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Letter Report

Severe Accident Initiated Steam Generator Tube Ruptures Leading to Containment Bypass – Integrated Risk Assessment

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FOREWORD

This report documents the methods, analyses, and results for a hypothetical plant application of an integrated probabilistic risk assessment (PRA) that has been performed to evaluate the risk associated with SG tube failure following low-probability severe accidents in pressurized water reactors (PWRs). The U.S. Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) initiated this work in support of a User Need request by the Office of Nuclear Reactor Regulation (NRR) concerning the behavior of steam generator (SG) tubes during postulated severe accidents. The concern is that a severe accident-induced SG tube rupture (SAI-SGTR) could result in containment bypass for fission products (denoted as SAI-SGCB). This concern resulted in the NRC developing a Steam Generator Action Plan (SGAP) which led to the subject NRR User Need, which requested improved modeling methods of SG tube integrity. The RES response to the subject NRR User Need required integration among three technical disciplines: probabilistic risk assessment (PRA) for assessing risk significance, thermal hydraulics (T-H) for addressing the fluid conditions within the reactor coolant system (RCS), and materials engineering (ME) for the work concerning the integrity of the SG tubes and other reactor coolant system components.

The overall objective of the work documented in this report was to develop improved methods to identify and model severe accident scenarios resulting in SGTRs and containment bypass. The developed methodology provides a framework for integrating the results of the PRA with those from supporting T-H and ME analyses. The ME analyses of the SG tube integrity and the materials response of other RCS components determined the pressure and temperature regimes of interest. The T-H analyses then determined how to get to those regimes. The PRA examines system and component failures that would put the reactor system in those conditions and identifies the operator recovery actions that can mitigate the accident progression. The integrated methodology then provides a framework that logically combines the results from all of these elements, including uncertainties, to provide the risk perspective for the SAI-SGCB issue. The work addressed in this report supports Task 3.5a through 3.5d of the SGAP.

The improved methodology was applied to an example plant. One objective of the example plant application was focused on identifying scenarios leading to SAI-SGTRs and developing the logical framework to calculate the resulting frequency of containment bypass events. A second objective of the application was to improve PRA modeling of high and dry sequences. Although numerical results were generated, those results are illustrative only since the full methodology was not implemented. Furthermore, the results are not applicable to any specific plant.

Both the developed methodology and its limited application went through an informal review by the NRC staff.

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TABLE OF CONTENTS

Page

LIST OF FIGURES

LIST OF TABLES

ACKNOWLEDGEMENTS

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NOMENCLATURE

AC	alternating current
ADV	atmospheric dump valve
AFW	auxiliary feed water
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ATHEANA	A Technique for Human Event Analysis
ATWS	anticipated transient without scram
BWR	boiling water reactor
CDF	core damage frequency
CFD	computational fluid dynamics
DC	direct current
DFC	Diagnostic Flow Chart
ECCS	emergency core cooling system
EOC	error of commission
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERG	emergency response guideline
FWIV	feed water isolation valve
HEP	human error probability
HFE	human factor event
HRA	human reliability analysis
IGA/SCC	intergranular attack/stress corrosion cracking
IPE	Individual Plant Examination
ISL	Information Systems Laboratories, Inc.
LERF	large early release frequency
LOCA	loss of coolant accident
LOSP	loss of offsite power
ME	materials engineering
MSIV	main steam line isolation valve
MSLB	main steam line break
MSSV	main steam safety valve
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Regulation
ODSCC	outer diameter stress corrosion cracking
PDS	plant damage state
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSF	performance shaping factor
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research

SACRG	Severe Accident Control Room Guide
SAIC	Science Applications International Corporation
SAI-SGCB	severe accident-induced steam generator tube ruptures resulting in containment bypass
SAI-SGTR	severe accident-induced steam generator tube ruptures
SAG	Severe Accident Guideline
SAMG	severe accident management guideline
SBO	station blackout
SG	steam generator
SGAP	Steam Generator Action Plan
SGTR	steam generator tube rupture
SHR	secondary heat removal
SLOCA	small loss of coolant accident
SNL	Sandia National Laboratories
SR	Supporting Requirement
SRV	safety/relief valve
SSC	structure, system, or component
TDAFW	turbine-driven auxiliary feed water
T-H	thermal hydraulic
TTS	top of tube sheet
TSC	Technical Support Center
TSP	tube support plate
VSLOCA	very small loss of coolant accident

1. INTRODUCTION

This report documents the methods, analyses, and results for a hypothetical plant application of an integrated probabilistic risk assessment (PRA) that has been performed to evaluate the risk associated with steam generator tube rupture (SGTR) following low-probability severe accidents in pressurized water reactors (PWRs). The U.S. Nuclear Regulatory Commission's (NRC) Office of Nuclear Regulatory Research (RES) initiated this work in support of a User Need request by the Office of Nuclear Reactor Regulation (NRR) concerning the behavior of steam generator (SG) tubes during postulated severe accidents. The concern is that a severe accident-induced SG tube rupture (SAI-SGTR) could result in containment bypass for fission products (denoted hereafter as SAI-SGCB). This concern resulted in the NRC developing a Steam Generator Action Plan (SGAP) [Collins, 2001], which led to the subject NRR User Need, which requested improved modeling methods of SG tube integrity. The RES response to the subject NRR User Need required integration among three technical disciplines: probabilistic risk assessment (PRA) for assessing risk significance, thermal hydraulics (T-H) for addressing the fluid conditions within the primary system, and materials engineering (ME) for the work concerning the integrity of the SG tubes and other primary system components.

In order to complete the work required to address the subject NRR User Need, RES contracted with two Department of Energy (DOE) National Laboratories and two commercial companies. Information Systems Laboratories, Inc. (ISL) performed analyses that have subsequently been used to evaluate the T-H plant response during postulated accident event sequences for the PRA. Argonne National Laboratories (ANL) performed the ME modeling and analyses of SG tube integrity and materials response of other reactor coolant system (RCS) components. Sandia National Laboratories (SNL) and Science Applications International Corporation (SAIC) developed the PRA methodology and a framework within which to integrate the results of the PRA with those of the T-H and ME analyses. The PRA effort addressed in this report supports Task 3.5a through 3.5d of the SGAP, the elements of which are shown in Figure 1, along with the integration points for the T-H and ME efforts.

1.1 Background and History

Steam generator tubes constitute a substantial fraction of the RCS pressure boundary in a PWR. SGTR is important to consider in nuclear plant risk assessments because radionuclides released from the primary system through the ruptured tube(s) could bypass the containment building and escape to the environment through openings in the secondary system. Previous risk assessments have typically addressed SGTRs that occur during normal operation or during an accident, but prior to core damage. Very few risk assessments have considered SGTRs that occur after core damage. This has been due to a limited understanding of the phenomena that govern these SAI-SGCBs.

Loss of structural integrity of the SG tubes could result from elevated tube temperatures and elevated differential pressures across the tubes. The temperatures and differential pressures required to cause tube rupture depend on the characteristics of any flaws that may exist in the tubes due to postulated tube degradation mechanisms. Consequently, an assessment of SAI-SGCB must consider the potential initial tube flaw characteristics, the pressure and temperature histories experienced by the tubes during such an accident, and the response of the tubes to these pressures and temperatures.

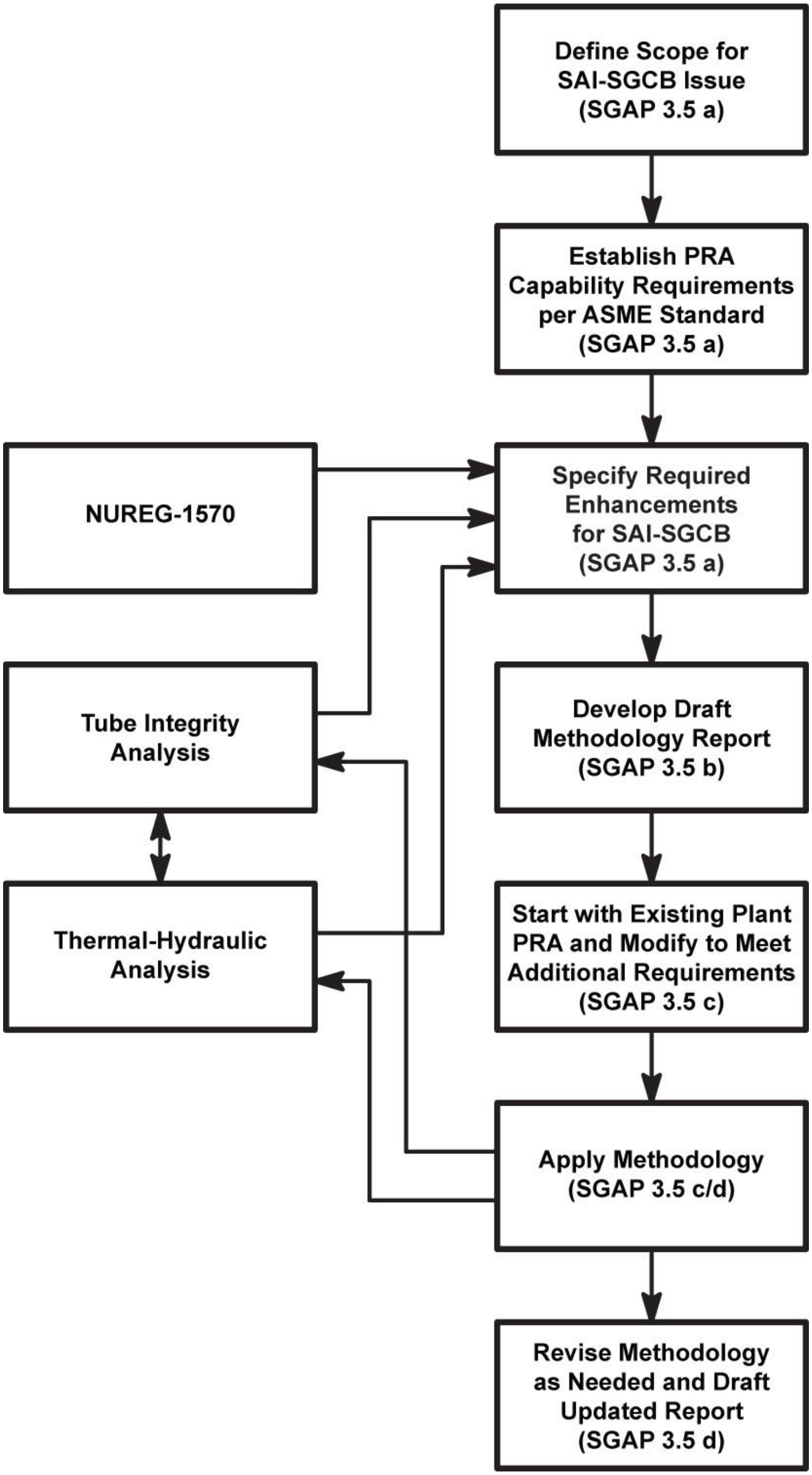

Figure 1. SAI-SGCB PRA project outline and integration points for T-H and ME analyses

The current approach to SGTR risk assessment links the results of accident frequency analyses (NUREG-1150 [USNRC, 1990]) with accident progression analyses (NUREG-1570, [USNRC, 1998a]). For the purposes of the integrated risk assessment, this approach appears to be applicable as well and has been used as the basis for the development of this effort. Several shortcomings, however, have been identified with the NUREG-1150 analyses [USNRC, 2002]. These include the use of expert opinion in the absence of phenomenological analyses and the need for improved analysis of human reliability, particularly in operator responses during severe accidents.

Since NUREG-1150 was issued in 1990, on-going research has resulted in an increased understanding of the T-H phenomena associated with severe accidents. In addition, considerable progress has been made in human reliability analysis (HRA). A Technique for Human Event Analysis (ATHEANA) is a second-generation HRA method that addresses shortcomings in the former approaches [Forester, 2007]. Application of the ATHEANA method in this work also includes consideration of human actions in the use of Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs). In addition, since the issuance of NUREG-1150, there is now a better understanding of component performance and severe accident phenomenology, which has resulted in the use of a more realistic PWR PRA as a starting point for the integrated assessment effort.

Key references for addressing SGTRs include: NUREG-1150, NUREG-1570, NUREG/CR-6365 [USNRC, 1996], NUREG/CR-5750 [USNRC, 1999], NUREG-1560 [USNRC, 1997], and Idaho National Engineering Laboratory Technical report INEL-95/0641 [INEL, 1996]. Review of these references has provided an identification of the functional failures that have been found to lead to an SAI-SGCB and a basis for an assessment of other potential functional paths.

1.2 Objectives and Scope

The overall objective of this work is to provide the NRC staff with a tool (i.e., an analysis method) by which it can determine the containment bypass frequency arising from a postulated SAI-SGCB. This work focused on the development and application of an integrated PRA methodology to a hypothetical PWR in order to better evaluate the risk associated with possible SG tube failure(s) following low-probability severe accidents.

The primary objectives of this PRA are:

1. To develop improved methods to identify those low-probability severe accident scenarios that could lead to a high primary system temperature and a high primary-to-secondary system pressure differential, such that there is a high likelihood of challenging the integrity of the RCS pressure boundary through an SAI-SGCB, and

2. To develop an improved PRA/HRA method and tools for modeling these scenarios (i.e., determining their expected frequency), including the effects of operator actions, uncertainties, and differences in plant design.

To meet these objectives, this PRA methodology focuses on assessing the frequency of accidents resulting in an SAI-SGCB, and includes approaches for identifying and screening accident sequences, modifying accident progression event trees and fault trees, performing HRA, and developing uncertainty distributions. An initial version of the PRA methodology was previously documented.[1] T-H and ME analyses are combined with the risk analyses to provide an overall coordinated methodology that has been applied in this work to a hypothetical plant to provide a demonstration of the methodology and allow for modifications as needed to address any identified weaknesses and gaps.

Early in this effort, it was determined that high primary system temperature, a high primary-to-secondary system pressure differential, and a dry SG secondary that result in "high and dry" conditions challenged the SG tubes the most. Station blackout (SBO) scenarios can create those conditions and, at most plants, SBOs are major risk contributors, if not the dominant risk sequences. The application presented in this report is therefore limited to SBO scenarios.

The methodology provides a framework for integrating the results of the PRA with those from the T-H and ME analyses. The ME analyses of the SG tube integrity and the materials response of other RCS components determined the pressure and temperature regimes of interest. The T-H analyses then determines what is required to get to those regimes. The PRA examines system and component failures that would put the reactor system in those conditions and identifies the operator recovery actions that can mitigate the accident progression. The integrated methodology then provides a framework that logically combines the results from all of these elements, including uncertainties, to provide the risk perspective for the SAI-SGCB issue.

The methodology presented in Chapter 2 of this report includes improved guidance and methods for identifying accident initiators and related severe accident scenarios that lead to high-temperature and high-pressure conditions such that there is a high likelihood of challenging the integrity of the reactor coolant pressure boundary through SGTR during a core damage scenario.

1.3 Report Organization

Section 2 of this report discusses the PRA methodology and framework for integrating the PRA results with those from the T-H and ME analyses. Descriptions of the required T-H and ME modeling are also presented.

Section 3 describes the application of the methodology to a hypothetical plant, including the data and results of contributing T-H and ME analyses.

Section 4 presents the risk results from the application of the PRA methodology.

A summary and the conclusions from the analyses are presented in Section 5

[1] David R. Bradley and Paul J. Amico, June 2003. "Methodology for Assessing Severe Accident-Induced Steam Generator Tube Ruptures," Draft Letter Report, ADAMS ML031810770, U.S. Nuclear Regulatory Commission.

2. RISK ASSESSMENT METHODOLOGY

This section describes improved guidance and methods for identifying accident initiators and related severe accident scenarios that can lead to SGTRs and evaluating the frequency of containment bypass for those scenarios. The methodology provides a framework, shown in Figure 2, for integrating the results from the T-H and ME analyses into the PRA evaluation. The ME analyses of the SG tube integrity and the materials response of other RCS components determined the pressure and temperature regimes of interest. The T-H analyses then determines which accident scenarios belong in those regimes. The PRA/HRA examines system and component failures and operator actions that would potentially influence the conditions which determine the potential for tube rupture. The T-H analyses, as necessary, confirm those conditions. The integrated methodology then provides a framework that logically combines the results from all of these elements, including uncertainties and sensitivity analyses, to provide the risk perspective for the SAI-SGCB issue. The details of the PRA and HRA methods and the modeling and methods for the T-H and ME analyses are discussed in the following sections.

2.1 Identification of Severe Accident Challenges to Steam Generator Tube Integrity

Steam generator tubes constitute a substantial fraction of the RCS pressure boundary in a PWR. Failure of one or more of the SG tubes could provide a pathway for release to the environment through the secondary side of the steam generator. This release path would bypass the barrier provided by the containment. Substantial radionuclide retention may occur within the steam generator, but a significant amount of the radionuclides released from the primary system could escape to the environment. In addition, this release could occur relatively early in an accident and thus the event could contribute to the large early release frequency (LERF).

SGTRs can occur during normal operation (i.e., a SGTR initiating event), after another initiator and before core damage (e.g., following a main steam line break), or after another initiator and after core damage. The last of these, which is often referred to as severe accident-induced steam generator tube rupture (SAI-SGTR) that leads to containment bypass (SAI-SGCB), is the subject of this assessment. For the purposes of this assessment, it is assumed that, until core damage occurs, the SG tubes are still intact.

Loss of structural integrity of the SG tubes during a severe accident could result from elevated tube temperatures and/or elevated differential pressures across the tubes. The temperatures and differential pressures required to cause tube rupture depend on the characteristics of any flaws that may exist in the tubes due to postulated tube degradation mechanisms (e.g., axial or circumferential stress corrosion cracking, or damage from loose parts).

A pressure-induced tube failure can be caused by an increase in differential pressures across SG tubes when both primary and secondary sides are at <u>normal</u> temperatures. A failure of this nature could be caused from either secondary side depressurization or primary side over-pressurization. Secondary side depressurization could occur from a main steam line break (MSLB) or a transient with a stuck-open atmospheric dump valve (ADV). Primary side over-pressurization can occur from large pressure excursions caused by an anticipated transient without scram (ATWS). Evaluation of MSLB, ATWS, and other sequences that result in a pressure-induced tube failure with the SG at normal temperatures were not included in the scope of the example plant evaluation documented in Chapters 3 and 4 of this report due to resource limitations.

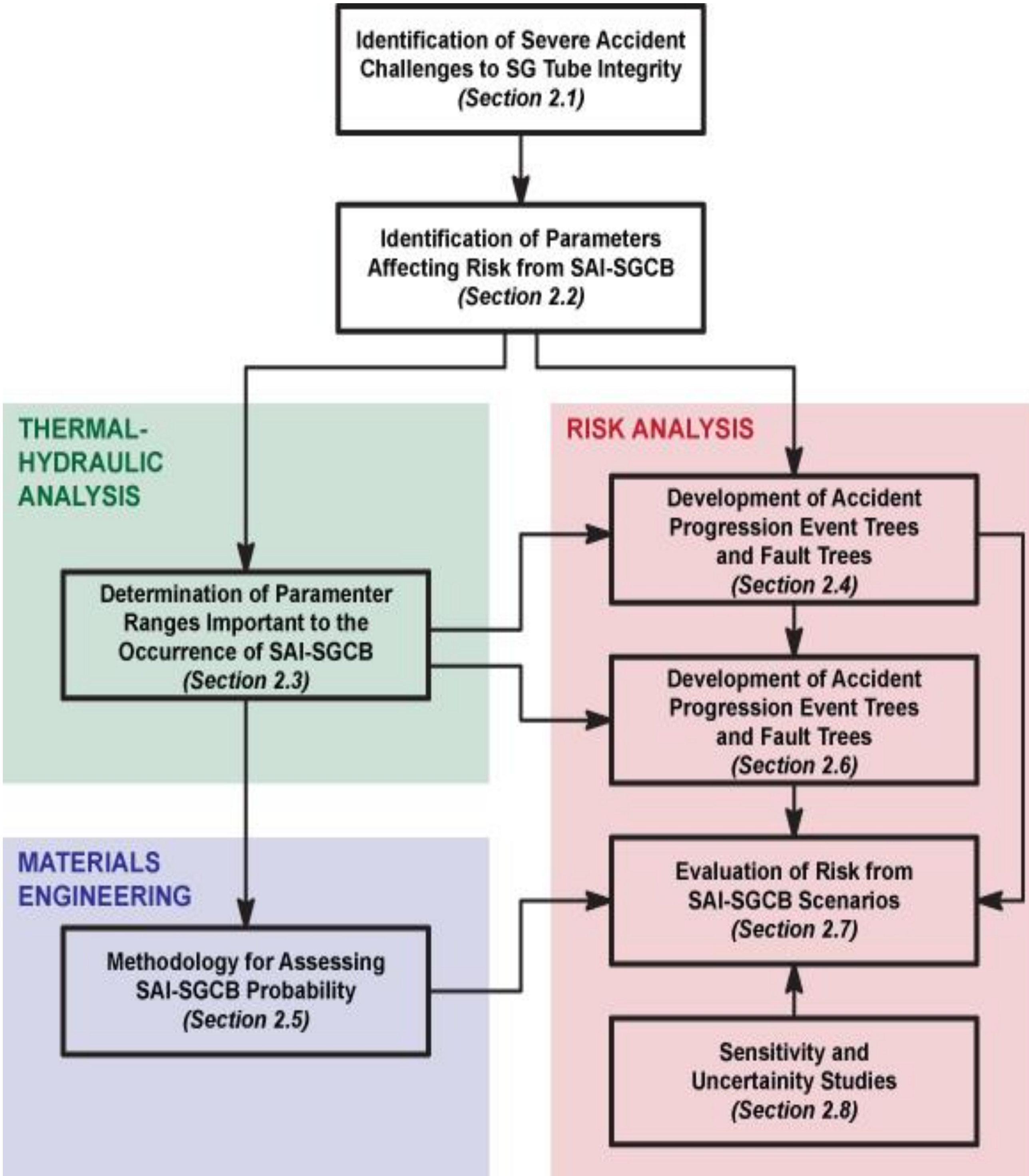

Figure 2**.** SAI-SGCB methodolog

A temperature-induced failure can be caused by the combination of high differential pressure across the tubes and excessively high SG tube temperatures. These conditions, commonly referred to as a “high and dry” condition, are likely to occur during the core damage phase of certain severe accidents. The conditions for temperature-induced SG tube failures are achievable when the secondary side is dry (i.e., no auxiliary feed water is available) and there is an elevated primary-to-secondary system differential pressure. Events resulting in core damage with the RCS pressure near either the pressurizer power-operated relief valve (PORV) or safety relief valve (SRV) set points and with the secondary side dry and depressurized are generally considered to pose the greatest threat of temperature-induced SGTR. SBO event sequences account for the majority of the events that can result in these conditions. However, events where the RCS is at intermediate pressures (i.e., below normal operating pressure) and the SG’s secondary side is dry and depressurized may also pose a substantial threat to tube structural integrity. The RCS can be partially depressurized due to the failure of PORVs/SRVs to close or reseat or by a loss-of-coolant accident (LOCA) caused by the failure of reactor coolant pump (RCP) seals. The degree of depressurization depends on the timing and leak area associated with valve failures or RCP seal LOCAs, and with the accumulator injection set points. The focus of the example application of the methodology described in this report was on SBO sequences with the RCS pressure at both high and intermediate pressures since these sequences were determined to be major contributors to an SAI-SGCB.

2.2 Identification of Parameters Affecting the Risk from SAI-SGCB

The temperatures and differential pressures required to cause SG tube rupture depend on the characteristics of any flaws that may exist in the tubes due to the postulated tube degradation mechanisms (e.g., axial or circumferential stress corrosion cracking, or damage from loose parts). Thus, the assessment of an SAI-SGCB must consider the initial tube flaw characteristics, the pressure and temperature histories experienced by the tubes during an accident, and the response of the tubes to these pressures and temperatures. The pressure and temperature histories are dependent upon the specific severe accident sequence that can vary depending on the impact of an accident initiator on the availability of mitigating systems. In addition, the response of other RCS components during a severe accident is important since their failure prior to an SAI-SGTR could preclude containment bypass through the ruptured tubes.

2.2.1 Characterize Existing Tube Flaws

Steam generator tubes exhibit a variety of flaw types, including circumferential cracks at the top of the tube sheet, axial primary water stress corrosion cracks (PWSCC) at roll transitions, freespan cracks, axial outer diameter stress corrosion cracks (ODSCC) at tube support plates (TSPs), circumferential cracks at TSP dents, axial cracks due to intergranular attack/stress corrosion cracking (IGA/SCC) in sludge pile areas, and flaws due to damage caused by loose parts. Existing tube inspection procedures are designed to detect most flaws before they reach a condition that could lead to tube failure under full power operation. Some flaws, however, are difficult to detect due to their location. In addition, because of human error, it is always possible that significant flaws may go undetected.

Because the probability of an SGTR depends on the characteristics of the existing tube flaws, it is important to accurately determine the flaw type, location, size, and depth. When doing this, the analyst should consider the time since the last tube inspection, the history of tube flaw growth at the plant, and the demonstrated effectiveness of tube inspection procedures at the plant.

Researchers at ANL and Dominion Engineering, Inc. have developed methods for estimating the number and size distributions for flaws of various types in SG tubes [Gorman et al., 1998]. These methods estimate the number of tubes with detectable flaws as a function of time, and the size distribution for defects in these tubes. The probability of detection during in-service inspections was considered when determining the distribution of flaw sizes. Sample distributions were provided for lightly-affected, moderately-affected, and severely-affected plants. Distributions were provided for six defect types: circumferential stress corrosion cracking, circumferential ODSCC at TSPs, free span ODSCC, IGA/SCC in hot leg sludge piles, axial ODSCC at TSPs, and flaws due to loose parts.

Although plant-specific flaw distributions should be used in the risk analysis whenever possible, in the absence of sufficient data to develop such distributions, use of the hypothetical distributions discussed above may be acceptable. The distributions should be selected based on plant-specific factors such as the age of the SGs and past experience with SG degradation.

2.2.2 Develop Screening Criteria for the Conditions Needed for SAI-SGCB

A conservative set of screening criteria should be developed to characterize the pressure and temperature conditions that could lead to an SAI-SGCB. The recommended approach for establishing these criteria include:

- Identify anticipated worst-case tube flaw characteristics for both axial and circumferential tube flaws.
- Use these worst-case flaw characteristics in tube integrity models to identify pressure and temperature conditions that could lead to tube failure.

The worst-case flaw characteristics should be selected so that they could reasonably be expected to exist in a steam generator near the end of a cycle (i.e., just prior to tube inspection).

In applying the tube integrity models, the analyst should recognize the uncertainty in the models and use conservative lower-bound parameter values (i.e., values that would lead to conservative pressure and temperature estimates for the conditions at tube failure). This ensures that conditions that could lead to an SAI-SGCB are not screened prematurely.

2.2.3 Determine Important Factors Influencing Pressure and Temperature History of Tubes

Severe accidents can lead to primary-to-secondary tube leakage through already present flaws. The highest driving force for leakage occurs under high RCS and low secondary system pressures.
Based on the high tube temperature and high SG pressure differential conditions that contribute to an SAI-SGCB, the following safety functions are important:

- RCS pressure control (affects differential pressure across the tubes);
- Secondary side pressure control (affects differential pressure across the tubes);

- Secondary side inventory control (affects differential pressure across the tubes, temperature of the tubes, and scrubbing of the release); and

- RCS inventory control (affects temperature across the tubes).

Certain parameters affect the above functions in ways that could influence the SAI-SGCB probability. Some examples are provided in the subsequent sections. The analyst must examine the plant's procedural guidance, unique system configurations, and scenario-specific impacts on the severe accident progression in order to identify important factors that can influence the potential for an SAI-SGCB.

2.2.3.1 RCS Pressure Control

An identified parameter of importance is the probability of partial depressurization of the RCS at the time of core damage. Two RCS states must be considered: a stuck-open pressurizer PORV or SRV and an RCP seal LOCA. The occurrence of an RCP seal LOCA can be influenced by the accident scenario (e.g., it is likely for scenarios where seal injection or cooling is unavailable such as SBO scenarios). The size of an RCP seal LOCA is important in determining if the RCS pressure will decrease sufficiently such that a high primary/secondary pressure differential will occur at the time of core damage.

If an RCP seal LOCA does occur, the tube heating in a depressurized SG increases significantly if loop seal clearing occurs in the same loop. Tube rupture is likely to occur under these conditions. In an analysis of a reference plant evaluated in NUREG-1570, loop seal clearing did not occur for sequences other than those involving RCP seal LOCAs. Thermal-hydraulic analyses performed since NUREG-1570 was completed indicate a larger margin to loop seal clearing than previously thought. In the reference plant analyses performed for this study, the loop seals always remained filled with water.

The use of PORVs to depressurize the RCS after core damage is procedularized in SAMGs and may be a key action in preventing an SAI-SGTR. Its effectiveness will vary depending on RCP leak rate. The procedures allow use of either one or two PORVs. The time delay for performing this action depends on whether or not the Technical Support Center (TSC) is in place to help guide the operator actions. The use of PORVs to depressurize the RCS can be affected by the specific accident scenario. For example, in an SBO scenario, battery depletion prior to core damage would rule out the remote opening of the PORV as a measure to preclude an SAI-SGCB.

2.2.3.2 Secondary Side Pressure Control

Another important factor is the secondary system pressure at the onset of an accident. The secondary side can be depressurized due to mechanisms such as:

- operator actions to depressurize using Atmospheric Dump Valves (ADVs),
- failure to manually reclose or block a stuck-open ADV,
- failure to isolate steam flow to a turbine-driven auxiliary feedwater (TDAFW) pump, and
- failure of a Main Steam Safety Valve (MSSV) to re-close.

Another important factor is the potential that the secondary side of one or more isolated SGs will depressurize due to leakage. Secondary leakage can occur out of the system (e.g., through valve stem seals) or internally through the main steam isolation valves (MSIVs) into downstream piping. The issue of secondary side leakage is discussed further in Appendix A.

Per procedural guidance, the operators may blow down the SGs under certain conditions in order to depressurize the RCS and inject the accumulators. Injection by the accumulators will help prevent or delay core damage. However, during scenarios such as SBO, battery depletion could preclude the potential for secondary pressure control.

2.2.3.3 Secondary Side Inventory Control

Secondary side inventory control affects the differential pressure across the tubes and the temperature of the tubes. Secondary inventory control is provided by either normal or auxiliary feed water. In many accident scenarios, main feed water may not be available and auxiliary feed water (AFW) may be limited. An example is an SBO where only TDAFW would be available. In this case, TDAFW could only be automatically controlled until the batteries deplete. After battery depletion, if local manual control is not successfully accomplished it is possible that a TDAFW pump would either trip immediately or continue operating until it overfills the SG to where water enters the pump turbine. In the latter case, the time to dry out the SG is substantially longer. The potential for recovery of feed water after core damage should also be considered since it can reduce the potential for an SAI-SGCB given that core damage has occurred.

2.2.3.4 RCS Inventory Control

Failure to provide water to the RCS will result in core damage. The progression of the core damage provides the temperature transient that could cause the high SG tube temperatures necessary to result in SAI-SGTRs. The availability of adequate RCS inventory control is dependent upon the type of accident scenario. For the SBO scenarios examined in the example analysis provided in Chapter 3, the only RCS inventory makeup possible is by the accumulators. Injection by the accumulators requires depressurization of the RCS.

2.2.4 RCS Component Failure

If failure of any part of the RCS pressure boundary occurs before the SG tube(s) rupture, an SAI-SGCB is assumed to be precluded. Thus, analysis of RCS component failures is an equally important part of the SAI-SGCB assessment. As with the tube failure analysis, creep failure of the hot leg and surge line is an important consideration and depends on the pressure and temperature at these two locations. Because the pressure-temperature histories at the hot leg and surge line are generally controlled by the same factors that control the pressures and temperatures experienced by the tubes, the times for hot leg and surge line failure are typically close to the times for tube failure. Thus, it is important to consider the uncertainty in the hot leg and surge line failure times.

2.3 Determination of Parameter Ranges Important to the Occurrence of SAI-SGCB

To support this PRA application, it is necessary to perform extensive T-H analysis. These analyses can be completed with computer models such as SCDAP/RELAP or MELCOR that have been accepted for use in severe accident analysis by the NRC. Codes that have not been accepted by the NRC (e.g., MAAP) may be subject to certain conditions or limitations on their use to evaluate SAI-SGCB scenarios.

The T-H analyses help determine which accident sequences need to be analyzed in detail and which systems or operator actions need to be addressed in the event trees. The T-H analysis supports event tree development by providing a basis for screening and grouping accident sequences. Sequences are screened (i.e., eliminated from consideration) if they would not lead to the conditions necessary for tube failure. Sequences are grouped if they would result in similar temperature and pressure conditions in the tubes and at the hot leg or surge line. The T-H analysis also supports the containment bypass analysis by providing the pressure and temperature histories at the tubes and at other RCS components that could fail.

As indicated in Section 2.1, the first step in preparing for the T-H analysis is to determine what accident scenarios to model. In the current PRA application, the focus was on SBO sequences only. Given an SBO scenario, several different factors could influence whether the SGs reach the high differential pressure and high temperature conditions needed for tube failure. These include factors that affect the pressures in the RCS and secondary side during core damage, and factors that affect the timing of core damage. As indicated in Section 2.2, a list of system conditions and operator actions that could influence these factors is prepared. Reasonable variations in these system conditions and operator actions are then defined. For example, RCP seal leakage was identified as a critical factor in determining the RCS pressure during an SBO-initiated core damage sequence. A range of potential RCP seal leak rates was then identified based on published studies.

In many cases, the number of required T-H analyses can be reduced by judicious ordering of the calculations. For example, if a 240 gallon per minute per pump (gpm/p) RCP seal leak is sufficient to fully depressurize the RCS prior to core damage (thereby preventing tube failure), it is not necessary to run calculations with higher leak rates. In addition, because tube failure would be precluded under these conditions, any sequence with 240 gpm/p leakage or greater can be screened from further consideration in the analysis.

The results from the T-H analysis are reviewed to determine whether groups of accident sequences result in similar pressure-temperature conditions at the tubes, hot leg, and surge line. Any such sequences would be expected to have similar SAI-SGCB probabilities and can therefore be grouped for the purposes of this analysis. For example, if extending battery depletion from 4 to 8 hours simply delays onset of core degradation, but does not significantly change the pressures and temperatures in the RCS or secondary side, then there is no need to distinguish between 4 and 8-hour battery depletion in the analysis from a T-H perspective. However, extending the battery depletion time can increase the potential for recovery of power and thus would affect the overall risk.

In some cases, it may not be obvious that two sets of T-H results are sufficiently similar to allow their corresponding accident sequences to be grouped. Under these conditions, two options are considered: (1) treat the accident sequences separately, or (2) do more detailed analyses to determine whether the containment bypass probability would be sufficiently close for the sequences to be grouped. The latter option requires that more detailed models be available when the sequence grouping is performed.

When developing the list of accident scenarios, the analyst should consider complete and partial failures (e.g., partially stuck-open PORV), human errors of omission or procedure-driven errors of commission, and changes in the state of a component as the accident progresses (e.g., reclosure of a stuck open valve). The analyst should also recognize that the timing of failures or human actions may be important. For example, the T-H analysis may show that a stuck-open PORV before time “x” or a human action to depressurize the primary system before time “y” will prevent an SAI-SGCB or delay it sufficiently that hot leg or surge line failure would nearly always occur first. These factors should be considered when developing success criteria for prevention of an SAI-SGCB.

2.4 Development of Accident Progression Event Trees and Fault Trees

The probabilistic analysis of an SAI-SGCB must start with an existing PRA that is of sufficient capability to reasonably model the conditions that can lead to an SAI-SGCB. Accident progression event trees and fault trees provide a framework for assessing the frequency of postulated accident scenarios. They provide an estimate of the frequency of primary and secondary system conditions that could challenge tube integrity and the overall frequency that a tube rupture will occur. They also provide a means for characterizing the progression of accidents and the resulting evolution of challenges to tube integrity. Since standard PRAs do not evaluate this possibility in detail, it is necessary to enhance the PRA for this purpose.

The entry point of an SAI-SGCB event tree is core damage (i.e., the termination point of the Level 1 PRA) with the plant in a condition that would not preclude an SAI-SGCB or for which an SAI-SGCB is no longer relevant. An example of the former condition might be a large or medium LOCA in which the RCS has been depressurized such that an SAI-SGCB is not possible. An example of the latter condition would be a tube rupture that occurs before core damage.

Initial plant states important to an SAI-SGCB may not be adequately captured in the Level 1 PRA. Thus, the analyst may have to do supplemental analyses to better define the initial plant states. Once the initial plant states have been determined, the analyst must identify complete or partial failures of systems or components, or human actions that could lead to pressure and temperature conditions that could challenge tube integrity. These conditions become top events in the event tree. The branches of the event tree are then selected to correspond to the conditions of interest for an SAI-SGCB.

The approach for determining what enhancements to the model are required is based on treating this assessment as if it were a risk-informed application under Regulatory Guide 1.174 [USNRC, 1998b]. This triggers specific requirements for both the supporting PRA and the supplemental calculations that must be performed to support quantification of the SAI-SGCB frequency. In order to meet the requirements for a risk-informed application, the PRA must meet certain standards. ASME standard RA-Sb-2005 [ASME, 2005] can be used to define the requirements for a PRA to be used to support an assessment of an SAI-SGCB. The ASME standard establishes the required capabilities of the PRA for risk-informed application and provides a framework for identifying the need for PRA enhancements or special studies.

Chapter 3 of the ASME PRA standard provides a flowchart illustrating the steps to follow when using a PRA for a risk-informed application. The methodology outlined in this report is adapted from the flowchart. As outlined in the standard, the general steps are:

1. Describe the issue to be assessed.
2. Identify the safety functions affected by the issue.

3. Identify the PRA scope and risk metrics needed to assess the issue.
4. Determine the Capability Category needed for each part of the PRA to support application.
5. Review the existing PRA to determine whether it meets the Capability Categories established in step 4.
6. Review the Supporting Requirements (SRs) identified in the standard to determine whether they are sufficient in scope and level of detail to support the application.
7. Review the existing PRA to determine whether it satisfies the SRs at the appropriate Capability Category.
8. If the scope of the PRA or the SRs is insufficient, supplementary analyses or requirements are identified.
9. Complete the required supplementary analyses.
10. Upgrade the PRA to meet the Capability Categories for all identified SRs and to incorporate all new analyses.
11. Re-quantify the PRA to determine the impact of any changes on the risk metrics chosen in step 3.
12. Provide risk input to the decision makers.

Using the results of prior studies of SAI-SGTR, the analyst can identify complete or partial failures of systems or components or human actions that could lead to pressure and temperature conditions that could challenge tube integrity (see Sections 2.1 and 2.2 of this report). The relevant timing of these failures is also identified. The event trees and fault trees are modified to incorporate these failures and actions, along with their timing, into the overall model.

To limit the number of accident sequences that must be developed and quantified, the T-H analyses are used to group accident sequences based on similar pressure and temperature histories and, consequently, the similar likelihood of an SAI-SGCB. Accident sequence binning continues throughout the risk assessment process as new results from T-H and event tree analyses become available.

After the event trees and fault trees are modified, the next step is the assignment of probabilities for basic events. In most cases, the existing PRA will already contain the necessary information to quantify the basic events. In those instances where the existing PRA may provide an inadequate basis for determining important basic event probabilities, supplementary analyses are performed. These supplementary analyses primarily relate to failures of reactor coolant boundary components (e.g., SG tubes, hot legs, and the pressurizer surge line) and human factor events (HFEs). These are discussed in Sections 2.5 and 2.6, respectively, of this report.

One of the strengths of the PRA framework is the ability to characterize uncertainties. When assigning basic event probabilities, the analyst should recognize the uncertainty in these probabilities and supply uncertainty distributions to the PRA model that adequately reflect the full range of uncertainty. The aggregation of these uncertainties will be reflected in the overall uncertainty in the calculated SAI-SGCB frequency. The breadth of the uncertainty distribution is a measure of how robust the SAI-SGCB frequency estimates are.

2.5 Methodology for Assessing SAI-SGCB Probability

The probabilistic response of RCS pressure boundaries and SG tubes is discussed in this section. The result of the analytical approach described below is the probability that sufficient SG tubes fail to cause a containment bypass (SAI-SGCB).

2.5.1 Failure of Steam Generator Tubes under Severe Accident Conditions

Failure of SG tubes under severe accident conditions is modeled by a combination of creep rupture and limit load analyses. Tubes with flaws that are part-through-wall are assumed to fail by creep failure of the remaining ligament. The initial ligament pop-through failure is then followed by either a widening of the opening due to continued creep or by a sudden rupture if a limit load failure condition is reached. Tubes with an initial through-wall flaw are assumed to have the flaw widen either by creep or by sudden rupture if the limit load condition is reached.

As discussed below, tube failure depends on the pressure difference across the tubes and the tube temperature. The pressure and temperature histories are calculated using T-H analyses, as discussed in Section 2.3.

2.5.1.1 Failure of Part-Through Flaws Due to Creep

Ligament pop-through is calculated using an analytical approach developed by researchers at ANL. The ANL model, which is described in NUREG/CR-6575 [Majumdar et al., 1998], assumes that pop-through occurs when the creep damage index reaches a value of 1. Written in the form of an equation, this condition is given by:

$$\int_0^{t_f} \frac{dt}{t_R(T, m_p\sigma)} = 1 \qquad [1]$$

where:

T = the temperature experienced by the tubes (a function of time and location within the tube)
m_p = the stress magnification factor
σ = the stress on the ligament, and
t_f = the failure time (the time at which the equality is satisfied).

The function in the denominator of Equation (1) is

$$t_R = 10^{\frac{P_{LM}}{T} - 15} \qquad [2]$$

where the mean value for the Larson-Miller parameter, P_{LM}, is given by the following correlation for Alloy 600

$$\begin{aligned} P_{LM} &= (24.3 - 3.0\ln(m_p\sigma)) \times 10^3 & m_p\sigma \le 5.7\,\text{ksi} \\ P_{LM} &= (23.2 - 2.4\ln(m_p\sigma)) \times 10^3 & m_p\sigma \ge 5.7\,\text{ksi} \end{aligned} \qquad [3]$$

Different equations are used for the stress magnification and ligament stress depending on whether the crack is an axial or a circumferential crack. In the latter case, different equations are used depending on whether the crack can be considered as constrained or free to deform. The equations used in each of these cases are presented in NUREG/CR-6575.

Once ligament pop-through has occurred, the crack may widen due to creep or the tube may rupture if the pressure inside the tube is sufficiently high. Analysis of crack opening and tube rupture are described in the following section, which also presents the tube failure models for cracks that are through-wall.

2.5.1.2 Failure of Through-Wall Flaws

Flaws that are initially through-wall or flaws that were initially part-through-wall but then had ligament pop-through due to creep are treated using the same analytical approach. Unstable failure of the tubes is predicted using limit load analysis. In addition, widening of the crack due to continued creep is also calculated.

Limit load analysis is used to predict the pressure differential at which the through-wall flaw would undergo unstable failure. The models used for the limit load analysis were developed by ANL [Majumdar et al., 1998]. The critical pressure at which unstable failure occurs is calculated using the following equation for axial flaws:

$$P_{cr} = \frac{\bar{\sigma} h}{m r_t} \qquad [4]$$

or for circumferential flaws:

$$P_{cr} = \frac{2\bar{\sigma} h}{m r_t} \qquad [5]$$

where:

m	=	the magnification factor for through-wall cracks
h	=	the thickness of the tube
r_t	=	the mean radius of the tube, and
$\bar{\sigma}$	=	the flow stress.

The magnification factor m is calculated using different equations for axial and circumferential flaws, and different equations for constrained and unconstrained circumferential flaws (see NUREG/CR-6575). The value for m depends on the flaw length. Longer flaws have higher values for m and lower calculated failure pressures.

Two options are used for calculating the flow stress in the tube failure model. One option is to use the following equation:

$$\bar{\sigma} = k(\sigma_y + \sigma_u) \qquad [6]$$

where:

k	=	a multiplication factor (typically assumed to be between 0.5 and 0.6)
σ_y	=	the yield stress (temperature dependent), and
σ_u	=	the ultimate stress (temperature dependent).

The flow stress also can be calculated using a correlation developed by ANL based on high temperature test data [Majumdar et al., 1998]. The ANL flow stress correlation yields higher flow stress values for the temperatures of interest.

At the high temperatures experienced by the tubes during severe accident conditions, through-wall flaws will also open wider due to creep. Models for crack opening due to creep have been developed by ANL and have been benchmarked against test data [Majumdar et al., 2002]. The ANL crack opening models predict slow crack opening rates at low temperatures but very rapid opening at high temperatures. At the temperatures of greatest interest (approximately 1000 K, or about 1340°F), cracks tend to open very rapidly, reaching their maximum crack opening displacement in less than 1 to 2 minutes. The ANL models are used to calculate the growth of each crack during the transient, and the total crack area is determined at each time step.

The maximum crack opening displacement is assumed to correspond to a maximum crack opening angle (i.e., the angle formed at the crack tip as the crack widens) of 60 to 90 degrees. The former value is the maximum angle observed in the tests at ANL [Majumdar et al., 1998]. The larger angle is provided as an alternative because the ANL tests could not maintain pressures after tube failure. Thus, it is possible that a larger crack opening angle would have occurred had the pressure been maintained.

It should be noted that the circumferential cracks considered in this analysis are located at either the top of the tube sheet or at the tube support plates and are expected to be surrounded by a buildup of sludge. Tests at ANL have shown that this sludge may significantly restrict flow through the flaw [Majumdar, 2004]. To account for this, the maximum crack opening displacement for circumferential flaws was artificially set to 1 millimeter in the current analysis.

The tube failure model calculates the time at which the maximum crack opening displacement is reached and also the time at which unstable failure would be predicted. The earlier of these two values is assumed to be the time at which the crack would open to its maximum displacement.

2.5.2 Estimation of the Crack Opening Area Required for Early Containment Bypass

Ideally, the aggregate crack opening area required for an SAI-SGCB would be determined by a detailed severe accident analysis. In the absence of such a study, it was assumed that (1) flow through the cracks is choked, and (2) early containment bypass occurs if the contents of the RCS would be released through the cracks in less than 4 hours. An uncertainty distribution for the required crack opening area was determined by considering the uncertainties in (1) the release time for containment bypass, (2) the temperature of the gas exiting the break, (3) the specific heat ratio for the gas mixture, and (4) the average molecular weight for the gas mixture. Using this analytical approach, the mean crack opening area for containment bypass is calculated to be 0.081 in^2. The lower and upper 90-percent confidence limits for this value were calculated to be 0.053 in^2 and 0.124 in^2, respectively.

2.5.3 Estimation of the Early Containment Bypass Time

The models discussed in the preceding sections are applied to the analysis of each tube flaw to determine at what time the aggregate crack opening area would be sufficient to cause an early SAI-SGCB. The failure time and maximum crack opening time are calculated for each flaw. The total crack opening area is then estimated as a function of time by assuming that the crack area for each flaw increases linearly with time from the initial through-wall failure until the maximum crack opening displacement is reached. The time at which the total crack opening area exceeds the critical area for containment bypass is assumed to be the containment bypass time. If this time is reached prior to any other failure in the RCS pressure boundary, an SAI-SGCB is assumed to occur. The next section of this report discusses how the failure time of other RCS components was estimated in the current analysis.

2.5.4 Estimation of the Failure Time for Other RCS Components

If failure of any part of the RCS pressure boundary occurs before the calculated SAI-SGCB time, it is assumed that an SAI-SGCB would be precluded. Thus, analysis of RCS component failures is an equally important part of the SAI-SGCB assessment.

Ideally, failure of other RCS components would be determined based on detailed materials response analyses performed with computer models such as ABAQUS. Limited detailed analyses were completed during this project and the results compared to the predicted creep-induced hot leg and surge line failure times calculated by SCDAP/RELAP [Siefken et al., 2001]. Based on this comparison, the SCDAP/RELAP failure times were judged to be adequate for the example application documented in Section 3.5.

2.5.5 Calculation of the SAI-SGCB Probability

The tube failure models described in the preceding sections were programmed into an Excel spreadsheet. Uncertainty distributions for key model inputs were then developed using an Excel add-in, Crystal Ball. The Crystal Ball software samples from the uncertainty distributions for the input parameters and runs Monte Carlo analyses to generate probability distributions for key model outputs.

Model parameters for which uncertainty distributions are provided include the following:

- the length and depth of each flaw (distribution from [Gorman et al., 1998])
- the Larson-Miller parameter P_{LM} (distribution from [Majumdar et al., 1998])
- the stress magnification factor m_p (distribution from [Majumdar et al., 1998])
- the axial location of the flaw
- the tube inlet temperature (from [Boyd et al., 2004])
- the crack area required for containment bypass (discussed above)
- the hot leg and surge line failure time (discussed above)

The following steps are performed in order to estimate the probability of containment bypass:

1. For each Monte Carlo sample, calculate the time at which the critical crack opening area is reached.

2. For each Monte Carlo sample, sample from the uncertainty distributions for the hot leg and surge line failure times.
3. If the critical crack opening time is before the earlier of the hot leg or surge line failure times, assume that containment bypass would have occurred.

Steps 1 through 3 are repeated until sufficient samples have been run for an adequate estimate of the containment bypass probability, which is calculated as the fraction of the Monte Carlo samples in which containment bypass is predicted to occur. The higher the predicted containment bypass probability, the lower the number of samples needed for an adequate statistical confidence level.

2.6 Evaluation of Human Error Probabilities for Accident Scenarios

The ATHEANA HRA method, as described in the ATHEANA User's Guide (NUREG-1880) [Forester, 2007], along with additional considerations described in this section, is recommended for performing the HRA for SAI-SGCB scenarios. However, it should be noted that only an abbreviated version of the ATHEANA HRA process was used to perform the HRA analysis for the present study. An abbreviated approach was used for several reasons. First, even though a particular plant and its PRA were chosen to serve as the "generic" plant for purposes of the analysis, it was decided that a plant visit was not needed for the level of analysis required at this time. Thus, no simulator exercises could be observed and no questions could be asked directly to plant operators and trainers. In addition, even though the Westinghouse Owners Group Emergency Response Guidelines (ERGs) and SAMGs were available for the analysis, plant-specific procedures were not. Finally, it was decided that for the present analysis, questions regarding plant-specific performance shaping factors (PSFs) such as operating crew training and biases, crew understanding of procedures and their usage, crew dynamics and characteristics, key instrumentation and cues for the scenario from the crews perspective, expected workload, human-system interface characteristics, and "informal rules" that might influence their decisions, could not be submitted to the plant at this time. Thus, much of the plant-specific information required by the ATHEANA HRA method (and many other methods for that matter) to perform a realistic analysis could not be obtained.

Nevertheless, while the HRA analysis process used in the present study does not correspond exactly to that described in the ATHEANA User's Guide, the process (see Section 3.6) and the results and related discussion (Section 3.6 and Appendix B), do provide a good illustration of the issues that need to be considered in an SAI-SGCB PRA and the general, albeit abbreviated, ATHEANA quantification process.

While the ATHEANA approach provides detailed guidance for performing an HRA analysis in the context of a PRA that can be used for any plant condition (e.g., full-power, low power and shutdown, fire, and severe accident conditions), it focuses on performing a full-power analysis, prior to core damage. Thus, a few factors should be noted as being particularly relevant to the SAI-SGCB scenarios, which may not be explicitly addressed by the ATHEANA guidance, even though the general guidance should lead analysts to address such factors. They include:

- Degree and type of training the operating crews receive on ERGs and SAMGs and the likely difficulty of these procedures given the scenarios being addressed. These procedures may rely more on knowledge-based decisions (including evaluating trade-offs between actions) than the EOPs and other procedures used prior to core damage.

The degree of training and operator familiarity on the steps relevant to the scenarios being examined will be important to their likely success.

- The relationship between the operating crews and the TSC, including the plan for and expectations of how they will interact, who will be in charge, and the likely efficiency of the process (e.g., will too many people be involved, which could significantly slow down the decision making process, etc.). In addition, do the operating crews and TSC receive training on interacting with one another?

- When would the TSC be expected to be in place? Due to the timing of the scenario, which decisions will be made by the crew alone and which are likely to be made jointly with the TSC.

- If it is desirable to take credit for non-proceduralized actions in long term scenarios, what is the basis for assuming that the appropriate knowledge would be available, both with respect to making the decision and to completing the action? In the context of PRA/HRA, this type of action has not traditionally been credited, and therefore reasonable investigation would be required before allowing any credit to be taken. Other actions may also benefit the scenarios, such as aligning diesel-driven fire water pumps for AFW, that are proceduralized at some plants (i.e., in the SAMGs), but such actions are really not expected to ever be needed, and therefore would require careful analysis. Thorough documentation of the basis for such actions should be performed before credit is taken and there should be a plan to develop appropriate procedures if the actions are important.

- Very limited data are available bearing on how operating crews will respond under severe accident conditions. Experts participating in the analysis will have to take this situation into account and factor it into their judgments based on the knowledge obtained from the analysis and from their own experiences.

As can be seen from review of the above issues, the types of additional information needed for the PRA/HRA will require a plant-specific analysis. Important sources for this information will include observations of simulator exercises to the extent possible and discussions with plant personnel, particularly operators, trainers, and procedure writers (if available). The ATHEANA method provides detailed guidance on how to collect relevant information and the factors that need to be considered. This general guidance will be applicable to the SAI-SGCB scenarios and should be followed to the extent possible.

Finally, in most cases, pre-initiator events will already be included in the Level 1 PRA analysis being used as the basis for the SAI-SGCB analysis and in general can be left “as is” in the models. However, if systems, structures, or components (SSCs) important for responding to severe accident conditions have not been included in the models, then analysts should include them as necessary. The ATHEANA method focuses on post-initiator HFEs, but in general its processes for identifying events to be modeled and quantifying those events can be generalized to pre-initiators. Alternatively, if other methods such as THERP (NUREG/CR-1278) [Swain, 1983] or ASEP (NUREG/CR 4772) [Swain, 1987] were used in the original PRA models, these methods can also be used. THERP and ASEP are acceptable for pre-initiators and tend to be simpler and more straightforward to apply to pre-initiator events.

2.7 Evaluation of Risk from SAI-SGCB Scenarios

The risk evaluation process combines inputs and outputs from the initiating event analysis, fault tree analysis, and the event tree model. The event tree is the logic model used to assemble each accident sequence analysis. All sequences, regardless of their origin, are modeled as one or more top events in the event tree. The result of the event tree quantification is a set of accident progression sequences describing the events that occur and the characteristics needed to determine that an SAI-SGCB has occurred. Each accident sequence has a frequency (F) that is stored as one of the two elements of the risk equation, $R = F \times C$. C, or consequences, is represented as a plant damage state, which for this study is the occurrence of an SAI-SGCB, and all sequences that result in this damage state are quantified and summed. Once the risk is assembled, the relationships of the model inputs are carefully evaluated for insights. Insights are derived from the quantitative assessment of the importance of various plant features, operations, or individual failures. The release characterization process yields insights concerning mitigation features.

The risk model was quantified using the Electric Power Research Institute (EPRI) Risk & Reliability workstation suite of codes. The individual modules of the suite used were as follows:

- ETA – This was used to model the accident progression event trees. Each branch of the tree was assigned a fault tree top event matching the system failure appropriate to the sequence.

- CAFTA – This was used to model the system fault trees. In addition to containing the Boolean logic for each top event, it also contains the failure rate data and quantifies the basic event failure probabilities. It also contains the initiating event frequency data.

- PRAQUANT – This takes the ETA and CAFTA models and automatically creates the necessary event sequence top logic to perform the quantification. This includes the top logic for both the system failures and for the system successes in the sequence. PRAQUANT also calls the quantification engine and recovery post-processor (FORTE and QRECOVER, see below), applies the flag and rules files, and creates and saves the resulting cut set file and quantification results.

- FORTE – This is the quantification engine that performs the Boolean reduction to develop the cut sets for each sequence.

- QRECOVER – This contains the recovery model and checks each cut set against a set of recovery rules and applies recovery basic events as appropriate and appends them to the cut sets generated by FORTE. Since this is really just a cut set post-processor, it was also used in this study to append the appropriate SGCB probability to all the cut sets in each given sequence.

The quantification process was iterative and followed a set of rules for establishing cut set truncation limits to assure that the risk profile was reasonably established, as follows:

- Initially, all sequences were quantified with a truncation limit of 1E-10/yr. Sequences with overall frequencies of 1E-7/yr or greater were left with this truncation limit.

- Sequences with frequencies of less than 1E-7/yr were assigned progressively lower truncation limits until the truncation limit was at least three orders of magnitude below the overall sequence frequency (e.g., sequences with frequencies of 1E-8/yr were truncated at 1E-11/yr). This truncation limit was used as long as there were over 100 cut sets returned. If not, the truncation limit was lowered by one order of magnitude.

- The lower truncation limit was set at 1E-13/yr regardless of the overall sequence frequency. This was the lowest truncation limit used, regardless of whether cut sets were returned.

This process provides reasonable assurance that sequences down to 1E-9/yr were quantified accurately. Results are only reported for these sequences. Sequences with calculated frequencies less than 1E-9/yr are simply reported as “low”.

2.8 Sensitivity and Uncertainty Analysis

Two types of uncertainty are currently discussed in the literature: epistemic uncertainty and aleatory variability. The term epistemic uncertainty is used to refer to uncertainties in the model that are due to a lack of perfect knowledge. Thus, the term is generally applied to uncertainties in the model parameter values, such as physical parameter values, failure frequencies, etc. Aleatory variability is a term used to describe the variability associated with a phenomenon that is random in nature (e.g., the wind speed at any arbitrary point in time). The basic difference between the two types of uncertainty is whether increased knowledge about the subject could reduce the imprecision in the model used in the PRA. For example, even if hundreds of years of data were taken on the wind speed at a specific location, accuracy in the prediction of the wind speed at a specific (future) time would benefit very little due to the inherent randomness of the phenomenon. Thus, wind speed has aleatory variability. Values for the failure rates of specific components used at the plant, however, could be improved significantly by observing many years of failures of those components. Theoretically, the uncertainty on the failure rate value for failures of identical components could be reduced to zero (i.e., the failure rate could be known exactly). In practice, this is unachievable. Note also that as modeled in the PRA, the actual failure time of a component is random because it is assumed to be governed by a Poisson process. Thus, the failure rate has epistemic uncertainty, while the failure time has aleatory variability.

These two types of uncertainties are included in the PRA in different ways. Aleatory variability is included in the basic structure of the models and in assumptions such as the Poisson nature of the failure processes. In contrast, epistemic uncertainty is treated as unknown parameters that are sampled using probability distributions and a Monte Carlo process.

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3. EXAMPLE APPLICATION OF METHODOLOGY

The PRA methodology described in Section 2 was developed to provide a framework that logically combines the results from PRA/HRA, T-H and ME analyses, including uncertainties, to assess the frequency of accidents resulting in an SAI-SGCB and to provide a risk perspective for the SAI-SGCB issue. To determine whether the methodology could be practically implemented, to refine the methodology, and to estimate the risk significance of the SAI-SGCB issue, the methodology was applied to a hypothetical plant. This section discusses the scope of the example application, the plant selection and selected sequences, and the details of the application to the selected plant.

3.1 Scope of Application

The objective of the example plant application was focused on identifying scenarios leading to SAI-SGTRs and developing the logical framework to calculate the resulting frequency of containment bypass events. The scenario and logic framework developments correspond to Task 3.5c and the frequency calculation corresponds to Task 3.5d of the SGAP, respectively. In addition, ensuing from the specific application, refinements to the draft methodology developed in 2003 were addressed, as required by Task 3.5d of the SGAP. Specifically, detailed calculations and sensitivity analyses were requested by the PRA team from the T-H and ME teams to support both the identification of the accident sequences that can lead to an SAI-SGCB and their frequencies. As detailed analyses became available, refinements were made to the draft methodology. In particular, attention has been paid to making the method as user friendly as possible for the NRC staff to apply.

The second objective of this PRA work, to improve PRA modeling of high and dry sequences, was also addressed in the application. The entry condition for the accident progression event tree is the frequency of high and dry sequences (and potentially others of interest identified through the detailed analyses). The current approach calculates this frequency by decomposition of the results of the PRA Level 1 frequency analysis. The results of the decomposition are subsequently inspected for system conditions in which feedwater has been lost, but the SGs have not yet boiled dry when the core uncovers.

Hence, during the application, a systematic approach to generating the entry conditions for the accident progression analysis (i.e., binning of high and dry states) was developed. This systematic approach includes a set of rules that could be used during the decomposition and binning of plant states from the Level 1 PRA. Development of this rule-based approach will increase the efficiency of the NRC staff in performing future risk-based license reviews and eliminate the subjective aspects of the methodology used in NUREG-1150.

Since the developed model may serve as a baseline for future analyses, the sources for all input such as recovery actions, operator actions, and the source of plant information are well documented in Chapter 3. The results of the application (i.e., the identification of the dominant SBO accident sequences that can lead to an SAI-SGCB and their frequencies) are presented in Chapter 4.

3.2 Plant and Accident Sequence Selection

When the draft methodology was under development in 2003, it was intended that the example plant application of the methodology would use the PRA/HRA analyses for the same plant for which the T-H and ME models were being constructed. It was desired, however, that the PRA models be modified to meet, at a minimum, the specific Capability Categories in accordance with the ASME PRA standard [ASME, 2005]. The details of the desired Capability Categories are delineated in the 2003 draft methodology report. A recently upgraded PRA that meets the input needs for applying the methodology was made available by the owners of a Westinghouse, 4-loop PWR. The T-H and ME analyses are based on modeling and calculations for the Zion plant, a similar, but different Westinghouse 4-loop PWR. Hence, two different, but similar plant models have been used in this integrated assessment, one for the PRA and another for the T-H and ME analyses. The PRA models were made available for the plant application, provided that the results from this assessment were not directly applied to the plant. Because a detailed PRA for this plant was not performed, and the T-H and ME analyses were not done for this specific plant, the results of this assessment should not be expected to reflect the risk effect of the SAI-SGCB issue for this specific reactor. So that any results from the example application in this work are not ascribed to the reactor for which the PRA models are constructed, the actual name of the plant is withheld and referred to in this report as WPWR-A.

In addition to the availability of the models, the choice to use the WPWR-A reactor was made for other reasons as well. The utility owners of the reactor completed an Independent Plant Evaluation (IPE) for it approximately 10 years ago. Since then, the PRA has undergone two "major" updates. While the updates did not affect most of the models, the updates did significantly affect the PRA models in a number of areas. For example, the HRA, while not performed using ATHEANA, has been significantly upgraded, which is very important for this current assessment. The HRA changes resulted from both the fact that the plant is operated differently and the fact that HRA techniques have advanced significantly since the IPE was originally performed. Operator actions are very important to the SAI-SGCB issue, and it makes the integrated assessment more straight-forward if the HRA for the PRA is as current as possible, both in technique and in procedures. In addition, additional Level 2 analyses have been performed for potential severe accidents. One disadvantage to using this upgraded plant PRA is that the systems models use CAFTA software, not the NRC-preferred SAPHIRE. This was not a major concern since the CAFTA software was available to the analysis team. The availability of the models and their updated status meant fewer resources were required to expand these models.

Additionally, the PRA methodology used for this integrated assessment was developed based on the assumption that such an assessment would be a risk-informed application under Regulatory Guide 1.174 [USNRC, 1998b]. As indicated in the initial version of this SAI-SGCB methodology developed in 2003, the plant PRA model to be used for this application must be modified to meet, at a minimum, specific Capability Categories in accordance with the ASME PRA standard [ASME, 2005]. Because of the upgrades, the PRA for the WPWR-A plant is much closer to meeting the current ASME standards.

Altogether, these considerations led to the use the WPWR-A models for the application of the integrated PRA methodology.

The WPWR-A PRA model has been modified to enhance the modeling of SAI-SGCB using various generic (i.e., not specific to WPWR-A) T-H, steam generator tube conditions, emergency procedures, and SAMGs. The modifications, particularly the successive focusing on the specifics of the accident sequences needing detailed study, were not an easy straightforward effort.

In a PRA, millions of potential severe accident pathways can be analyzed, each varying by at least a little from other pathways. However, not all potential pathways are equally risk significant. Neither is each possible variation among risk significant pathways equally important to risk. Thus, only certain types of sequences need to be analyzed in detail with only a few of the possible variables being exercised. Knowing which these are, however, is not readily apparent and requires supporting analyses. Preliminary T-H and ME analyses provided results to support the rationale for identifying which accident sequences need further development.

Preliminary phenomenological and material research efforts showed that SG tube integrity is most threatened when high pressure and temperature differences across the tubes exists. This most likely happens when the primary system pressure is high and the steam generator secondary side is dry and depressurized. One accident scenario that leads to these conditions is SBO, the loss of all AC electric power, both offsite and onsite. At most PWRs, including the example plant, SBO scenarios are major risk contributors, both in terms of core damage frequency (CDF) and LERF. Hence, the overall SAI-SGCB effort concentrated resources on examining SBO scenarios, accident sequences that all have an SBO, but have different combinations of other system failures, timings, and operator actions. An additional set of scenarios that could contribute to the SAI-SGCB issue involves the loss of main feedwater and a number of other, independent system and operator failures. However, for most PWRs, including the example plant, these scenarios have a much lower frequency of occurrence than SBO scenarios.

Within the current Level 1 PRA for the WPWR-A reactor, a substantial number of questions regarding specifics for the PRA modeling of SAI-SGCB needed to be addressed. First, the definition of initiating event groups and the structure of the event trees in the PRA from the plant (as would be true for all Level 1 PRAs) are based on the success criteria for preventing core damage. For the example application, consideration needed to be given to the impact on the probability of SAI-SGCB. For example, in the PRA, the very small LOCA (VSLOCA) spans a range of RCP seal leak sizes where the core cooling success criteria are the same. For the consideration of core cooling only (and even for containment response excluding SAI-SGCB), this serves to adequately define the plant damage states (PDSs) for interface with the Level 2 accident progression analysis. This is not necessarily true for SAI-SGCB. Different leak rates result in different RCS pressure time histories after the onset of core damage. If different RCP seal leak rates within the definition of the VSLOCA result in different SAI-SGCB probabilities, then the original PDS definitions in the PRA would not be sufficient for modeling SAI-SGCB, and it would be necessary to sub-divide the VSLOCA initiating event.

Determining what initiating events and sequences can be screened out from the risk assessment was a major goal of the T-H analysis. Certain T-H runs focused on determining what conditions resulted in pressure-temperature time histories that would not result in SAI-SGCB (probability essentially zero). With that knowledge, the system models for those initiating events and sequences did not need to be enhanced. The T-H analyses also provided the rationale for identifying any required enhancements to the event trees. Consideration of these factors led to the identification of the set of SBO scenarios to be analyzed in the example application.

Finally, for those sequences that are included in the model, detailed representative pressure-temperature time histories for each bin were required in order to determine the probability of SAI-SGCB. These histories were needed both for the SG tubes and also for each RCS component that could fail before the tubes.

3.3 Evaluation of Important Parameters for Selected Scenarios

To support this PRA application, it was necessary to perform extensive T-H analysis. The T-H analysis helped determine which accident sequences needed to be analyzed in detail and which systems or operator actions needed to be addressed in the event trees. The T-H analysis permitted some sequences to be screened (that is, eliminated from consideration) because they would not lead to the conditions necessary for tube failure, and other sequences to be grouped together because they would result in similar temperature and pressure conditions in the tubes and at the hot leg or surge line. The T-H analyses also provided the pressure and temperature histories at the tubes, which enabled the material response of the tubes to be calculated.

The T-H analyses were performed using the SCDAP/RELAP computer code. The SCDAP/RELAP system model used for the current study simulated accidents in the Zion nuclear power station, a four-loop Westinghouse plant. All of the T-H analyses were performed as sensitivities to the base case, which is an SBO simulation with the following conditions:

- Initial RCP seal leakage of 21 gpm/p[2]
- No steam generator blow down before core damage[3]
- TDAFW fails
- No RCS depressurization after core damage[4]
- 0.5 in^2 leakage out of isolated steam generator[5]

The conditions simulated in the base case lead to a high and dry condition with a large pressure differential across the tubes and high tube temperatures. These are conditions that favor early creep rupture of the tubes, particularly tubes that are severely flawed.

In the base case scenario, it was assumed that the batteries deplete in 4 hours. This assumes that no operator action is taken to shed non-critical DC loads and thus extend battery life. The timing of battery depletion does not affect the base case because no operator actions are assumed either to use the pressurizer PORVs to control primary pressure or to use the TDAFW to control secondary inventory (and thus both SG tube temperature and secondary pressure).

[2] This is the minimum leakage expected in the blackout case and represents "normal" leak rates for seals when seal injection is lost. The leak rates in the RCP seal model are expressed in gpm at the initial RCS pressure. These are converted to equivalent diameter "holes" in each pump, so that the leak rate changes as the RCS pressure changes.

[3] A procedure calls for the operators to blow down the steam generators under certain conditions in order to depressurize the RCS and inject the accumulators. The base case assumes this does not occur even if those conditions are present.

[4] The SAMGs direct the operator to open pressurizer PORVs once the core exit thermocouples indicate 1200ºF. The base case assumes this action is not taken.

[5] Small leaks (either externally or even internally to piping downstream of the SG isolation valves) can cause depressurization of the steam generators once they go dry and no additional steam is generated. The base case leak rate is sufficient to depressurize the steam generators prior to significant core damage and while the RCS pressure is still high. See Appendix A for further discussion.

However, the timing of battery depletion is important to some of the other scenarios that were analyzed.

A number of variations on these system conditions and operator actions were simulated in the T-H analyses. Table 1 shows the variations addressed in the T-H analyses.

It should be noted that, in previous analyses such as those performed for NUREG-1570, loop seal clearing was an important factor in the assessment of SAI-SGTR. More recent T-H analyses have indicated that loop seal clearing is much less likely than previously thought. In all of the T-H analyses performed for this study, the loop seals remained filled with water.

Table 1. System conditions and operator actions simulated in the T-H analyses.

System Condition or Operator Action	Variations Analyzed	Comments
Primary Side Leakage	21 gpm/p, 60 gpm/p, 180 gpm/p, and 480 gpm/p	Range of values based on Westinghouse Owners Group Report WCAP 10541 [WOG, 1984] and subsequent NUREG/CR-4294 report [Boardman et al., 1985]
Secondary Side Leakage	Equivalent hole sizes of 0.1, 0.5, or 1.0 square inches or stuck open ADV	Leakage of isolated steam generator out of the secondary system (e.g., through valve seals) or internal to the secondary
Battery depletion	After 4 or 8 hours	
Steam generator Depressurization before core damage	Depressurization using ADVs at 30 minutes	The EOPs call for rapid depressurization of the RCS through the steam generators , even in a SBO.
TDAWF Status	Fails to run or runs until battery depletion	
Stuck-Open PORV Re-Closes before core damage	Valve recloses or remains stuck open	
RCS Depressurization after core damage	Operator opens 1 or 2 PORVs	Depressurization indicated by SAMGs using either 1 or 2 PORVs. Timing depends on whether or not the TSC is in place.
PORVs Reclose at Battery Depletion	Always close at battery depletion	PORVs return to their “failed” state

If all the logical combinations of these parameters are taken into account, over 1,000 T-H runs would have had to be performed. Based on a judicious ordering of analyses, it was possible to screen and group sequences so that a manageable number of T-H runs were identified and run. The resulting list of T-H runs is provided in Table 2. Insights from the T-H analysis are discussed in the following sections.

Table 2. SAI-SGTR SBO scenarios included in the PRA analysis.

Case #	RCP Seal LOCA (at t=13 min) or PORV LOCA (at t=10th PORV lift)	SG[1] Depress before CD[2] (t=30 min)	TDAFW runs until battery depletion	PORV recloses @1000 Psi[3]	RCS depress after CD[4]	Valves reclose at battery depletion	SG leakage (Nominal = 0.5 in^2) or ADV
Variations in RCS Leakage (RCP Seals or PORV)							
69	21 gpm/p	n/a	N	n/a	No	4 hr	Nominal
70	60 gpm/p	n/a	N	n/a	No	4 hr	Nominal
71	180 gpm/p	n/a	N	n/a	No	4 hr	Nominal
72	480 gpm/p	n/a	N	n/a	No	4 hr	Nominal
83	PORV	N	N	N	No	4 hr	Nominal
185	PORV	n/a	N	Y	No	4 hr	Nominal
Variations in Secondary Side Leakage (Equivalent hose size or ADV)							
177	21 gpm/p	N	N	n/a	No	4 hr	0.1 in^2
178	480 gpm/p	N	N	n/a	No	4 hr	0.1 in^2
179	21 gpm/p	N	N	n/a	No	4 hr	1.0 in^2
180	480 gpm/p	N	N	n/a	No	4 hr	1.0 in^2
183	21 gpm/p	N	N	n/a	No	4 hr	ADV
184	480 gpm/p	N	N	n/a	No	4 hr	ADV
Variations in SG Depressurization Actions with TDAFW Available until Battery Depletion							
153	21 gpm/p	Y	Y (4 hr)	n/a	No	4 hr	Nominal
154	60 gpm/p	Y	Y (4 hr)	n/a	No	4 hr	Nominal
155	180 gpm/p	Y	Y (4 hr)	n/a	No	4 hr	Nominal
157	21 gpm/p	N	Y (4 hr)	n/a	No	4 hr	Nominal
159	180 gpm/p	N	Y (4 hr)	n/a	No	4 hr	Nominal
160	480 gpm/p	N	Y (4 hr)	n/a	No	4 hr	Nominal
161	21 gpm/p	Y	Y (8 hr)	n/a	No	8 hr	Nominal
163	180 gpm/p	Y	Y (8 hr)	n/a	No	8 hr	Nominal
165	21 gpm/p	N	Y (8 hr)	n/a	No	8 hr	Nominal
167	180 gpm/p	N	Y (8 hr)	n/a	No	8 hr	Nominal
Variations in RCS Depressurization Actions after Core Damage							
5	21 gpm/p	n/a	N	n/a	1P, no TSC	4 hr	Nominal
7	180 gpm/p	n/a	N	n/a	1P, no TSC	4 hr	Nominal
21	21 gpm/p	n/a	N	n/a	1P, TSC	4 hr	Nominal
23	180 gpm/p	n/a	N	n/a	1P, TSC	4 hr	Nominal
37	21 gpm/p	n/a	N	n/a	2P, no TSC	4 hr	Nominal
39	180 gpm/p	n/a	N	n/a	2P, no TSC	4 hr	Nominal
53	21 gpm/p	n/a	N	n/a	2P, TSC	4 hr	Nominal
55	180 gpm/p	n/a	N	n/a	2P, TSC	4 hr	Nominal
Variations with No Secondary Side Leakage							
192	21 gpm/p	n/a	N	n/a	1P, TSC	4 hr	No Leakage
198	21 gpm/p	n/a	N	n/a	No	4 hr	No Leakage
199	180 gpm/p	n/a	N	n/a	No	4 hr	No Leakage
200	PORV	n/a	N	Y	No	4 hr	No Leakage
205	180 gpm/p	Y	Y (4 hr)	n/a	No	4 hr	No Leakage
206	180 gpm/p	N	Y (4 hr)	n/a	No	4 hr	No Leakage
207	180 gpm/p	Y	Y (8 hr)	n/a	No	8 hr	No Leakage
208	180 gpm/p	N	Y (8 hr)	n/a	No	8 hr	No Leakage
190	21 gpm/p	n/a	N	n/a	1P, no TSC	4 hr	No Leakage
191	180 gpm/p	n/a	N	n/a	1P, no TSC	4 hr	No Leakage

Table 2. SAI-SGTR SBO scenarios included in the PRA analysis.

Case #	RCP Seal LOCA (at t=13 min) or PORV LOCA (at t=10th PORV lift)	SG[1] Depress before CD[2] (t=30 min)	TDAFW runs until battery depletion	PORV recloses @1000 Psi[3]	RCS depress after CD[4]	Valves reclose at battery depletion	SG leakage (Nominal = 0.5 in^2) or ADV
193	180 gpm/p	n/a	N	n/a	1P, TSC	4 hr	No Leakage
194	21 gpm/p	n/a	N	n/a	2P, no TSC	4 hr	No Leakage
195	180 gpm/p	n/a	N	n/a	2P, no TSC	4 hr	No Leakage
196	21 gpm/p	n/a	N	n/a	2P, TSC	4 hr	No Leakage
197	180 gpm/p	n/a	N	n/a	2P, TSC	4 hr	No Leakage
201	PORV	n/a	N	Y	1P, no TSC	4 hr	No Leakage
202	PORV	n/a	N	Y	1P, TSC	4 hr	No Leakage
203	PORV	n/a	N	Y	2P, no TSC	4 hr	No Leakage
204	PORV	n/a	N	Y	2P, TSC	4 hr	No Leakage

Notes:
1. SG=steam generator
2. CD=core damage
3. After reclosing, PORV re-sets itself to its original set point, and will not reopen until that pressure is reached.
4. Entry indicates number of PORVs opened (1P or 2P) and timing (no TSC=@1200ºF; TSC=@1200ºF+12 minutes).

3.3.1 Sensitivity to Primary Side Leakage

Runs were conducted for variations in the induced small LOCA (SLOCA) and VSLOCA cases included in the PRA. Three "off-normal" RCP pump seal leakage cases (60 gpm/p, 180 gpm/p, and 480 gpm/p) as well as a stuck-open PORV were run. All other base-case parameters were unchanged. These runs determined how the various leak rates affect RCS pressure at the time of steam generator heat-up, which directly affects the stress on the SG tubes and the likelihood of SAI-SGCB.

At the highest leak rate of 480 gpm/p, the leak depressurizes the RCS sufficiently that no tube failures are predicted. The stuck-open PORV depressurizes the RCS sufficiently that only tubes that are significantly degraded (high stress multipliers) are threatened and only when these tubes are in hottest portion of the hot plume. A 180 gpm/p leak was found to delay tube failures as well as hot leg/surge line failures relative to the base case, but tube failures were predicted under a wide range of conditions. A 60 gpm/p leak also delays tube failures and hot leg/surge line failures relative to the base case, but the difference from the 21 gpm/p condition in the base case was relatively minor. Therefore, from the perspective of SAI-SGCB probability, the 21 gpm/p and 60 gpm/p leaks are nominally the same.

The results of these T-H runs caused certain changes to the systems models. In the PRA, a 21 gpm/p (normal leakage) is categorized as a transient, whereas 60 gpm/p, 180 gpm/p, and stuck open PORV are all categorized as a VSLOCA[6] and a 480 gpm/p is categorized as an SLOCA. Based on the preceding discussion, SLOCA is not quantified as part of SAI-SGCB risk since no tube ruptures are predicted to occur. In addition, VSLOCA is subdivided into VSLOCA-1 (size equivalent to stuck-open PORV), VSLOCA-2 (size in the range of 180 gpm/p

[6] The stuck open PORV is the upper limit of this VSLOCA size

seal leak, and VSLOCA-3 (size in the range of 60gpm/p). Only VSLOCA-2 and VSLOCA-3 were evaluated for the SAI-SGCB probability since tubes were not assumed to be severely degraded in the example analysis.

3.3.2 Sensitivity to Secondary Side Leakage

Runs were conducted for three cases of leakage out of an isolated steam generator when the TDAFW was not operating. This was done to simulate conditions under which secondary side pressure cannot be maintained in a dry steam generator. In addition to the base case leakage case, which assumed a 0.5 in^2 leak from each steam generator, additional cases were run with a 0.1 in^2 and 1 in^2 equivalent leakage. Cases were also run to simulate a stuck-open ADV. All other base-case parameters were unchanged. These runs were performed to determine the amount of steam generator leakage that will depressurize a steam generator prior to tube heat-up.

Leak sizes of 0.5 in^2 and greater were found to result in essentially full depressurization of the steam generator and therefore yield essentially the same SAI-SGCB probability. In contrast, leak sizes of 0.1 in^2 or smaller do not result in full depressurization. The T-H analysis showed that leakage of this size delays most tube failures by 1,000 to 2,000 seconds relative to the base case, but only delays hot leg/surge line failure by about 450 to 600 seconds. The net result is a much lower SAI-SGCB probability compared to base case.

Based on these results, secondary side leakage was characterized by two distinct leak rates: one for leaks greater than 0.1 in^2 equivalent area and one for leaks less than 0.1 in^2 equivalent area.

3.3.3 Sensitivity to TDAFW Availability (with and without Steam Generator Depressurization before Core Damage)

Calculations were conducted to evaluate cases in which TDAFW is available for either 4 or 8 hours, delaying the onset of core damage. Calculations were also run with or without the operators taking action to depressurize all steam generators using the ADVs. Depressurizing the steam generators lowers the RCS pressure rapidly to permit accumulator injection. The RCP seal leak rate was equal to the base case value of 21 gpm/p in each of these runs.

This series of calculations determined if long-term core damage scenarios caused by battery depletion could have a significant impact on the probability of SAI-SGCB. Battery depletion disables key operator actions that can be credited in short-term core damage scenarios, such as opening the primary PORVs to reduce RCS pressure. The resulting change in T-H conditions later in the event could change the pressure-temperature histories sufficiently that the probability of SAI-SGCB could change.

While the T-H calculations showed that core damage is delayed as expected, the pressure and temperature histories from the onset of core damage through failure of SG tubes and other RCS components are not significantly different from the base case. Based on this result, the probability of SAI-SGCB is not significantly affected.

In addition, these T-H runs show that the ramp in temperature during core degradation occurs after battery depletion, so that there will be no DC power available at the point where the

SAMGs call for the operators to open the pressurizer PORVs to reduce RCS pressure. Therefore, there is no need to include this action in the model.

Based on these results, since the TDAFW operation does not eliminate or significantly reduce SAI-SGCB probability, these scenarios cannot be eliminated from the risk model as potential contributors to SAI-SGCB. Consequently, the systems model will need to retain the late core damage sequences and the delayed timing taken into account in the sequence modeling.

3.3.4 Sensitivity to RCS Depressurization when TDAFW Available and Steam Generator Depressurization Before Core Damage

This series of T-H runs looked at the combined effect of the operator depressurizing the steam generators when TDAFW is available for various cases of RCP leakage. TDAFW is assumed to run until batteries deplete at 4 hours. In addition, the operators are assumed to depressurize the steam generators using the ADVs. The runs considered three different RCP seal leak rates: 60 gpm/p, 180 gpm/p, and 480 gpm/p. All other base-case parameters were unchanged.

The objective of this series of runs was to investigate the same leakage rates addressed in other analyses, but this time with delayed core damage. The results from these runs were used to determine whether the conclusions for the short-term scenarios are also valid for long-term scenarios.

The calculations show that, as expected, core damage is shifted later in time. The calculations also show that the post-core damage pressure and temperature histories are very similar to the corresponding cases with early core damage. Based on these results, it was determined that conclusions discussed previously for early core damage scenarios are equally valid for late core damage scenarios.

3.3.5 Sensitivity to RCS Depressurization after Core Damage

This series of T-H calculations looked at the effectiveness of opening primary PORVs after the onset of core damage in order to reduce RCS pressure. This action is called for in the SAMGs when the core exit thermocouples reach 1200°F. In some calculations, it is assumed that the TSC has been assembled and directs the operators to open either 1 or 2 PORVs. The timing of this action is estimated to be 12 minutes after the required temperature is reached. This interval allows for the time it takes for the TSC to walk through their decision process. In other calculations, it is assumed that the TSC has not yet been assembled and the operators immediately open 1 or 2 PORVs upon reaching 1200°F. All other base-case parameters were unchanged.

In all cases, the opening of the PORVs arrests tube creep, but this effect ends when the valves close upon battery depletion. Based on these results, tube failures would occur either before a PORV is opened or after one re-closes, and this is all in the 4-hour time frame. Taken in conjunction with other runs, it is clear that by the time the 8-hour time frame comes around, PORV re-closure would no longer be relevant. Therefore, additional T-H runs to look at these same scenarios for 8-hour battery depletion would not provide any additional information required for the SAI-SGCB analysis.

3.3.6 Sensitivity to Both Secondary Side Steam Generator Leakage and RCS Depressurization before Core Damage

A set of T-H runs was made with various combinations of secondary side leakage and RCS depressurization (either due to RCP seal leakage or operator actions). Seal leakage cases included 60 gpm/p, 180 gpm/p, and 480 gpm/p. Secondary side leakage corresponded to equivalent hole sizes of 0.1 in^2 and 1 in^2 in each steam generator, as well as a stuck-open ADV. The objective of this study was to determine whether the interaction between steam generator leakage and RCS depressurization could have a significant impact on the SAI-SGCB probability.

For the cases where the operator opens either one or two primary PORVs and there is only normal leakage from the primary (the most restrictive conditions), the primary pressure at the time of the temperature ramp up is below the normal secondary pressure. Therefore, by implication, if the secondary side is at normal pressure there will be no differential pressure across the tubes and SAI-SGCB will not occur.

However, these events happen at around four hours and the PORVs may reclose upon battery depletion. This would cause re-pressurization of the RCS. Although re-pressurization occurs, since the time for the onset of core damage does not change, the relative timing of the temperature and pressure ramps could be different from the base case resulting in a different probability of SAI-SGCB.

3.3.7 Additional Analyses Using a Control Room Simulator

In addition to the T-H analysis, the Westinghouse control room simulator at the NRC's Technical Training Center in Chattanooga, TN, was used to determine whether or not the operator would blow down the steam generators in a SBO if the TDAFW pump failed to start (as opposed to initially working and then failing when the batteries deplete). The simulator is for the Trojan nuclear plant, which is also a 4-loop Westinghouse system. Therefore, the results should be generally applicable to the example plant used in the current study.

In the simulation, the operators follow appropriate procedures, but cannot recover power or TDAFW. Operators perform the step to depressurize steam generators using the ADV, taking into account all required conditions and cautions and terminating depressurization as called for in the procedure.

Based on discussions with Westinghouse personnel, the procedural steps would be performed as follows:

- If at the time steam generator depressurization is called for (Step 16 of ECA 0.0) at least one steam generator has a level reading above 5% on the narrow range indicator, the operators will depressurize through all steam generator PORVs at the maximum possible rate.

- The operators will depressurize the steam generators to 270 psig and then try to control the pressure at around that level (stopping depressurization if the pressure falls below 170 psig).

- If the cold leg temperatures drop below 280°F during the depressurization, the operators will close the PORVs (e.g., return all of them to their nominal settings).

The results from the simulation were that the narrow range steam generator level went off scale low before the simulator operators got to the step to depressurize the steam generators. Thus, the conditions for depressurization were not met.

Next, the simulator operators ran a case where they made an "executive decision" to depressurize the steam generators anyway in an attempt to inject accumulators, using the minimum allowable steam generator pressure as a stopping point for depressurization and ignoring the steam generator level requirement. Following this action, the accumulator injection achieved was minimal, although the RCS pressure did decrease to about 600 psi. The final steam generator level was about 8% wide range in the simulation. These results suggest that this combination of system events and operator actions will not be important to the SAI-SGCB analysis.

3.4 Accident Progression Event Trees

The event trees and fault trees from the WPWR-A PRA were modified in order to address the SAI-SGCB issue. A major goal of this study is to determine how difficult it is to modify the systems models (the event trees and fault trees) to enable quantification of SAI-SGCB. The principle reason for amending the model is that a Level 1 PRA examines up to the onset of core damage with minimal treatment of release categories (primarily just differentiating between LERF and non-LERF states) whereas the examination of SAI-SGCB is a more detailed accounting of a specific release scenario that develops well after core melt begins. In fact, most Level 1 PRAs equate uncovering the top of the active fuel with core damage. The significance of this is that there is more time to recover systems and perform operator actions for the SAI-SGCB issue than there is for simply estimating the frequency of core damage. As a simple example, consider loss of offsite power (LOSP) events. It has already been stated that SBO scenarios are important to SAI-SGCB—they potentially result in the right conditions of a high pressure primary and dry secondary and are risk significant at most plants. To prevent core damage, there is a certain amount of time to recover power. If it is not recovered in that amount of time, there will be core damage. There is more time to recover power, however, to prevent full scale core melting, which precedes possible threatening of the SG tubes. Hence, the model for SBO needs to be modified to reflect this greater amount of available time.

Of course, the situation is not as simple as adding one more recovery factor. Each cut set generated during model quantification can present nearly unique timing characteristics, or at least as unique as the cases listed in Table 2. For an example of additional cut set amending, in the original PRA, all of the SBO cut sets (with one exception) were essentially lumped into the category of a VSLOCA because the success criteria for preventing core damage was the same for all of the SBO sequences. However, in this study it was necessary to divide those cut sets containing the RCP seal LOCA event into the subsets that varied with the size of the seal LOCAs because the different sizes affect the success criteria for preventing SAI-SGCB during the core damage progression. Hence, the challenge is in differentiating the conditions that affect this specific phenomenon, which in many cases are not particularly important for CDF (or even the PDS frequencies as generally defined in most current PRAs). The following changes were made to the PRA models for the WPWR-A plant.

1. It was necessary to create a finer differentiation of primary leak sizes (LOCAs) than is in the existing model. From the standpoint of core damage, the existing model is in keeping with current PRA practice. The leak sizes correspond to the success criteria of the front line systems required to prevent core damage. However, the post-core damage pressure-temperature time histories are quite different across the range of possible sizes, and these time histories greatly impact the SAI-SGCB probability. Based on the leak size model used in the existing PRA and the T-H analysis performed for this study, the leak sizes fell into three ranges:

    - a break equivalent to a leak of 60-150 gpm/p at full pressure (represented by a 60 gpm/p leak, which would have an SAI-SGCB probability essentially equivalent to a non-LOCA transient),
    - a break equivalent to a leak of 150-250 gpm/p at full pressure (represented by a 180 gpm/p leak, which would have a lower, but still credible, SAI-SGCB probability depending on the specific sequence), and
    - a break equivalent to a leak greater than 250 gpm/p at full pressure (which includes large RCS pump leaks and stuck-open PORV and has a negligible probability of SAI-SGCB).

    The event trees for the first two cases would have the same structure, since the success criteria for core damage are the same.

2. A top event for SBO was added to the event trees. In the existing model, the SBO frequency was extracted at the PDS level by parsing out the cut sets that resulted in SBO. For this study, some of the unique timing features associated with SBO needed to be modeled in greater detail in order to address SAI-SGCB, in particular those associated with battery depletion. In addition, the scope of this study is limited to SBO sequences, so highlighting them on the event trees seemed appropriate. So even though the system failure combinations were not different than the sequences for non-SBO, having specific SBO event tree sequences provided a clear delineation on the SAI-SGCB outcome.

3. Early secondary heat removal (SHR) failure during SBO was separated from late SHR in order to separate failures that occurred after battery depletion versus those that occur before battery depletion. This SHR failure differentiation does not affect CDF since both instances result in core damage and the simplification doesn't impact dominant core damage cut sets. The availability of the batteries, however, is very important to possible operator actions to prevent SAI-SGCB. Specifically, sequences with early SHR failure will have battery power available to open the primary PORV after core damage, which will prevent SAI-SGCB. This action is not possible for late SHR failures with the batteries depleted. The model was altered to include separate top events for early SHR and late SHR failure on the event trees, and to make changes to the fault tree models to separate SHR failures related to battery depletion from those that occur earlier. Since SBO fails all SHR sources except the TDAFW pump, the modeling changes only needed to be done for this SHR train.

4. The use of secondary depressurization to depressurize the primary and inject the accumulators was considered in the existing model, but only for cases where it meaningfully affected the frequency of core damage, which was not the case for non-LOCA transients. However, this action, if performed for transients, has the potential to affect the SAI-SGCB probability, so it was added to the transient tree.

5. The existing PRA handled the issue of battery depletion in a simplified manner. In looking at the model, it appears that in many (if not all) cases the depletion of the batteries at four hours was treated the same as the loss of the batteries at time zero. Also, in many cases no credit was given for operator action to shed nonessential DC loads to prolong battery life to eight hours. The simplified way battery depletion and off-site power recovery was modeled in the PRA did not appear to affect the dominant core damage cut sets. However, when trying to model the additional potential for such things as secondary depressurization and post-core damage actions such as primary depressurization using primary PORVs, these simplifications will affect the dominant cut sets for SAI-SGCB. Therefore, battery depletion was separated from other DC failures. The availability of the batteries themselves (from the standpoint of discharge) was modeled directly on the event trees by adding an event for whether the operator shed nonessential loads (8-hour battery life) or did not (4-hour battery life). A small fault tree for this top event was added. Battery depletion was removed from the support system model, leaving the other faults that could cause loss of DC power, which were still assumed to cause loss of DC at time zero.

6. The existing PRA addressed offsite power recovery in a very detailed fashion, but of course limited the analysis to achieving recovery in time to prevent core damage. It is also possible to recover offsite power after it is too late to prevent core damage, but before the conditions for possible SAI-SGCB occur. Realistically, if this is accomplished, the restoration of systems will almost assuredly prevent SAI-SGCB. As alluded to above, it is worth noting that some of the offsite power recovery timeframes used for preventing core damage were relatively conservative, based on the usual PRA definition of the onset of core damage coupled with the broad range of actual timeframes for recovery that exist for any given sequence. The more detailed modeling of SBO sequences that related to the SAI-SGCB issue allowed this study to be more precise. For each SBO sequence, a sequence-specific recovery model was created that addressed long-term recovery of offsite power. The available timeframe for recovery was set at the time of battery depletion or the time of the pressure-temperature spike that is associated with the potential SAI-SGCB conditions, whichever came first.

7. As discussed in the section on HRA, a key potential action that can be performed during the progression of core damage prior to reaching the conditions associated with SAI-SGCB is for the operators to open the primary PORVs, which will reduce the pressure differential across the SG tubes. This action was incorporated into the sequence-based recovery model for those sequences where this action is possible (i.e., where there is still battery power available at the time the action would be taken). The automated recovery model logic was instructed to add this event to the cut sets as appropriate.

8. As also discussed in the section on HRA, very long time periods may be available between the time when the batteries deplete and when the conditions associated with SAI-SGCB are reached. The timeframe may be on the order of hours. During that timeframe, it may be possible (despite the extreme conditions resulting from total loss of all DC power) to take actions that would restore one or more functions before the conditions required for SAI-SGCB are reached (it may also be possible to accomplish this before the onset of core damage, but the existing PRA did not take credit for it and it is doubtful that taking credit would reduce the overall CDF for the plant). This action was incorporated into the sequence-based recovery model for those sequences where this

action is possible. The specifics of the action are not precisely defined, as the possibilities are too numerous to mention. It simply represents the possibility of any action restoring a function that could affect the potential for SAI-SGCB. The automated recovery model logic was instructed to add this event to the cut sets as appropriate.

9. It was necessary to add the probability of SAI-SGCB to each sequence cut set. This was also done by using the sequence-based recovery model. Although occurrence of SAI-SGCB is not a recovery failure, the recovery model can be used as a way to add additional logic to the sequence top logic, and to add additional failure events. It is convenient to use this approach to add the SAI-SGCB event, since it depends on everything that has occurred before it.

10. Some other changes had to be made to the existing master recovery model because of the changes made to the sequence logic. The way the changes were made and the sequences were parsed and defined had some unintended effects, causing some recovery failures to be improperly applied. Again, while the logic that caused this could be considered a flaw in the recovery model it is not believed to have any meaningful affect on the core damage or PDS frequencies of the existing base PRA model for the plant or on the dominant cut sets. However, it does have an affect on the parts of the model that were created to assess the SAI-SGCB frequency associated with SBO sequences. This was addressed by putting logic that would correct the cut sets in the sequence-based recovery models created for this study.

In summary, while it was challenging to adapt the existing PRA model to analyze SAI-SGCB scenarios for which it was not designed and that would have no particular effect on CDF, it was possible to do so in a logical and effective fashion. There is no reason why this could not be done for any PRA model developed for a nuclear power plant.

From the results of the T-H calculations, the Level 1 PRA event trees of the example WPWR-A plant were modified to reflect the SAI-SGCB issue. These trees are shown as Figures 4, 6, and 7 for transients, VSLOCA (RCP seal LOCA of 60 gpm each at RCS operating pressure and temperature), and VSLOCA (RCP seal LOCA 180 gpm each), respectively. Note that even though these figures show non-SBO sequences, only the SBO sequences were analyzed in the example application. Figures 3 and 5 are the transient and VSLOCA trees from the original PRA.

In order to increase the level of understanding between the models, the sequence numbering system links back to the sequence numbers from the original PRA. Each sequence in the original PRA had a designator (e.g., TRA-1, VSL-1). Each of these is associated with a specific set of conditions that lead to core damage. In the revised trees, the sequence number is retained at the beginning of each sequence name to signify that it is a variation of the original sequence (i.e., the reason for reaching core damage is the same), but has different conditions relative to SAI-SGCB. For example, all sequences starting with TRASGCB-1 are variations of TRA-1. VSL60-1 and VSL-180-1 are variations of VSL-1 (60 and 180 being the leakage size in initial flow rate in gpm). The other nomenclature describes the specifics of the variation.

- An “L” means that SHR failed late and secondary depressurization failed.
- A “D” means that SHR failed late and secondary depressurization succeeded.
- If there is no “D” or “L” it means that SHR failed early (secondary depressurization is not available in this case).

- A “4” means the batteries deplete in 4 hours.
- An “8” means the batteries deplete in 8 hours.

In the figures, the T-H case numbers are provided at the end of the accident sequences, in the column labeled “core damage bin,” for those SBO sequences where directly applicable T-H runs were performed. These correspond to the case numbers used in Table 2. For SBO sequences where there is no directly applicable T-H run (i.e., where an extrapolation needed to be made based on a run for another sequence) this column is left blank.

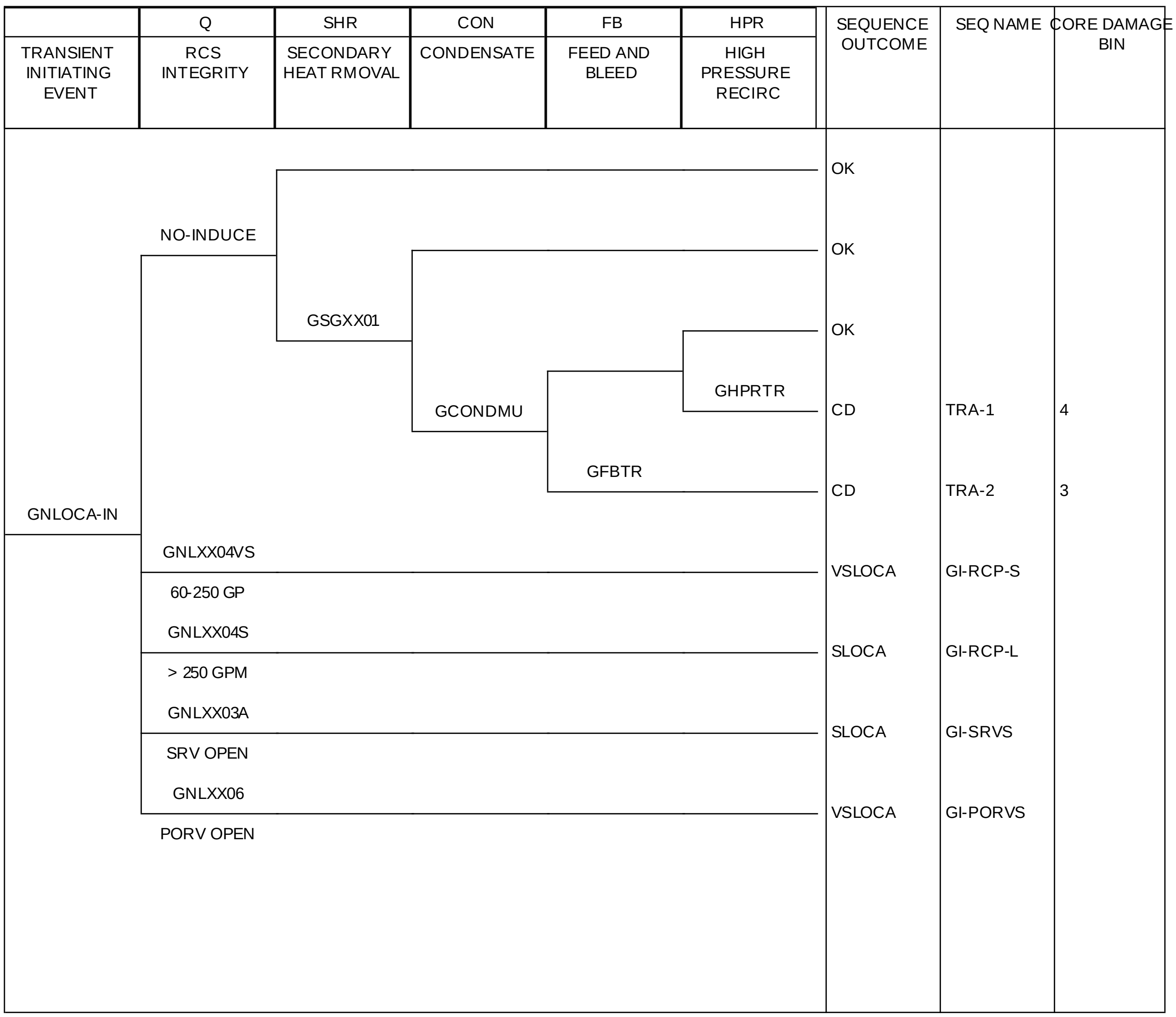

Figure 3. Original transient event tree.

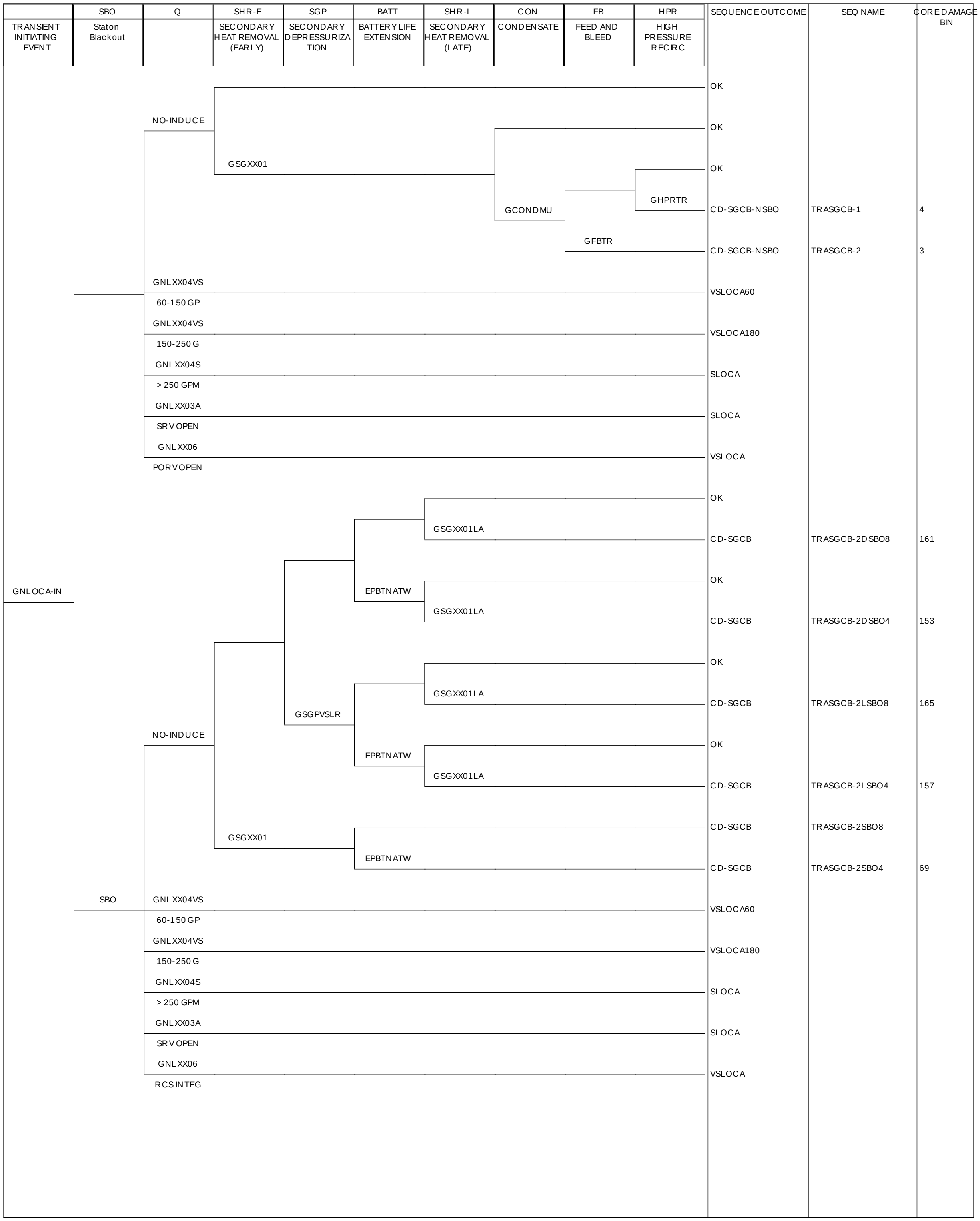

Figure 4. Transient event tree for analyzing SAI-SGCB.

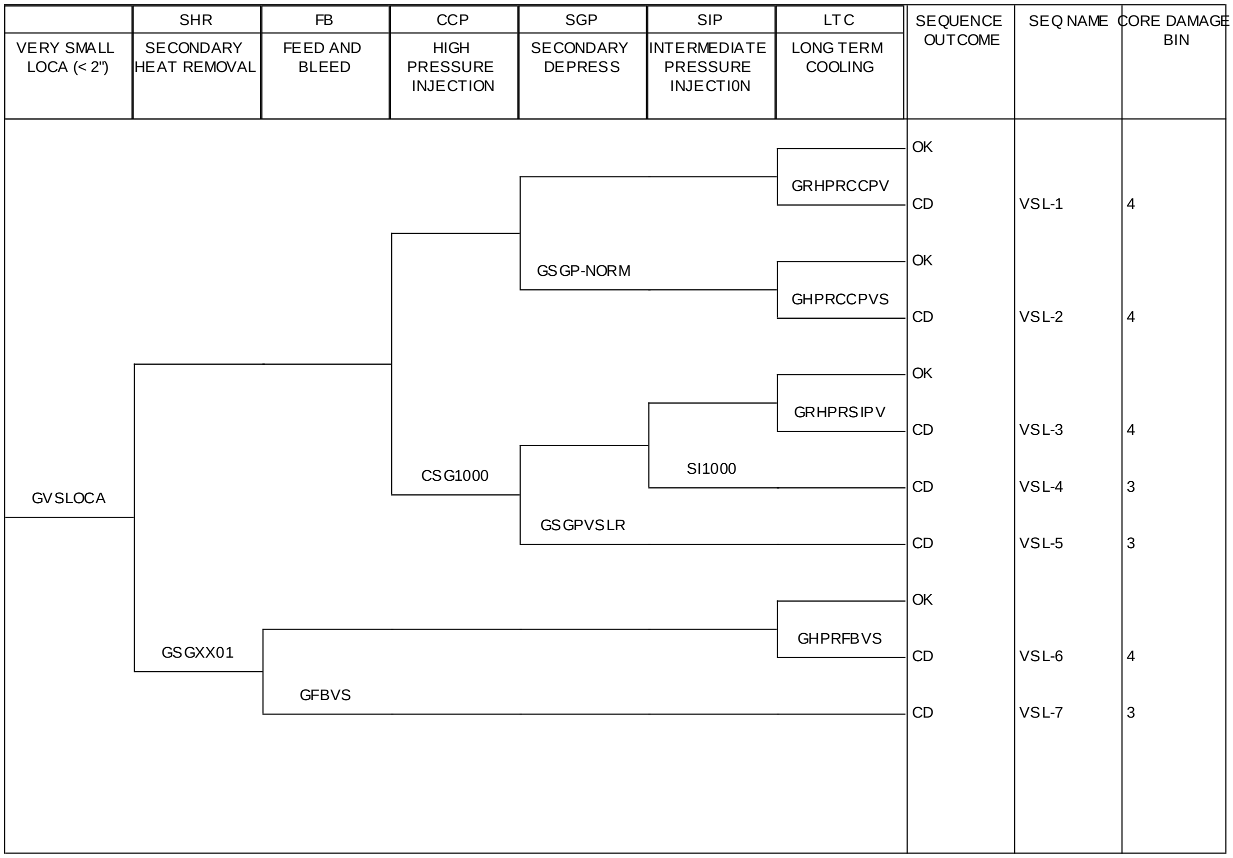

Figure 5. Original VSLOCA (VSL) event tree.

3.5 Evaluation of the SAI-SGCB Probability

The probabilistic containment bypass model discussed in Section 2.5 requires the following plant-specific inputs:

- probability distributions for the length and depth of each flaw and the number of flaws
- the time-dependent pressure difference and temperature experienced by the flaw, and
- probability distributions for the failure time of other RCS components.

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- probability distributions for the failure time of other RCS components.

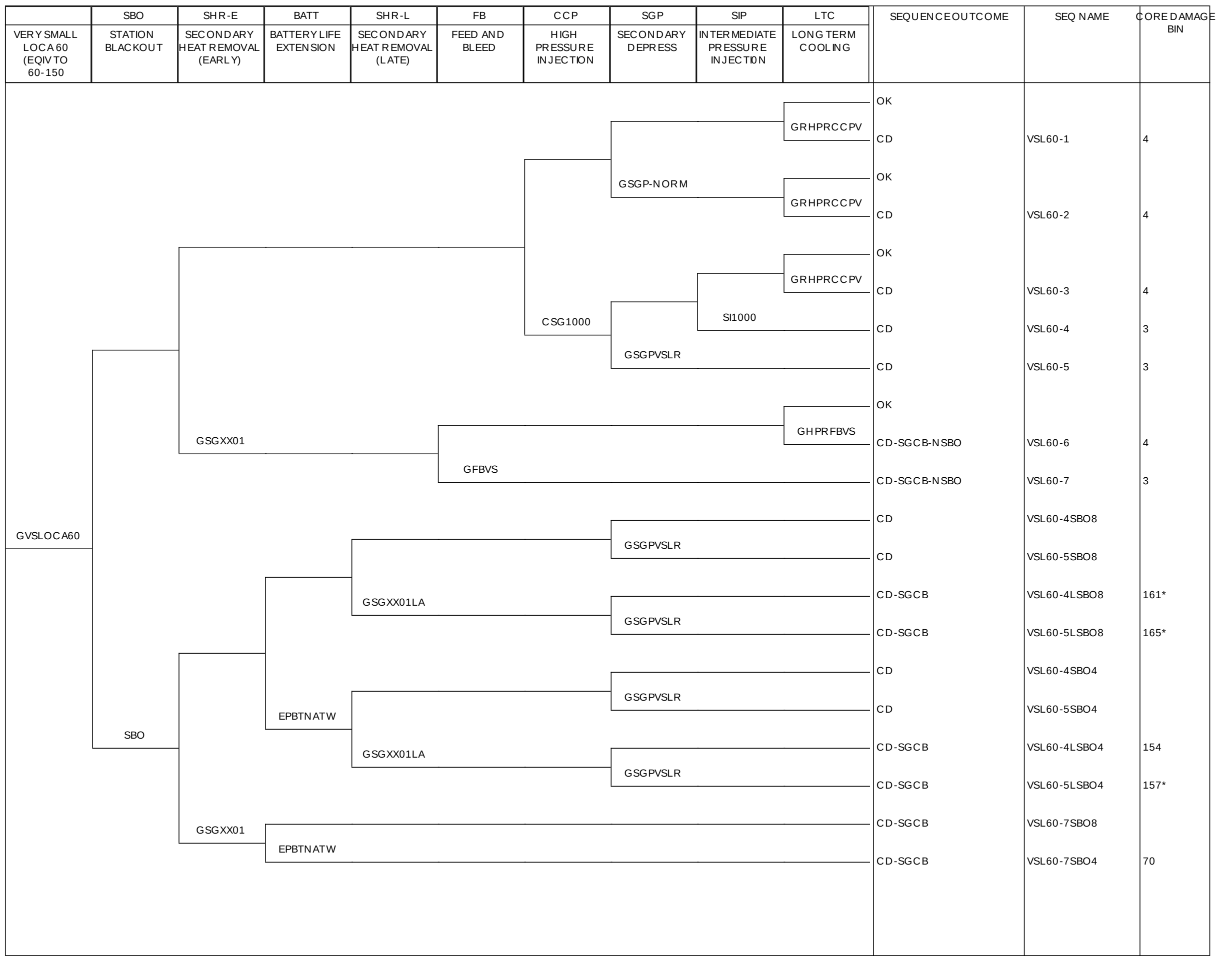

Figure 6. VSL 60 gpm/p event tree for analyzing SAI-SGCB.

NUREG/CR-6521 also provides estimates for the number of flaws of each type that would be present in the steam generators of lightly degraded, moderately degraded, and severely degraded steam generators. For the example plant analysis, a moderately degraded steam generator was assumed.

Only flaws located in tubes that see the hot plume are considered in the analysis because only those tubes would experience the high temperatures needed for creep rupture. For the same reason, only hot tube flaws located between the hot leg inlet and the top of the U-bend were considered.

It is possible to greatly reduce the number of flaws considered in the analysis by recognizing that only more severe flaws will contribute to the containment bypass probability. For example, though the flaw distribution developed by Gorman for axial ODSCC at TSPs shows 570 flaws for moderately degraded steam generators or more than 6,000 flaws for severely degraded steam generators, most of these flaws are relatively shallow. Calculations using the Gorman distribution have shown that only 0.26 percent of these flaws have stress magnification factors (m_p in the creep failure models) greater than 1.2 and only 0.07 percent have values greater than

1.25. Because flaws with values for m_p less than about 1.2 or 1.25 are not likely to fail before other RCS components, these flaws can be safely excluded from the analysis.

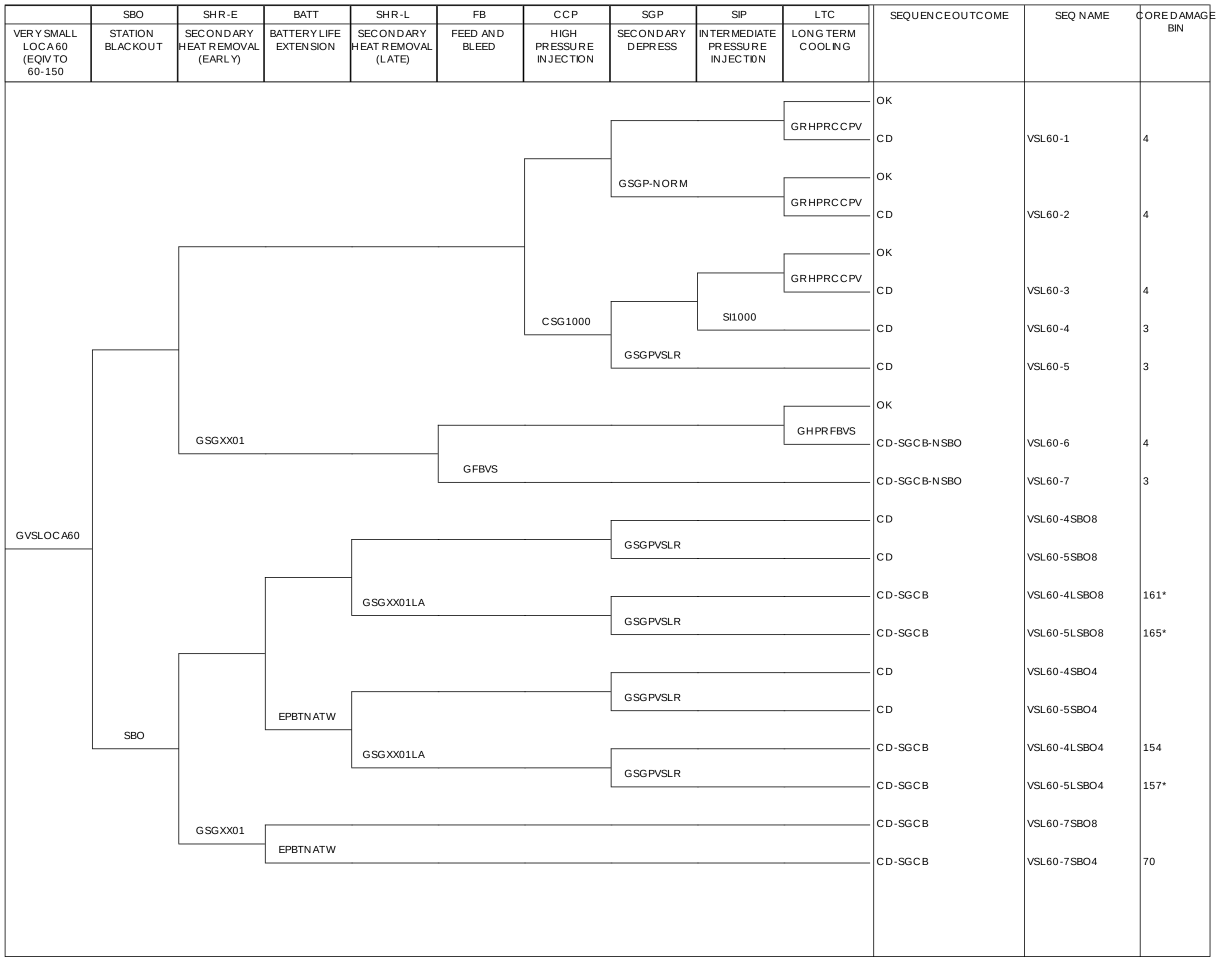

Figure 7. VSL 180 gpm/p event tree for analyzing SAI-SGCB.

3.5.2 Pressure Difference and Temperature Histories at the Flaw Location

Pressure and temperature histories for each accident sequence were calculated using the SCDAP/RELAP T-H computer code [Fletcher, 2004; 2005]. The T-H runs that correspond to each accident sequence shown in Figures 3 through 7 are listed in the column labeled “Core Damage Bin”. The parameters varied in each T-H run are listed in Table 2.

Figure 8 provides two examples of pressure and temperature histories calculated by SCDAP/RELAP. The only difference between these two T-H cases is that case 69 assumes reactor coolant pump leakage at 21 gpm/p whereas case 71 assumes 180 gpm/p leakage. Because primary coolant leakage is considerably smaller for case 69 the pressures and temperatures in the primary system remain higher than in case 71. Consequently, the conditions experienced by the SG tubes are much more severe for case 69 than for case 71. Thus, one would expect case 69 to have a much higher probability of tube failure and containment bypass than case 71.

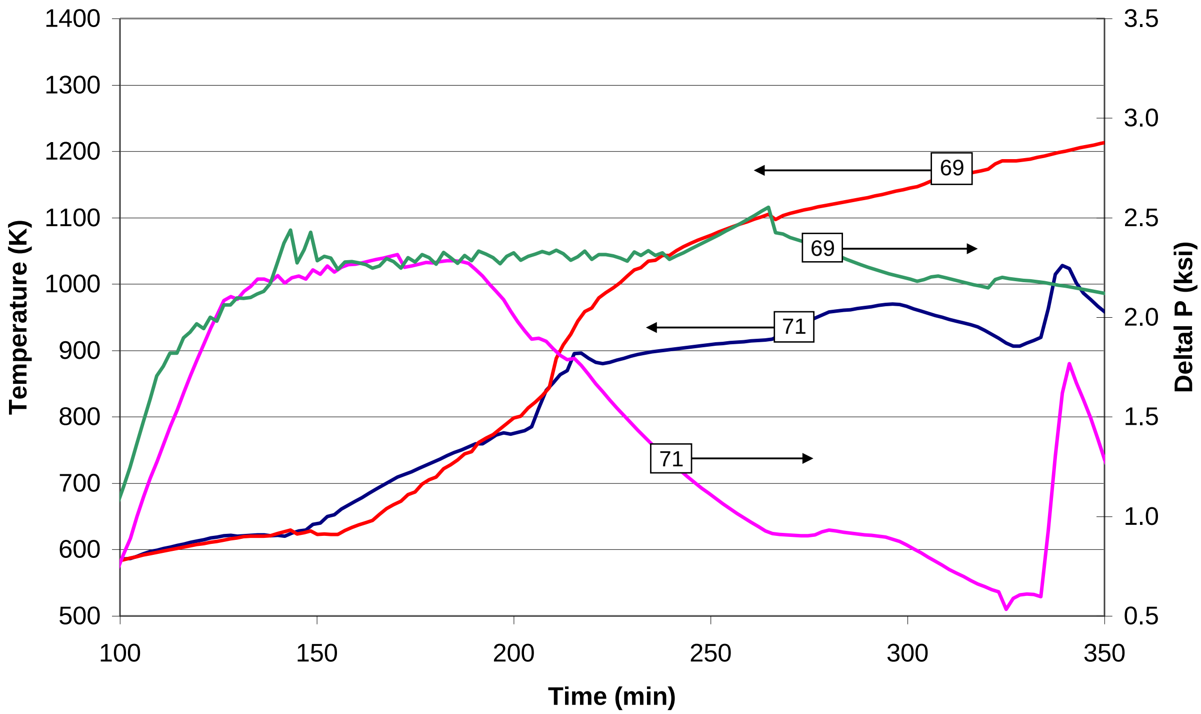

Figure 8. Comparison of the pressure and temperature histories for Cases 69 and 71.

After reviewing the T-H results calculated by SCDAP/RELAP, it became clear that many of the basic events considered in the event trees (Figures 3 through 7) simply delay the onset of the rapid increase in temperature shown in Figure 8. This sudden temperature increase, which is caused by the start of rapid fuel oxidation, initiates the most severe phase of core degradation. Events that just change the timing of the core degradation process and do not significantly alter the magnitude of the pressures and temperature at the tubes, do not significantly affect the SAI-SGCB probability.

For example, consider the list of T-H runs shown in Table 3. Each of these runs was found to result in temperature and pressure histories similar to case 69 (illustrated in Figure 8.) The important characteristics shared by these runs are as follows:

- RCP coolant loss is slow enough (21 or 60 gpm/p) to keep the RCS pressure high during the rapid temperature increase.
- There is no RCS depressurization after core damage.
- Secondary side leakage is sufficient to depressurize the secondary side before the temperature increase begins.

The high temperatures and pressure differences calculated for these conditions present the most severe challenge to the tubes and result in the highest probability of early tube failure and containment bypass.

Table 3. List of T-H conditions that produce similarly severe tube conditions.

Case Number	RCP Seal LOCA (gpm/p)	SG Depress before CD	TDAFW Runs Until Battery Depletion	Valves Reclose before CD	RCS Depress after CD	Battery Depletion Time	Secondary Side Leakage
65	21	Y	N	n/a	No	4hr	Leakage
66	60	Y	N	n/a	No	4hr	Leakage
69	21	N	N	n/a	No	4hr	Leakage
70	60	N	N	n/a	No	4hr	Leakage
73	21	Y	N	n/a	No	8hr	Leakage
74	60	Y	N	n/a	No	8hr	Leakage
77	21	N	N	n/a	No	8hr	Leakage
78	60	N	N	n/a	No	8hr	Leakage
153	21	Y	Y	n/a	No	4hr	Leakage
154	60	Y	Y	n/a	No	4hr	Leakage
157	21	N	Y	n/a	No	4hr	Leakage
158	60	N	Y	n/a	No	4hr	Leakage
161	21	Y	Y	n/a	No	8hr	Leakage
162	60	Y	Y	n/a	No	8hr	Leakage
165	21	N	Y	n/a	No	8hr	Leakage
166	60	N	Y	n/a	No	8hr	Leakage
179	21	N	N	n/a	No	4hr	1 in^2 Leak
181	21	Y	N	n/a	No	4hr	ADV
183	21	N	N	n/a	No	4hr	ADV

The T-H calculations show that one of the major uncertainties in the T-H analysis is the magnitude of the normal secondary side leakage (i.e., normal leakage at valves, flanges, etc.). The majority of the calculations performed for this study assumed secondary side leakage characterized by a 0.5 in^2 hole. A few calculations were performed with lower secondary leakage (equivalent to a 0.1 in^2 hole) or no secondary leakage.

Figures 9 and 10 show the calculated pressure and temperature histories for three T-H runs that differ only in the magnitude of the secondary side leakage. Figure 10 shows that at the highest of these three leak rates (case 69), the secondary leakage is sufficient to depressurize the secondary side so that the pressure difference across the tubes is equal to the primary side pressure. For many of the T-H calculations performed for this study (such as those shown in Table 2), the pressure difference was greater than 2,200 psi. In contrast, if no leakage is assumed (case 198), the secondary side pressure remains high and the pressure difference across the tubes is generally lower than 1,250 psi. As will be shown, tube failure and containment bypass are much less likely for the cases with lower secondary leakage.

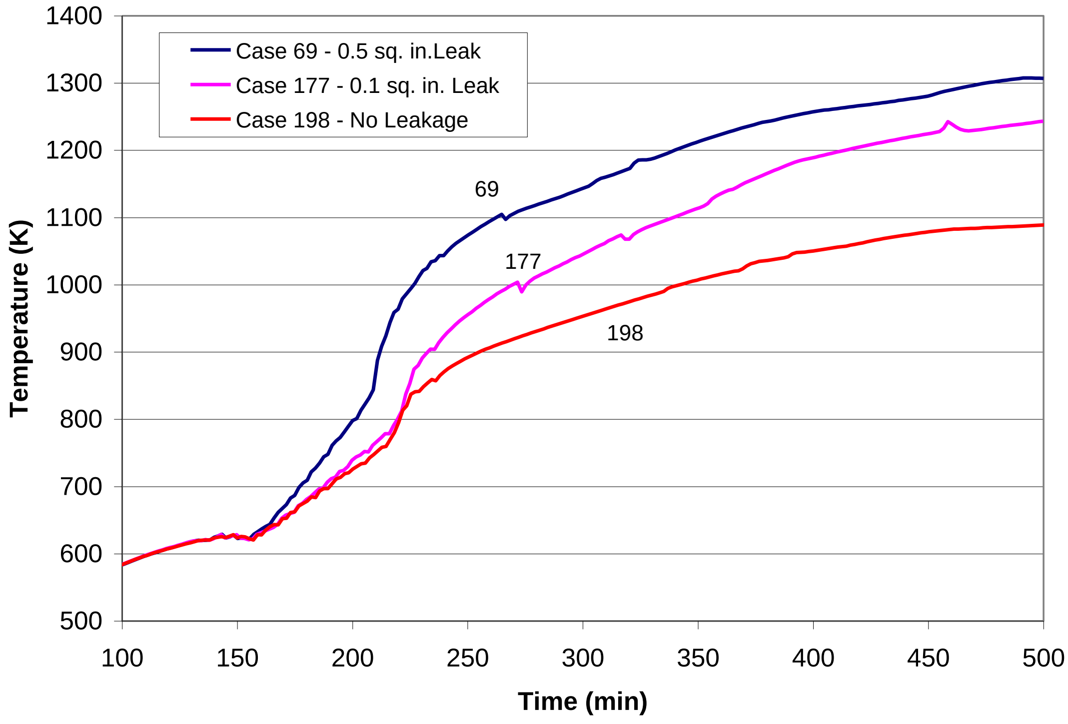

Figure 9. Sensitivity of calculated tube temperature to assumed secondary side leakage.

Ideally, the containment bypass analysis should consider uncertainty in the T-H results. This could be accomplished by performing a series of T-H analyses to address uncertainties in model inputs. The result from this series of calculations would be a family of curves that represent the uncertainty in the T-H models or the uncertainty in model inputs. The containment bypass analysis would then be performed using pressure-temperature histories representative of this range of uncertainty. The example analysis presented here does not consider T-H uncertainty, but rather uses a single pressure-temperature history for each accident sequence.

The temperatures shown in Figures 8 and 9 are the average temperatures experienced by tubes in the hot plume (that portion of the inlet region with up flow from the hot leg) rather than the temperature of any specific tube. Thus, in order to realistically model tube failure it is necessary to consider variations in temperature across the tubes in the hot plume region. The tube failure model considers lateral temperature variations across the hot plume based on computational fluid dynamics (CFD) calculations performed by NRC staff (Boyd et al., 2004). The CFD calculations have been benchmarked against scaled-experiments performed by Westinghouse.

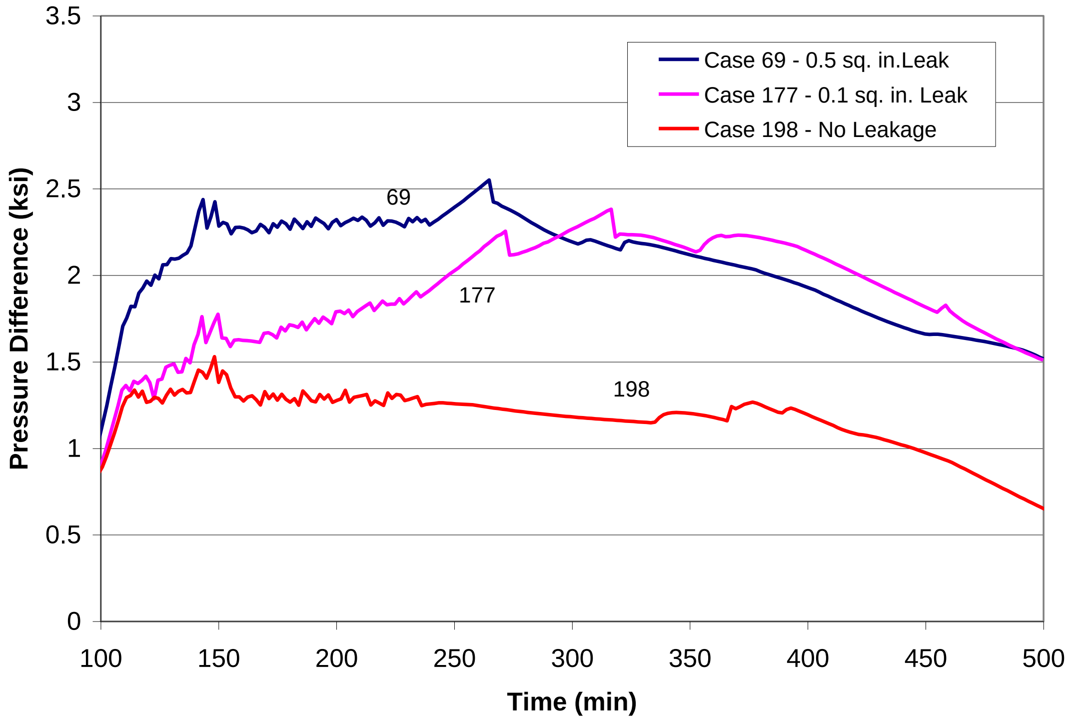

Figure 10. Sensitivity of calculated pressure difference to assumed secondary side leakage.

The CFD analyses showed that the number of tubes exposed to the hot plume and the temperature distribution in the plume vary between plant designs. For the Westinghouse Model 44/51 steam generator, the CFD analysis indicated that 44% of the tubes would be in up flow and that these tubes would have temperatures characterized by the distribution shown in Figure 11. The figure shows the distribution as the ratio of the tube temperature to the average hot plume temperature.

The tube temperature at the flaw also depends on the axial location of the flaw relative to the inlet plenum, with temperatures decreasing with distance from the plenum. The axial temperature gradient used in the tube failure model is based on the results from the SCDAP/RELAP T-H analyses. These calculations showed a temperature drop of approximately 70 K between the tube inlet and the top of the U-bend.

3.5.3 Failure Time for Other RCS Components

For this example application of the containment bypass analysis, only failure of the hot leg and surge line are considered. Creep-induced failure times for the hot legs and surge line are calculated by SCDAP/RELAP for each T-H run. As with the tube failure analysis, creep failure of the hot leg and surge line depend on the pressure and temperature at these two locations. Because the pressure-temperature histories at the hot leg and surge line are generally controlled by the same factors that control the pressures and temperatures experienced by the tubes, the times for hot leg and surge line failure are typically close to the times for tube failure. Thus, it is important to consider the uncertainty in the hot leg and surge line failure times.

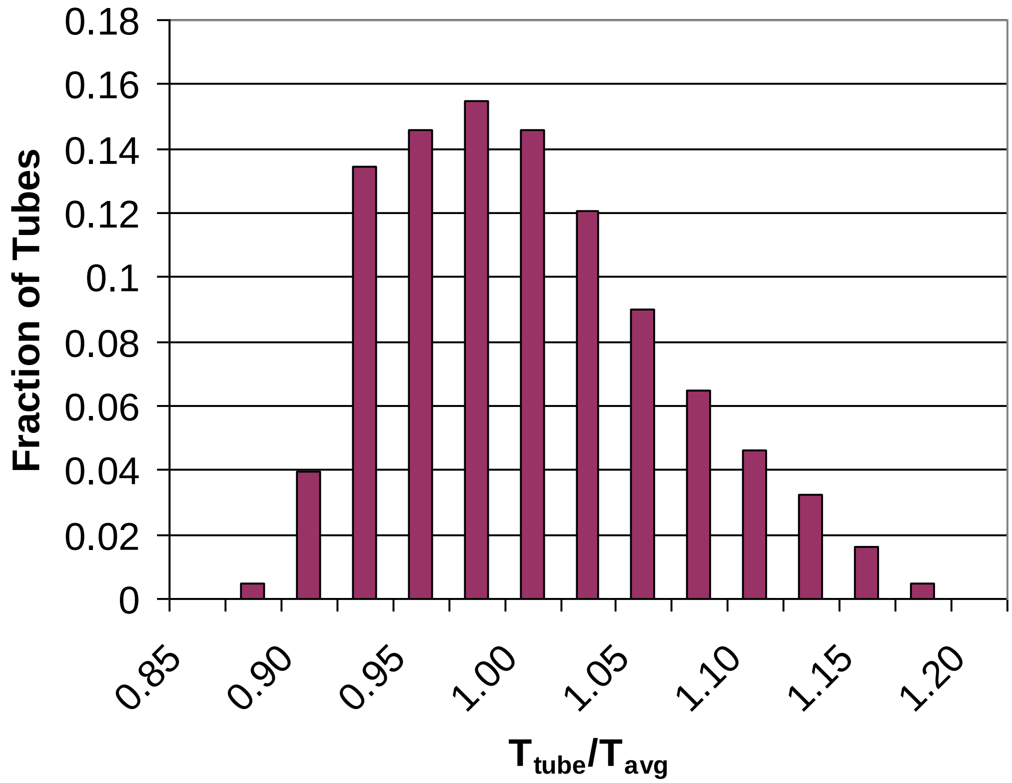

Figure 11. Temperature distribution for tubes in the hot plume.

Based on a review of the SCDAP/RELAP analyses and on detailed failure calculations performed using ABAQUS, researchers at ANL estimated that the probability distributions for the failure times of the hot leg and surge line could be approximated by a normal distribution with standard deviation of 3 minutes. This uncertainty reflects uncertainty both in material properties of the hot leg and surge line and in the T-H results.

3.5.4 Calculated Results

The containment bypass probability has been calculated for each accident sequence shown in the event trees presented in Figures 3 through 7. This analysis uses the probabilistic approach outlined in Section 2.5. In this methodology, the containment bypass model is repeatedly solved using Monte Carlo analysis. In the Monte Carlo analysis, the probability distributions for uncertain model inputs are sampled and the time to the critical crack opening area is calculated. If this time occurs before the time of hot leg or surge line failure (determined by sampling from the probability distributions discussed in the preceding section), then it is assumed that containment bypass has occurred for that sample. The containment bypass probability is simply the fraction of the Monte Carlo samples in which bypass would be predicted.

The calculated containment bypass probabilities are shown in Table 4. The table shows that the containment bypass probability is 0.4 for many of the accident sequences. In these sequences, the RCP seal leakage is 60 gpm/p or less and the secondary side leakage area is 0.5 in^2. Under these conditions, the secondary side is fully depressurized when the RCS is at high pressure. Consequently, the pressure difference across the tubes is greater than 2,200 psi when the average tube temperature increases to greater than 1000 K and multiple early tube failures are predicted to occur. The table also shows that the containment bypass probability is considerably smaller if the assumed RCP seal leakage is 180 gpm/p or higher.

As noted in Table 4, the assumed nominal secondary side leakage was 0.5 in^2 in these calculations. A limited set of T-H runs has been completed assuming smaller secondary side leakage. The sensitivity of the containment bypass probability to the assumed secondary side leakage can be inferred by considering the three cases shown in Figures 9 and 10. These three cases assumed the following equivalent secondary side leakage areas: 0.5 in^2, 0.1 in^2, and no leakage. The calculated containment bypass probabilities for these three cases were as follows: 0.4 for the case with 0.5 in^2, 0.08 for the case with 0.1 in^2, and 0.01 for the case with no leakage. Clearly, in any future application of this methodology to a given plant, it will be very important to determine the level of secondary side leakage for the plant.

3.6 Assessment of Human Error Probabilities

This section summarizes the human reliability analysis (HRA) performed to support this initial SAI-SGCB PRA. The HRA performed for this study addresses only post-initiator HFEs. Pre-initiator HFEs may have been included in the PRA model used for this study, but were not further analyzed and left "as is" in the model. The post-initiator HFEs addressed in the current HRA analysis were limited to operating crew actions associated with SBO scenarios, in which either an early or (at least potentially) a late loss of TDAFW occurs. In the scenarios addressed, the RCS remains at high pressure during core melt. If the integrity of the secondary coolant system is lost, for whatever reason, and steam egresses from it, then the secondary sides of the steam generators become dry and the pressure drops causing larger pressure differentials across the tubes, increasing the chance of tube failure.

The PRA examined system and component failures that would put the reactor system in those conditions. The PRA and HRA analysts working together and using results from the T-H analyses, identified operating crew actions that could mitigate the accident progression. Many system events and operator actions can affect the timing and the progression of such accident scenarios. These include such things as the availability of the TDAFW pump, demand of PORVs, operator shedding of DC loads to extend battery life, operator depressurization of the secondary side and/or primary side per emergency procedures or severe accident management guidelines, secondary side leakage, and power recovery.

The following subsections describes the process used to obtain information needed to understand the likely scenario conditions that could influence crew behavior and the process used to quantify the failure probabilities of the modeled HFEs. A summary of the results of the quantification process for each of the modeled HFEs is also provided. A detailed discussion of the derivation of each of the human error probabilities (HEPs) for the modeled HFEs is presented in Appendix B.

Table 4. Calculated containment bypass probabilities.

Accident Sequence	T-H Case #	SAI-SGCB Probability	RCP Seal Leakage	SG Depress before CD	TDAFW runs until battery depletion	RCS depress after CD	Valves reclose at battery depletion	SG Leakage[a]
TRASGCB-2DSBO8	161	0.4	21 gpm/p	Y	Y (8 hr)	No	8 hr	Nominal
TRASGCB-2DSBO4	153	0.4	21 gpm/p	Y	Y (4 hr)	No	4 hr	Nominal
TRASGCB-2LSBO8	165	0.4	21 gpm/p	N	Y (8 hr)	No	8 hr	Nominal
TRASGCB-2LSBO8	157	0.4	21 gpm/p	N	Y (4 hr)	No	4 hr	Nominal
TRASGCB-2SBO8	69[b]	0.4	21 gpm/p	n/a	N	No	8 hr	Nominal
TRASGCB-2SBO4	69	0.4	21 gpm/p	n/a	N	No	4 hr	Nominal
VSL60-4LSBO8	161[c]	0.4	60 gpm/p	Y	Y (8 hr)	No	8 hr	Nominal
VSL60-5LSBO8	165[c]	0.4	60 gpm/p	N	Y (8 hr)	No	8 hr	Nominal
VSL60-4LSBO4	154	0.4	60 gpm/p	Y	Y (4 hr)	No	4 hr	Nominal
VSL60-5LSBO4	157[c]	0.4	60 gpm/p	N	Y (4 hr)	No	4 hr	Nominal
VSL60-7SBO8	70[b]	0.4	60 gpm/p	n/a	N	No	8 hr	Nominal
VSL60-7SBO4	70	0.4	60 gpm/p	n/a	N	No	4 hr	Nominal
VSL180-4LSBO8	163	0.02	180 gpm/p	Y	Y (8 hr)	No	8 hr	Nominal
VSL180-5LSBO8	167	0.00	180 gpm/p	N	Y (8 hr)	No	8 hr	Nominal
VSL180-4LSBO4	155	0.02	180 gpm/p	Y	Y (4 hr)	No	4 hr	Nominal
VSL180-5LSBO4	159	0.07	180 gpm/p	N	Y (4 hr)	No	4 hr	Nominal
VSL180-7SBO8	71[b]	0.14	180 gpm/p	n/a	N	No	8 hr	Nominal
VSL180-7SBO4	71	0.14	180 gpm/p	n/a	N	No	4 hr	Nominal

Notes:

[a] All T-H runs listed in this column assumed nominal secondary side leakage of 0.5 in^2.

[b] Equivalent short term run (4-hour battery depletion) used because longer battery life does not significantly affect pressures and temperatures experienced by the tubes.

[c] These VSL60 sequences used the equivalent runs for 21 gpm/p RCP seal leakage

3.6.1 Summary of the HRA Process

An abbreviated version of the ATHEANA HRA process (NUREG-1880) [Forester, 2007] was the basic approach used to perform this HRA analysis. An abbreviated approach was used for several reasons. First, even though a particular plant and its PRA were chosen to serve as the “generic” plant for purposes of the analysis, it was decided that a plant visit was not needed for the level of analysis required at this time. Thus, no simulator exercises could be observed and no questions could be asked directly to plant operators and trainers. In addition, even though the Westinghouse Owners Group ERGs and SAMGs were available for the analysis, plant - specific procedures were not. Finally, it was decided that for the present analysis, questions regarding plant-specific performance shaping factors (PSFs) such as operating crew training and biases, crew understanding of procedures and their usage, crew dynamics and characteristics, key instrumentation and cues for the scenario from the crews perspective, expected workload, human-system interface characteristics, and “informal rules” that might influence their decisions, could not be submitted to the plant at this time. Thus, much of the plant-specific information required by the ATHEANA HRA method to perform a realistic analysis (and many other methods for that matter) could not be obtained.

In lieu of the opportunity to obtain this kind of information, many assumptions were made (rather than searches for information as is prescribed by the ATHEANA process) regarding the scenario conditions and how operating crews would respond in the scenarios being analyzed. However, as noted above, the Westinghouse Owners Group ERGs and SAMGs were available and the PRA/HRA team included several individuals with many years experience in performing PRA and HRA, including engineers and psychologists. These individuals are very familiar with plant control rooms and plant operations and in addition, a former senior reactor operator from a PWR was available to support the team.

Thus, even though an ATHEANA HRA analysis could not be performed for this work, it was believed that reasonable assumptions could be made about the factors and conditions likely to influence operating crew behavior in the scenarios being examined. In this respect, attempts were made to at least “hypothetically” obtain the information that would be needed for an ATHEANA analysis and use it during the quantification process. While it is believed that in general the results of the analysis are probably “realistically conservative” (which was the initial goal), it should be realized that without plant-specific information, it is possible that incorrect assumptions have been made that could significantly alter the probabilities of the modeled HFEs. It should also be recognized that very little is known about how crews will respond under severe accident conditions. Thus, the results of the HRA analysis should be considered in this light and it should be realized that a more thorough analysis is needed to increase confidence in the results.

3.6.2 Process for Identification of HFEs and Scenario Context for the Analysis

Since plant-specific information could not be obtained, an early step in the analysis was to study and understand the ERGs and SAMGs relevant to the scenarios of interest. To facilitate this process, flow charts of the relevant procedures were developed (per the ATHEANA process) and critical decision points were identified. This information, in conjunction with the PRA modeling of the accident scenarios, was used to identify human actions with the potential to be important (including errors of omission and errors of commission). In turn, this information, in conjunction with the results of the T-H analyses (including scenario specific timing information and plots of the expected behavior of critical parameters), was then used to develop the important accident scenarios and the list of HFEs to be quantified. This information was also used to identify, to the extent possible, aspects of the scenario context (including PSFs and plant conditions, where the plant conditions were taken from the output of the T-H analyses) that would be faced by the operating crews responding to the accident scenarios and that could influence the probability of the HFEs.

3.6.3 Quantification Process

The HRA quantification approach used for the SAI-SGCB analysis is described in six steps below. The quantification approach relies on a facilitator-led, expert opinion elicitation.

Step 1: Describe the HFE and associated context

The purpose of Step 1 is to:

1. Collect (or make assumptions about) any additional information that is not already collected and that is needed to describe and define the HFEs (and associated contexts),
2. Review all information for clarity, completeness, etc., and
3. Interpret and prioritize all information with respect to relevance, credibility, and significance.

Table 5 provides examples of information that would normally be identified using the ATHEANA method and that would serve as inputs to the quantification process, whether collected during the HFE and context identification process or as part of Step 1 of quantification.

As discussed above, for the present analysis, assumptions about such information had to be made in many cases.

The third item from Step 1 above is especially important if:

- some information is applicable only to certain scenarios, HFEs, or contexts
- there are conflicts between information sources
- information is ambiguous, confusing, or incomplete
- information must be extrapolated, interpolated, etc.

All of the three items in Step 1, and especially the third item, were performed as part of an open discussion among the experts (in this case, the HRA team for the study) involved in the expert opinion elicitation process. The goal of this discussion was not to achieve a consensus but, rather, to advance the understanding of all the experts through the sharing of distributed knowledge and expertise. In each case, the scenario (or group of similar scenarios) and the HFE in question are described and the vulnerabilities and strong points associated with taking the right action are discussed openly among the team.

Table 5. Examples of information useful to HFE quantification.

Information Type	Examples
Plant conditions and behavior for scenario/context	T-H conditions as a function of time, expected plant indications as a function of time, system/equipment operations, expected operator actions.
Critical plant functions for accident mitigation	Specific equipment operation, requirements for operator action, possible operator recovery actions.
Operating crew characteristics (i.e., crew characterization)	Crew structure, communication style, emphasis on crew discussion of “big picture”, behaviors observed in simulator exercises and/or identified by training staff.
Features of procedures	Structure, how implemented by operating crews, opportunities for “big picture” assessment and monitoring of critical safety functions, emphasis on relevant issue (e.g., SAI-SGCB), priorities, any potential mismatches with deviation scenarios (see NUREG-1880).
Relevant informal rules	Experience, training, practice, ways of doing things - especially those that may conflict with informal rules or otherwise lead operators to take inappropriate actions.
Timing	Plant behavior and requirements for operator intervention versus expected timing of operator response in performing procedure steps, etc.; input from training staff and results of simulator exercises; based upon perceived needs of the PRA, multiple times or time frames may need to be considered for each HFE.
Relevant vulnerabilities	Any potential mismatches between the scenarios and expected operator response with respect to timing, formal and informal rules, biases from operator experience and training, and so forth.
Performance shaping factors	Those deemed associated with or triggered by the relevant plant conditions and including whether they are positive or negative influences and the strengths of their influence on operator performance for the context (e.g., missing or misleading indications, complex situations, timing mismatches and delays, procedural ambiguities, workload, and human-machine interface concerns).
Recovery potential	Possible recovery actions if the initial error should be made. Consideration of cues for doing so, time available before undesired consequences, staff resources for doing so, and so forth.

Step 2: Identify the key or driving factors of the scenario context

The purpose of Step 2 is to identify the key or driving factors on operator behavior/performance for each HFE and associated context. Each expert participating in the elicitation process individually identifies these factors based on the expert’s own judgment. Usually, these factors are not formally documented until Step 4.

Typically, multiple factors will be deemed most important to assessing the probability for the HFE in question. This is due to the focus of the ATHEANA search process on combinations of factors that are more likely to result in an integrated context. When there is only a single driving factor, it is usually one that is so overwhelming that it alone can easily drive the estimated probability. For example, if the time available is shorter than the time required to perform the actions associated with the HFE, quantification becomes much simplified and does not need to consider other factors.

Step 3: Characterize the context for the HFE and determine an HEP

In Step 3, each expert participating in the elicitation process must answer the following question for each HFE: Based upon the factors identified in Step 2, how difficult or challenging is this context relative to the HFE being analyzed?

Answering this question involves independent assessments by each expert. In order to perform this assessment, the specifics of the context defined for a HFE must be generalized or characterized. These characterizations or generalizations then must be matched to general categories of failures and associated failure probabilities.

To assist the analysts (who may not have strong backgrounds in probability) in making their judgments regarding the probability of events, some basic guidance is provided. In thinking about what a particular probability for a HFE will be, they are encouraged to try to imagine how many times out of 10, 100, 1000 etc. would they expect crews to commit the HFE, given the identified context, The following examples of what different probabilities mean are provided to the analysts:

- "Likely" to fail ~ 0.5 (5 out of 10 would fail)
- "Infrequently" fails ~ 0.1 (1 out of 10 would fail)
- "Unlikely" to fail ~ 0.01 (1 out of 100 would fail)
- "Extremely unlikely" to fail ~ 0.001 (1 out of 1000 would fail)

The analysts are allowed to select any values to represent the probability of the HFE. That is, other values (e.g., 3E-2, 5E-3) can be used. However, the analyst must provide numeric probabilities. The qualitative descriptions above are provided initially to give analysts a simple notion of what a particular probability means.

For exceptional cases, the quantification approach also allowed an HEP of 1.0 to be used when failure was deemed essentially certain.

Step 4: Discuss and justify the characterizations of the context and the HEP estimates made in Step 3

In Step 4, each expert was asked to independently provide his/her estimate for each HFE. Once all the expert estimates were recorded, each expert was asked to describe the reasons why he/she chose a particular failure probability. In describing his/her reasons, each expert should identify what factors (positive and negative) were thought to be key to characterizing the context and how this characterization fit the failure category description.

After the original elicited estimates were provided, a discussion was then held that addressed not only the individual expert estimates but also differences and similarities among the context characterizations, key factors, and failure probability assignments made by all of the experts. This discussion allowed the identification of any differences in the technical understanding or interpretation of the HFE versus differences in judgment regarding the assignment of failure probabilities. Examples of factors important to HFE quantification that might be revealed in the discussion include:

- differences in key factors and their significance, relevance, etc., based upon expert-specific expertise and perspective
- differences in interpretations of context descriptions
- simplifications made in defining the context
- ambiguities and uncertainties in context definitions

A consensus opinion was not required following the discussion.

Step 5: Refinement of HFEs, associated contexts, and assigned HEPs (if needed)

Based upon the discussion in Step 4, the experts formed a consensus on whether or not the HFE definition must be refined or modified based upon its associated context. If the HFE must be refined or re-defined, this is done in Step 5. If such modifications were necessary, the experts "re-estimated" the HEP based upon the newly defined context for the HFE (or new HFEs, each with an associated context).

The experts participating in the elicitation process also were allowed to change their estimate after the discussion in Step 4, whether or not the HFE definition and context were changed. Once again, a consensus was not required.

Step 6: Determine final HEP for HFE and associated context

The final probability estimate (from the initial quantification process) that will be incorporated into the PRA for each HFE is determined in Step 6.

The failure probabilities assigned in the HRA quantification for the SAI-SGCB analysis are, based on the assumptions about the context that were made (see discussion above), assumed to be "realistically conservative." To help ensure this conservatism, if consensus could not be reached, the final failure probability that was assigned to each HFE was determined by choosing the highest assigned probability among the final estimates of the experts participating in the expert elicitation process.

It is important to note how dependencies among multiple HFEs appearing in the same scenario were handled. Given the detail provided in the contextual development provided for each HFE, the HRA team was careful to identify cases where the HFE being considered was one of multiple HFEs that were part of the overall scenario of concern. For example, if it was known that one scenario involved multiple relevant functional failures that would require multiple human actions in the same sequence of events versus another scenario where this was not the case for the HFE of interest, then human error values were estimated both for cases where the HFE of interest would appear among other multiple HFEs and where it would not. This was possible since (1) the PRA model, being largely an event tree model, easily displayed where multiple HFEs would appear in a given scenario and (2) given the close integration of the PRA and HRA efforts, the HRA team was well aware of where these multiple HFEs applied. In this way, the HRA team was able to know, in large part, where there might be multiple and dependent HFEs needing to be quantified within the same scenario context and the resulting elicited estimates already accounted for such possible dependencies.

3.6.4 Summary of HRA Results

The results of the HRA are presented in this section. First, the initial list of HFEs identified for modeling are presented and discussed. Reasons are provided for why some of these HFEs were retained and modeled and why others were not. Second, the results of the quantification of the retained HFEs are provided.

Initially Identified HFEs that were Not Modeled

A. Error of commission (EOC) to inappropriately depressurize the steam generators.

The issue for this event is whether the crew might choose (for some reason) to inappropriately depressurize the steam generators in SBO scenarios with a loss of TDAFW. While it may be possible to identify reasons why the crew might choose to take this action (fairly early in the scenario when steam generator pressure might still be somewhat high), the understanding from team discussions was that whether they did so or not would not make a significant difference in the outcome of the scenario. That is, since the steam generators are expected to depressurize anyway due to leakage, and since in the long term the timing for severe core damage would not change significantly whether the steam generators are depressurized early or allowed to depressurize slowly over time as a result of a leak, manual depressurization on the part of the crew would not significantly change the timing of the scenario or its effects on the likelihood of tube failure and therefore it is not necessary to model the action.

Potential dependencies between the EOC and later actions in the scenario were also considered. That is, if the crew took such an inappropriate action, would that fact need to be considered in evaluating later operator actions. It was decided that due to the eventual presence of the TSC and probably several hours between when the action might occur and later important actions that need to be taken, e.g., depressurizing the RCS, it was very unlikely that significant dependencies would be expected. In other words, it was decided that the event was not needed in the model in order to address potential dependencies with later operator actions.

B. Operator failure to isolate main steam path and main feedwater path, including any faulted or ruptured steam generators (in the SBO procedure ECA-0.0).

As in Item #1 above, based on the information received through team discussions, the crew failing to isolate the main steam path and main feedwater path under the conditions being addressed (SBO with loss of TDAFW) would not significantly change the timing or outcome of the scenario. Again, the steam generators will depressurize anyway due to leakage. Also, there is little reason to expect a long-term linkage between the failure of this event and later operator actions, mainly because of the eventual presence of the TSC and the long time-frame available for the crew to be able to evaluate plant status before later important actions must be taken.

C. EOC to throttle back or shutdown an operating TDAFW system

Since this would be a very significant error, the analysts asked whether there are any reasons the crew might severely throttle or terminate TDAFW. It would seem that many instruments would have to fail in order to confuse the crew enough to lead them to terminate their only source of heat removal. Given that the SBO would be clear (and there are no extenuating circumstances such as a fire), a very unique set of instrument failures (in very unlikely patterns) would have to occur before one might consider that the crew would turn off or throttle back their only source of heat removal. Even if most relevant instruments failed, the SBO conditions would seem to dictate that the crew should maintain TDAFW as long as possible. Thus, this action was not modeled.

General HFEs that Were Modeled

1. Operator failure to depressurize the steam generators when TDAFW is running.

In this scenario, a SBO has occurred, but TDAFW is still available. In ECA 0.0 (SBO procedure), Step 16 directs the crew to depressurize the steam generators if/when a specified narrow range level is reached. This is an important event in this scenario. Depressurizing the steam generators will get rid of a lot of heat at once, which would cool down the RCS and allow significant additional recovery time to restore offsite power or for recovering the diesels, etc. In other words, depressurizing the steam generators while TDAFW is available will extend the scenario time, lengthening the time to core damage and changing the overall impact of the event (reducing CDF) since more recovery potential subsequently exists. Depressurizing the steam generators will also allow the accumulators to dump, which would also contribute to extending the scenario time.

2. Operator failure to shed all large non-essential DC loads

In Step 14 of ECA 0.0, operators are directed to shed all large non-essential DC loads. The goal is to extend DC battery life, which as in item #1 above, would extend the scenario time, lengthening the time to core damage and changing the overall impact of the event (reducing CDF). If neither offsite power nor the diesels are restored, then this action would not directly affect the potential for creep rupture etc., but it does extend the time to allow relevant recoveries. Hence this event was added to the model with a corresponding HEP.

3. Operator failure to provide DC power to allow opening of the PORVs (or other potential long-term recovery actions, e.g., aligning diesel driven fire water pumps for AFW).

In the SBO scenarios of interest, the DC batteries have depleted, but at <u>least</u> several hours are available (at least if the TDAFW pump was running before DC power is lost and so a significant core cooling period has occurred) for the TSC and crew to diagnose the need, and identify, locate, and hook-up an alternate source of DC power in order to allow opening of the PORVs to depressurize the RCS. The goal is to provide means for avoiding creep rupture of the steam generators in these core damage scenarios. This action may not be proceduralized, but given the amount of time available, there is at least some reason to believe that the TSC and crew would diagnose the need and execute the relevant action. However, in the context of PRA/HRA, this type of action has not traditionally been credited, and therefore reasonable investigation would be required before allowing any credit to be taken. Other actions may also benefit these scenarios, such as aligning diesel driven fire water pumps for AFW. Aligning the fire water system may be proceduralized at some plants (i.e., in the SAMGs), but such actions are really not expected to ever be needed, and therefore would require careful analysis.

4. Operator Failure to open PORVs to depressurize the RCS

In the basic SBO scenario with no TDAFW, it is assumed that the TSC has been formed and that the crew enters Severe Accident Control Room Guideline 2 (SACRG-2), the SAMG for the case where the TSC has been formed. This procedure essentially directs the crew to follow the direction of the TSC, which will be guided by the Diagnostic Flow Chart (DFC). SACRG-2 directs the crew to perform several checks and evaluations while the TSC is following the DFC, including checking whether any instruments have failed or are behaving in an unexpected manner. The operating crew is to notify the TSC of any status changes, determine whether there are any special monitoring requirements, and follow the TSC direction regarding the operation of equipment.

In this basic scenario, it is assumed that RCS pressure will exceed the threshold for entering Severe Accident Guideline 2 (SAG-2), requiring the TSC to address whether or not to depressurize the RCS. This is a critical decision with respect to preventing potential creep rupture of the SG tubes, since depressurizing the RCS reduces the differential pressure between the primary and secondary sides of the tubes. The question is what is the probability that the TSC and the operating crew would fail to depressurize the RCS, given that RCS pressure has reached the appropriate level and there has been a loss of all FW (which leads to the steam generators boiling dry)?

Another potential scenario is similar, except that the TSC has not yet formed and the operating crew must make the decision to open the PORVs, which is covered in Step 9 of SACRG-1. This possibility is also addressed during quantification.

3.6.5 Results of the HFE Quantification Process

Table 6 presents the basic results of the quantification of the HFEs. Note that for all the HFEs, it was necessary to separately quantify sub-cases of the four actions (numbers 1 through 4) described above. This was necessary to meet various scenario requirements of the PRA model. Appendix B provides a detailed discussion of the derivation of the HEPs for each of the events and cases presented in Table 6.

3.7 Additional Data Evaluation

The SAI-SGCB sequence frequencies were quantified using the data developed by the WPWR-A plant for their original PRA (e.g., failure of the TDAFW pump to start). However, some additional data values were generated including the probabilities for the HFEs discussed in Section 3.6. In addition, the probabilities used in the original PRA for the recovery of offsite power and for different sizes of RCP seal LOCA were changed or modified. This section presents these revised probabilities.

3.7.1 Recovery of Offsite Power

The probability of recovering offsite power as a function of time was modeled using the latest information that was published in draft form in 2004 and later finalized in NUREG/CR-6890 [Eide, 2005]. This draft report updated earlier work on the same topic, published in 1996 as NUREG/CR-5496 [Atwood, 1998]. The final data values published in NUREG/CR-6890 are different than the values shown below and used in the example plant analysis because the data was subsequently updated to include data from 2004.

Table 6. Basic results of HFE quantification.

HFE Description	HEP
(1A) Operator failure to depressurize the steam generators when TDAFW is running. Note that complete dependence was assumed between this action and the crew failing to shed non-essential DC loads. That is, this action is assumed to fail (HEP =1.0) if the crew fails to shed non-essential DC loads. Something must be wrong if the crew fails to follow procedure to shed DC loads and it is therefore difficult to give credit for the subsequent action to depressurizing.	0.01
(1B) Operator failure to depressurize the steam generators when TDAFW is running (but in this case at least one diesel generator starts and runs for awhile, but then stops). Note that complete dependence was assumed between this action and the crew failing to shed non-essential DC loads. That is, this action is assumed to fail (HEP =1.0) if the crew fails to shed non-essential DC loads.	0.01
(2A) Operator failure to shed all large non-essential DC loads (SBO is assumed to occur immediately in the scenario)	0.01
(2B) Operator failure to shed all large non-essential DC loads (but in this case at least one diesel generator starts and runs for 15-30 minutes, but then stops). Failure to shed loads results in battery depletion in 4 hours as opposed to 8 hours if loads are shed.	0.01
(3A) Operator failure to provide DC power to allow opening of the PORVs (early secondary heat removal fails, operator sheds nonessential DC loads) No HEP assessed because lower head of reactor vessel fails before batteries are assumed to deplete at eight hours	NA
(3B) Operator failure to provide DC power to allow opening of the PORVs (early secondary heat removal fails and operator fails to shed nonessential DC loads).	1.0
(3C) Operator failure to provide DC power to allow opening of the PORVs (early secondary heat removal succeeds [does not depend on success in shedding nonessential DC loads]). It is assumed that the plant has no pre-planned and specific contingencies for restoring DC power. This was used as the base case since the team was not able to do a plant-specific analysis and did not have industry information regarding the use of specific and pre-planned contingencies to restore power to the battery chargers, or to provide DC power directly to the DC busses. Thus, the more conservative value was assigned to the base case.	0.5
(3D) Operator failure to provide DC power to allow opening of the PORVs (early secondary heat removal succeeds [does not depend on success in shedding nonessential DC loads]). It is assumed that the plant has pre-planned and specific contingencies for restoring DC (used as sensitivity case).	0.1

Table 6. Basic results of HFE quantification.

HFE Description	HEP
(4A) Operator Failure to open PORVs to depressurize the RCS (early secondary heat removal succeeds). No HEP assessed because the action is implicitly included under cases 3C and 3D above (i.e., if the crew fails to restore DC power, they cannot open PORVs anyway; if they do restore DC power, it is assumed PORVs will be opened by the operator since this would be one of the primary purposes for restoring DC power).	NA
(4B) Operator Failure to open PORVs to depressurize the RCS (early secondary heat removal fails and operator fails to shed nonessential DC loads. HEP not assessed since case #3B above (Failure to restore DC = 1.0) precludes the ability to open the PORVs since there will be no DC power.	NA
(4C) Operator Failure to open PORVs to depressurize the RCS (early secondary heat removal fails and operators shed nonessential DC loads) TSC is not yet activated.	0.1
(4D) Operator Failure to open PORVs to depressurize the RCS (early secondary heat removal fails and operators shed nonessential DC loads) TSC is activated, which is the most likely case since we assume that a prolonged SBO would trigger TSC activation.	0.5

The new report updates both the frequency and duration of offsite power losses. Events are partitioned into five categories for analysis: plant-centered, switchyard, grid, severe weather, and extreme severe weather. Frequencies and durations are estimated for each category. The frequencies estimated for critical operation were used in the example analysis since only the risk associated with power operation was calculated. These frequencies are shown in Table 7.

Table 7. LOSP frequency, by category (per Rx-crit-yr).

Plant-centered	Switchyard	Grid-related	Severe weather	Extreme severe weather
2.38×10^{-3}	8.74×10^{-3}	1.67×10^{-2}	2.98×10^{-3}	2.32×10^{-3}

Recovery for each of these categories was modeled with a Weibull distribution. Table 8 shows the estimated probability of not recovering offsite power by time t, for each of the categories modeled.

The final nonrecovery probability that is used in the SAI-SGCB model is a frequency-weighted average of these values, using the frequencies from Table 7. This gives the unconditional probability of not recovering offsite power by time t. The final results are shown in Table 9 below, and graphically in Figure 12.

Table 8. Probability of not recovering offsite power by time t, by category.

Time (mins.)	Plant Centered	Switchyard	Grid-related	Severe weather	Extreme severe weather
60	0.140	0.363	0.617	0.464	0.998
75	0.105	0.308	0.559	0.430	0.997
110	0.058	0.217	0.452	0.371	0.995
130	0.043	0.181	0.403	0.345	0.994
300	5.25E-03	0.049	0.164	0.220	0.981
330	3.87E-03	0.040	0.142	0.207	0.979
430	1.45E-03	0.021	0.089	0.172	0.969
480	9.19E-04	0.016	0.070	0.158	0.964
600	3.32E-04	8.01E-03	0.041	0.132	0.952
720	1.29E-04	4.27E-03	0.025	0.112	0.938
800	7.13E-05	2.83E-03	0.018	0.101	0.929
1000	1.77E-05	1.09E-03	7.92E-03	0.081	0.904
2000	5.60E-08	1.81E-05	1.96E-04	0.034	0.766
3000	5.17E-10	5.73E-07	6.80E-06	0.018	0.624
4000	8.55E-12	2.59E-08	2.88E-07	0.011	0.494
5000	2.08E-13	1.49E-09	1.41E-08	7.00E-03	0.382
6000	6.66E-15	1.03E-10	7.65E-10	4.71E-03	0.289
7000	0	8.18E-12	4.54E-11	3.28E-03	0.214
8000	0	7.28E-13	2.89E-12	2.36E-03	0.156
9000	0	7.13E-14	1.96E-13	1.73E-03	0.112
10000	0	7.55E-15	1.41E-14	1.30E-03	0.079

Table 9. Unconditional probability of not recovering offsite power by time t.

Time (mins.)	Pr(T > t)
60	0.529
75	0.480
110	0.393
130	0.354
300	0.185
330	0.170
430	0.134
480	0.122
600	0.102
720	0.089
800	0.084
1000	0.075
2000	0.057
3000	0.045
4000	0.036
5000	0.027
6000	0.021
7000	0.015
8000	0.011
9000	7.98E-03
10000	5.64E-03

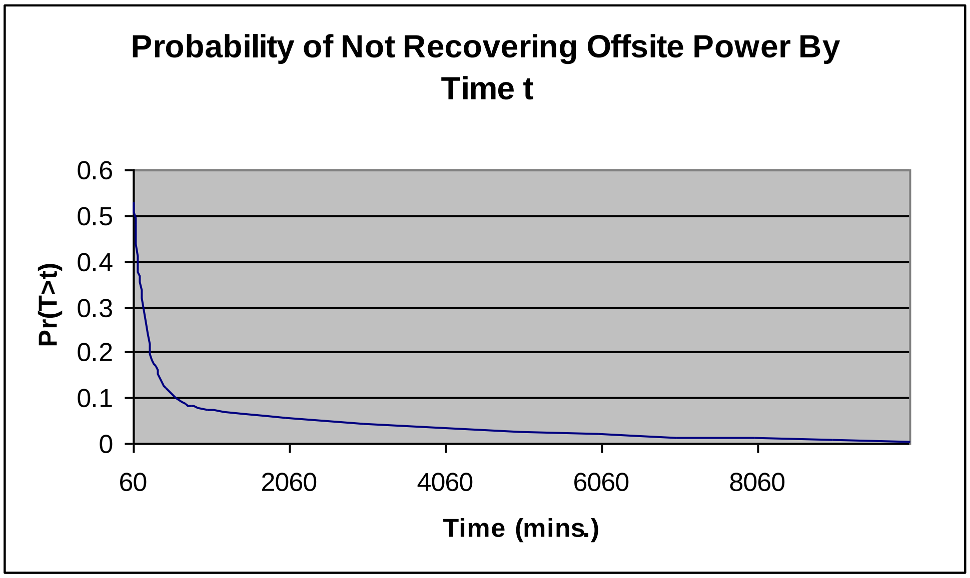

Figure 12. Unconditional probability of not recovering offsite power by time t.

3.7.2 RCP Seal LOCA

As discussed previously, the T-H calculations showed that there could be a difference in the potential for SAI-SGCB for different RCP seal LOCA sizes. Thus, it was necessary to obtain probabilities for each of the RCP seal LOCA ranges modeled in the SAI-SGCB evaluation: 60-150 gpm/p, 150-250 gpm/p, and 250-480 gpm/p. The RCP seal LOCA model used in the WPWR-A PRA was used in this evaluation but required some modification for use in the example plant SAI-SGCB evaluation. The WPWR-A PRA used a probability of 0.094 for 60-250 gpm/p RCP seal LOCA (this range was categorized in the original PRA as a VSLOCA). Subdividing this probability, the associated probabilities for the 60-150 gpm/p and 150-250 gpm/p RCP seal LOCAs used in the example application are 0.052 and 0.043, respectively. The probability of having a seal LOCA of 480 gpm/p is 0.0025 according to the WPWR-A PRA, but this size LOCA precludes an SAI-SGCB in that the primary side will depressurize sufficiently fast to not threaten the tubes (the “high” part of the “high and dry” would not exist). The probability of having a 21 gpm/pump leak is 1.0 minus these values or 0.9.

4. RESULTS OF APPLICATION

This chapter presents the results of the quantification of the SAI-SGCB frequency for the example plant analysis. Since the scope of the evaluation only included SBO sequences, the calculated SAI-SGCB frequency is not complete. Furthermore, the results are approximate since the methodology discussed in Chapter 2 was not completely exercised. The results should only be interpreted as representative for the WPWR-A plant since the PRA models and data do not completely reflect the plant. The results of selected sensitivity studies are also presented and discussed. Although the uncertainty in the results was not quantified, it is qualitatively evaluated.

4.1 Risk Analysis Results

The revised model for the example plant was quantified with all of the modifications discussed in Chapter 3. Only those sequences that could lead to SAI-SGCB and involved SBO were quantified, as this was the limited scope defined in this study. Cut sets were developed and quantified for each of these sequences in accordance with the quantification process described in Section 2.8. The results are shown in Table 10 for sequences down to 1E-9/yr.

Table 10. Dominant accident sequences.

Sequence Designator	SAI-SGCB Frequency
TRASGCB-2DSBO8	4.3E-6/yr
TRASGCB-2SBO8	6.6E-7/yr
VSL60-4LSBO8	2.0E-7/yr
TRASGCB-2DSBO4	7.7E-8/yr
TRASGCB-2LSBO8	5.3E-8/yr
VSL60-7SBO8	3.2E-8/yr
VSL180-7SBO8	1.0E-8/yr
VSL180-4LSBO8	9.0E-9/yr
TRASGCB-2SBO4	6.6E-9/yr
VSL60-4LSBO4	3.9E-9/yr
VSL60-5LSBO8	2.4E-9/yr
TRASGCB-2LSBO4	1.0E-9/yr
Total	5.4E-6/yr

For the purpose of this study, dominant SAI-SGCB sequences are defined as those sequences with frequency estimates greater than 1E-7/yr. There are three such sequences, and they are described in greater detail below.

(1) TRASGCB-2DSBO8N (4.3E-6/yr) – (This is Transient Case 161 in Table 2.) LOSP followed by SBO. SHR is successfully provided by the TDAFW pump. Following procedure, the operators depressurize the secondary side to reduce RCS pressure and inject the accumulators, an action intended to extend the time available before the onset of core damage. In addition, in accordance with procedure, the operators shed

nonessential loads from the DC busses. This extends the battery lifetime to about 8 hours. Efforts to recover off-site power over the ensuing eight hours are unsuccessful, and the batteries begin to fail. With the loss of indications and control, the only way to maintain AFW flow is by taking local manual control of the TDAFW pump. This action is unsuccessful, and AFW is lost. Boil-off begins and reaches the onset of core damage at about 16 hours after the initiation of the accident with the conditions for potential SAI-SGCB being reached about two hours later. Efforts to perform heroic actions to restore safety functions after depletion of the batteries fail. The SG tubes fail prior to other RCS components, and SAI-SGCB occurs.

(2) TRASGCB-2SBO8 (6.6E-7/yr) – (This is Transient Case 69 in Table 2.) LOSP followed by SBO. SHR fails. In accordance with procedure, the operators shed nonessential loads from the DC busses. This extends the battery lifetime to about eight hours. Boil-off begins and reaches the point of the onset of core damage at about two hours after the initiation of the accident with the conditions for potential SAI-SGCB being reached at about four hours. Efforts to recover off-site power within two hours (to prevent core damage) and within four hours (to prevent SAI-SGCB) are unsuccessful. Because the batteries have sufficient power at this point, the operators have the ability to open the PORVs in accordance with the SAMGs. However, the TSC (which has been formed by this time) fails to direct this action in time. The SG tubes fail prior to other RCS components, and SAI-SGCB occurs.

(3) VSL60-4LSBO8 (2.0E-7/yr) – (This is 60 gpm/p VSL Case 161 in Table 2.) LOSP followed by SBO. The RCP seals fail resulting in a leak in each pump sufficient to allow the loss of about 60 gpm at normal operating temperature and pressure. Secondary heat removal is successfully provided by the TDAFW pump. Following procedure, the operators depressurize the secondary side to reduce RCS pressure and inject the accumulators, an action intended to extend the time available before the onset of core damage. In addition, in accordance with procedure, the operators shed nonessential loads from the DC busses. This extends the battery lifetime to about 8 hours. Efforts to recover off-site over the ensuing eight hours are unsuccessful, and the batteries begin to fail. With the loss of indications and control, the only way to maintain AFW flow is by taking local manual control of the TDAFW pump. This action is unsuccessful and AFW is lost. Boil-off begins and reaches the point of the onset of core damage at about 16 hours after the initiation of the accident with the conditions for potential SAI-SGCB being reached about two hours later. Efforts to perform heroic actions to restore safety functions after depletion of the batteries fail. The SG tubes fail prior to other RCS components, and SAI-SGCB occurs.

Note that the first and third sequences occur over a relatively long time period. Their current frequencies are highly dependent on the limited credit given for “extraordinary actions” in these extended periods, including such things as being able to manually control the TDAFW pump after battery depletion, providing some means to extend or replace battery power beyond the 8 hour depletion time, or other such actions to either prevent core damage or get the PORVs open after core damage to relieve RCS pressure and prevent SAI-SGCB. As discussed in the HRA evaluation in Section 3.6, the current values used for the failure of such actions is 0.5, and represent a “generic” plant that has no specific plans or procedures that could reasonably assure success of these actions.

The second sequence is a much shorter term sequence and more representative of the types of early melt sequences that commonly dominate SBO core damage risk. Its frequency is highly dependent on the actions of the TSC in directing the operators to open the PORVs in the short timeframe between core exit temperature exceeding 1200° F and the point at which SAI-SGCB conditions would exist. As discussed in the HRA section, the current values used for the failure of such actions is 0.5, representing a “generic” plant that has little specific guidance beyond the direction provided in the generic SAMGs that the TSC consider this action.

To put these results in perspective, the PRA for the example plant had an overall internal event CDF of nearly 2E-5/yr of which approximately 70% was caused by SBO scenarios. Of this, this study estimates that approximately 35% of the SBO scenarios (or 25% of the total CDF) could result in SAI-SGCB, or about 5E-6/yr.

4.2 Results of Sensitivity Studies

As noted in the discussion of results above, the results are sensitive to a number of assumptions and judgments. Based on the results of the quantification process and comments by reviewers, a series of sensitivity analyses were performed on specific assumptions and values used in the quantification. Each sensitivity was addressed by assuming an alternative value for that specific parameter. Sensitivities were considered one at a time. This section discusses the extent of some of the sensitivities.

4.2.1 Restoring DC Power and Opening PORVs after Battery Depletion

The concept of recovering plant systems and power after the batteries are depleted is treated in the model by a single human action representing the restoration of DC power and opening the PORVs, which will then prevent SGCB. This generically covers all actions that could theoretically be taken, regardless of how difficult they may be. The base case HEP for this action (HFE 3C) is 0.5. In section 2.7, a sensitivity case for a plant that had a specific plan to accomplish this was considered, with a revised HEP of 0.1. Applying this revised HEP reduces the SAI-SGCB probability by 70% relative to the baseline. The resulting SAI-SGCB probabilities for each accident sequence are shown in Table 11 (the ordering of the sequences is left as in the baseline and the values that change are shown in bold).

It is worth noting that since three of the sequences are not affected by this action, the reduction in SGCB frequency that can be achieved is limited. Even if all contribution from this human action is eliminated, the residual risk from SGCB would still be on the order of 7E-7/yr.

4.2.2 Failure Mode of TDAFW after Battery Depletion

The baseline model assumes that the TDAFW fails immediately upon battery depletion unless manual action is taken to control AFW locally. Studies have shown that TDAFW will run uncontrolled until steam generator overfill occurs and then will fail from water intrusion into the pump turbine. If it is assumed that the latter occurs, then the time of AFW loss is later and also the steam generators have more water in them. This will stretch out the time before the steam generators dry out and core damage occurs, leaving more time for recovery action. This change does not alter the dynamics or timing of the SAI-SGCB phenomenology itself. T-H calculations done for this study to consider other timing sensitivities have shown that while the time to onset of core damage can change the time from the onset of core damage to the occurrence of SAI-SGCB (if it occurs) is unaffected. Therefore, this sensitivity will only serve to provide more time for the human action addressed in the previous sensitivity and illustrated by the results in Table 11. The impact of this change on the SAI-SGCB probability is likely to be comparable to the results presented in Table 11; however, without a more detailed analysis, the actual impact cannot be quantified.

Table 11. Results of DC power restoration and opening PORV sensitivity.

Sequence Designator	New SAI-SGCB Frequency	Original SAI-SGCB Frequency
TRASGCB-2DSBO8	**8.3E-7/yr**	4.3E-6/yr
TRASGCB-2SBO8	6.6E-7/yr	6.6E-7/yr
VSL60-4LSBO8	**3.3E-8/yr**	2.0E-7/yr
TRASGCB-2DSBO4	**1.4E-8/yr**	7.7E-8/yr
TRASGCB-2LSBO8	**8.6E-9/yr**	5.3E-8/yr
VSL60-7SBO8	3.2E-8/yr	3.2E-8/yr
VSL180-7SBO8	1.0E-8/yr	1.0E-8/yr
VSL180-4LSBO8	**1.6E-9/yr**	9.0E-9/yr
TRASGCB-2SBO4	**1.0E-9/yr**	6.6E-9/yr
VSL60-4LSBO4	**small**	3.9E-9/yr
VSL60-5LSBO8	**small**	2.4E-9/yr
TRASGCB-2LSBO4	**small**	1.0E-9/yr
Total	**1.6E-6/yr**	5.4E-6/yr

4.2.3 Leakage from Isolated Steam Generators

The baseline assumes that the various isolation valves on the secondary side of the steam generators (e.g., MSIV, blow down isolation valve, feedwater isolation valve (FWIV)) do not form a perfect seal (since they are not a primary system boundary and are neither qualified not tested for such a seal) and thus allow some leakage into the secondary side systems. The leakage assumed in the T-H analysis is the equivalent of a 0.5 in^2 leak from each steam generator. This leak was large enough to depressurize a dried-out steam generator prior to the onset of SGCB conditions. In contrast, T-H calculations performed for a 0.1 in^2 leak indicated that this leak size was not sufficient to fully depressurize the steam generator. Interpolating between these two limits suggests that an equivalent leak on the order of a 0.2 in^2 to 0.3 in^2 size is sufficient.

Therefore, to perform a sensitivity analysis on this assumption it is necessary to set a condition that ***no*** steam generator has a leak past the isolation valves that exceeds about 0.2 in^2 equivalent size (i.e., all four steam generators are sealed to below this value). In that case, a steam generator relief valve would have to fail to fully close (again, to less than 0.2 in^2 leakage) in order to depressurize a steam generator. This could occur on any of the series of lifts that would take place during the steam generator boil-off process. Two sensitivities are considered:

Sensitivity 1: Secondary side PORV failure to fully close with a probability of 0.05 per steam generator. Failure of any one out of four to fully close is 0.2. This affects each sequence equally and is a direct multiplier. The revised total SAI-SGCB frequency is 5.4E-6 * 0.2 = 1.1E-6/yr, a reduction of 80%.

Sensitivity 2: Secondary side PORV failure to fully close with a probability of 0.01 per steam generator. Failure of any one out of four to fully close is 0.04. This affects each sequence equally and is a direct multiplier. Revised total SAI-SGCB frequency is 5.4E-6 * 0.04 = 2.2E-7/yr, a reduction of 96%. A summary of the results of this sensitivity is shown below.

Original SAI-SGCB Frequency	Sensitivity 1 SAI-SGCB Frequency	Sensitivity 2 SAI-SGCB Frequency
5.4E-6/yr	1.1E-6/yr	2.2E-7/yr

Note that the relationship between the probability of leakage and the frequency of SAI-SGCB is linear, so that any sensitivity result can be obtained by multiplying the probability of a leak of sufficient size times the number of steam generators times the original SAI-SGCB frequency.

4.2.4 Manual Local Control of TDAFW after Battery Depletion

Taking local manual control of the TDAFW after battery depletion will allow TDAFW to continue running during the core damage progression, and thus prevent SAI-SGCB. The baseline value for failure to do this is 0.5. A sensitivity analysis has been conducted at a value of 0.1. As shown in Table 12, this change results in a 70% reduction in the SAI-SGCB probability relative to the baseline.

It is worth noting that, since three of the sequences are not affected by this action, there is a limit to the achievable reduction in SAI-SGCB frequency that can be achieved. Even if all contribution from this human action is eliminated, the residual risk from SAI-SGCB would still be on the order of 7E-7/yr.

Table 12. Results of TDAFW manual local control sensitivity.

Sequence Designator	New SAI-SGCB Frequency	Original SAI-SGCB Frequency
TRASGCB-2DSBO8	**8.3E-7/yr**	4.3E-6/yr
TRASGCB-2SBO8	6.6E-7/yr	6.6E-7/yr
VSL60-4LSBO8	**3.3E-8/yr**	2.0E-7/yr
TRASGCB-2DSBO4	**1.4E-8/yr**	7.7E-8/yr
TRASGCB-2LSBO8	**8.6E-9/yr**	5.3E-8/yr
VSL60-7SBO8	3.2E-8/yr	3.2E-8/yr
VSL180-7SBO8	1.0E-8/yr	1.0E-8/yr
VSL180-4LSBO8	**1.6E-9/yr**	9.0E-9/yr
TRASGCB-2SBO4	**1.0E-9/yr**	6.6E-9/yr
VSL60-4LSBO4	**small**	3.9E-9/yr
VSL60-5LSBO8	**small**	2.4E-9/yr
TRASGCB-2LSBO4	**small**	1.0E-9/yr
Total	**1.6E-6/yr**	5.4E-6/yr

4.2.5 Open PORVs after Core Damage Begins

This is an action that the operator would perform as directed by the TSC. Note that this action does not apply to cases where DC power needs to be recovered (see first sensitivity) since, as discussed in the HRA, the conditional probability of failing to open the PORVs after working to recover DC power for that purpose is negligible. The current baseline value for HFE 4D is 0.5. A sensitivity analysis has been performed for a value of 0.1.

As shown in Table 13, this change results in an 11% reduction in the SAI-SGCB probability relative to the baseline. It is worth noting that since most of the sequences are not affected by this sensitivity, including the most dominant, the residual risk cannot be significantly reduced further by improving this action.

Table 13. Results of sensitivity for opening PORVs after core damage begins.

Sequence Designator	New SAI-SGCB Frequency	Original SAI-SGCB Frequency
TRASGCB-2DSBO8	4.3E-6/yr	4.3E-6/yr
TRASGCB-2SBO8	**1.1E-7/yr**	6.6E-7/yr
VSL60-4LSBO8	2.0E-7/yr	2.0E-7/yr
TRASGCB-2DSBO4	7.7E-8/yr	7.7E-8/yr
TRASGCB-2LSBO8	5.3E-8/yr	5.3E-8/yr
VSL60-7SBO8	**4.8E-9/yr**	3.2E-8/yr
VSL180-7SBO8	**1.7E-9/yr**	1.0E-8/yr
VSL180-4LSBO8	9.0E-9/yr	9.0E-9/yr
TRASGCB-2SBO4	6.6E-9/yr	6.6E-9/yr
VSL60-4LSBO4	3.9E-9/yr	3.9E-9/yr
VSL60-5LSBO8	2.4E-9/yr	2.4E-9/yr
TRASGCB-2LSBO4	1.0E-9/yr	1.0E-9/yr
Total	**4.8E-6/yr**	5.4E-6/yr

4.3 Uncertainty Analysis

As discussed in Chapter 2, the consideration of uncertainties is an important part of the methodology to promote a full understanding of the results. However, due to resource limitations, no quantitative assessment of accident sequence uncertainties was performed.[7]

Instead, the uncertainties were examined from the systems analysis perspective and assessed qualitatively. Only the systems analysis perspective is considered because the phenomenological parts of the work (the T-H and the RCS component response investigations) have no uncertainty analyses to incorporate into an integrated analysis. Hence, the importance of variables for two-thirds of the overall project, which contribute the most to the uncertainty can not be assessed quantitatively. However, some qualitative insights can be identified.

[7] Uncertainties in model parameters were considered in determining conditional SAI-SGCB probability, but only the mean of the resulting distribution was used in the accident sequence quantification. All basic events in the accident sequence model were represented by a single value (the mean) and no uncertainty distributions were used in the quantification of the sequences.

The primary insight is that the SAI-SGCB frequency calculated in the current study is likely to bound the true value. In other words, consideration of uncertainties will likely decrease the calculated SAI-SGCB frequency. This insight ensues from systems, operator action, and phenomenological considerations. First, from a system perspective (but linked to a phenomenological result), it is currently assumed that the secondary side of at least one steam generator has a leak large enough to depressurize it by the time the pressure and temperature spike occurs. The T-H analysis shows that a leak somewhere between 0.1 in^2 and 0.5 in^2 is sufficient to depressurize the steam generator secondary (see the Appendix A). In this context, the leak can either be an external leak (e.g., a leak through a valve stem, flange, or ADV seat) or an internal leak (e.g., a leak through an isolation valve that allows steam to exit the steam generators into the main steam line or main feed water piping). Either of these two types of leaks will depressurize the steam generators. Based on discussions among knowledgeable engineers both within and outside the analysis team, it was determined that this is a very small leak when compared to the total area available for leakage to occur and that it would be extremely difficult to get a good enough seal on every potential pathway to keep the steam generators pressurized. However, it is very important to note that the results are highly sensitive to this assumption, and that the SAI-SGCB frequency would drop directly with the probability that leakage would occur. For example, if there was not secondary leakage other than the possibility of a stuck-open ADV, then the SAI-SGCB frequency could drop more than an order of magnitude. There is no way to make the probability of sufficient secondary side leakage go higher. Hence it is prudent to understand what phenomenological uncertainties might change this result.

Another reason the above uncertainty statement can be made is the nature of the final results; namely, that 80% of the answer comes from one sequence. This sequence involves a number of things succeeding, at least initially. For example, (1) the TDAFW pump initially starts and runs and does not fail until the batteries deplete (2) the operators successfully extend battery depletion time and, (3) initially the operators successfully depressurize the secondary. All three of these successes have close to probability of 1.0 of happening. If there is an order of magnitude of uncertainty in these values it could decrease the dominant sequence by about 30%, but also might increase some now non-dominant sequences by one or two orders of magnitude so that they would then be in the 1 E-6/yr frequency range. Overall, however, the result would change little.

In the dominant sequence, there is a 0.5 probability of the operators performing some “heroic” (non-procedural) action over the 18 hour duration of the accident to stop its progression. Certainly, this value has the potential to go down more than it could go up. It is also possible to vary several of the variables concurrently, but logically they cannot go higher overall by much. Therefore, it is likely that a more detailed analysis of these “heroic” actions will lower the SAI-SGCB frequency.

The preceding are relatively minor uncertainties compared to what is common in most PRAs. In contrast, the uncertainties related to the modeling of SAI-SGCB phenomena are substantial. Previous analyses have shown that accident calculations performed using different T-H computer codes, such as SCDAP/RELAP or MAAP, can produce very different results even when inputs are controlled to simulate identical accident conditions. In addition, for a given computer code, substantial variability in the calculated results can occur due to uncertainty in code inputs. Given that the calculated SAI-SGCB probability for many sequences is already very high (0.4 in many cases), it is likely that consideration of phenomenological uncertainties will decrease the calculated SAI-SGCB probability.

4.4 Impact of Limited Risk Assessment Methodology Application

The example application of the SAI-SGCB methodology did not identify the need for any modifications to the methodology outlined in Chapter 2. Although the example application presented in this report did not completely apply the recommended methodology, it demonstrated that the methodology outlined in Chapter 2 can provide credible estimates of SAI-SGCB. However, the areas of the methodology that were not applied are significant to the calculated results. Specifically:

- The HRA methodology requires interaction with plant operations and management staff in order to properly assess the HEPs. At least one and, ideally, two visits to the plant are required by the methodology to provide the necessary interactions. This is especially true in this study, where the procedures for actions taken after the onset of core damage are not as prescriptive and clear as for other EOPs. Without such interactions, the HRA herein cannot (and thus does not) meet the requirements of the ASME PRA Standard and the ATHEANA methodology and so any conclusions of the example plant analysis would likely not be accepted in a risk-informed application.

- There are known deficiencies in the T-H analysis in that issues were identified that required the generation of a new base case. While this new base case was generated, none of the other cases required for the other SGCB sequences were run. Therefore, the SAI-SGCB probabilities in the study are based on older runs with known errors. In addition, significant modeling differences between the SCDAP/RELAP code and the industry-sponsored MAAP code have been identified, but not fully resolved. These differences are likely to result in substantial differences in the calculated SAI-SGCB probability. Accurate T-H analysis and treatment of related phenomenological uncertainties are clear deficiencies in the example analysis results.

- During the course of the example plant analysis, it was determined that the assessment of the RCS pressure boundary failure was not adequate to properly implement the methodology. Work was underway to provide better models of hot leg and surge line failure and to address areas of importance identified in the PRA integration that could provide other paths for RCS depressurization, in particular the nozzle safe ends and the RCP seals. This work was terminated before the results were determined, and so it was necessary to use the simplified RELAP/SCDAP failure models. It is unknown how the current results would be affected by applying an appropriate ME methodology. Uncertainties in this area also were not rigorously assessed.

In summary, the example plant application supports the validity of the methodology. However, programmatic decisions to not implement key aspects of the methodology have a significant impact on the results, and as such renders any conclusions regarding the risk significance of SAI-SGCB based on these example results questionable.

5. SUMMARY AND CONCLUSIONS

Steam generator tubes constitute a substantial fraction of the RCS pressure boundary in a PWR. SGTR is important to consider in nuclear plant risk assessments because radionuclides released from the primary system through the ruptured tube(s) could escape to the environment through openings in the secondary system and bypass the containment building. Most previous risk assessments have addressed SGTRs that occur during normal operation or during an accident, but prior to core damage. Very few risk assessments have considered SGTRs that occur after core damage. This is due to a limited understanding of the phenomena that govern SAI-SGTRs that result in containment bypass (denoted as SAI-SGCB).

The previously utilized approach for SGTR risk assessment links the results of accident frequency analyses (NUREG-1150) with accident progression analyses (NUREG-1570). This approach appears to be applicable today. Several shortcomings, however, have been identified with this approach that could be addressed. These include the use of expert opinion in the absence of phenomenological analyses and the need for improved analysis of human reliability, particularly in operator responses during severe accidents. Since NUREG-1150 was issued in 1990, research has resulted in increased understanding of the T-H phenomena associated with severe accidents. In addition, considerable progress has been made in HRA. ATHEANA is a second-generation HRA method that addresses shortcomings in the former approaches. Concern over the importance of SAI-SGTRs resulted in a SGAP, which identified the need for improved modeling methods of SG tube integrity. The improved method was to integrate analyses from three disciplines: PRA for assessing risk significance, T-H for addressing the fluid conditions within the primary system, and ME for determining the integrity of the SG tubes and other primary system components.

This report documents the effort to establish an improved approach for analyzing the potential for SAI-SGCB. The overall objective of the documented work was to develop improved methods to identify severe accident scenarios resulting in SGTRs and containment bypass. In addition, the project included development of an improved PRA/HRA method and tools for modeling these scenarios (i.e., determining their expected frequency), including the effects of operator actions, uncertainties, and differences in plant design. The developed methodology provides a framework for integrating the results of the PRA with those from supporting T-H and ME analyses. The ME analyses of the SG tube integrity and the materials response of other RCS components reveal the pressure and temperature regimes of interest. The T-H analyses then determines how to get to those regimes. The PRA examines system and component failures that would put the reactor system in those conditions and identifies the operator recovery actions that can mitigate the accident progression. The integrated methodology then, provides a framework that logically combines the results from all of these elements, including uncertainties, to provide the risk perspective for the SAI-SGCB issue.

The improved methodology was demonstrated by applying it to a hypothetical plant. Unfortunately, the application could not exercise all of the recommended methods or evaluate the importance of SAI-SGTRs to all types of PWRs. Specifically, the application was limited to one plant type – a Westinghouse four loop PWR. Furthermore, the analysis did not examine all accident sequences that could contribute to SAI-SGCB – only the contribution from SBO sequences was evaluated. Since the analysis was applied to a hypothetical plant it did not include a detailed plant-specific HRA – the HRA was based on generic emergency operating procedures and severe accident management guidelines. Although detailed T-H analyses were used to support the evaluation, the materials engineering support was limited. Specifically, analysis of all potential RCS boundary failure modes was not provided. Finally, there was no detailed uncertainty analysis particularly in the T-H and ME arenas. Although limited and incomplete, this application of the recommended methodology did not identify any needed modifications. Moreover, it demonstrated the methodology provides credible results and could be more certain if the entire methodology was utilized.

The PRA for the example plant used in the methodology demonstration had an overall internal event CDF of nearly 2E-5/yr of which approximately 70% was caused by SBO scenarios. The SAI-SGCB methodology demonstration for this plant indicated that approximately 35% of the SBO scenarios (or about 5E-6/yr) could result in SAI-SGCB. These values should only be viewed as representative for the example plant and are likely to be bounding. Three SBO sequences contribute to this containment bypass frequency, and two of them, including one that is 80% of the total, are very long sequences (18 hours until the SG tubes fail). In these two sequences, systems such as the TDAFW system initially work but fail, or at least become uncontrollable, when the batteries fail. The operators have done their jobs correctly, but they run out of responses. During this time operators would continue to respond to prevent or mitigate the accident progression, but without specific procedural guidance, it was not possible to provide a robust quantification for “heroic” measures (such as assembling an alternate DC power source from available batteries). In addition, T-H calculations performed for the study indicate that very small leakages in the secondary side can in fact depressurize it over the time frame of interest to help lead to the high differential pressure conditions across the tubes needed for SAI-SGCB; no additional system failures are necessary. The occurrence of secondary side leakage is thus an important factor in determining the SAI-SGCB frequency.

An integrated uncertainty analysis was not performed. However, some insights about the uncertainty can be addressed that are fairly robust. First, the frequency of SAI-SGCB cannot go much higher than was calculated. Some of the parameters contributing to uncertainty are close to unity: the systems originally succeed, the operators perform as required, and the T-H conditions produce tube failures before other RCS component failures about 35% of the time. Thus, there is little “upward pressure” on the risk results from the uncertainty in these parameters. The factors that could decrease parameter values cannot decrease them much, with several exceptions identified in the report. The most important factor is the current understanding that the secondary side needs no component failure to depressurize. From the standpoint of phenomenological uncertainty, it may be that only small, and perhaps likely, changes in the inputs to the models or the models themselves could reduce significantly the containment bypass frequency due to SAI-SGTRs.

6. REFERENCES

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APPENDIX A. SECONDARY SIDE LEAKAGE

Since the prevention of SAI-SGCB is extremely sensitive to maintaining pressure on the secondary side of each steam generator, the question of how much secondary side leakage could be tolerated was the subject of intense analysis early in the project. Originally, the base case risk assessment models included the sticking open of an ADV based on the assumption that this was the only way that the secondary could be depressurized sufficiently enough to result in a significant probability of SAI-SGCB. Because the probability of having an ADV sticking open is on the order of 0.01 to 0.1, the frequency of SAI-SGCB would be reduced by one to two orders of magnitude. The possibility was raised that, with the steam generators dried out and with no inventory makeup being provided, the amount of leakage that would depressurize the steam generator secondary might be small. Hence, the issue of secondary side leakage was further investigated.

The T-H analysts performed sensitivity analyses of a SBO sequence with secondary leak areas of 0.1, 0.5, and 1 in^2 to determine the approximate leak area necessary to depressurize the steam generator secondary after the termination of auxiliary feedwater. The location of the leak was not specified; it was just assumed that there was no back pressure against the leak. These analyses were reviewed to determine whether the secondary side was depressurized during the severe accident-induced temperature increase, conditions under which the potential for SAI-SGCB would be relatively high. The findings from the T-H analyses were as follows:

- Secondary leak areas of 0.5 and 1.0 in^2 result in essentially full depressurization of the steam generator by the time the severe accident-induced temperature ramp occurs. Therefore, leaks of this size will have essentially the same SAI-SGCB probability as for a stuck-open ADV.

- A leak area of 0.1 in^2 results in depressurization to approximately 500 psia in the secondary at the time of the temperature ramp, which will delay the tube failure significantly relative to the RCS boundary failure. Leaks of this size will have essentially the same SGCB probability as the case of no leakage.

Based on these results, it was estimated that a leakage area between 0.2 and 0.3 in^2 would depressurize the steam generator secondary sufficiently to encourage the potential for SAI-SGCB. This area corresponds to an equivalent diameter of approximately 0.5 to 0.6 inches in any one steam generator.

To further address the issue and confirm the results, a set of iterative calculations were conducted to determine an estimate of the minimum leak area that would result in reaching 500 psi and atmospheric pressure at the time of severe accident-induced temperature ramp up. The results of the calculations were as follows:

- To reach atmospheric pressure, a leakage flow area of approximately 0.35 in^2 (an equivalent leak area of 0.0025 ft^2) is required. With this leakage flow area, the calculation indicated a steam generator secondary pressure of 22.3 psia at the time of pressurizer surge line failure.

- To reach 500 psia, a leakage flow area of approximately 0.1 in^2 (an equivalent leak area of 0.0007 ft^2) is required. With this leakage flow area, the calculation indicated a steam generator secondary pressure of 494.2 psia at the time of pressurizer surge line failure.

Somewhere between these secondary pressures and associated leak areas is the point at which the SAI-SGCB probability will rapidly increase.

Given the very small leak size that could be tolerated, consideration was given to where these leaks could occur. There are two primary leak types:

- ***Leaks directly to atmosphere***. Given closure of the MSIVs, FWIVs, and steam generator blow down valves, such leaks would need to be in the stems or seals of these valves; the stems or seals of other valves or ports upstream of these valves; or the stems, seals, or seats of the secondary side PORVs or SRVs. Such leaks would be present during normal operation. Another potential leakage source could occur if a secondary side PORV or SRV re-closes, but does not re-close completely (e.g., allows a small amount of leakage).

  Industry personnel who were familiar with plant enthalpy balances were contacted by the analysis team, and their response was that a leak this small would likely not even show up in the balance. Even if such a small leak were detected, the effect on power production would be so small that it could cost more to shut down to repair than it would be to continue to operate with the leak until it could be fixed as part of another maintenance outage. Furthermore, there is no *safety* reason for a reactor to shut down if there is steam leakage on the secondary side (unless it is from a degraded pipe segment, which is not considered here).

- ***Leaks into the secondary piping***. Perhaps more significant is the potential for leakage past the isolation valves into the downstream piping in the secondary system. The long, large runs of piping have a significant volume and so could accept small leakage rates without themselves pressurizing to provide any backpressure. The amount of leakage past the valve seats would be very small relative to the total size of the valve. A 20" diameter MSIV would have a total flow area of over 300 in^2. Therefore, an MSIV that is 99.9% closed will still not be sufficient to maintain secondary pressure. Steam generator isolation valves are not part of the containment boundary, and so are not required to meet containment isolation leak rate requirements. The performance requirements for these valves are established based on maintaining pressure in the steam generators when they are full, and so they are not required (nor are they designed, qualified, or tested) for this kind of leak tightness.

The analysis team did attempt to research the issue of secondary side leakage, including contacting various people within NRC and also individuals at EPRI and some utilities. The summary of these contacts is as follows:

- There is no useful data for leakage rates for PWR MSIVs. Tests to determine leakage rates for these valves are not required and are not done.

- People with operational experience were universal in the belief that boiling water reactor (BWR) isolation valve data could not be used to represent PWRs, and that it was extremely unlikely that the integrity and leak-tightness of the PWR valves would approach that of the BWR valves.

- People with operational experience are not able to quantify their intuition, but they told the analysis team that they don't think it would be possible to maintain pressure in an isolated, dry steam generator. One individual stated specifically that he thinks it would depressurize through the ADVs, and perhaps through the MSIVs.

- One utility did conduct an “inadvertent” test of the isolation leak tightness on one steam generator. They were looking for a hole in a SG tube. They intended to pressurize the steam generator to several hundred psi with nitrogen, which should have been possible with a tube leak. However, they were not able to do so as the pressure stopped rising at about 60 psi despite the continued addition of nitrogen. They found that there was a leak in an isolation valve. The leak was determined to be on the order of 0.2 in^2.

- Small amounts of steam leakage are common in nuclear power plants and generally in any industrial facility with a considerable amount of steam piping. A small amount of steam results in a lot of condensed water vapor that can be observed when walking around in a plant. No one on the team could remember being to a plant (nuclear or otherwise) and not observing this.

- Failure rate data for valve leaks are focused on leaks large enough to fail the valves’ safety function (e.g., divert significant amounts of flow, allow extensive backflow, etc.). These data are not valid for the very small leak rates at issue here.

The analysis team considered building a fault tree for all the different ways in which a leak could occur in the secondary side. No PRA, however, has modeled the secondary side of a PWR at this level of detail, including the PRA for the example plant used in this study. Further, given the lack of data on leak rates or even the occurrence of leaks (since these are entirely too small to be reportable), there is no way to assign probabilities of leaks in this range for each potential leak path, or even to determine split fractions for the equivalent leak size being above or below the critical size. In addition, the issue is the total leakage in any one steam generator, not the leakage from any specific source. Instead of attempting to build such a model the team used engineering judgment (much akin to the approach used in the HRA portion of the example analysis), to determine the probability of secondary leakage in at least one steam generator.

It was the consensus judgment of the team, made in accordance with an appropriate collection of available information and an appropriate expert elicitation process conducted within the team, that the weight of the available evidence and opinion leads to the conclusion that it is a near certainty that the sum of all leakage will exceed the critical flow area for depressurization in at least one steam generator. This consensus was reached after due consideration of all of the input received by the team on this issue.

Based on this consensus, it was decided that the base case for the risk quantification would be represented by a 0.5 in^2 leakage since a number of T-H runs were already done using that value and since the specific leak size does not matter as long as it is large enough to depressurize the steam generator. This was adequately demonstrated by reviewing and comparing the equivalent T-H runs for 0.35, 0.5 and 1.0 in^2 and the stuck open secondary side PORV. It was further decided to perform sensitivity analysis on this base case by considering other probabilities for secondary side leakage.

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APPENDIX B. DETAILED DESCRIPTION OF QUANTIFICATION OF HUMAN FACTOR EVENTS

This appendix presents detailed descriptions of the HFEs modeled in the example plant analysis describe in chapter 3 of this report. The HFEs considered are sorted into four categories. Sections B.1 and B.2 describe operator failures to depressurize the steam generators. Sections B.3 and B.4 address operator failures to shed all large nonessential DC loads. Section B.5 discusses operator failures to provide DC power to allow opening of the pressurizer PORVs. Section B.6 describes operator failure to open PORVs to depressurize the RCS. Note that the numeric and alphabetic designations for the HFEs below correspond to those in Table 6 in the main body of the report.

B.1 Event 1A - Operator Failure to Depressurize the Steam Generators When TDAFW is Running

In this scenario, an SBO has occurred, but TDAFW is still available. In ECA 0.0 (SBO procedure), Step 16 directs the crew to depressurize the steam generators if/when a specified narrow range level is reached. This is an important event in the SAI-SGTR scenario. Depressurizing the steam generators will get rid of a lot of heat at once, which would cool down the RCS and allow significant additional recovery time to restore offsite power or for recovering the diesels, etc. In other words, depressurizing the steam generators while TDAFW is available will extend the scenario time, lengthening the time to core damage and changing the overall impact of the event (reducing CDF and a subsequent rupture of SG tube(s) leading to containment bypass event) since more recovery potential subsequently exists. Depressurizing the steam generators should also allow the accumulators to dump, which would also contribute to extending the scenario time.

General Context

As noted above, in this scenario, a SBO has occurred, but TDAFW is still available. In ECA 0.0 (SBO procedure), Step 16 directs the crew to depressurize the steam generators if/when a specified narrow range level is reached.

There will be many indications of the presence of a SBO including no voltage on the busses, failed diesel generators, and only auxiliary power in the control room and the plant (e.g., only back-up lighting available). The crew would enter the SBO procedure from Step 3 of E-0 and begin working through the steps in the procedure. Given the SBO, the operating crew would be carefully monitoring the TDAFW system since it is the only available cooling for the vessel (obviously a very important function) until power is restored or alternate sources of power are identified. It was estimated that it would take the crew approximately 10 to 15 minutes to reach Step 14 in the SBO procedure, which asks them to strip non-essential DC loads to maintain the batteries as long as possible. It was expected that the crew would reach Step 16, which calls for depressurizing the steam generators, after no more than another 5 to 10 minutes.

In discussions among the analysts participating in the expert opinion elicitation process, it was agreed that this action would be very non-controversial. The procedure is clear, operating crews receive regular training on SBO scenarios, there is no obvious reason why they would not want to take the action (i.e., no obvious tradeoffs), and the action would provide significant immediate cooling to the core. In addition, even though depressurizing the steam generators might create minor instabilities in the system, there is little reason to expect a loss or problems with the system due to conducting the action.

A number of issues were noted/discussed before estimates of the HEP were obtained from the subject matter experts. They included the following:

- The procedure is essentially written assuming TDAFW will be running, which is the case for this scenario.
- The action is clearly indicated by procedure – it takes up two pages with cautions relevant to control, etc. While the procedure warns against depressurizing the steam generators so much that the nitrogen accumulators dump, it was decided that there was nothing special about this caution that would create any problems for operating crews. Their only concern would be to avoid depressurizing so low that nitrogen injection from accumulators would occur, which is not difficult to prevent from occurring.
- Normal timing for the action would be acceptable. A regular pace of working through the procedure would fall within the expected time frame.
- No reasons are identified for why the crew would be hesitant to conduct the action and it would be relatively simple to execute? No adverse effects would be expected.
- There are multiple redundant indicators of steam generator level and it was agreed that there was little likelihood that there would be enough failures of instrumentation to confuse the crew.
- It was decided that potential dependencies between this action (Step 16) and Step 14 would be expected. That is, if they fail to strip non-essential DC loads (Step 14), then it would be hard to give them credit for completing this action. Something must be wrong.

Estimation of HEP (HFE 1A in Table 6)

The three analysts making a judgment on the HEP for the operating crew failing to depressurize the steam generators in Step 16 of ECA 0.0 all agreed on the first polling of estimates. **Each analyst proposed an HEP of 0.01.** All agreed on the basic reasons for their choice of the HEP, as described above. They all agreed that the value is conservative, but reasonable, given that a visit to an actual plant was not possible at the time. Note that if shedding non-essential DC loads has failed, it is assumed that this event will also fail (complete dependence)

B.2 Event 1B - Operator Failure to Depressurize the Steam Generators When TDAFW is Running (One Diesel Generator Starts and Runs, but Subsequently Stops)

This event is obviously similar to Event 1A above, but in this case the SBO is delayed for a short time (e.g., 15 to 30 minutes). One issue concerns how the crew would be led into the SBO procedure. The pathway into the SBO procedure might not be crystal clear in this type of scenario. After completing much of E-0, the crew would circulate back into Step 19 of E-0, but this would not directly take them to the step that would lead them the SBO procedure. However, the presence of the SBO would be clear and it was thought that entry into the SBO procedure would likely occur even if there is no explicitly direct transition step from E-0 to the SBO procedure in this case.

Again, it was agreed that the need to depressurize the steam generators would be clear to the crew, particularly if they had successfully stripped all non-essential DC loads in Step 14 (which is assumed to be the case for this HEP). If shedding non-essential DC loads has failed, it is assumed that this event will also fail (complete dependence).

Estimation of HEP (HFE 1B in Table 6)

The three analysts making a judgment on the HEP for the operating crew failing to depressurize the steam generators in Step 16 of ECA 0.0 in a delayed SBO scenario all agreed on the first polling of estimates. **Each analyst proposed an HEP of 0.01.** However, it should be remembered that this action is assumed to fail if the crew fails to shed non-essential DC loads.

B.3 Event 2A - Operator Failure HFE to Shed All Large Nonessential DC Loads (Immediate SBO Scenario)

In this scenario, a SBO has occurred, but TDAFW is still available. In Step 14 of ECA 0.0 (SBO procedure), operators are directed to shed all large non-essential DC loads. The goal is to extend DC battery life until power can be restored. This is an important event in the SAI-SGCB scenario because performing the action would extend the scenario time (lengthening the time to core damage), extend the time to allow relevant recoveries, and change the overall impact of the event (reducing CDF).

General Context

As noted above, in this scenario, a SBO has occurred, but TDAFW is still available. There will be many indications of the presence of a SBO, including no voltage on the busses, failed diesel generators, and only auxiliary power in the control room and the plant (e.g., only back-up lighting). The crew would enter the SBO procedure through E-0 (Step 3 of E-0) and begin working through the steps in the procedure. Given the SBO, in addition to carefully monitoring the TDAFW system (since it is the only available cooling for the vessel until power is restored), the crew would also be very concerned about recovering power and keeping the core protected for as long as possible. To this end, they would want to extend DC battery life as long as possible. It was estimated that it would take the crew approximately 10 to 15 minutes to reach Step 14 in the SBO procedure, which directs them to strip non-essential DC loads to maintain the batteries as long as possible.

In discussions among the analysts participating in the expert opinion elicitation process, it was agreed that this action would be very non-controversial. The procedure is clear, operating crews receive regular training on SBO scenarios, there is no obvious reason why they would not want to take the action (i.e., no obvious tradeoffs), and the action would provide them with significant additional time to restore power and prevent core damage. In such scenarios, the crew is probably focusing on riding-out the SBO until power can be restored, so it would seem that on this basis alone, the action would be a high priority.

A number of issues were noted/discussed before estimates of the HEP were obtained from the subject matter experts. They included the following:

- The action is clearly indicated by procedure and each individual necessary action is specified in the plant specific procedures. Operators train regularly on the relevant actions (e.g., once per year) and would practice SBO scenarios in the simulator. It was agreed that crew members would be knowledgeable about how to perform the actions, even though it might be the first time the crew has been in an actual SBO.

- Given the existence of the SBO, it was agreed that the crews would be motivated to execute the action and that they would want to start conserving DC power as soon as possible. It was thought that there would be little interest in delaying in hopes of getting AC power back. This is not an action that would cause problems for the plant, while not doing it might.

- It was agreed that the number (probably 5 or 6 major actions) and the nature of actions involved would not have an impact on the decision to respond. It was thought that many of the actions could be accomplished from the control room and that any required ex-control room action should not create problems.

- Although there would be some potential for loss of instrumentation, it was not thought that it would be significant enough to create confusion. As discussed above, the SBO would be obvious and the need for the action clear.

- It was thought that staffing would not generally be a problem, but some random influences could effect staffing negatively on some occasions, e.g., unexpectedly short-staffed in middle-of-the-night, event occurs during a shift change etc.

- Nuisance alarms might distract the crew somewhat, but not enough to significantly delay this important action.

- Steps 10 to 13 of ECA 0.0 direct the crew to deal with (isolate) any potentially faulted or ruptured steam generators etc. While this could slow the crews down somewhat, it was not expected to significantly delay them from reaching Step 14 in a timely manner. In addition, a faulted steam generator is unlikely to occur more than approximately 1 time in 100.

- Normal timing for the action would be acceptable. A regular pace of working through the procedure would fall within the expected time frame.

Estimation of HEP (HFE 2A in Table 6)

The three analysts making a judgment on the HEP for the operating crew failing to shed all non-essential DC loads in Step 14 of ECA 0.0 all agreed on the first polling of estimates. **Each analyst proposed an HEP of 0.01.** All agreed on the basic reasons for their choice of the HEP, as described above. They all agreed that the value is conservative, but reasonable, given that a visit to an actual plant was not possible at the time.

B.4 Event 2B - Operator Failure HFE to Shed All Large Nonessential DC Loads (One Diesel Generator Starts and Runs, but Subsequently Stops)

This event is obviously similar to Event 2A above, but in this case the SBO is delayed for a short time (e.g., 15 to 30 minutes). One issue concerns how the crew would be led into the SBO procedure. The pathway into the SBO procedure might not be crystal clear in this type of scenario. After completing much of E-0, the crew would circulate back into Step 19 of E-0, but this would not directly take them to the step that would lead them the SBO procedure (Step 3of ECA 0.0. However, the presence of the SBO would be clear and it was thought that the entry into the SBO procedure would likely occur even if there is no explicitly direct transition step from E-0 to the SBO procedure in this case.

Again, it was agreed the need to strip DC loads would be very apparent given the crew's training, etc.

Estimation of HEP (HFE 2B in Table 6)

The three analysts making a judgment on the HEP for the operating crew failing to shed all non-essential DC loads in Step 14 of ECA 0.0 (given a delayed SBO) did not all agree on the estimated HEP on the first polling. The three analysts initially offered the following HEPs:
Analyst 1 - 0.05, Analyst 2- 0.1, and Analyst 3 - 0.01. After additional discussion of the basis for the various HEPs, two of the three analysts revised their initially suggested HEPs with the following HEPs: Analyst 1 - 0.03, Analyst 2- 0.01, and Analyst 3 - 0.01.

The main reason for the changes was the eventual agreement that the action should be very obvious to crews in such scenarios because of their training, their basic understanding of the situation they are in, and the clear need to be in the SBO procedure in which the action is clearly specified. Also, given that we were assuming that the crew would be 15 to 30 minutes into the scenario before the diesel stopped, it is likely that additional help would be available to assist in the decision process within the time for the action. Analyst 2 initially chose 0.1 due to concern about crews not knowing which procedure to enter. However, after additional discussion, it seemed clear that his initial value was too conservative given the knowledge base of most operating crews, but that an HEP 0.01 still provided for possible exceptions (random "bad" crew or unusual quirks of the scenario). Analyst 1 was also willing to shift his estimated HEP to some extent, but initially not completely to 0.01, mainly because the lack of a direct procedure link still bothered him somewhat. In the end, given this is a preliminary, "conservative, but reasonable" analysis, a consensus value of 0.01 was agreed upon.

B.5 Event 3 - Operator Failure to Restore DC Power after Battery Depletion

In this scenario, an SBO is in progress, the TDAFW pumps have failed, and the emergency batteries have depleted. Without battery power, it is assumed that the operators have no indication of vital plant parameters, and further may not be able to control TDAFW injection to the steam generators. They also may have difficulty in restoring offsite power to the emergency busses, should it become available, or in restarting a repaired emergency diesel generator. Further, they will not be able to open the pressurizer PORVs to reduce RCS pressure.

For purposes of this assessment, if DC power is not restored, the PORVs cannot be opened or re-opened (had they been opened earlier but then failed closed on loss of DC power), and it is assumed that no other relevant actions can occur either, such as restoring the TDAFW train. If DC power is restored, at this stage of performing a "simplified" analysis, the experts being elicited were to assume that completing this action also encompassed other "heroic" measures to save the plant, such as restoring compressed air to the PORV accumulators and opening the PORVs, or even restoring other equipment that will prevent a significant release from failed SG tubes. For example, TDAFW might be restored or an alternate source of FW might be found so that they can cover the tubes. Hence success of recovering DC power was assumed to lead directly to success of preventing failure of the SG tubes.

General Context

There will be many obvious indications of battery depletion. Valves will change position, lights will go out, instruments will fail, etc. However, depending upon the plant, there may or may not be methods at hand to restore power to the DC busses. Some plants, probably a minority, may have contingencies in place to restore DC, such as small portable generators, which can be connected to the battery chargers. Procedures may also specifying when and how this is to be done. However, the majority of plants likely have no specific contingencies in place.

Because of this difference in plant preparedness, the experts decided to analyze two cases, one where equipment and procedures are in place to restore DC power, and the other where no such contingencies exist. Note that in the first case, where contingent equipment and procedures are assumed to be in place, the experts did not use any specific information from a particular plant in arriving at their decision about the HEP.

B.5.1 Failure of Early Heat Removal Scenarios

B.5.1.1 Event 3A - Operator Failure to Provide DC Power to Allow Opening of the PORVs (Operator Sheds Nonessential DC Loads)

T-H calculations indicate that the core exit thermocouples will reach 1200°F and the onset of core damage will begin in about three hours following the los of TDAFW and all electrical power. Depending upon whether the operators have shed nonessential DC loads in Step 14 of ECA-0.0, the remaining battery life after the onset of core damage is assumed for this analysis to be either an additional one (no load-shed) or five hours. For the case where the nonessential DC loads are stripped (making DC power last longer), and the PORVs have been opened (addressed in the second elicitation below), by the time the DC power is sufficiently depleted, the core will have experienced extensive damage because there has been no early heat removal. The reactor vessel bottom head will have suffered a failure that will depressurize the primary system and avoid failure of the SG tubes. Hence this case is irrelevant from the standpoint of operator actions being needed to recover DC power after it is depleted (about 8 hours into the scenario) so that they will be able to open or maintain open the PORVs. Thus, it is not addressed further.

Estimation of HEP (HFE 3A in Table 6) - NA

B.5.1.2 Event 3B - Operator Failure to Provide DC power to Allow Opening of the PORVs (Operator Fails to Shed Nonessential DC Loads)

For the case with early secondary heat removal failure along with a failure to shed nonessential DC loads (hence DC is assumed to be lost in about 4 hours), the analysis also indicates that SG tubes will begin to fail because of creep rupture about 40 to 60 minutes after the onset of core damage, unless the RCS PORVs are opened to reduce the differential pressure across the tubes. Because the onset of core damage begins about three hours after the initiating event with no early heat removal, the tubes may fail at about the same time that DC power is depleted. Because such a short time window is available to restore DC power and prevent SAI-SGCB, and because in this scenario the DC loads were not shed (indicating a lack of appropriate attention to the degrading condition of the DC electrical system), the experts decided to give no credit for restoration in this sequence.

Estimation of HEP (HFE 3B in Table 6)

Per the above, HEP = 1.0.

B.5.2 Success of Early Heat Removal Scenarios

In the cases where secondary heat removal is successful early, but fails at the time of battery depletion, there is an additional 5 to 9 hours available from the time of battery depletion until the core exit thermocouples reach 1200^{o}F and core damage begins. Furthermore, there is an additional 40 to 60 minutes after that before SG tube integrity is lost. Note that this time window of 40 to 60 minutes after core damage is insensitive to how long the batteries last before they deplete (i.e., 4 or 8 hours).

Given the above, the only HEPs to be assessed involve those scenarios where early heat removal has been successful and thus there is considerable time to become aware of and take necessary actions to prevent DC power loss or otherwise restore DC so as to be able to take other actions (such as opening the PORVs) that will prevent SG tube failure.

B.5.2.1 Event 3C - Operator Failure to Provide DC Power to Allow Opening of the PORVs (Plant Has No Pre-Planned and Specific Contingencies for Restoring DC Power)

This event does not depend on success in shedding nonessential DC loads. It is assumed that the plant has no pre-planned and specific contingencies for restoring DC power (used as base case).

Estimation of HEP (HFE 3C in Table 6)

It is assumed for this case that the plant has no specific and pre-planned contingencies to restore power to the battery chargers, or to provide DC power directly to the DC busses. In this case, two of the experts arrived at a failure probability of 0.5, based on an almost total lack of information about what the crew would/could do in this situation. For similar reasons, the other expert initially arrived at a value of 0.7 for this action. This expert also considered that with no specific plans available, plant personnel would have little to guide them (i.e., their situation assessment and decision processes will be entirely “knowledge-based” under highly stressful conditions). Nevertheless, with so little difference among the estimates, it was agreed that a consensus value 0.5 should be used.

B.5.2.2 Event 3D - Operator Failure to Provide DC Power to Allow Opening of the PORVs (Plant Has Pre-Planned and Specific Contingencies for Restoring DC Power)

This event does not depend on success in shedding nonessential DC loads. It is assumed that the plant has pre-planned and specific contingencies for restoring DC power (used as sensitivity case).

Estimation of HEP (HFE 3D in Table 6)

It is assumed that the plant has specific and pre-planned contingencies for restoring DC power.Two of the experts chose 1/10 of their “high” value for case 3C above (i.e., 0.05) as the value for this case, and the other expert chose 0.1. Wanting to remain somewhat conservative at this stage of analysis (i.e., given that there was no direct contact with plants having such contingencies or any review of those contingencies so as to better understand them), the consensus among the experts was to use 0.1.

B.6 Event 4 - Failure to Open Pressurizer PORVs

When 1200°F is reached on the core exit thermocouples (CETs), the operators are directed into SACRG-1, part of the Westinghouse SAMGs. At this point, the TSC is to be activated, if it has not been activated earlier because of the continuing SBO condition. As the operating crew enters SACRG-1, if the TSC is not yet activated (they may still be in the process of assembling), Step 9 of SACRG-1 directs the control room operators to depressurize the RCS if pressure is above a certain point (believed to be about 400 psig). If the TSC is activated, the operators will be in SACRG-2, and the action to depressurize is transferred to the TSC staff which will be using the DFC to decide what to do. The DFC will direct them to enter SAG-2 to depressurize the RCS if pressure is above a certain point (again believed to be about 400 psig), because of the desire to prevent high pressure melt ejection from the reactor vessel, although negative tradeoffs are to be considered.

B.6.1 Success of Early Heat Removal Scenarios

B.6.1.1 Event 4A - Operator Failure to Open PORVs to Depressurize the RCS (Early Secondary Heat Removal Succeeds)

For long-term sequences, where early secondary heat removal is successful until the time of battery depletion and then secondary heat removal is lost, T-H calculations indicate that 1200°F is not reached until five to nine hours after battery depletion. The discussion above on failure to restore DC power indicated that failure to open the PORVs during these long-term scenarios is irrelevant. This is due to the fact that if DC power is lost and is not restored, then the PORVs cannot be opened anyway. If DC power is recovered, it was assumed in the context of the above elicitation, that other follow-on actions such as recovering the TDAFW train, opening the PORVs, or other actions would be taken and would be successful. While such an assumption may be slightly optimistic, it is not thought to be significant to the overall results given the somewhat “high” HEPs from the previous elicitation (restoring DC power). Hence only failure of early heat removal scenarios needs to be considered under this elicitation.

Estimation of HEP (HFE 4A in Table 6) - NA

B.6.2 Failure of Early Heat Removal Scenarios

B.6.2.1 Event 4B - Operator Failure to Open PORVs to Depressurize the RCS (Operator Fails to Shed Nonessential DC Loads)

In short-term sequences, where secondary heat removal fails early, the batteries will deplete about one hour after reaching 1200°F (core damage), unless nonessential DC loads were shed in Step 14 of ECA-0.0. If nonessential DC loads were shed, it is assumed for this analysis that the batteries will last until about five hours past the time at which core damage begins. As already discussed in the previous elicitation, for cases where unnecessary DC loads were not stripped, battery power is assumed to be lost soon after the onset of core damage and hence the PORVs cannot be opened (or maintained open for very long) anyway. Thus, this event is not of interest.

Estimation of HEP (HFE 4B in Table 6) – NA

B.6.2.2 Event 4C - Operator Failure to Open PORVs to Depressurize the RCS (Operators Shed Nonessential DC Loads, TSC Is Not Yet Activated)

For cases where the unnecessary DC loads are shed, allowing DC power to last an assumed eight hours, it is important that the PORVs be opened for a while following the onset of core damage at three hours (which is possible because DC power is available). Accomplishing this allows SG tube failure to be avoided while core damage takes place. Severe accident melt-progression calculations indicate that, for the case where the nonessential DC loads have been shed, the lower head of the reactor vessel will fail about three hours after the onset of core damage, that is, before the batteries deplete at eight hours. However, if the PORVs are not open during this time period, there will be a large differential pressure across the SG tubes. This occurs because the RCS will be pressurized while the secondary is assumed to depressurize through leakage. Thus, the integrity of the tubes cannot be assured and there could be failure of the tubes before the lower head of the reactor vessel fails. It is in consideration of this context, that the HEP was assessed.

Estimation of HEP (HFE 4C in Table 6)

In this case, the TSC has not been activated and the control room operators are implementing SACRG-1. Step 9 of SACRG-1 directs the operators to depressurize the RCS if pressure is above a specific level, which is believed to be 400 psig. In the second case, the TSC is activated. If RCS pressure is above the assumed value of 400 psig, the TSC is directed by the DFC to enter SAG-2, Depressurize the RCS. SAG-2 discusses benefits and potential negative impacts of RCS depressurization. Foremost among the benefits is preventing high pressure melt ejection, followed by preventing creep rupture of SG tubes, and allowing accumulator injection to the RCS. Potential negative impacts include release of hydrogen to the containment, and containment overpressurization due to steam.

Two of the experts arrived at an HEP of 0.1, the third expert arrived at a value of 0.5. After some discussion, the consensus HEP was determined to be 0.1. The evaluations considered the positive aspects of the SACRG-1 being very clear in its direction to depressurize. However, the experts also considered that the operators may be aware of the potential negative impacts identified in SAG-2, and this awareness might cause them to hesitate in executing Step 9. In addition, the experts cited the high stress of the situation, and the possibility that the operators might feel that the TSC would be staffed shortly and thus might defer the decision to the TSC.

B.6.2.3 Event 4D - Operator Failure to Open PORVs to Depressurize the RCS (Operators Shed Nonessential DC Loads, TSC Is Activated)

Estimation of HEP (HFE 4D in Table 6)

With the TSC activated, the experts all arrived at an HEP of 0.5. The basis for choosing this value was the almost total lack of information available to the experts on how the TSC might weigh the benefits and negative impacts of depressurizing, along with the heightened workload of being in other SAGs in parallel with SAG-2. It should be noted that this value is not intended to indicate in any way that it is better to operate without the TSC under these circumstances. The higher value chosen for this case merely reflects the lack of information available to the experts at this time, with regard to how the TSC makes decisions and how the TSC and operating crew interact in such a situation.