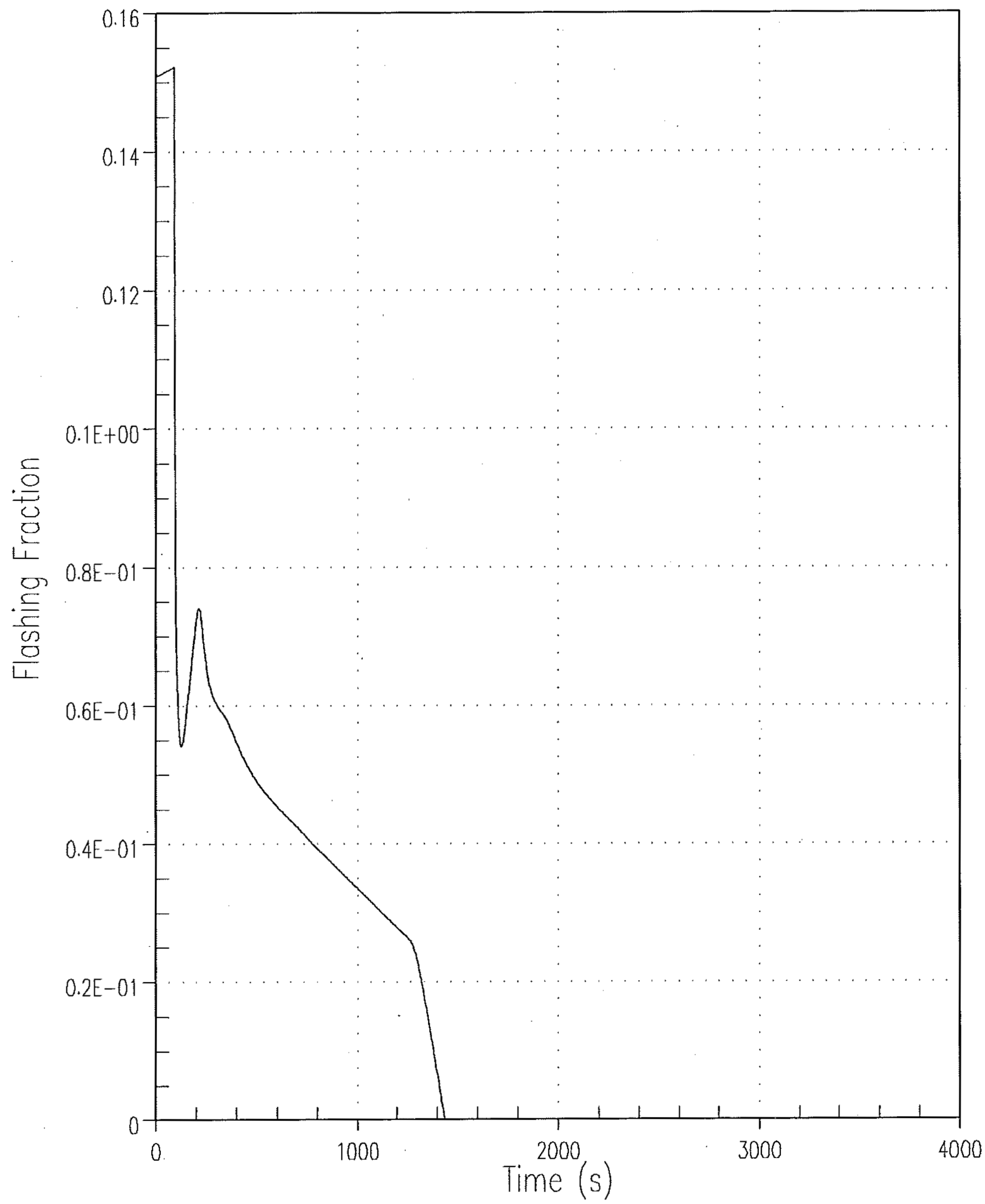


**Figure 15: SGTR Supplemental Input to Dose Evaluation
Integrated Primary-to-Secondary Flashed Break Flow**

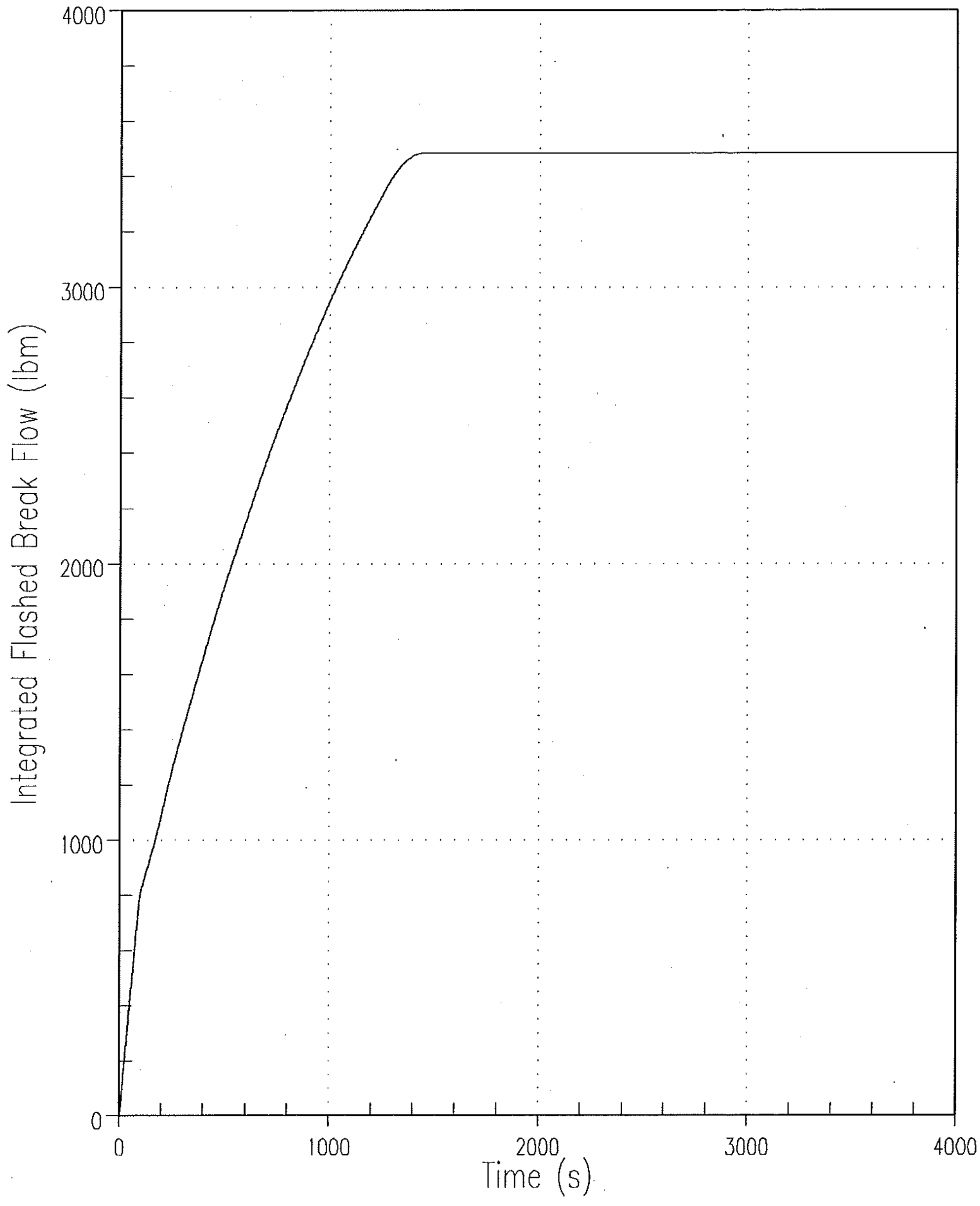
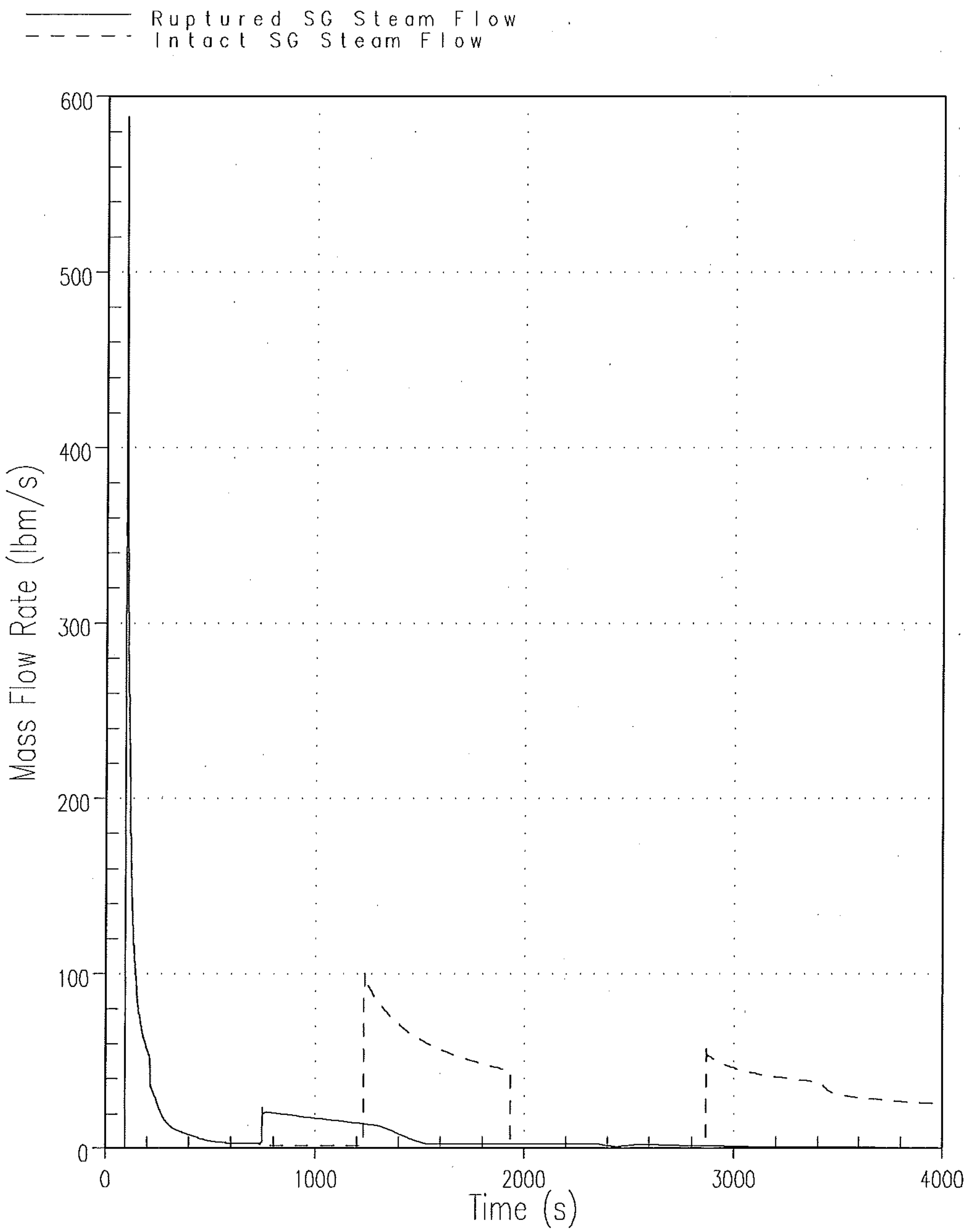


Figure 16: SGTR Supplemental Input to Dose Evaluation Steam Generator Atmospheric Steam Releases



B.2 Page 116 of the LAR indicates that, based on the current PINGP licensing basis, the "termination of release from ruptured SG" was completed within 30 minutes from initiation of the SGTR event.

Discuss the "current licensing basis" for the event termination time of 30 minutes, its effect on and acceptability of the radiological release analysis, and its relationship with the break flow termination time of 30 minutes assumed in the MTO analysis.

Response

As noted above in the response to Part B.1 of this question, the licensing basis SGTR analysis is based on the use of simplified calculations to determine the integrated break flow and the steam release from the SGs to the atmosphere for the assumed 30 minute duration of the accident. This methodology is also used to perform the licensing basis SGTR analyses for many other plants and the results are included in the plant FSARs which are approved by the NRC. The overall calculation provides conservative estimates of the break flow and steam release to the atmosphere for the SGTR radiological consequences analysis, as shown by a comparison to the transient thermal hydraulic analysis in the Part B.1 response.

Following the R.E. Ginna Nuclear Power Plant SGTR event, which occurred in January 1982 the NRC raised several SGTR licensing issues with plants with license applications pending at that time. These included justification of the 30 minute operator action time assumed, qualification of the equipment assumed to be used for SGTR recovery, single failure considerations for the SGTR analysis, and the potential for steam generator overfill. A subgroup of the affected utilities in the Westinghouse Owners Group (WOG) was formed to resolve the SGTR issues. The resolution included the development of a new SGTR analysis methodology which is based on modeling the operator actions for SGTR recovery using design basis operator action times derived from plant specific simulator studies. The revised analysis was to be performed using the LOFTTR2 computer code. The LOFTTR2 program is based on the Westinghouse LOFTRAN code and includes the capability to model operator actions, an improved steam generator secondary side model, and a more realistic tube rupture break flow model. The NRC approved the revised SGTR analysis methodology in 1987 and the methodology has been applied for the SGTR analyses for plants licensed after the R.E. Ginna Nuclear Power Plant SGTR event.

The improved break flow model which incorporates frictional losses in the tube and pressure losses at the break site, results in a lower break flow rate at a given RCS pressure. The transient calculation results in delay from the time of safety injection actuation, until the equilibrium between injection flow and break flow is reached, and in many cases this equilibrium is never reached. Modeling of the operator actions to cool the RCS results in reduced primary pressures and associated break flow rates, as does the operator action to depressurize the RCS. After the RCS depressurization, SI is terminated and break flow gradually decreases to zero. The transient analysis calculation of the RCS temperatures, which includes the post-trip cooldown, cooling due to safety injection flow and manual cooling using the intact SG, results in lower flashing fractions, in comparison to the licensing basis hand calculation. Ruptured SG steam releases are reduced due to the consideration of energy absorption by the cold safety injection flow and the cooldown performed using only the intact SG.

The plant analyses which have been performed using the LOFTTR2 SGTR analysis methodology have utilized operator action times based on simulator studies which have typically resulted in delaying break flow termination beyond the 30 minutes assumed in earlier analyses. The benefits provided by the improved break flow model and the modeling of the operator actions and transient affects, have tended to offset the penalties associated with using longer operator action times leading to later break flow termination.

Although the NRC initially indicated that the SGTR issues raised by the R. E. Ginna Nuclear Power Plant SGTR event were considered to be generic and would likely become backfit issues, the NRC has not required the older plants that used the earlier methodology with the 30 minute break flow termination time to update their SGTR analyses. The NRC has also approved alternative source term submittals and power uprates for plants that continue to use this methodology.

Details of the calculation of the input to the dose analysis are provided in response to Part B.1 of this question. A number of simplifying assumptions are in the calculation that support the conclusion that it provides acceptable input for the radiological analysis despite the assumed 30 minute duration. The most significant are (1) the use of a constant break flow at the equilibrium RCS pressure, (2) the use of a constant hot leg temperature in the calculation of the flashing fraction, (3) the application of this flashing fraction to all of the break flow, (4) neglecting cooling due to safety injection and (5) equal participation in decay heat and stored energy removal by the ruptured SG for the 30 minute duration. As discussed in response to Part B.1 of this question, a detailed analysis has been performed to confirm that radiological consequences analyses performed using the input developed with the simplified modeling and 30 minute break flow termination time are significantly more limiting than those performed using more accurate transient modeling with computer code such as LOFTTR2 incorporating expected operator actions time leading to break flow termination after 30 minutes.

As previously stated, the radiological licensing basis SGTR analysis is based on the use of simplified calculations to determine the integrated break flow and the steam release from the SGs to the atmosphere for the assumed 30 minute duration of the accident in order to determine the onsite and offsite consequence of the accident. The licensing basis SGTR analysis did not consider steam generator overfill. There is no direct connection between the input to dose calculations with the assumed break flow termination time of 30 minutes and the PINGP margin to overfill assessments.

The supplemental margin-to-overfill analysis discussed in the response to A.2, models operator actions leading to break flow termination beyond 30 minutes and demonstrates that overfill does not occur, in consideration of extended operator action times. This demonstrates that break flow is terminated prior to the potential for a liquid release via the PORV and/or Safeties. In addition, the supplemental input to dose analysis discussed in Part B.1 of this question, demonstrates that the event consequence is bounded by the original hand calculation method that utilizes the 30 minute break flow termination assumption, since a larger amount of mass is released to the environment when more accurate transient modeling incorporating extended operator action times beyond 30 minutes are utilized.

C. RAI Related to Update of Updated Safety Analysis Report

Discuss PINGP's plans to reflect the information provided in response to above items A and B in an update of the Updated Safety Analysis Report (USAR), pursuant to the requirements of 10 CFR 50.71, "Maintenance of records, making of reports."

Response

The PINGP Updated Safety Analysis Report (USAR) will be updated to reflect information provided in response to Items A and B above consistent with the requirements of 10 CFR 50.71(e), and NRC Exemption letter dated May 22.2006 (ADAMS Accession No. ML061110032), (Reference 10).

References

1. NSPM Letter to US NRC, "License Amendment Request (LAR) to Adopt the Alternative Source Term Methodology," dated October 27, 2009 (ADAMS Accession No. ML093160583).

2. US NRC Letter to Xcel Energy Energy, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Request for Additional Information (RAI) Associated with Adoption of the Alternative Source Term (AST) Methodology (TAC Nos. ME2609 and ME2610)," dated May 12, 2011 (ADAMS Accession No. ML103540433).

3. NSPM Letter to US NRC, "Response to Requests for Additional Request RE: License Amendment to Adopt the Alternative Source Term Methodology (TAC Nos. ME2609 and ME2610)," dated August 12, 2010 (ADAMS Accession No. ML102300295).

4. US NRC letter to NextEra Energy Point Beach, LLC, "Point Beach Nuclear Plant (PNBP), Units 1 and 2 – Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)," May 3, 2011 (ADAMS Accession Nos. ML110880039 and ML110450159).

5. US NRC letter to Consolidated Edison Company of New York, Inc., "Indian Point Nuclear Generating Unit No. 2 – RE: Issuance of Amendment Affecting Containment Air Filtration, Control Room Air Filtration, and Containment Integrity During Fuel Handling Operations (TAC No. MA6955)," July 27, 2000 (ADAMS Accession No. ML003727500).

6. US NRC letter to Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MB0739 and MB0740)," October 24, 2001 (ADAMS Accession No. ML012690136).

7. US NRC letter to Nuclear Management Company, LLC, "Kewaunee Nuclear Power Plant – Issuance of Amendment Regarding Implementation of Alternate Source Term (TAC No. MB4596)," March 17, 2003 (ADAMS Accession No. ML030210062).

8. US NRC letter to Entergy Nuclear Operations, Inc., "Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendments RE: Full Scope Adoption of Alternative Source Term (TAC No. MC3351)," March 22, 2005 (ADAMS Accession No. ML050750431).

9. US NRC letter to PSEG Nuclear LLC – X04, "Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendments RE: Alternate Source Term (TAC Nos. MC3094 and MC3095)," February 17, 2006 (ADAMS Accession No. ML060040322).

10. NRC Exemption Letter, "Point Beach Nuclear Plant, Units 1 and 2, and Prairie Island Nuclear Generating Plant, Units 1 and 2 – Exemptions to 10 CFR 50.71(e)(4) (TAC Nos. MC8654, MC8655, MC8656, and MC8657), dated May 22, 2006 (ADAMS Accession No. ML061110032).