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State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

Appendix B
Surry Integrated Analysis

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ABSTRACT

New analyses of severe accident progression and consequences were performed to develop more realistic estimates of the likely outcomes. This study has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for the Surry Nuclear Power Station. By using the most current emergency preparedness (EP), plant capabilities, and best-available modeling, these analyses are more detailed, integrated, and realistic than past analyses. These analyses also consider all mitigative measures, contributing to a more realistic analysis.

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ACRONYMS

AFW	Auxiliary Feedwater
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CST	Condensate Storage Tank
EAL	Emergency Action Levels
ECCS	Emergency Core Cooling System
ECST	Emergency Condensate Storage Tank
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
GE	General Emergency
LLNL	Los Alamos National Lab
LOCA	Loss Of Cooling Accident
LOOP	Loss Of Offsite Power
LTSBO	Long-Term Station Blackout
MSIV	Main Steam Isolation Valve
NG	Noble Gas
NPP	Nuclear Power Plant
OREMS	Oak Ridge Evacuation Modeling System
ORO	Offsite Response Organizations
PORV	Power operated relief valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SAE	Site Area Emergency
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SOARCA	State-of-the-Art Reactor Consequence Analysis Project
SORV	Stuck Open Relief Valve
SPAR	Standardized Plant Analysis Risk
SRV	Safety Relief Valve
STSBO	Short-Term Station Blackout
TAF	Top of Active Fuel
TDAFW	Turbine Driven Auxiliary Feedwater
UE	Unusual Event

1. INTRODUCTION

The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC) over the last several decades. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated and realistic than at any time in the past. A desire to leverage this capability to address excessively conservative aspects of previous reactor accident analysis efforts was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project seeks to provide a body of knowledge that will support an informed public understanding of the likely outcomes of severe nuclear reactor accidents.

The primary objective of the SOARCA project is to develop a body of knowledge of the realistic outcomes of severe reactor accidents in the U.S. civilian nuclear reactor sites. To accomplish this objective the SOARCA project utilized integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community. This report documents the analysis of the Surry Power Station for the risk dominant but extremely low likelihood accidents that could progress to radiological release.

1.1 Outline of the Report

Section 2 of this report briefly summarizes the method used to select the specific accident scenarios subjected to detailed computational analysis. Additional details of this method can be found in Surry Report of this report. Section 3 then describes the results of the mitigation measures assessment process when it was applied to Surry. Section 4 describes the key features of the MELCOR model of the Surry Power Station. Section 5 describes for each case the results of MELCOR calculations of thermal hydraulics, and, when core damage was predicted, accident progression and radionuclide realease into the environment. Section 6 describes the way in which plant-specific emergency response actions were represented in the calculations of offsite consequences, and Section 7 describes the calculations of offsite consequences for each accident scenario, and also describes analysis of off-site consequences comparing SOARCA results to consequence results from earlier studies. References cited in this report are listed in Section 8.

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2. ACCIDENT SCENARIO DEVELOPMENT

In the SOARCA Program, accident sequences that have an estimated frequency greater than 1×10^{-6} per year of reactor operation[1] are retained as candidate sequences for further evaluation. Once candidate accident sequences are identified, realistic opportunities for plant personnel to respond to the observed failures of control and safety systems are evaluated. Possibilities for mitigation included the licensee's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs) and mitigation measures developed specifically for response to security concerns that arose from the events of September 11, 2001 and codified in Title 10, Section 50.54(hh) of the Code of Federal Regulations (10CFR50.54(hh)). The manner in which mitigation measures were evaluated for each accident sequence is described in Section 2.2. The end result of this process was a list of accident scenarios (i.e., event sequence plus options for mitigation), which were subjected to detailed analysis of radionuclide release to the environment (described in Sections 4 and 5) and offsite radiological consequence (Section 6 and 7).

2.1 Sequences Initiated by Internal Events

This scenario selection process was used to determine the scenarios for further analyses:

1. Candidate accident sequences were identified in analyses using plant-specific, SPAR models (Version 3.31).
    a. Initial Screening– Screened out initiating events with low core damage frequencies (CDFs $<10^{-7}$) and sequences with CDFs $<10^{-8}$. This step eliminated 4% of the overall CDF.
    b. Sequence Evaluation– Identified and evaluated the dominant cutsets for the remaining sequences. Determined system and equipment availabilities and accident sequence timing.
    c. Sequence Grouping– Sequences determined to have similar equipment availabilities (i.e., details of individual component or support system failures might differ, but the functional capability of key systems was similar) and result in a similar time for the onset of core damage were aggregated into a single 'sequence group.'

2. Containment systems availabilities for each sequence were assessed using system dependency tables which delineate the support systems required for performance of the target front-line systems and from a review of existing SPAR model system fault trees.

3. Core-damage sequences from the licensee PRA model were reviewed and compared with the scenarios determined by using the SPAR models. Differences were discussed during meetings with licensee staff.

4. The screening criteria (CDF $< 10^{-6}$ for most scenarios, and $< 10^{-7}$ for containment bypass sequences) were applied to eliminate sequences from further analyses.

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[1] 1×10^{-7} per reactor-year for sequences involving bypass of the containment pressure boundary or a perceived possibility of a large-early release.

This process identified two sequences groups that met the screening criteria of 1×10^{-7}/reactor-year for bypass events that have the potential to result in significant early releases to the environment.

- spontaneous steam generator tube rupture – 5×10^{-7}/reactor-year
- interfacing systems loss-of-coolant accident – 3×10^{-8}/reactor-year[2]

This scenario selection process identified two sequences groups that met the screening criteria of 1×10^{-7}/reactor-year for bypass events that have the potential to result in significant early releases to the environment.

- spontaneous steam generator tube rupture – 5×10^{-7}/reactor-year
- interfacing systems loss-of-coolant accident – 3×10^{-8}/reactor-year[3]

This process provides the basic characteristics of each scenario. However, it is necessary to have more detailed information about the scenario than is contained in a PRA model. The emergency operating procedures (EOPs) were reviewed to evaluate the subsequent progression of events, which was beyond the scope of actions typically treated in current PRA models. In particular, the mitigation measures treated in SOARCA include the licensee's EOPs, the severe accident management guidelines (SAMGs), and mitigation measures codified in 10CFR50.54(hh) following the events of September 11, 2001. The mitigation measures assessment for internal events also included mitigation measures codified in 10CFR50.54(hh), but these measures were subsequently shown to be redundant to the wide variety of equipment and indications available for mitigating them. The identified internal events involve few equipment failures and are controlled by postulated operator errors.

2.2 Sequences Initiated by External Events

Seismic-initiated sequences were found to be the most restrictive in terms of the ability to successfully implement onsite mitigative measures and offsite protective actions. In addition, the seismic-initiated sequences were found to be important contributors to the external event core damage and release frequencies. As a result, representative external event sequences were assumed to be initiated by a moderate to large seismic event resulting in wide-spread damage to important plant support systems (primarily electric power sources).

2 This following scenario does not meet the SOARCA screening criterion of 1×10^{-7} per reactor-year for a bypass event. The SPAR model assigns it a frequency of 3×10^{-8}/reactor-year, and the licensee's PRA assigns it a frequency of 7×10^{-7}/reactor-year. The SPAR model's frequency does not meet the SOARCA screening criterion for bypass events of 1×10^{-7}/reactor-year, but the licensee's PRA frequency does. Therefore, it was retained for analysis.

3 This following scenario does not meet the SOARCA screening criterion of 1×10^{-7} per reactor-year for a bypass event. The SPAR model assigns it a frequency of 3×10^{-8}/reactor-year, and the licensee's PRA assigns it a frequency of 7×10^{-7}/reactor-year. The SPAR model's frequency does not meet the SOARCA screening criterion for bypass events of 1×10^{-7}/reactor-year, but the licensee's PRA frequency does. Therefore, it was retained for analysis. The main reason for the difference is that the licensee assumed the likelihood of sunsequent low head injection piping ruptute was 1, while the NRC estimated it to be 0.1.

The sequence selection process identified two sequences groups that met the screening criteria of $1x10^{-6}$ per reactor-year for containment failure events and one event that met the screening criteria of $1x10^{-7}$/reactor-year for events that have the potential to result in significant early releases to the environment:

- long-term station blackout – $1x10^{-5}$ to $2x10^{-5}$/reactor-year
- short-term station blackout – $1x10^{-6}$ to $2x10^{-6}$/reactor year
- short-term station blackout with thermally induced steam tube rupture – $1x10^{-7}$ to $8x10^{-7}$/reactor year. This is a bypass event, which has a screening criterion of $1x10^{-7}$/reactor-year. (This assumes a conditional tube failure probability of 0.25 [27].)

It was noted earlier that the initiating event for external event sequences was assumed to be a seismic event, because it was judged to be limiting in terms of how much equipment would be available to mitigate. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event. For these sequence groups, the seismic PRAs provided information on the initial availability of installed systems. Next, judgments were made concerning the general state of the plant to assess the availability of the mitigation measures codified in 10CFR50.54(hh) and the additional time to implement mitigation measures and activate emergency response centers (e.g., Technical Support Center and Emergency Operations Facility).

The seismic events considered in SOARCA result in the loss of offsite and onsite AC power and for the more severe seismic events, loss of DC power. Under these conditions, the use of the turbine-driven system turbine-driven AFW system pump is an important mitigation measure. The Surry emergency procedure guidelines include operation of the TDAFW without electricity to cope with station blackout conditions. The 10CFR50.54(hh) mitigation measures have taken this a step further, and also include the long-term starting of the TDAFW pump without electricity and the methods to establish instrumentations and control valves using a portable generator to supply indications such as reactor pressure and level indications. If the DC buses are available, the TDAFW can be used to cool the core until battery exhaustion. After battery exhaustion, black run of the TDAFW can continue to cool the core. Severe accident code calculations are used to demonstrate core cooling under these conditions.

The external events PRA does not describe general plant damage and accessibility following a seismic event. NUREG/CR-4334 was consulted to assess the potential viability of safety-related piping after a 1.0 peak ground acceleration (pga) event [41]. For the short term station blackout (0.5-1.0 pga) the damage was assumed to be sufficiently widespread such that accessibility would be difficult. The TDAFW system was judged not initially available and was judged not recovered under these circumstances prior to fuel damage (i.e., fuel damage in 3 hours) due to failure of the immediate gross rupture of the ECST (Emergency Condensate Storage Tank). However, extrapolating results from NUREG/CR-4334, the low-pressure water injection and containment spray safety-related piping were judged to remain intact. Other studies, including a German study that physically simulated ground motion equal to 1.0 pga on an existing plant, also supported this evaluation.

In the less severe long-term station blackout (0.3 – 0.5 pga), the TDAFW system was available and the low-pressure safety injection and safety-related containment spray piping were also judged to remain intact. The integrity of the safety-grade piping provided a connection point for a portable, diesel-driven pump to inject water into the RCS or into the containment spray systems.
The 10CFR50.54(hh) mitigation measures include the application of portable equipment such as portable power supplies for the instrumentation, portable diesel-driven pumps, and portable air bottles to open air-operated valves. Applicable procedures have been written to implement these mitigative measures under severe accident conditions. At the time of the Surry site visit, the licensee stored the portable injection equipment and the site fire truck onsite in a structure away from the containment. A walk-down of the storage building and pathway to the plant suggested that the operators would be able to retrieve the equipment following a seismic event.

The time estimates to implement individual mitigation measures were provided by the licensee staff for each sequence group based on the sequence descriptions provided by the NRC. The time estimates take into account the plant conditions following the seismic event. The time estimates reflect exercises run by licensee staff that provided actual times to move and connect the portable, diesel-driven pump. The time estimates for manning the Technical Support Centers and the Emergency Operations Facilities also were provided by licensee staff and reflect the possible effects of the seismic event on roads and bridges.

The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps and power supplies from sister plants or from contractors, external spray systems), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented.

Finally, the following items are cited as relevant to the accident progression and mitigative measure response. Initially, no multi-unit accident sequences were selected for the SOARCA project. This was beyond the scope of the project and considered unrealistic (i.e., beyond the screening criteria) for internal event sequences. Therefore, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Surry is a two-unit site. In conclusion, Surry Unit 1 had an opening in the reactor cavity wall and Surry Unit 2 did not, which may affect ex-vessel debris cooling. No scenario relied on ex-vessel debris cooling through the cavity wall, so the difference was not pursued further.

3. ACCIDENT SCENARIO DEFINITIONS AND MITIGATIVE MEASURES

Various mitigated and unmitigated scenario initial and boundary conditions were developed for the severe accident code calculations. Sections 3.2 and 3.1 describe the short-term and long-term station blackouts scenarios, respectively, which were initiated by a seismic external event. The spontaneous steam generator tube rupture and interfacing systems loss-of-coolant accident scenarios are described in Sections 3.3 and 3.4, which were internal events initiated by piping failures. Each section describes the initiating event, the available systems, the pertinent mitigative actions, and the detailed initial and boundary conditions for the severe accident code calculations. The SOARCA mitigation measures assessment was a qualitative assessment of the likelihood of mitigation. Thes was sufficient to satisfy the SOARCA objective of developing a body of knowledge of the likely outcomes of severe reactor accidents. A quantitative assessment of reliability for B.5.b measures was not performed, because the additional precision did not justify the additional cost.

3.1 Long-Term Station Blackout

The long-term station blackout is initiated by a beyond-desgin-basis earthquake (0.3–0.5 peak ground acceleration - pga). It has an estimated frequency of $1x10^{-5}$ to $2x10^{-5}$/reactor-year, which meets the SOARCA screening criterion of $1x10^{-6}$/reactor-year.

Section 3.1.1 describes the initial status of the plant following the seismic event. The key system availabilities normally accessible during the course of the accident are summarized in Section 3.1.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.1.3. Section 3.1.4 delineates various scenarios based on the success of the mitigative actions. In particular, mitigated scenarios are defined where the mitigative actions are successful. Unmitigated scenarios are also defined where certain key mitigative measures are not successfully implemented.

For station blackout scenarios, boiling in the RCP seal could cause the spring-loaded part of the seal to pop open and stay open. As such, MELCOR modeling for Surry includes seal failure when conditions in the seal approach saturation. The hole size for this failure mode is that which produces a 180 gpm/pump flow rate at normal RCS temperature and pressure. Also, it has been hypothesized that seal failure could occur as early as 13 minutes into a station blackout scenario due to the loss of seal cooling; seal cooling requires AC power. The conditional probability this early seal failure (as early as 13 minutes) has been estimated by the industry to be 0.2 [36]. Applying this 0.2 probability to the Surry long-term station blackout scenario frequency of 1 to $2x10^{-5}$/reactor-year results in an event frequency of 2 to $4x10^{-6}$/reactor-year, which meets the SOARCA screening criteria of $1x10^{-6}$/reactor-year. While seal failure could occur as early as 13 minutes into the scenario and could include seal failures in as many as all 3 RCPs, such early and multiple seal failures would have a lower probability. However, to examine the potential range of system response, the project staff analyzed with MELCOR long-term station blackout mitigated and unmitigated sensitivity cases assuming that the seals of all 3 RCP seals failed 13 minutes into the scenario.

3.1.1 Initiating Event

The seismic event results in the loss of offsite power (LOOP) and failure of onsite emergency alternating current (AC) power resulting in a station blackout (SBO) event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray and fan coolers). The TDAFW pump is available initially. In the long term, the loss of the TDAFW pump may occur due to battery depletion and loss of direct current (DC) power for sensing and control. Nominal RCP leakage occurs due to the loss of pump seal cooling (21 gpm/pump). The unmitigated and mitigated base cases includes the potential for a later thermo-mechanical RCP seal failure. In addition, unmitigated and mitigated sensitivity cases are performed that include an early RCP seal failure (i.e., at 13 minutes).

3.1.2 System Availabilities

The TDAFW pump is available until the emergency condensate storage tank empties. The station batteries give instrumentation until they exhaust at 8 hr. The secondary PORVs are initially available for a manual 100°F/hr system cooldown. The secondary PORVs are assumed to close following battery failure. No other systems are available.

3.1.3 Mitigative Actions

The LTSBO event results in the loss of offsite and onsite AC power. Under these conditions, the TDAFW pump is an important mitigation measure. PWR emergency procedure guidelines include operation of the TDAFW pump without AC power to cope with station blackout conditions. The mitigation measures codified in 10CFR50.54(hh) have taken this a step further and also include long-term operation of TDAFW pump without DC power, methods to establish instrumentations and control valves using a portable generator to supply indications such as reactor pressure and level indications. The TDAFW pump is used to cool the core until battery exhaustion. After battery exhaustion, black run of TDAFW pump is used to remove heat from the primary system.

The external events PRA does not describe general plant damage and accessibility following a seismic event. The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems. The emergency condensate storage tank initially supplies the TDAFW pump but has finite resources (i.e., empty in 6 hours). However, it was assessed that the operators would have sufficient time, access, and resources to make-up water for injection.

The severity of this seismic event is lower than the short-term station blackout. Consequently, the low-pressure injection and safety-related containment spray piping were also judged not likely to fail for this scenario. The integrity of this piping provided a connection point for a portable, diesel-driven pump to inject into the RCS. Licensee staff estimated that transporting the pump and connecting it to plant piping takes about two hours. Hence, the availability of the vessel injection was assessed to occur at 3.5 hours, or 2 hours after the action was recommended by the operators and support staff. Companion unmitigated analyses were also performed to quantify the response without successful mitigation by a portable pump.

In summary, the following actions are credited in the mitigated scenario calculations.

- Provide vessel injection using a portable, high-pressure, diesel-driven (Kerr) pump through three drain lines on the residual heat removal piping
- Use portable air bottles to operate the steam generator power-operated relief valves, which allows for depressurization and cooldown of the RCS
- A portable power supply is used to restore SG and RCS level indication
- Manual operation of the TDAFW pump without DC power is credited
- A portable, diesel-drive, low-pressure (Godwin) pump is used to refill the emergency condensate storage tank.

While not used in the mitigated scenario calculations, the following additional mitigative measures were identified as additional options for consideration.

- Use firewater or pumper truck for the charging pump oil cooler and use an alternative power source for high head safety injection pump RCS makeup.
- Lineup the portable, diesel-driven (Godwin) pump and firewater system for auxiliary feedwater makeup to the steam generators.

3.1.4 System Boundary Conditions

Section 3.1.4.1 lists the sequence of events in the unmitigated long-term station blackout calculation. Section 3.1.4.2 summarizes the sequence of events in the mitigated long-term station blackout calculation which credits additional manual actions. Mitigated and unmitigated sensitivity cases were also performed that include an early failure of the RCP seals.

3.1.4.1 Sensitivity Case without B.5.b Equiptment

There is one unmitigated base case and one unmitigated sensitivity case. The unmitigated sensitivity case includes early failures of the RCP seals. The boundary conditions for the two cases are listed below.

Unmitigated Case (without portable mitigation equipment)

Event Initiation

- Loss of offsite power followed by a station blackout
- The reactor trips and the MSIVs close
- The DC buses are available, at minimum loading, currently being used for instrumentation, PORV operation, and TDAFW system operation
- The TDAFW system auto-initiates providing make-up to the steam generators (SGs). The TDAFW system takes suction water from the emergency condensate storage tank (ECST)
- RCP seal leakage begins at 21 gpm per pump. The RCP seals may subsequently fail due to high temperatures causing a leak of 182 gpm per pump (see Section 4.6.2 for a description of the failure model).

- Emergency core cooling systems are inoperable
- Containment cooling systems are inoperable
- Containment is isolated
- Recovery of offsite power is not expected during the mission time

15 minutes

- Initial Operations assessment of plant status complete, initiate the following actions:
  - Attempt manual start of EDGs
  - Operation of SG PORVs available for 30 minutes using a dedicated internal battery for RCS pressure control (Main steam code safety valves also are available for RCS pressure control.)
- The SG level is being maintained by AFW, RCS is cooling down, no RCS makeup currently available

1 hour

- Manual start of EDGs assumed to be unsuccessful due to initiating event
- The TSC is manned and operational. The primary function of the TSC would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.

1.5 hours

- The EOF is manned. The primary function of the EOF would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff members are the primary users of SAMGs and mitigation measures codified in 10CFR50.54(hh). Shift supervisors and TSC supervisors are trained on these procedures.
- Operations initiate a controlled depressurization of the SGs to approximately 120 psi to achieve an RCS cooldown of <100°F per hour by manually opening the SG PORVs.
- The TSC and EOF review actions taken by Operations and determine the availability of the portable, diesel-driven pumps and the station pumper truck stored outside the protected area. Recommend the following actions:

  - Connect the portable, high-pressure, diesel-driven (Kerr) pump to three drain lines of the residual heat removal piping for RCS makeup and use portable bottles for manual operation of primary PORVs, as needed.
  - Hook up portable power supply to power instrumentation
  - Use the 2 firewater storage tanks (250,000 gallons per tank), the portable, low-pressure, diesel-driven (Godwin) pump, and the fire pumper truck to supply AFW suction, if necessary
  - Setup to provide the firewater system or a portable, low-pressure, diesel-driven (Godwin) pump to the containment spray header in preparation for containment cooling

1.75 hours

- Operations assesses and concurs with TSC and EOF recommendations. Operations prioritizes recommendation based on plant conditions..

>1.75 hours

- All mitigative actions are unsuccessful including connecting a portable, diesel-driven pump for vessel injection, refilling the water supply for the TDAFW (i.e., the emergency condensate storage tank), and maintaining instrumentation using a portable power supply

8 hours

- DC station batteries are exhausted[4]
- SG PORVs reclose
- Loss of control and instrumentation for the TDAFW

Unmitigated Case (without portable mitigation equipment) + early RCP seal failure

Identical sequence of events as the unmitigated base case but includes an early RCP seal failure at 13 minutes.

13 minutes

- All three RCP seals fail and leak at a nominal rate of 182 gpm per pump.

3.1.4.2 Base Case

There is one mitigated base case and one mitigated sensitivity case. The mitigated sensitivity case includes an early failure of the RCP seal on all three pumps. The boundary conditions for the two cases are listed below.

Mitigated Case (using portable mitigation equipment)

Event Initiation

- Loss of offsite power followed by a station blackout
- The reactor trips and the MSIVs close
- The DC buses are available, at minimum loading, currently being used for instrumentation, PORV operation, and TDAFW operation
- The TDAFW system auto-initiates providing make-up to the steam generators. The TDAFW system takes suction water from the emergency condensate storage tank with additional makeup from the condensate storage tank.
- RCP seal leakage begins at 21 gpm per pump. The RCP seals may subsequently fail due to high temperatures causing a leak of 182 gpm per pump (see Section 4.6.2 for a description of the failure model).
- Emergency core cooling systems are inoperable
- Containment cooling systems are inoperable
- Containment is isolated
- Recovery of offsite power is not expected during the mission time

[4] The Surry DC station batteries are required to last for 2 hours. Initially, the licensee estimated a best-estimate life of 8 hours. Following completion of the analysis, 6 hours was thought to be more realistic. However, the ECST ran out of water at 5 hours and stopped the TDAFW pump. Consequent, the most significant benefit of the station batteries failed at 5 hours, which was less than the best-estimate battery life.

15 minutes

- Initial Operations assessment of plant status complete, initiate the following actions:
  - Attempt manual start of EDGs
  - Operation of SG PORVs available for 30 minutes using a dedicated internal battery for RCS pressure control (Main steam code safety valves also are available for RCS pressure control.)
- Station batteries available. (Batteries typically last for approximately 2 to 8 hours under normal loading conditions depending on life cycle of battery. At the beginning of its life, the battery duration is 8 hours. At the end of its life, the battery duration is 2 hours. It was assumed that the battery life for a seismic-initiated long-term station blackout was 8 hours due to the minimum loading conditions caused by the initiating event and the minimum loading expected throughout the event due to the limited equipment available.)

- SG level being maintained by AFW, RCS is cooling down, no RCS makeup currently available

1 hour

- Manual start of EDGs assumed to be unsuccessful due to initiating event
- The TSC is manned and operational. The primary function of the TSC would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.

1.5 hours

- The offsite EOF is manned. The primary function of the EOF would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff members are the primary users of SAMGs and mitigation measures codified in 10CFR50.54(hh). Shift supervisors and TSC supervisors are trained on these procedures.
- Operations initiate a controlled depressurization of the SGs to approximately 120 psi to achieve an RCS cooldown of < 100°F per hour by manually opening the SG PORVs.
- TSC and EOF review actions taken by Operations and determine the availability of the portable, diesel-driven pumps and the station pumper truck stored outside the protected area. Recommend the following actions:

  - Connect the portable, high-pressure, diesel-driven (Kerr) pump to three drain lines of the residual heat removal piping for RCS makeup and use portable bottles for manual operation of primary PORVs, as needed.
  - Hook up portable power supply to power instrumentation
  - Use the 2 firewater storage tanks (250,000 gallons per tank), the portable, low-pressure, diesel-driven (Godwin) pump, and the fire pumper truck to supply AFW suction, if necessary
  - Setup to provide the firewater system or a portable, low-pressure, diesel-driven (Godwin) pump to the containment spray header in preparation for containment cooling

1.75 hours

- Operations assesses and concurs with TSC and EOF recommendations. Operations prioritizes recommendation based on plant conditions and begins implementation.

3.5 hours

- The Kerr pump provides emergency 150 gpm makeup flow to the RCS
- A portable power supply provides to power to the instrumentation
- TDAFW pump maintaining S/G level
- Pre-staging and lineups are ongoing for other mitigation measures:
  - Setup to provide the firewater system or the a portable, low-pressure, diesel-driven (Godwin) pump to the containment spray header in preparation for containment cooling
  - Use the 2 firewater storage tanks with the portable, low-pressure, diesel-driven (Godwin) pump or the fire pumper truck to supply AFW

Mitigated Case (using portable mitigation equipment) + early RCP seal failure

Identical sequence of events as the unmitigated base case but includes an early RCP seal failure at 13 minutes.

13 minutes
All three RCP seals fail and leak at a nominal rate of 182 gpm per pump.

3.2 Short-Term Station Blackout

The short-term station blackout is initiated by a large earthquake (0.5–1.0 pga). It is more severe than the long-term station blackout and has an estimated frequency of $1x10^{-6}$ to $2x10^{-6}$/reactor year, which meets the SOARCA screening criterion of $1x10^{-6}$/reactor-year.

Section 3.2.1 describes the initial status of the plant following the seismic event. The key system availabilities during the course of the accident are summarized in Section 3.2.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.2.3. Section 3.2.4 delineates various scenarios based on the success of the mitigative actions. In particular, mitigated scenarios are defined where the mitigative actions are successful. Unmitigated scenarios are also defined where certain key mitigate measures are not successfully implemented. In addition, mitigated and unmitigated scenarios are defined that include a thermally-induced steam generator tube rupture.

3.2.1 Initiating Event

The seismic event results in a loss-of onsite power (LOOP) and failure of onsite emergency AC power resulting in a station blackout (SBO) event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including all active emergency core cooling systems (ECCS) and the containment engineered safety systems (e.g., the containment sprays and fan coolers). The seismic event also causes a loss of DC power, which makes it impossible to remotely control the TDAFW pump. The reactor coolant system (RCS) and containment are undamaged and the containment is isolated. No instrumentation is available. Significant structural damage is assumed, including structural failure of the turbine building and loss of access to the condenser blow down valves is expected.. Auxiliary building accessibility is difficult, due to fallen piping and cabling, steam and water leaks, and damaged stairways. Following the loss of the seal cooling flow, the reactor coolant pump (RCP) seals will nominally leak at 21 gpm (i.e., at normal operating pressure and temperature). The RCP seals may fail later in the accident if the RCP seal region heats to saturated conditions. However, the ECCS accumulators, portable power supplies, portable air bottles, and portable high-pressure (Kerr) and low-pressure (Goodwin) diesel driven pumps are available.

Both unmitigated and mitigated sensitivity cases are performed that include a thermally-induced steam generator tube rupture(s). Thermally-induced steam generator tube ruptures are a known risk contributor and has been investigated by industry and the NRC. The short-term station blackout has an estimated frequency of $1x10^{-6}$ to $2x10^{-6}$/reactor-year, and the conditional probability of tube rupture has been estimated by the NRC to be in the range of 0.1 to 0.4 [27]. Therefore, the overall frequency of this sequence group is 1 to $8x10^{-7}$/reactor-year, which meets the SOARCA screening criterion of $1x10^{-7}$/reactor-year for bypass events. In the context of the short-term station blackout sequence evaluations, sensitivity studies are performed to examine the response with a thermally-induced steam generator tube rupture.

3.2.2 System Availabilities

No systems are available except as noted in the mitigative actions.

3.2.3 Mitigative Actions

The STSBO results in a loss of offsite and onsite AC power and the loss of DC power. Under these conditions, operation of the TDAFW pump is an important mitigation measure. PWR emergency procedure guidelines include operation of the TDAFW pump without AC power to cope with station blackout conditions. The mitigation measures codified in 10CFR50.54(hh) have taken this a step further and also include long-term operation of TDAFW pump without DC control power. Procedures have been developed to provide instrumentation (e.g., reactor pressure and level indications) and to operate valves using a portable generator. The external events PRA does not describe the general plant damage and accessibility following a seismic event. The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems. The TDAFW pump was assumed not to be available initially and was judged not recovered under these circumstances prior to fuel damage (i.e., fuel damage in <3 hours) due to failure of the ECST. However, there was sufficient time, access, and resources to establish containment sprays with the portable emergency pump by 8 hours. Once activated, the operator could inject as much as one million gallons of water into the containment. This action both mitigates the release and delays containment failure.

NUREG/CR-4334 was consulted to assess the potential viability of safety-related piping after a 1.0 pga event [41]. Extrapolating results from NUREG/CR-4334, the low-pressure water injection and safety-related containment spray piping were judged to remain intact. Other studies, including a German study that physically simulated ground motion equal to 1.0 peak ground acceleration (pga) on an existing plant, also supported this evaluation. The integrity of the safety-grade piping provided a connection point for a portable, diesel-driven pump to inject water into the RCS or into the containment spray systems. The licensee staff estimated that transporting the pump and connecting it to plant piping takes about two hours. Because of difficult accessibility, the set-up of the containment spray system following a large seismic event was assumed to require 8 hours. Hence, the availability of the emergency containment spray injection was assumed to occur after the vessel failure (i.e., the MELCOR results indicate 3 hours to core damage and 7 hours to lower head failure). Additional unmitigated analyses were performed to quantify the response without successful mitigation by a portable pump.

3.2.4 Scenario Boundary Conditions

Section 3.2.4.1 lists the sequence of events in the unmitigated short-term station blackout calculation. Section 3.2.4.2 summarizes the sequence of events in the mitigated short-term station blackout calculation that credits one additional manual action. Sensitivity cases for the mitigated and unmitigated thermally-induced steam generator tube ruptures are also described.

3.2.4.1 Unmitigated Cases

There is one unmitigated base case and two unmitigated sensitivity cases. The unmitigated sensitivity cases include thermally-induced steam generator tube ruptures prior to creep rupture in any other RCS location. Since the sensitivity cases include a stuck open secondary relief valve, there is an open containment bypass pathway to release radionuclides to the environment. In the base case, the secondary relief valve closes when the pressure falls below the closing setpoint.

Unmitigated base case

<u>Event Initiation</u>

- Loss of offsite power followed by the failure of all diesel generators and a station blackout
- Successful reactor trip and Main Steam isolation Valves (MSIVs) close
- RCS and containment are undamaged and the containment isolates
- Failure of TDAFW system due to failure of the ECST
- An early RCP seal failure during subcooled conditions is not included in this scenario, but late RCP seal failures may occur[5]
- Active ECCS equipment is inoperable due to electrical and physical system damage
- Containment cooling systems are inoperable due to electrical and physical system damage
- Recovery of offsite and onsite power is not expected during the mission time

[5] See discussion in Section 4.6.2 of MELCOR's RCP seal failure model.

30 minutes

- Initial Operations assessment of plant status complete; Operations initiates the following action:
    - Attempt manual start of the Emergency Diesel Generators (EDGs) and Station Blackout (SBO) diesel generator
- RCS pressure being maintained by code safety valves, power operated relief valves (PORVs)
- not currently available because loss of instrument air and backup air

1 hour

- Manual start of EDGs and SBO diesel generator assumed to be unsuccessful due to initiating event
- Offsite Emergency Operations Facility (EOF) is manned. The primary function of the EOF is review of initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff members are the primary users of SAMGs and mitigation measures codified in 10CFR50.54(hh). Shift supervisors and TSC supervisors are trained on these procedures.

1.5 hours

- Offsite EOF reviews actions taken by operations. Recommend the following actions:
    - Maintain portable power supply for instrumentation
    - Connect the portable, high-pressure, diesel-driven (Kerr) pump to three drain lines of the residual heat removal piping for RCS makeup and use portable bottles for manual operation of SG PORVs, as needed
    - Connect the portable, diesel-driven (Godwin) pump for containment spray or containment flooding

1.75 hours

- Operations assesses and concurs with offsite EOF recommendations. Operations prioritizes recommendations based on plant conditions and begins implementation.

2 hours

The TSC is manned and operational. Because of the magnitude of the seismic event, a one hour delay in reporting of TSC members was assumed. The primary function of the TSC would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.

- Onsite EOF is manned and operational 60 minutes later.

3 hours

- Onsite EOF is operational.

3.5 hours

- Determine the availability of the portable, diesel-driven (Godwin) pump, portable air bottles, and portable power supply (currently supplying instrumentation)
- Portable air bottles ready to be connected to the steam generator PORVs for depressurizing RCS
- The portable diesel-driven pumps are being positioned and the connections are being assessed.

6.5 hours

- RCS can be depressurized using portable air bottles to control the appropriate air-operated valves. The accumulators will provide make-up to the RCS once depressurized. However, since the RCS hot leg failure at 3.75 hours had already depressurized the RCS (i.e., see Section 5.2.1), this mitigation effort is not successful.

>6.5 hours

- Unable to connect portable injection systems
- No other mitigation attempts are successful

<u>**Unmitigated sensitivity cases with thermally-induced steam generator tube ruptures**</u>

The unmitigated sensitivity cases have identical sequence of events as the unmitigated base case but includes a stuck open relief valve on the secondary side that leads to a thermally-induced steam generator tube rupture (TI-SGTR).

3 hours

- The lowest-pressure safety relief valve sticks open on the steam generator with the tube rupture.

At a time calculated by MELCOR to be 3 hr 33 min

- A thermally-induced SGTR is assumed to occur when the hot leg C cumulative creep damage index exceeds 5% (i.e., a criteria to identify hto conditions in the RCS piping and ensure that the steam generator tube fails before the hot leg nozzle).
    - Sensitivity Case 1 – rupture area is the equivalent of 100% of the tube flow area
    - Sensitivity Case 2 – rupture area is the equivalent of 200% of the tube flow area

3.2.4.2 Mitigated Cases

There is a mitigated base case and a mitigated sensitivity case. The mitigated sensitivity case includes a thermally-induced steam generator tube rupture prior to any other RCS creep rupture failure. Since the sensitivity case includes a stuck open secondary relief valve, there is an open containment bypass pathway to release radionuclides to the environment. In the base case, the secondary relief valve closes when the pressure falls below the closing setpoint.

Mitigated base case

Event Initiation
- Loss of offsite power followed by a station blackout
- Successful reactor trip and MSIVs close
- RCS and containment undamaged and the containment isolated
- Failure of TDAFW pump due to failure of the ECST
- An early RCP seal failure during subcooled conditions is not included in this scenario, but late RCP seal failures may occur[6]
- Emergency core cooling systems are inoperable due to electrical and physical system damage
- Containment cooling systems are inoperable due to electrical and physical system damage
- Recovery of offsite power is not expected during the mission time

30 minutes
- Initial Operations assessment of plant status complete; Operations initiates the following action:
  - Attempt manual start of EDGs and SBO diesel generator
- The RCS pressure is maintained by code safety valves. The pressurizer PORVs are not currently available because loss of instrument air and backup air

1 hour
- Use portable power supply to restore minimum instrumentation (RCS level, RCS pressure, SG level)
- Manual start of EDGs and SBO diesel generator assumed to be unsuccessful due to initiating event
- The EOF is manned. The primary function of the EOF would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff members are the primary users of SAMGs and mitigation measures codified in 10CFR50.54(hh). Shift supervisors and TSC supervisors are trained on these procedures.

[6] See discussion in Section 4.6.2 of MELCOR's RCP seal failure model.

1.5 hours

- Offsite EOF reviews actions taken by operations. Recommend the following actions:
  - Maintain portable power supply for instrumentation
  - Connect the portable, high-pressure, diesel-driven (Kerr) pump to three drain lines of the residual heat removal piping and use portable bottles for manual operation of SG PORVs, as needed
  - Connect the portable, diesel-driven (Godwin) pump for containment spray or containment flooding

1.75 hours

- Operations assesses and concurs with offsite EOF recommendations. Operations prioritizes recommendations based on plant conditions and begins implementation.

2 hours

- The TSC is manned and operational. Because of the magnitude of the seismic event, a one hour delay in reporting of TSC members was assumed. The primary function of the TSC would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.
- Onsite EOF is manned and operational 60 minutes later.

3 hours

- Onsite EOF is operational.

3.5 hours

- Determined the availability of the remotely located portable, diesel-driven (Godwin) pump, portable air bottles, and portable power supply (currently supplying instrumentation)
- Portable air bottles ready to be connected to PORVs for depressurizing RCS
- The portable diesel-driven pumps are being positioned and the connections are being assessed.

6.5 hours

- RCS can be depressurized using portable air bottles to control the appropriate air-operated valves. The accumulators will provide make-up to the RCS once depressurized. However, since the RCS hot leg failure at 3.75 hours had already depressurized the RCS (i.e., see Section 5.2.1), this mitigation effort is not successful.

8 hours

- The portable, diesel-driven (Godwin) pump is connected to containment spray system and injection starts. Injection continues for 1,000,000 gallons. This mitigates the release and delays containment failure.

<u>Mitigated case with thermally-induced steam generator tube rupture</u>

The mitigated sensitivity case has an identical sequence of events as the mitigated base case but includes a stuck open relief valve on the secondary side that leads to a thermally-induced steam generator tube rupture.

3 hours

- The lowest-pressure safety relief valve sticks open on the steam generator with the tube rupture.

At a time to be calculated by MELCOR (which was 3 hr 33 min)

- A thermally-induced SGTR is assumed to occur when the hot leg C cumulative creep damage index exceeds 5% (i.e., a criteria to identify hot conditions in the RCS piping and ensure that the steam generator tube fails before the hot leg nozzle).
- The steam generator tube rupture area is the equivalent of 100% of the tube flow area
-

3.3 Spontaneous SGTR

Section 3.3.1 describes the initial status of the plant following the tube rupture. The key system availabilities during the course of the accident are summarized in Section 3.3.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.3.3. Section 3.3.4 delineates various scenarios based on the success of the mitigative actions. In particular, a mitigated scenario is defined where the mitigative actions are successful. Two unmitigated scenarios are also defined where certain key operator actions are not successfully performed.

3.3.1 Initiating Event

This sequence group consists of a spontaneous rupture of a steam generator tube equivalent to 100% of the tube flow area. The leak occurs near the steam generator inlet-side tube sheet. The operator fails to isolate damaged steam generator (SG), fails to implement procedures ECA 3.1 and 3.2, and fails to depressurize and cooldown RCS. The spontaneous SGTR sequence group results in core damage because of operator errors.

3.3.2 System Availabilities

The full complement of systems is considered functional in this scenario including all systems associated with engineered safeguards and instrumentation and control as well as all auxiliary and emergency systems. The operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown the RCS, and 3) refill the RWST or cross-connect to the unaffected unit's RWST. The ISLOCA results in core damage because of ineffective operator responses. In particular, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST.

3.3.3 Mitigative Actions

The SPAR model and the licensee's PRA concluded that the spontaneous SGTR event proceeds to core damage because of the above errors. However, the PRA models do not appear to have credited the significant time available for the operators to correct their mistakes. They also do not appear to credit technical assistance from the TSC and the EOF. The subsequent accident simulation showed that 27 to 46 hours are available for mitigative actions before the core damage begins, see Section 5.4. Therefore, the licensee provided realistic estimates of the times by which the operators would respond to the event. These time estimates included consideration of indications that the operators would have of the bypass accident, operator training on plant procedures for dealing with bypass accidents and related drills, and assistance from the TSC and EOF, which were estimated to be manned and operational by 1 to 1.5 hours into the event.

Operator actions in this scenario are essentially those expected per training and procedure. Specifically, the operators are trained to perform the following actions to mitigate to sequence:

- Secure AFW delivery to the steam generator with the broken tube (the faulted steam generator)
- Secure 1 of the 3 total high head safety injection (HHSI) pumps
- Isolate the faulted steam generator, i.e., close the MSIVs serving the faulted steam generator
- Secure the remaining HHSI pumps once the faulted generator is isolated, which will end the RCS leakage
- Perform a 100°F/hr cool-down of the RCS
- Establish long-term cooling with residual heat removal

The following other mitigation measures were identified but not used.

- Use the pressurizer PORVs to depressurize the RCS to get an accumulator injection at low pressure
- Cross-connect to the unaffected unit's RWST
- Use firewater makeup to RWST from the firewater header at ~300 gpm from the two 500,000 gallons firewater storage tanks, then the James River
- The portable, low-pressure, diesel-driven (Godwin) pump is available to makeup to the RWST and the CST at ~2000 gpm at 120 psi
- ~190,000 gallons available from the SFP for rapid RWST makeup
- Procedures exist to align firewater to the suction of the AFW pump via installed piping and valves from firewater storage tanks and James River
- Two portable, high-pressure, diesel-driven (Kerr) pumps are available to inject into RCS using firewater at 2.5 hours (assumes guidance from TSC and EOF at 1.5 hours and an hour to implement)

3.3.4 Boundary Conditions

Section 3.3.4.2 lists the sequence of events to be prescribed in the mitigated spontaneous steam generator tube rupture where the operator successfully performs the actions described in

Section 3.3.3. Section 3.1.4.2 summarizes the sequence of events in two unmitigated scenarios where the operator does not successfully perform the actions described in Section 3.3.3. The second unmitigated scenario uses the same failed operator actions but also includes the failure of the steam generator secondary relief valve to create a sustained containment bypass pathway for fission products.

3.3.4.1 Unmitigated Cases

The sequence of events for the unmitigated cases is the same as the mitigated case until 2.5 hours. No other successful operator actions are credited after 2.5 hours.

Unmitigated Case 1

2.5 hours
- Fail to isolate the faulted steam generator
- Fail to depressurize and cool the RCS
- Fail to extend the available duration of ECCS injection by refilling the RWST or cross connecting to the other unit's RWST

Unmitigated Case 2

Exactly the same boundary conditions as Unmitigated Case One but include an additional equipment failure.

At a time to be calculated by the severe accident analysis code (which was 44 min)
- Fail the secondary relief valve open when water first reaches the valve. The stuck-open valve creates an open bypass containment pathway to the environment (see note below).

2.5 hours

Note: There was some uncertainty whether water could reach the secondary relief valve. The utility stated that the secondary system would never go solid due to the large volume of piping and 12 steam traps (eight 1.5" lines and four 1" lines) open to the main condenser. The MELCOR model did not include models of the steam traps or steam dump valves to the condenser. The calculation conservatively neglected any leakage pathways for water from the steam line except for the cycling relief valve.

3.3.4.2 Mitigated Case

There is one mitigated base case. Although the operator initially fails to implement the correct procedures, the errors are eventually identified by the technical support groups and the correct procedures are followed. The boundary conditions are listed below.

<u>Event Initiation</u>
- Spontaneous tube rupture equivalent to a double-ended break of a single tube
- The reactor trips

- The turbine stop valves automatically close
- The 8 steam dump valves automatically go to the full open position and then throttle open and close to maintain RCS T_{ave} at 547 °F
- Containment Phase 1 isolation auto-initiates
- The HHSI auto-initiates and all three pumps start and operate as designed. The operator secures one charging pump early in the event as required by procedure. The water source is the RWST (380,000 gallons)
- The one turbine-driven (TD) and two motor-driven (MD) auxiliary feedwater pumps automatically start on a low-level actuation signal. The initial water source is the emergency condensate storage tank (ECST) (110,000 gallons) but can be refilled from the CST, which has 300,000 gallons.
- Reactor coolant pumps continue to run
- Operators fail to implement Emergency Procedure (E-3), "Steam Generator Tube Rupture." More specifically, operators fail to 1) isolate the faulted SG, 2) depressurize and cooldown the RCS, and 3) refill the RWST or cross-tie to the unaffected unit's RWST.

10 minutes

- Initial Operations assessment of plant status complete

15 minutes

- RCS level being maintained by HHSI, operator secures one of the three HHSI pumps per procedure
- Operator takes control of AFW to maintain level in the SGs
- When level in the faulted SG reaches the top of fill range, AFW flow will be stopped to that SG

30 minutes

- Damaged SG continues to fill, overflowing into the TDAFW pump turbine causing it to shut down
- The two MDAFW pumps provide makeup to non-faulted SGs

1 hour

- The TSC is manned and operational. The primary function of the TSC would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.

1.5 hours

- RCS and SG levels being maintained by HHSI and AFW, respectively
- Offsite EOF is manned. The primary function of the EOF would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff members are the primary users of SAMGs and mitigation measures codified in 10CFR50.54(hh). Shift supervisors and TSC supervisors are trained on these procedures.

- The TSC and EOF recognize that the damaged SG is not isolated, the MCR is not implementing E-3, and the RCS is not being cooled down and depressurized. Recommends to the MCR that they implement the following actions:
  - Implement E-3
  - Isolate the damaged S/G
  - Cooldown and depressurize the RCS

1.75 hours

- Operations assesses TSC and EOF diagnoses, concurs with their determination, and implements procedure E-3, "Steam Generator Tube Rupture"

2.5 hours

- Within 45 minutes the damaged SG is isolated, HHSI is secured, and the RCS is undergoing a normal cooldown

Event Termination

- Establish long-term cooling using the residual heat removal system (closed-circuit cooling system)
  - RCS at 400-450 psi and ~350 °F (RHR entry conditions)
  - Operators verify RCS is 30 °F sub-cooled, pressure stabilized, pressurizer level in normal band and stabilized, and non-affected SG levels in normal band and stabilized

3.4 Interfacing Systems LOCA

This sequence group is initiated by a common mode failure of both LHSI inboard isolation check valve disks. The open pathway pressurizes and ruptures the low-pressure piping outside the containment, which opens a containment bypass LOCA. This sequence group consists of the bypass LOCA followed by operator failures to refill the RWST or cross-connect to the unaffected unit's RWST. The SPAR model calculated a frequency of $3x10^{-8}$/reactor-year and the licensee's PRA calculates a frequency of $7x10^{-7}$/reactor-year.

Section 3.4.1 describes the initial status of the plant following the tube rupture. The key system availabilities during the course of the accident are summarized in Section 3.4.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.4.3. Section 3.4.4 delineates various scenarios based on the success of the mitigative actions. In particular, a mitigated scenario is defined where the mitigative actions are successful. An unmitigated scenario is defined where certain key mitigate measures are not successfully implemented.

3.4.1 Initiating Event

The interfacing systems loss-of-coolant accident (ISLOCA) initiates with a common mode rupture of both of the inboard isolation check valve disks resulting in over-pressurization and failure of low-head safety injection (LHSI) piping outside of containment in safeguards building. The resulting double-ended guillotine pipe break permits back-flow of the high-pressure RCS water into the safeguards building. Water will also spill into safeguards building via forward flow through the LHSI pumps to the pipe break. The broken 6" LHSI line has a 2.57" venturi valve between the RCS and the break that will limit the backward break flow. Although LHSI pumps are initially available until the safeguards building floods and LHI pump motors are submerged, they are ineffective because all their flow goes out the pipe break.

3.4.2 System Availabilities

The full complement of systems is considered functional in this scenario including all systems associated with engineered safeguards and instrumentation and control as well as all auxiliary and emergency systems.

3.4.3 Mitigative Actions

The SPAR model and the licensee's PRA concluded that the ISLOCA proceeds to core damage because of the above errors. However, the PRA models do not appear to have credited the significant time available for the operators to respond adequately. The PRA model also does not appear to credit technical assistance from the TSC and EOF. The realistic analysis of thermal hydraulics in Section 5.5 subsequently estimated 3 hours until the RWST is empty and 10 hours until the fission product releases begin, providing considerable time for the operators to respond. The ISLOCA time estimates are based on a double-ended pipe rupture, which drains the RWST at the maximum rate.

Based on detailed discussions of the scenario with the plant operators during the site visit and subsequent phone calls, the following operator and mitigative actions and their associated timelines were used.

- Per procedure, only two HHSI pumps are required. All three will start and one is secured within 15 minutes.
- The operators will take control of the AFW pumps to maintain normal level in the steam generators after 15 minutes.
- To minimize backflow leakage to the safeguards building, the operators will shift HHSI injection from the cold leg to the hot leg at 45 minutes. Additional HHSI pumps can be secured if an adequate water level can be maintained to minimize the spill rate into the safeguard building. The second and third HHSI pumps were secured at 2 hour and 9 hour, respectively.
- The operators will start a 100°F/hr RCS cooldown at 1 hour and continue depressurizing the steam generators to atmospheric pressure to minimize the break flow.
- The unaffected unit's HHSI pumps and RWST are aligned to the affected unit through a series of operator actions.

The following other mitigation measures were identified but not used in the calculations.

- The RWST can be refilled using firewater makeup from the firewater header at ~300 gpm from two 250,000-gallon firewater storage tanks, then from the James River.
- The portable, low-pressure, diesel-driven (Godwin) pump is available to makeup to the RWST and the CST at ~1200 gpm at 120 psi
- 190,000 gallons available from the spent fuel pool for rapid RWST makeup

- The two portable, high-pressure, diesel-driven (Kerr) pumps are available to inject into RCS using firewater at 2.5 hours (assumes guidance from TSC and EOF at 1.5 hours and an hour to implement)

3.4.4 Boundary Conditions

Section 3.4.4.1 lists the sequence of events to be prescribed in the unmitigated interfacing systems loss-of-coolant accident calculation. Section 3.4.4.2 summarizes the sequence of events in the mitigated interfacing systems loss-of-coolant accident, which credits additional operator actions.

3.4.4.1 Unmitigated Interfacing Systems LOCA

Event Initiation

- The LHSI inboard isolation check valves fail causing a pipe break in the low pressure piping in the Safeguards Building
- The reactor trips on low pressure
- Containment Phase 1 isolation auto-initiates.
- All 3 high pressure injection (HHSI) pumps auto-initiate on the ECCS injection signal.
- LHSI initiates on the ECCS injection signal, which pumps water into the Safeguards Building through the pipe break until the LHSI pump motors become submerged
- The MSIVs close
- The RCPs trip due to cavitation
- The one turbine-driven (TD) and two motor-driven (MD) auxiliary feedwater pumps automatically start on a low-level actuation signal. The initial water source is the emergency condensate storage tank (ECST) (110,000 gallons) but can be refilled from the CST, which has 300,000 gallons.

2 minutes

- LHSI motors fail when they are flooded in the Safeguards Building (approximately 2 minutes into the event)
- LHSI outboard isolation valve submerged and inaccessible.
- After the LHSI motor failures, the RWST continues to gravity drain through the pipe break in the safeguards building.

15 minutes

- Initial Operations assessment of plant status complete, LOCA identified
- RCS level being maintained with two HHSI pumps, one HHSI pump secured per procedure
- SG level being maintained by AFW

30 minutes

- Auxiliary Building sump pump alarm sounds and the two Auxiliary Building sump pumps auto-initiate, pumping water from the auxiliary building basement at a rate of ~100 gpm (50 gpm per pump). The sump pumps will continue to operate as long as auxiliary basement does not flood more that 2' above the basement floor flooding the sump pump motors.

45 minutes

- Operations transfers HHSI to RCS hot legs.

50 minutes

- The TSC is manned. Primary function would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures.
- The EOF is manned. Primary function would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigation measures. The TSC staff are the primary users of SAMGs and EDMGs. Shift supervisors and TSC supervisors are trained on SAMGs and EDMGs.

1 hour

- Operators begin 100°F/hr cooldown

1.25 hours

- The TSC is operational.

1.5 hours

- The EOF is operational.
- The TSC and EOF review and concur with actions taken by operations. Recommends shifting to the unaffected unit's RWST while in operation to prevent running out of inventory.

1.75 hours

- Secure second HHSI pump and throttle remaining pump flow to maintain water level above the core

>1.75 hours

- Operations does not successfully implement actions to shift to unaffected unit's HHSI pumps and RWST. HHSI will terminate when the RWST empties.

3.4.4.2 Mitigated Interfacing Systems LOCA

The mitigated case has an identical sequence of events until 1.75 hours, the time assessed to implement the TSC and EOF recommendations. The operator successfully initiates the following actions, starting at 1.75 hours.

1.75 hours

- Operations assesses TSC and EOF recommendation to lineup the unaffected unit's RWST to the blender to makeup to the RWST at ~ 150 gpm while continuing to provide RCS makeup with the same RWST.
- Operations swaps HHSI flow to unaffected unit's RWST. This action also uses the unaffected unit's HHSI pumps.

- The affected unit's RWST is isolated, securing drain down of RWST into the Safeguards Building. However, water continues to flow from the RCS into the Safeguards Building and is controlled by the RCS pressure and hydrostatic water head in the Safeguards Building.

  Note: The HHSI pumps could trip off line if auxiliary building is allowed to flood to approximately 5 feet above of auxiliary building basement floor without mitigation measures. The volume of the Auxiliary Building basement that will result in flooding of the HHSI pumps is 530,000 gallons. Another mitigation option is to use portable submersible pumps to pump out the Auxiliary Building basement to preclude flooding of the HHSI pumps. This option is recognized by the licensee but is not included in plant procedures.

Event Termination (Started at 6 hours in the calculation)

- Establish long-term cooling using the residual heat removal system (closed-circuit cooling system)

4. Intergrated, Self-Consistent Modeling of Severe Accidents

The Surry MELCOR model applied in this report was originally generated at Idaho National Engineering Laboratories (INEL) in 1988 [5]. The model was updated by Sandia National Laboratories (1990 to present) for the purposes of testing new models, advancing the state-of-the-art in modeling of PWR accident progression, and providing support to decision-makers at the U.S. Nuclear Regulatory Commission (NRC) for analyses of various issues that may affect operational safety. Significant changes were made during the last twenty years in the approach to modeling core behavior and core melt progression, as well as the nodalization and treatment of coolant flow within the RCS and reactor vessel. Detailed reports have been prepared to discuss this model evolution as part of the MELCOR code development program [8], and these discussions will not be repeated here. It is simply noted that the model described herein is a culmination of these efforts and represents the state-of-the-art in modeling of potential PWR severe accidents.

In preparation for the SOARCA analyses described in this report, the model was further refined and expanded in two areas. The first area is an upgrade to MELCOR Version 1.8.6 core modeling. These enhancements include:

- a hemispherical lower head model that replaces the flat bottom-cylindrical lower head model,
- new models for the core former and shroud structures that are fully integrated into the material degradation modeling, including separate modeling of debris in the bypass region between the core barrel and the core shroud,
- models for simulating the formation of molten pools both in the lower plenum and the core region, crust formation, convection in molten pools, stratification of molten pools into metallic and oxide layers, and partitioning of radionuclides between stratified molten pools,
- a reflood quench model that separately tracks the component quench front, quench temperature, and unquenched temperatures,
- a control rod silver aerosol release model, and
- an application of the CORSOR-Booth release model for modern high-burn-up fuel.

The second area focused on the addition of user-specified models to represent a wide spectrum of plant design features and safety systems to broaden the capabilities of MELCOR to a wider range of severe accident sequences. These enhancements included:

- update of the containment leakage/ failure model (see Section 4.4),
- update of core degradation modeling practices,
- modeling of individual primary and secondary relief valves with failure logic for rated and degraded conditions,
- update of the containment flooding characteristics,
- heat loss from the reactor to the containment,
- separate motor and turbine-driven auxiliary feedwater models with control logic for plant automatic and operator cooldown responses,
- new turbine-driven auxiliary feedwater models for steam flow, flooding failure, and performance degradation at low pressure,
- nitrogen discharge model for accumulators,

- update of the fission product inventory, the axial and radial peaking factors, and an extensive fission product tracking control system, and
- improvements to the natural circulation in the hot leg and steam generator and the potential for creep rupture (see Section 4.2).

The model description is subdivided into description of the vessel and reactor coolant system (Section 4.1), the natural circulation modeling (Section 4.2), the containment (Section 4.3), the containment leakage model (Section 4.4), and the auxiliary building model (Section 4.5). Finally, Section 4.6 summarizes the best modeling practices applied to accident progression analyses conducted under the SOARCA project. The best practices include discussions of the base case approach to modeling key phenomena that have significant importance to the progression of the accident and uncertainty in their response.

4.1 Vessel and Reactor Coolant System

Figure 1 illustrates the configuration of the hydrodynamic model for the Surry RCS. The model includes explicit representation of the entire reactor coolant system including each of the three reactor coolant loops, steam generators, and reactor coolant pumps, the steam lines out to the isolation valves and associated safety and power-operated relief valves. On Loop C, the pressurizer and associated safety and power-operated relief valves, and the pressurizer relief tank are modeled. Boundary conditions are used to represent the turbine pressure and feedwater flow to allow direct calculation of the nominal, full-power steady state operating conditions.

Control system models with mass and energy sources and sinks model the accumulators, the emergency core cooling systems, the main feedwater, and the motor and turbine-driven auxiliary feedwater. An extensive set of control functions are used to represent the plant control systems such as the reactor scram logic, the emergency core cooling signal, the main feedwater control and trip logic, the turbine control valve isolation logic, reactor trip logic (i.e., based on assumed trip logic during following cavitation), the containment spray actuation, the containment recirculation functions and the residual heat removal, the containment fan cooler actuation, and the plant station batteries.

Following a loss of seal cooling, water will leak through the pump seals. Under degraded accident conditions, the pumps seals could fail and create a large leak. For each pump, three flow paths model the pump seal leakage. These leak paths describe chronic leaks from the RCS pump seals that are estimated to leak at 21 gpm at full reactor pressure[15]. The leakage model is also set up to mimic the seal failures in the pump using guidance from the utility's probabilistic pump seal leakage model [16]. For example, the failure of the second stage seals was modeled to occur coincidently with loss of liquid subcooling in the RCP pump (i.e., voiding of the RCP). The model is set up to include the following leak rates for each of the three loops:

- 21 gpm nominal leakage at 15.5 MPa with failure of the seal cooling system (i.e., no AC power)
- 182 gpm at 15.5 MPa (failure of #1 and #2 seal following change to saturated conditions in pump)[7]

[7] Upon failure of the #1 seal, the #2 seal is expected to also immediately fail [Dominion, ET-CME-05-0020].

- 500 gpm at 15.5 MPa (blowout of the seal internals with flow being controlled by the Labyrinth seal upstream of the seal package)

MELCOR's choked flow model will predict the change in seal leakage flowrate as a function of pressure, quality, and liquid and gas temperature.

Figure 2 shows a detailed illustration of the reactor vessel hydrodynamic nodalization and the corresponding spatial divisions of the core. The core is represented by five concentric rings of hydrodynamic control volumes and core structures (fuel assemblies, control rods, and support structures). Each ring is divided into five vertically stacked hydrodynamic control volumes. The axial length of the core fuel cells in each ring are represented by ten COR cells in each ring. The outer ring (i.e., Ring 5) in the active fuel region is further subdivided into two regions. The inside region of Ring 5 models the peripheral assemblies of the core. The outer region of Ring 5 models the bypass region between the core shroud around the fuel and the core barrel.

Figure 3 shows the radial core profile and the five ring nodalization. The radial power profile in the center of the core is relatively flat. However, the peripheral region has a sharp decease in the assembly powers. The inner four rings provide sufficient resolution and are similarly sized to the outer ring (i.e., important for the thermal response). The 5-ring nodalization balances the objectives of representing of the radial power variations versus complexity for computational efficiency. Once core degradation begins, the 5-ring nodalization provides representation of the regional fuel collapses and blockages.

As shown on the left-hand side of Figure 2, a 6-ring by 7-axial level nodalization is used in the lower plenum, offering a detailed radial spatial representation of the bottom of the vessel and associated structures. Ring 6, which is not included in the active fuel region, represents the outer radial region beneath the vessel downcomer. Separate axial levels represent the core plate, the flow mixer, and the lower core plate. Between the core supporting structures are the support columns, which transmit the load within the core to the lower core support plate. The vessel lower head is subdivided into 10-radial by 6-azimuthal segments for a two-dimensional conduction solution. The lower head failure is evaluated using a one-dimensional mechanical response model that determines the stresses and strains in the lower head to predict creep-rupture failure. The lower head structural creep (plastic strain) failure is calculated using the default Larson-Miller lifetime damage model.

A matrix of axial and radial flow paths simulates two-dimensional flow patterns in the core region. Each flow path in the core and lower plenum nodalization simulates the effects of flow blockages and changes in resistance during core degradation. Ring 5 also uses special flow paths to represent the hydraulic openings following the failure of the core shroud if such failure is predicted.

The five ring radial hydrodynamic nodalization from the core extends upward into the upper plenum of the vessel. The upper plenum is divided into two axial levels with radial flow between each ring. Each ring also includes a representation of the guide tubes. Gas or water can flow through the control rod guide tubes between the upper plenum and the upper head. In the outer radial ring, there are three axial levels to separate the natural circulation flow outward to the hot legs (CV-154) versus the returning flow (CV-153). The leakage pathways between the downcomer and the upper plenum and from the downcomer to the upper head are also represented.

The steam generators nodalizations are shown in Figure 4 and Figure 5. The red flowpaths are only active in natural circulation conditions. Both the hot legs and the steam generator tubes are split into two halves to permit counter-current natural circulation flows (see Section 4.2). The steam generator includes explicit modeling of the primary-side tubes, the steam generator inlet and outlet plenums, the secondary side of the steam generator, the steam lines, and the safety and power-operated relief valves. The hot leg and steam generator nodalization is somewhat complicated because it must simulate conditions ranging from (a) normal operating conditions, (b) single-phase liquid and two-phase accident conditions, and (c) single-phase gas natural circulation conditions. As will be discussed in Section 4.2, special flowpaths are activated to simulate some of the natural circulation phenomena.

The model includes the heat loss from the reactor system to the containment. Each external structure of the vessel, the recirculation looping, the steam generators, and the steam lines transfers heat to the containment. These heat structures are coupled to the appropriate control volumes representing different regions of containment. The total heat loss to the containment at rated conditions is 0.08% (1.97 MW), [48] (Table 5.3-2).

4.2 Natural Circulation Modeling

Three natural circulation flow patterns can be expected during a severe accident; (1) in-vessel circulation, (2) countercurrent hot leg flow, and (3) loop natural circulation (see Figure 6 [2]). Natural circulation is important in severe accident sequences because circulating steam from the core to upper reactor internals, the hot leg, and the SGs (1) transfers heat away from the core, (2) changes the core melt progression, and (3) changes in-vessel fission product distribution. More importantly, the resultant heating of the external piping could progress to a thermal stress (i.e., creep rupture) failure of the primary pressure boundary and a resulting depressurization prior to lower head failure. For example, a high-pressure station blackout accident without a severe RCP pump seal failures is not expected to result in a loop natural circulation flow (i.e., natural circulation pattern 3 shown on the left-hand side of Figure 6) at the start of the core degradation phase of the accident. Consequently, the prediction of the first two natural circulation flow patterns is most critical [2]. The first two natural circulation flow patterns have been studied experimentally in the $1/7^{th}$-scale natural circulation test program by Westinghouse Corporation for the Electric Power Research Institute (EPRI) [10][11], computationally using the FLUENT computational fluid dynamics computer program [3][4], and with plant application analyses using SCDAP/RELAP5 [6]. Subsequently, MELCOR was used to model the $1/7^{th}$-scale natural circulation tests [14].

More recently, NRC has continued improving natural circulation modeling as part of the steam generator tube integrity program [12][13]. The natural circulation modeling techniques used in MELCOR plant models were based on work performed as part of the code assessment of the $1/7^{th}$-scale tests [Wagner, 2001], which closely followed the previous work performed by Bayless [1993]. The natural MELCOR modeling approach in the Surry model was updated for the SOARCA project to incorporate some of the recent modeling advances used by Fletcher with the SCDAP/RELAP5 severe accident analysis code [12].

The key features of the updated MELCOR natural circulation models are,

- 5 radial rings in the vessel and upper plenum for natural circulation
  - Separate axial and radial flowpaths throughout the core and upper plenum
  - Radial and axial blockage models in the core during degradation

- modeling important modes of heat transfer in the internal vessel,
  - Convective heat transfer
  - Gas-structure radiation in the upper plenum
  - Structure to structure thermal radiation within the core
  - Variable zircaloy emissivity as a function oxide layer thickness
  - Variable steel emissivities in the core as a function temperature

- hot leg counter-current natural circulation tuned to a Froude Number correlation using results from a NRC FLUENT CFD analysis [3][4],

$$Q = C_D \left[g \left(\Delta\rho / \rho \right) D^5 \right]^{1/2}$$

where g is the acceleration due to gravity.
Q is the volumetric flow rate in a horizontal duct
ρ is the average fluid density (ρ)
$\Delta\rho$ is the density difference between the two fluids
C_D is the hot leg discharge coefficient
D is the pipe hydraulic diameter

  - Hot leg split into upper and lower halves
  - C_D from FLUENT = 0.12

- Steam generator mixing fractions based on FLUENT CFD analysis [3][4]
  - Inlet plenum subdivided into 3 regions for hot, mixed, and cold regions from plume analyses
  - Flow ratio from the inlet SG plenum into the hot SG tubes is 15% from the hot, unmixed plume and 85% from the mixed region
  - Flow ratio into the cold leg piping from the inlet SG plenum is 15% from the cold SG tubes and 85% from the mixed region
  - The SG is nodalized to have 50% of the SG tubes in upflow and 50% in downflow[8]
- Modeling important modes of heat transfer in the hot leg and steam generator
  - Convective heat transfer
    - Augmented in hot leg based on FLUENT turbulence evaluations

[8] Boyd, et al., and Fletcher and Beaton [4][12] used a 41%/59% flow split of hot tubes to cold tubes in the steam generator for natural circulation conditions. For simplicity, in non-naturalcirculation conditions, a 50/50 split was used in the MELCOR model.

- Gas to structure radiative exchange in the hot leg and steam generator tubes
- Heat loss through the piping and insulation

- Steam generator tube to hot leg flow ratio tuned to results from the FLUENT CFD analysis [3][4]
  - The SG flow rate to hot leg flow rate ratio was set to a value of 2 as in the FLUENT CFD analysis

- The pressurizer and steam generator PORV and safety valves were modeled individually to prevent non-physical disruptions of natural circulation flow when they operated.[9]

- Creep rupture modeling
  - Hot leg nozzle carbon safe zone
  - Hot leg piping
  - Surge line
  - Steam generator inlet tubes

The complexities of time-varying buoyant flows are impossible to resolve using MELCOR. Consequently, special flow paths are introduced to simulate natural circulation conditions measured in experiments and calculated using computational fluid dynamics codes. The red flow paths in Figure 4 and Figure 5 show the special flow paths in the hot legs and steam generators. As indicated in the legend, special flow paths are activated during natural circulation conditions to achieve the desired flow patterns. In particular, valves and additional head/drag terms are applied to match or the desired phenomena. During natural circulation conditions (i.e., single-phase gas flow into the hot leg and steam generator), the red flow paths are activated. The result is a counter circulation flow pattern in the hot leg that matches the Froude Number correlation, a counter-current tube flow rate that is twice the hot leg flow, and 85%/15% flow mixing between the mixture and hot and cold streams entering and leaving the steam generator inlet plenum. However, if conditions change that would preclude the natural circulation flow pattern (e.g., flooding by the accumulators or an injection system, a creep rupture piping failure, operation of multiple relief valves, etc.), the control logic reactivates MELCOR's normal two-phase thermal-hydraulic model with the base nodalization (i.e., the "black" flow paths in Figure 4 and Figure 5).

4.3 Containment

The containment is divided into a total of nine control volumes and seventeen flow paths. Figure 7 and Figure 8 show the hydrodynamic nodalization of the containment. The control volumes represent the internal regions of the containment, which are identified as the basement, the cavity under the reactor, the three separate steam generator cubicles, the pressurizer cubicle, the pressurizer relief tank (PRT) cubicle, the lower dome, and the upper dome. The basement region includes the bottom part of the containment as well as the surrounding cavity that lies between the outer wall and internal containment.

[9] Previously, the valves were lumped together to simplify the modeling representation. When the valves are lumped together, it creates a very large flow that non-physically disrupts natural circulation flow patterns.

Forty heat structures simulate the containment structures. Thirteen of these heat structures are composed of carbon steel-concrete, twenty are pure concrete, and seven are carbon steel. The heat structures include rectangular, cylindrical, and hemispherical geometries depending on the actual structure that is being modeled. The carbon steel structures represent miscellaneous steel (equipment and other structures) within various control volumes in the containment. The carbon steel-concrete structures represent the major exterior walls in the containment. Cylindrical and rectangular heat structures model the outer walls of the containment that are shared with the environment (1.371 m thick), the wall separating the reactor cavity and basement (1.371 m thick), and the wall separating the pressurizer cubicle and the outer cavity (0.61 m thick), the PRT cubicle floor (0.3 m thick), the outer wall separating the upper dome from the environment (1.371 m thick), and the wall separating the lower dome from the upper dome (0.762 m thick). The containment dome has a hemispherical geometry and is approximately 0.762 m thick. The remaining heat structures in this model have a rectangular geometry and are used to model walls and floors of control volumes along with miscellaneous steel within the control volumes.

The reactor cavity is represented using special physics models for core concrete interactions (CCI). The concrete floor is a combination of limestone aggregate and common sand concrete and has a 0.135 mass fraction of iron rebar. This concrete has an ablation temperature of 1650 K and an initial temperature of 311 K. The reactor cavity is represented with a flat-bottom cylindrical cavity that has an inner radius of 4.280 m, an outer radius of 5.582 m, and a height of 1.0 m. The thickness of the concrete below the bottom of the cavity is 3.04 m.

The reactor cavity connects to the basement through a 12" diameter hole bored through the shield wall at elevation -25'-0" (centerline).[10] The centerline of this hole is 2'-7" above the containment floor.

4.4 Containment Leakage Model

Extensive research and scale model testing of reinforced and prestressed concrete containments to determine behavior at beyond design basis accident pressure has been performed in the last 25 years at SNL [37] and (Central Electricity Generating Board) CEGB [38]. Testing has shown that concrete containments start to leak at leak rates much higher than design leakage well before a large rupture or gross failure would occur. This leakage could preclude the large rupture or failure. The relationship between containment leakage and internal pressures for reinforced concrete and prestressed concrete containment model tests is described in References 4 and 5. The details of the containment performance model developed by the NRC staff for use in this analysis is described in detail in Appendix B.1. The concrete containments start to leak appreciably once the liner plate yields and tears. The rate of leakage when the liner plate yields and tears is about 10 times more than normal leakage of 0.10 percent of containment air mass per day at the containment design pressure. The leakage rate increases appreciably with further increases in test pressure. Once the rebars yield, the leakage rate is about 10-15 percent per day. Thereafter, the leakage rate continues to increase and reaches to about 60-65 percent per day when the strain in the rebars is about 1-2 percent. The containment pressure does not increase significantly after leakage rate exceeds 60-65 percent 'per day. The liner welds and concrete

[10] The containment model is based on Unit 1. Note that Unit 2 does not have this hole. Water from the basement will flow through the hole when it >2'-7" deep.

crack after rebars and liner plate yields to create a path for leakage. The leakage occurs in areas such as equipment hatch, personnel airlocks and penetrations where local strains are substantially higher than the global strains.

The results in [39] and [40] are for scale model tests of two concrete containments. Rebar and concrete crack spacing, and aggregate size can affect the leakage rates in full size containments. However, based on the results of testing and analyses presented above, it is reasonable to conclude that all concrete containments start to leak once the rebars and liner plate yield. In addition, leakage becomes excessive once the strains in the reinforced and prestressed concrete containments reach about 2 and 1 percent, respectively. Based on information from the containment model test and analyses, it is reasonable to assume that containment leakage is about one percent of the containment mass per day when the liner plate yields. This increases to 13 percent of containment mass per day when rebars yield. Similarly, a leakage rate of 62 percent can be used in severe accident analysis when the containment global strains are 1-2 percent. The uncertainty in the leakage rate can be accounted for by conservatively reducing the yield and failure pressure calculated by simplified analysis to 85 percent of the calculated value.

The location of the leakage can have a significant effect on the results of the severe accident analysis and dose rates. For instance, if the containment leakage occurs through penetrations that are located inside adjoining plant buildings, the fission product release into atmosphere would be significantly less as compared to direct leakage to the environment. Previously, some of the severe accident analyses were based on the assumption that the leakage takes place at the top of the containment dome. A more realistic approach is to consider leakage to occur at the equipment hatch, which was done in SOARCA. Leakage through the equipment hatch discharges into the environment from the side of the containment dome.

The implementation into the Surry MELCOR model uses two containment failure mechanisms.

1. Nominal leakage per design specifications - 0.1% volume/day @ P_{Design}), see Figure 9.
2. Containment overpressure leakage as described above - see Figure 10.

The nominal leakage is always active but very small. The containment overpressure failure occurs at 2.17 times the design pressure, or 0.775 MPa (112.4 psia). This estimate of 2.17 times the design pressure is derived from a curve fit of the three data points shown in Figure 10. The leakage starts very small but grows as the pressure increases. If the containment pressure subsequently decreases, the leakage area will not decrease from the maximum value.

4.5 Auxiliary Building

A total of 9 control volumes and 17 flow paths represent the Auxiliary Building (see Figure 11). The auxiliary building is modeled on a floor-by-floor basis beginning with the basement floor and rising up through the main floors up to the fourth floor. The first floor, at a 2'0" elevation, is broken up into four control volumes. The first floor is subdivided to represent major rooms at that elevation. The HHSI pumps and motors are located at this elevation. The second floor, at 13' elevation, is divided into 3 control volumes. A large room in the middle of the second floor contains boric acid transfer pumps as well as part of the boric acid tanks. The other two rooms contain the cable vault, electrical tunnel, and electrical vault. The middle room is connected to the side rooms by doorways. This floor also connects with the first floor by three separate stairwells. The third floor, at a 27'-6" elevation, is represented by a single control volume. The third floor contains the volume control tanks and part of the boric acid tanks. The third floor connects with the second floor by the stairwell located next to the elevator. The fourth floor, at a 45'-10" elevation is also represented by a single control volume. The fourth floor is where the personnel hatches are located along with the heating and ventilation equipment. There are many potential leakage locations to the environment on the fourth floor through ventilation ducting and the blowout panels. The leakage is represented as a 0.65 m^2 (7 ft^2) flow path to the environment.

The auxiliary building model contains 39 heat structures. These heat structures represent the floors, walls, ceilings and miscellaneous steel structures present in the auxiliary building. The floors, ceilings and walls are rectangular in geometry and are composed of concrete. The miscellaneous steel masses were estimated from calculations performed by measuring the dimensions of some of the equipment in the rooms.

4.6 Best Modeling Practices

At the start of the SOARCA project, the scope of the severe accident modeling effort was formulated. The primary objective of the SOARCA project is to provide a best estimate prediction of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective, the SOARCA project utilizes integrated modeling of the accident progression and off-site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community.

The MELCOR 1.8.6 computer code [8] is used for the accident analysis. MELCOR includes capabilities to model the two-phase thermal-hydraulics, core degradation, fission product release, transport, and deposition, and the containment response. The SOARCA analyses include operator actions and equipment performance issues as prescribed by the sequence definition and mitigative actions. The MELCOR models are constructed using plant data, and the operator actions were developed based on table-top exercises performed with operators during site visits. The code models and user-specified modeling practices represent the current best practices.

The progression of events in a severe accident has uncertainties. The uncertainties are addressed in probabilistic risk assessments by evaluating the conditional probabilities of alternative phenomena and their responses, alternative equipment responses, or alternate operator actions. The conditional likelihood of the events or responses is evaluated, which leads to multiple possible outcomes. Consistent with the best-estimate approach in SOARCA, the most likely progression(s) of events was defined and evaluated. Through the process of defining a best-estimate accident progression that met the goals of the program, two important historical phenomena that could lead to early containment failure and release of fission products were not included. The two issues, steam explosions and direct containment heating, have been studied extensively and found to have very low likelihoods of occurrence [18]. Hence, they were not included in the SOARCA program. Section 4.6.1 summarizes the phenomena and some of the significant research that led the SOARCA program to neglect their inclusion.

Next, the accident progression analysts developed a list of other important uncertain phenomena that were likely. The issues and the proposed modeling approach were presented to an external expert peer review group during a public meeting sponsored by the NRC on August 21-22, 2006 in Albuquerque, New Mexico. Section 4.6.2 provides the recommended modeling approach or base case values for the uncertain issues. The MECLOR Best Modeling Practices [7] provides a more detailed discussion of the modeling approach.

For uncertainties in operator actions and recently added emergency mitigative equipment, the SOARCA program examined two possible responses. In the unmitigated response, some operator actions might be initially credited. However, the emergency equipment or a key operator action was not successful. This led to a severe accident that was generally consistent with classical severe accident evaluations of the sequence of events (i.e., see [17]). Alternately, a calculation was performed where the equipment or operator action was successful and the accident was partially or fully mitigated. As discussed in the mitigative measures discussion, it was beyond the scope of SOARCA to quantify the relative likelihood these outcomes.

Finally, a systematic evaluation of phenomenological uncertainties for a particular sequence is a separate task and not discussed in this report. The study will evaluate the importance and impact of alternative settings or approaches for key uncertainties.

4.6.1 Early Containment Failure Phenomena

The objective of SOARCA is to perform best-estimate evaluations of the accident progression and consequences from the most likely severe accident sequences for specific plants. Two phenomenological issues not included in the best-estimate approach used in SOARCA include (1) alpha-mode containment failure and (2) direct containment heating leading to containment failure. These severe phenomena leading to an early failure of the containment were included in some of the first studies to quantify the risks from nuclear reactors.

The alpha-mode event is characterized by the supposition that an in-vessel steam explosion might be initiated during core meltdown by molten core material falling into the water-filled lower plenum of the reactor vessel. The concern was that the resulting steam explosion could impart sufficient energy to separate the upper vessel head from the vessel itself and form a missile with sufficient energy to penetrate the reactor containment. This of course would produce an early failure of the containment building at a time when the largest mass of fission products is released from the reactor fuel. In the following years, significant research was focused on characterizing and quantifying this hypothesized response in order to attempt to reduce the significant uncertainty. A group of leading experts ultimately concluded in a position paper published by the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations that the alpha-mode failure issue for Western-style reactor containment buildings can be considered resolved from a risk perspective, posing little or no significance to the overall risk from a nuclear power plant.

Similarly, direct containment heating was another important event identified to cause early containment failure. NUREG-1150 [1] was an important risk study that included DCH as an early containment failure phenomenon. Extensive research was performed characterizing DCH as well as other phenomena that can preclude an early, energetic failure of the containment (e.g., natural circulation leading to creep rupture of the RCS boundary, see Section 4.2). First, the extensive natural circulation research shows that RCS failure prior to vessel failure due to RCS creep rupture is most likely. In the unlikely event there is a high-pressure vessel failure (i.e., not within SOARCA's objectives for best-estimate consequence evaluations), the resolution of the DCH issue found early containment failure to be very unlikely [18]. The issue resolution utilized a probabilistic framework that decomposes the DCH problem into three probability density functions that reflect the most uncertain initial conditions (UO_2 mass, zirconium oxidation fraction, and steel mass). Uncertainties in the initial conditions are significant, but the quantification approach established reasonable bounds that are not unnecessarily conservative. The phenomenological models in the probabilistic model were compared with an extensive database including recent integral simulations at two different physical scales (1:10-scale in the Surtsey facility at Sandia National Laboratories and 1:40-scale in the COREXIT facility at Argonne National Laboratory). The loads predicted by these models were significantly lower than those from previous parametric calculations. The containment load distributions do not intersect the containment strength (fragility) curve in any significant way, resulting in containment failure probabilities less than 10^{-3} for all scenarios considered. Sensitivity analyses did not show any areas of large sensitivity. Consequently, DCH is not a likely accident progression event and therefore not within SOARCA's best-estimate approach guidelines.

4.6.2 Base Case Approach on Important Phenomena

A review of severe accident progression modeling for the State-of-the-Art Reactor Consequence Analysis (SOARCA) project was conducted at a public meeting in Albuquerque, New Mexico on August 21-22, 2006 [9]. This review focused primarily on best modeling practices for the application of the severe nuclear reactor accident analysis code MELCOR for realistic evaluation of accident progression, source term, and offsite consequences. The scope of the meeting also included consideration of potential enhancements to the MELCOR code as well as consideration of the SOARCA project in general.

The review was conducted by five panelists with demonstrated expertise in the analysis of severe accidents at commercial nuclear power plants. The panelists were drawn from private industry, the Department of Energy national laboratory complex, and a company working on behalf of German Ministries. The review was coordinated by Sandia National Laboratories and attended by Nuclear Regulatory Commission staff.

The following important uncertain modeling practices and their baseline approach were identified to the SOARCA modeling practices peer review panel. For practicality due to the complexity of running severe accident calculations, base case approaches were identified for these uncertain and typically important parameters. A separate task in the SOARCA program is planned to address the importance of uncertainties in these modeling parameters.

- **Safety relief valve cycling and failure**

Mean opening and reclosing failure probabilities for the pressurizer and steam generator power operated relief (PORV) and safety valves (SV) were applied in the calculations. A high temperature thermal failure model was also applied.

- **Pump seal leakage and blowout**

The base case pump seal leakage model described in Section 4.1 was identified as the base case modeling approach.

- **Loop seal clearing and effects on the accident progression**

The most important impact from this event is an increased vulnerability of the steam generator tubes for failure due to a full loop circulation of hot gases from the core during core degradation. MELCOR has basic thermal-hydraulic model for calculating loop seal clearing. However, it is recognized that loop seal clearing is related to other complex and uncertain events, such sensitive system hydrodynamic pressure balances during core degradation events and pump seal leakage. NRC has a separate research program with examining thermally-induced steam generator tube failure. Due to the potential importance of steam generator tube failure (i.e., the most important consequence of loop seal clearing), sensitivity calculations were performed that included steam generator tube failure.

- **Fuel degradation and relocation treatment**

An additional model has been added to characterize the structural integrity of the fuel rods under highly degraded conditions. The new model acknowledges a thermal-mechanical weakening of the oxide shell as a function of time and temperature. As the local cladding oxide temperature increased from the Zircaloy melting temperature (i.e., represented as 2098 K in MELCOR) towards 2500 K, a thermal lifetime function accrues increasing damage from 10 hours to 1 hour until a local thermo-mechanical failure.

- **Lower plenum debris/coolant heat transfer**

Following the fuel-debris slump into the lower plenum, there may be fuel-coolant interactions. The lower plenum heat transfer settings were updated to reflect the end-state thermal condition of the debris in the deep pool FARO tests (i.e., significant thermal interaction with the water). The resultant behavior resulted in debris cooling if there was pool in the lower plenum. The subsequent heat up of the vessel lower head was delayed heat until the overlying water evaporated.

- **Core plate failure**

The timing of core plate failure affects the relocation of the degraded core materials from the core region into the lower plenum. The local thermal-mechanical failure of the lower core plate, the flow mixer plate, and the lower support forging are calculated within MELCOR using the Roark engineering stress formulae. The yield stress is calculated based on the loading and local temperature.

- **Fission product release, speciation, and volatility**

First, a new ORNL-Booth fission product release model is used that was adjusted to match the measured responses from the VERCORS Test 4. VERCOS Test 4 is representative of modern, high burn-up fuel. The previous default model was not representative of the high burn-up release physics.

Second, the predominant speciation of cesium was changed based on detailed chemical analysis of the deposition and transport of the volatile fission products in the Phebus facility tests. The chemical analysis revealed molybdenum combined with cesium and formed cesium molybdate. Previously, the default predominant chemical form cesium was cesium hydroxide. As consistent with past studies, all the released iodine combines with the cesium.

- **RCS natural circulation treatment**

The base case RCS natural circulation models described in Section 4.2 were identified as the base case modeling approach.

- **Vessel lower head failure and debris ejection**

The base case approach of modeling the vessel lower head failure and debris ejection includes some in MELCOR. First, all the solid debris in the lower plenum is in contact with water, if present. Previously, a restrictive one-dimensional counter-current flooding limitation criterion prevented penetration of water into the debris bed. Second, the vessel lower head fails using creep rupture model. A Larson-Miller failure criterion is calculated based on the one-dimensional conduction and stress profile through the lower head. The failure of a lower head penetration prior to gross head failure was judged unlikely based on observations from experimental studies at Sandia National Laboratories lower head failure (LHF) tests.

- **Ex-vessel phenomena - CCI**

The default model's ex-vessel debris surface heat flux to an overlying pool of water was enhanced to replicate the magnitude observed the MACE tests. The default model did not include multi-dimensional effects of fissures, other surface non-uniformities, and side heat fluxes.

- **Ex-vessel phenomena - Hydrogen combustion**

The default MELCOR ex-vessel combustion model was used with the modeling options to include horizontal and vertical propagation of burns and the time delay for the flame front to span the width of the control volume.

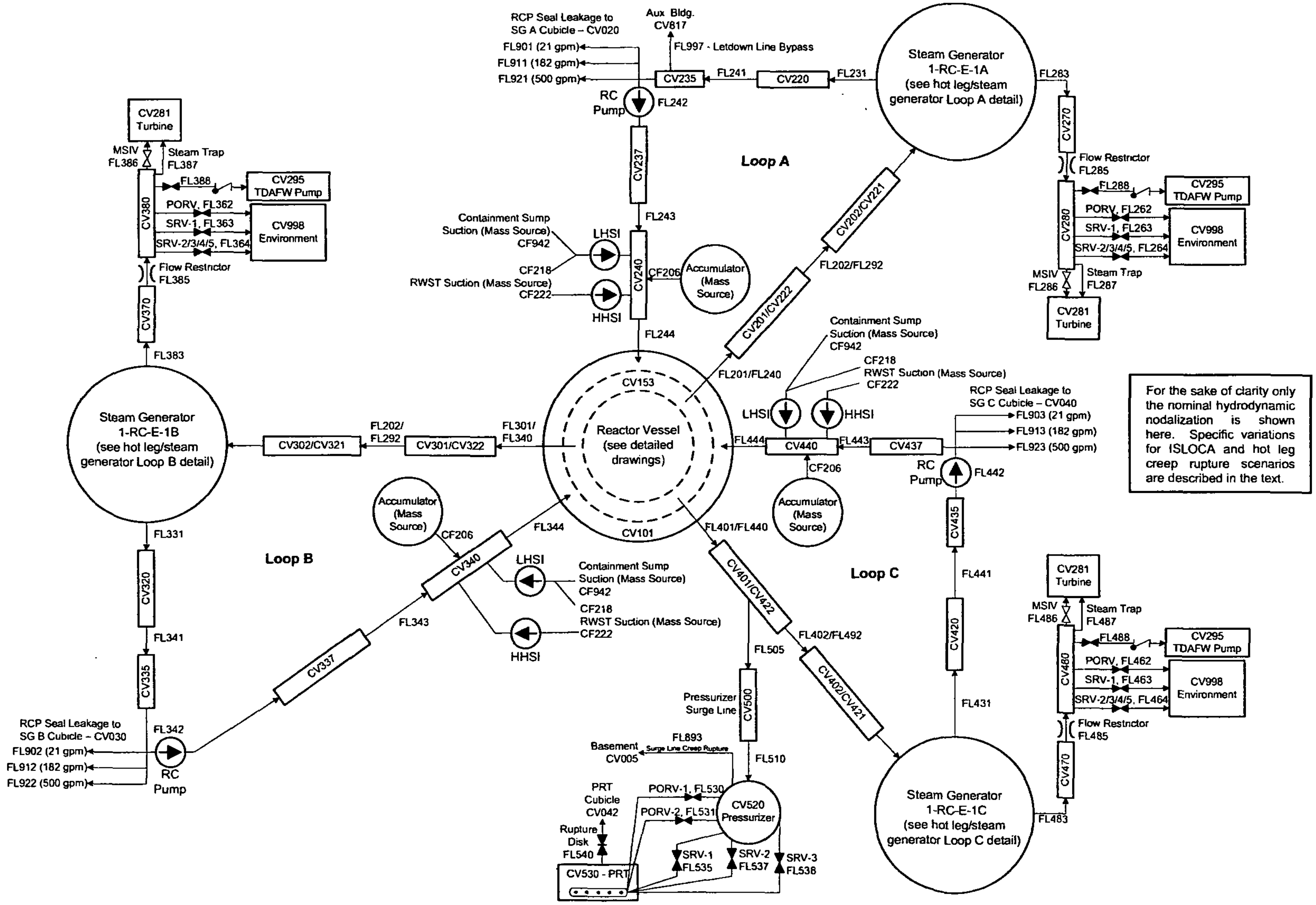


Figure 1 The Surry Reactor Coolant System Hydrodynamic Nodalization.

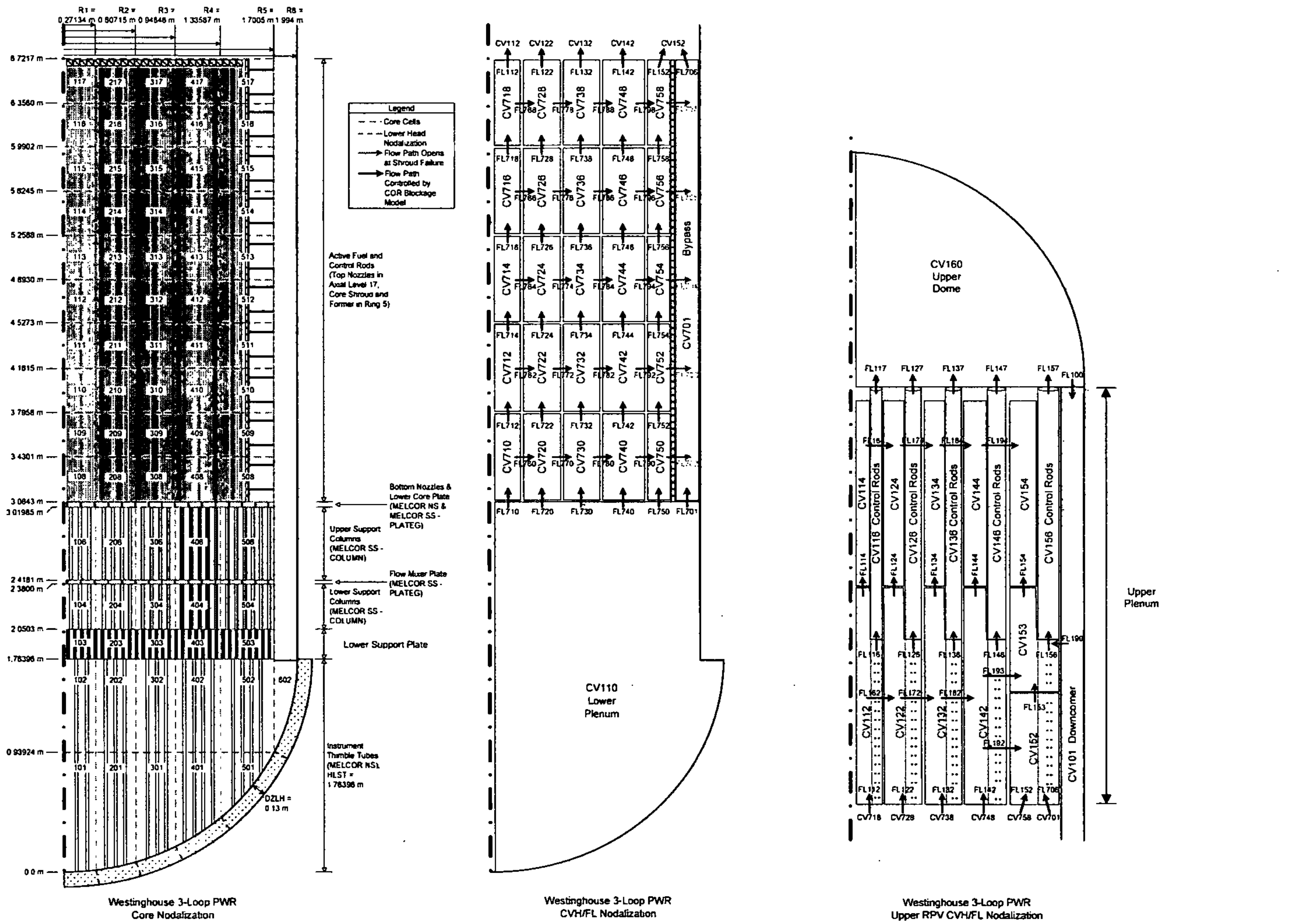
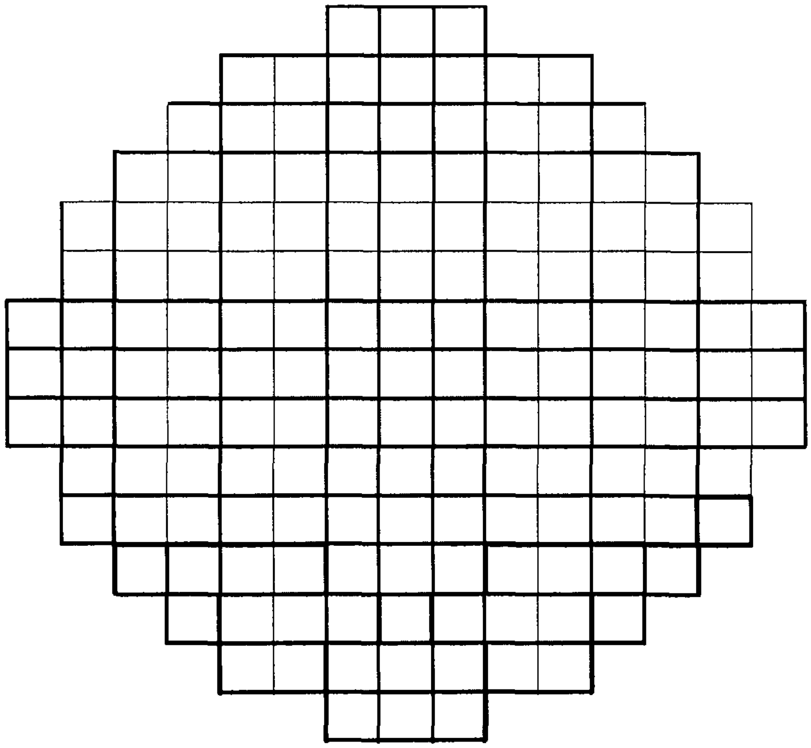


Figure 2 The Surry Reactor Vessel Core, Lower Plenum, and Upper Plenum and Steam Dome Hydrodynamic and COR Structure Nodalization.



MELCOR COR Radial Ring 1, Power Factor = 1.226

MELCOR COR Radial Ring 2, Power Factor = 1.293

MELCOR COR Radial Ring 3, Power Factor = 1.301

MELCOR COR Radial Ring 4, Power Factor = 1.110

MELCOR COR Radial Ring 5, Power Factor = 0.321

Figure 3 The Surry Reactor Core Radial Power Profile and Nodalization.

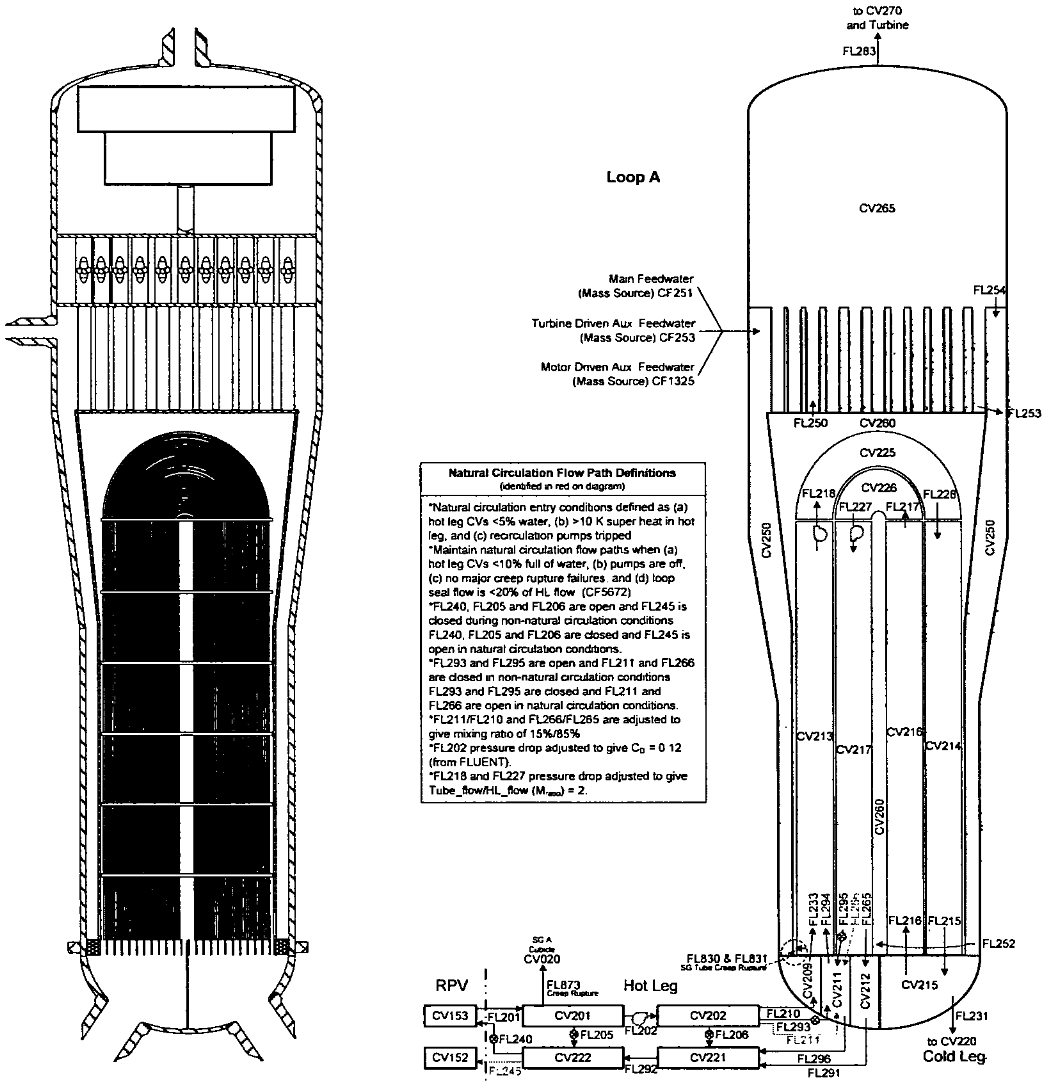


Figure 4 The Surry Steam Generator A Hydrodynamic Nodalization.

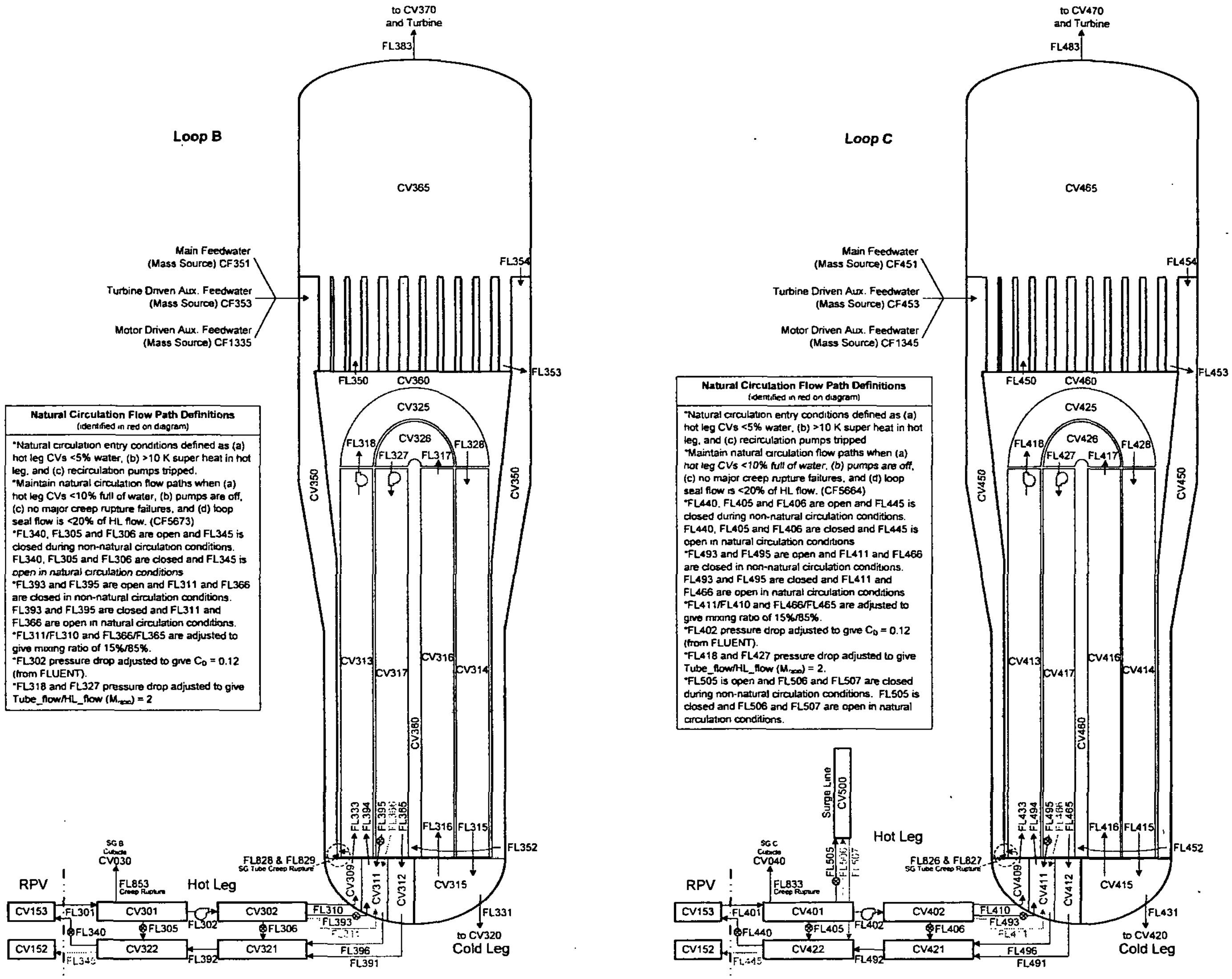


Figure 5 The Surry Steam Generator B and C Hydrodynamic Nodalization.

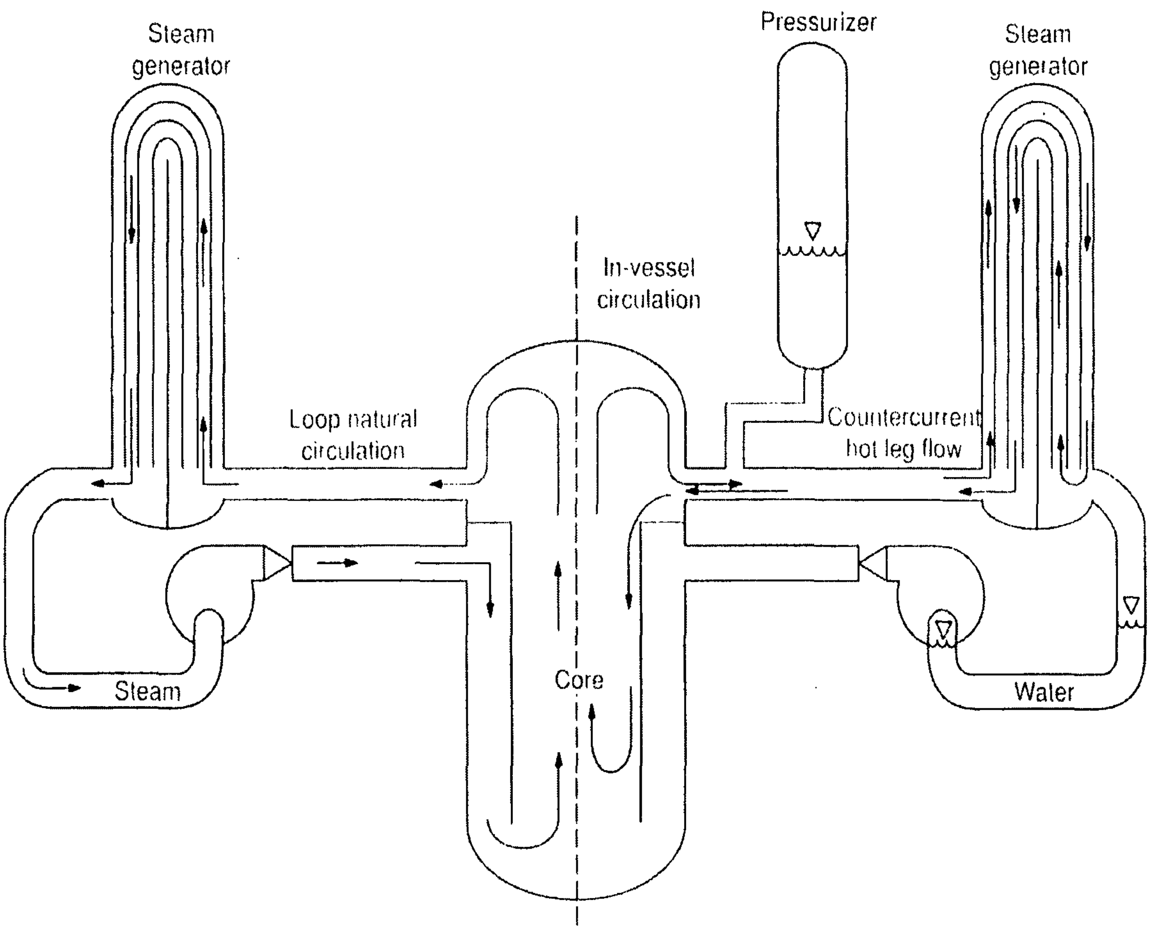


Figure 6 Natural Circulation Flow Patterns in a PWR.

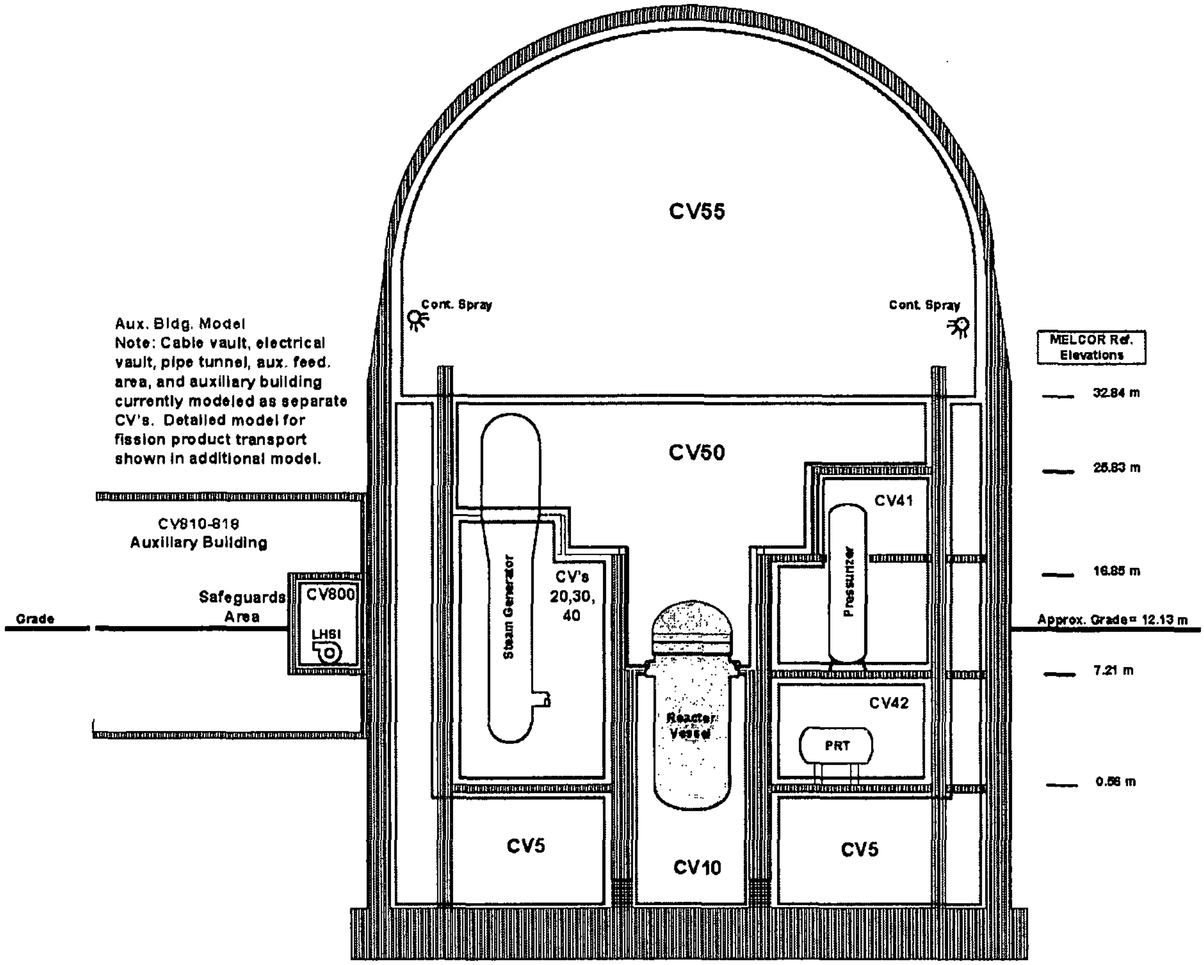


Figure 7 Containment Hydrodynamic Nodalization.

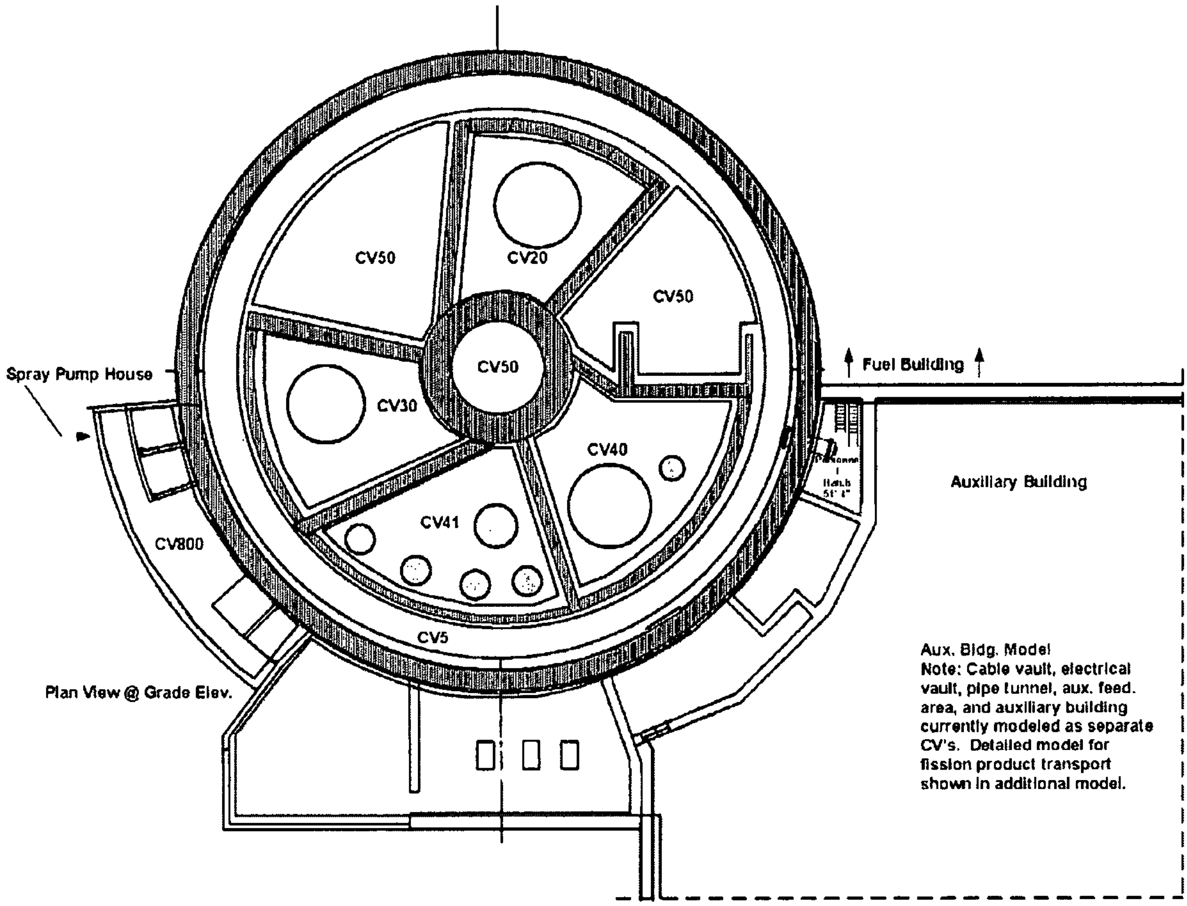


Figure 8 Containment Hydrodynamic Nodalization, Plan View.

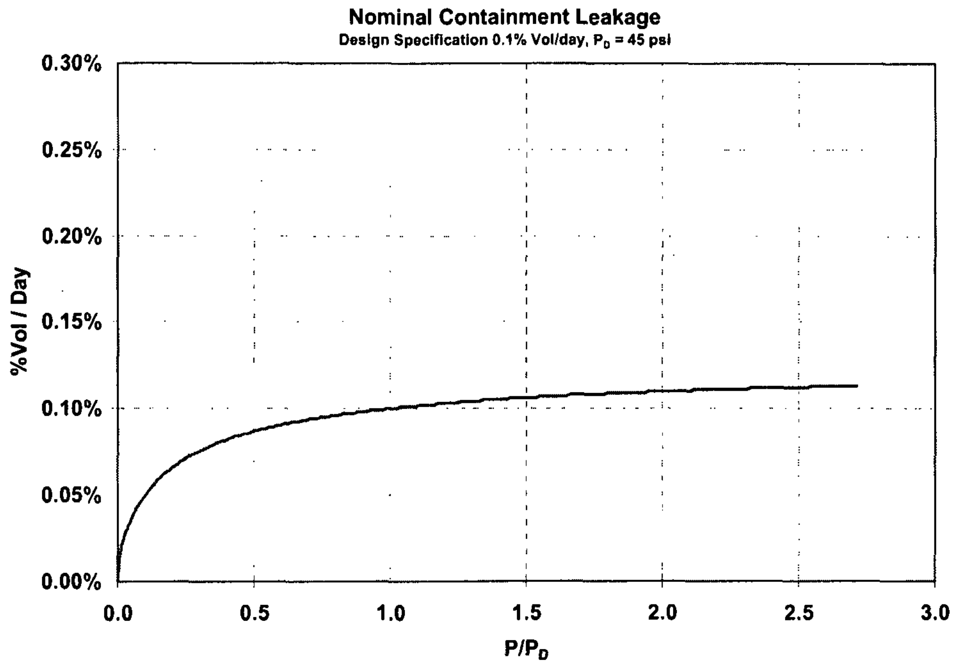


Figure 9 Nominal Containment Leakage Model.

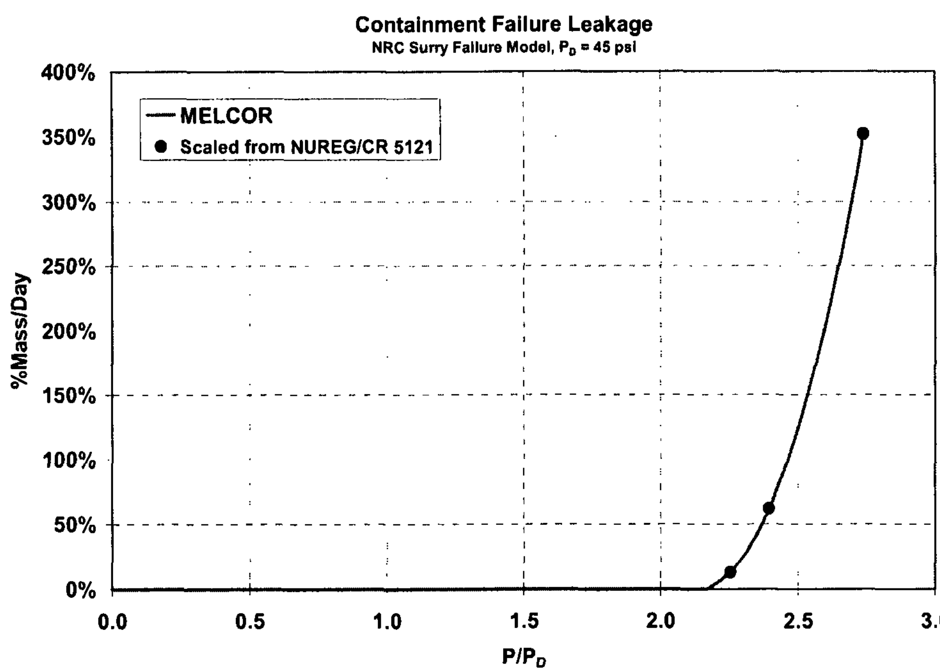


Figure 10 Containment Failure Leakage Model.

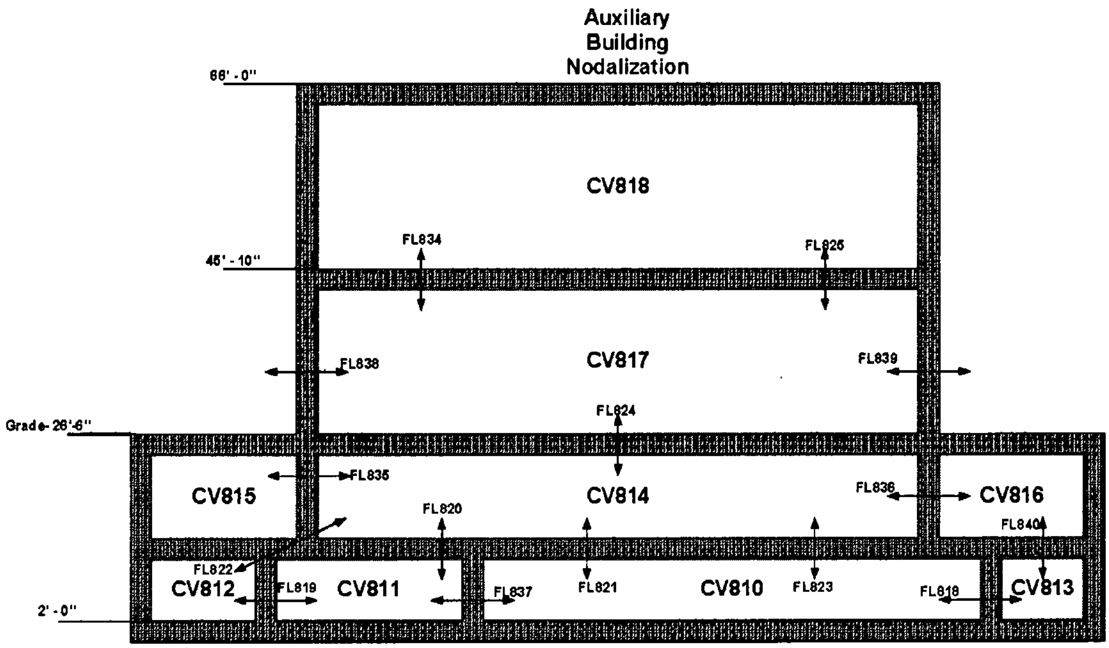


Figure 11 Auxiliary Building Hydrodynamic Nodalization.

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5. Integrated Thermal Hydraulics, Accident Progression, and Radiological Release Analysis.

This section describes the intergrated self-consisteeent analusis of each scenario using the MELCOR code. The analusus includes calculations to confirm the table-top exercise resilte that the timing and capacity of mitigation measures are sufficient to prevent core damage or delay or reduce fission product releases. This analysis also includes sensitivity calculations without B.5.b mitigation measures. Versoin 1.8.6YR of the MELCOR sever accident analysis code was used for the Surry analysis [8].

5.1 Long-Term Station Blackout

The long-term station blackout is assumed to be initiated by a beyond-design-basis seismic event. Section 5.1.1 presents the results of an unmitigated scenario with initially successful operator actions to depressurize the RCS and maintain TD-AFW flow. However, once the DC station batteries fail at 8 hours, no more operator actions are successful. For the mitigated scenario in Section 5.1.2, a portable emergency pump is connected to the RCS at 3.5 hours and a continuous supply of water is maintained.

The Surry SPAR model assessed an early RCP seal failure with the credited operator actions as less likely than a late failure (i.e., 20% versus 80% likely).

5.1.1 Unmitigated Long-Term Station Blackout

Table 1 summarizes the timing of the key events in the unmitigated LTSBO. As described in Section 3.1, the accident scenario initiates with a complete loss of all onsite and offsite AC power but the DC station batteries are available. The reactor successfully scrams and the containment isolates but all powered safety systems are unavailable except the TD-AFW. The timings of the key events are discussed further in Sections 5.1.1.1 and 5.1.1.2. However, it is worth noting that the fission product releases from the fuel do not begin until 16 hr and significant fission product releases to the environment do not begin until after 45 hr.

Table 1 The timing of key events for unmitigated long-term station blackout.

Event Description	Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC power	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump	00:00
TD-AFW auto initiates at full flow	00:01

Event Description	Time (hh:mm)
First SG SRV opening	00:03
Operators control TD-AFW to maintain level	00:15
Operators initiate controlled cooldown of secondary at ~100°F/hr	01:30
Vessel water level drains into upper plenum	01:57
Initial minimum vessel water level	02:30
Accumulators begin injecting	02:25
SG cooldown stopped at 120 psig to maintain TD-AFW flow	03:35
Emergency CST empty	05:08
DC Batteries Exhausted	08:00
S/G PORVs reclose	08:00
Pressurizer SRV opens	13:06
PRT failure	13:40
Start of fuel heatup	14:16
RCP seal failures (calculated)	14:46
First fission product gap releases	16:04
Creep rupture failure of the C loop hot leg nozzle	17:06
Accumulator empty	17:06
Vessel lower head failure by creep rupture	21:08
Debris discharge to reactor cavity	21:08
Cavity dryout	21:16
Containment at design pressure (45 psig)	28:00
Start of increased leakage of containment (P/P_{design} = 2.18)	45:32

5.1.1.1 Thermal-hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 12. At the start of the accident sequence, the reactor successfully scrams in response to the loss of power. The main steam line and containment isolation valves close in response to the loss of power. The reactor coolant and main feedwater pumps also trip due to the loss of power. In response to the loss of the main feedwater, the turbine-driven auxiliary feedwater automatically starts. The TD-AFW initiates at full flow but is subsequently controlled by the operator after 15 min to maintain level. The TD-AFW restores the steam generator liquid levels by about 30 min and is throttled thereafter. After the closure of the main steam isolation valves, the secondary system quickly pressurizes to safety relief valve opening pressure, which causes the safety relief valves to open and then subsequently close when the closing pressure criterion is achieve. The relief flow through the SG SRVs is the principle primary system energy removal mechanism in the first 90 min.

The heat removal through the steam generator depressurizes the primary system to 10.3 MPa by 90 min. At 90 min, the operator starts a controlled (100°F/hr) cooldown of the primary system by venting the steam generator power-operated relief valves (PORVs). As the secondary pressure decreases, the saturation temperature of the water in boiler section of the steam generator also decreases, which cools the primary system fluid. At about 3.5 hr, the steam generators reached 0.93 MPa (120 psig), where the secondary system pressure was stabilized. The TD-AFW can achieve full flow (700 gpm) at 600 psig, but degrades thereafter. It is described to work below 600 psig with an estimated lower limit of operability at 120 psig. Even with degraded performance at 120 psig, the TD-AFW adequately maintained the steam generator level until 5 hr 8 min when the emergency condensate storage tank (ECST) empties. In the unmitigated sequence, no operator actions were credited to replenish the ECST inventory. After 5 hr 8 min, the steam generator level starts to decrease and is empty by 12 hr 18 min.

By depressurizing the primary system to 120 psig via the secondary system cooldown, several beneficial results were achieved. First, the leakage through the RCP seals decreased from 21 gpm per pump at full operating pressure conditions to less than 7 gpm per pump. Furthermore, if a RCP seal should fail under these conditions, then the resulting leakage flow would be much lower than if the primary system pressure was not actively controlled to low pressure. Second, the accumulators begin injecting at 600 psig (4.1 MPa). The accumulators are a source of cold water to replace the losses due to RCP seal leakage and the volume shrinkage during the cooldown. By 8 hr, about 4500 gal had been discharged from each accumulator, or about two-thirds of the water inventory. Consequently, as shown in Figure 13, the inventory loss was minor during the first 8 hr.

At 8 hr, the station batteries were estimated to fail. At the same time, the steam generator relief valves closed and were no longer actively controlled. In response to the steam generator valve closure, both the primary and secondary systems rapidly pressurized to the secondary safety relief valve opening pressure. The primary system remained just above this pressure until about 12 hr 18 min, when the steam generators boiled dry. Subsequently, the primary system pressurized to the pressurizer safety relief valve opening set point and began to relieve steam and water. The fluid in the vessel heated to saturation conditions and then swelled in response to the heatup. Once the pressurizer safety relief valves began cycling, a significant amount of fluid is vented out of the RCS and the vessel level dropped quickly (see Figure 13). The top of the fuel was uncovered by 14.3 hr and the core heatups began (see Figure 14).

Shortly after the start of the core uncovery, the RCP seals failed when saturated water started flowing through the loop seals. The effective leak rate increased from 21 gpm at full operating pressure and temperature to 182 gpm at full operating pressure and temperature.[11] Once the two-phase water level drops below the core plate, the decrease in the vessel two-phase level slows because the water level is below the bottom of the fuel (see Figure 13).

[11] The leak model is tuned to these values at normal operating conditions. In a transient calculation, the leakage flow rate changes as a function of subcooling, quality, and pressure.

Similar to the STSBO (see Section 5.2.1), an in-vessel natural circulation flow develops between the hot fuel in the core and the cooler structures in the upper plenum. Hot gases rise out of the center of the core rise into the upper plenum and return down the cooler peripheral sections of the core. Simultaneously, a natural circulation circuit develops between the vessel and the steam generator. Due to its close proximity to the hot gases exiting in the vessel, the hot leg nozzle at the carbon steel interface region to the stainless steel piping was first predicted to fail by creep rupture at 17 hr 6 min.[12]

Following the accumulator injection, the decay heat from the fuel boiled away the injected water. By 18.2 hr, a large debris bed had formed in the center of the core. The debris bed continued to expand until 18.9 hr when all the fuel had collapsed and was resting on the core plate. The hot debris failed the core plate and fell onto the lower core support plate, which failed at 19.9 hr. Following the lower core support plate failure, the debris relocated onto the lower head. The small amount of remaining water in the lower head boiled away. As shown in Figure 15, the hot debris quickly heated the lower head to above the melting temperature of stainless steel (i.e., 1700 K) on the inside surface. As the heat transferred through the lower head, it eventually failed at 21 hr 8 min due to the creep rupture failure criterion (i.e., primarily due the thermal stress component due to the low differential pressure).

By 21.3 hr, nearly all the hot debris relocated from the vessel into the reactor cavity under the reactor vessel. The hot debris immediately boiled away the water in the reactor cavity started to ablate the concrete. The ex-vessel core-concrete interactions (CCI) continued for the remainder of the calculation, which generated non-condensable gases. In addition, the hot gases exiting the reactor cavity and the radioactive heating from airborne and settled fission products steadily evaporated the water on the containment floor outside the reactor cavity from 21.1 hr to 67 hr. The resultant non-condensable and steam production pressurized the containment (see Figure 16). At 45.5 hr, the containment failed due to liner tearing near the containment equipment hatch at mid-height in the cylindrical region of the containment. The containment continues to pressurize until the leakage flow balanced the steam and non-condensable gas generation. By 67 hr (2.8 days), all the water on the floor has evaporated. The containment depressurized thereafter due to only a smaller gas loading from the non-condensable gas generation. The conservatively assumed failure location was the around the equipment hatch, which is located on the side of the containment without a surrounding building (e.g., the auxiliary or safeguards buildings) other locations such as personnel airlocks and penetrations would result in lower releases due to a transportamd deposition inside adjacent buildings. Consequently, any released fission products are released directly to the environment.

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[12] Alternate failure locations could include the pressurizer surge line and the steam generator tubes. In the MELCOR calculation, the RCP seals had failed so hot gases were no longer flowing out the pressurizer when the core exit temperatures were hottest. Due to the relatively high pressure in the steam generators' secondary side, the resultant thermo-mechanical stresses across the steam generator tubes were less severe than the hot leg nozzle. Consequently, the most vulnerable location was calculated to be the hot leg nozzle. The initial failure location for this scenario is also part of an on-going investigation of another research program in the NRC.

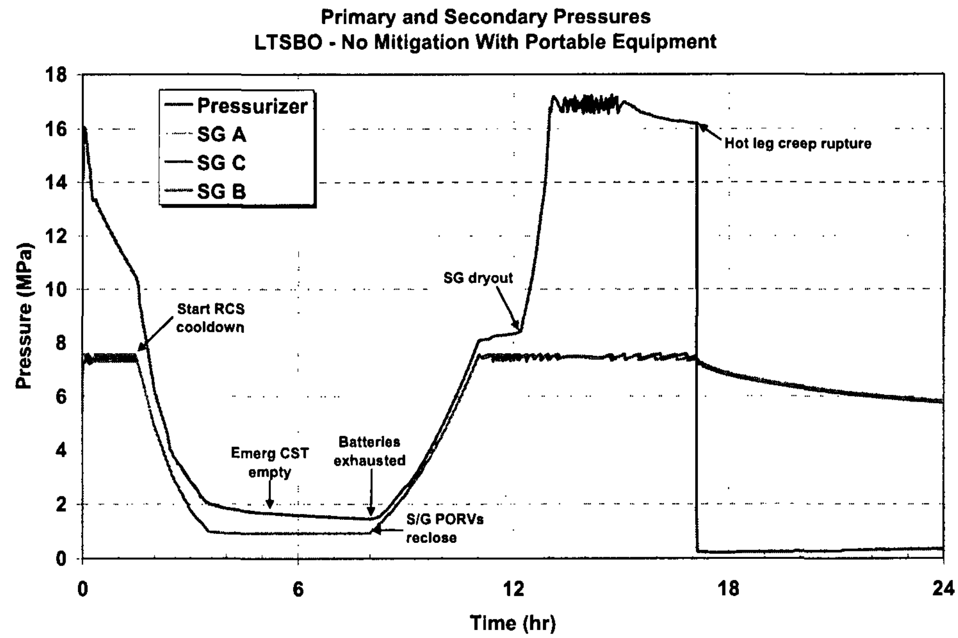

Figure 12 Unmitigated LTSBO primary and secondary pressure history.

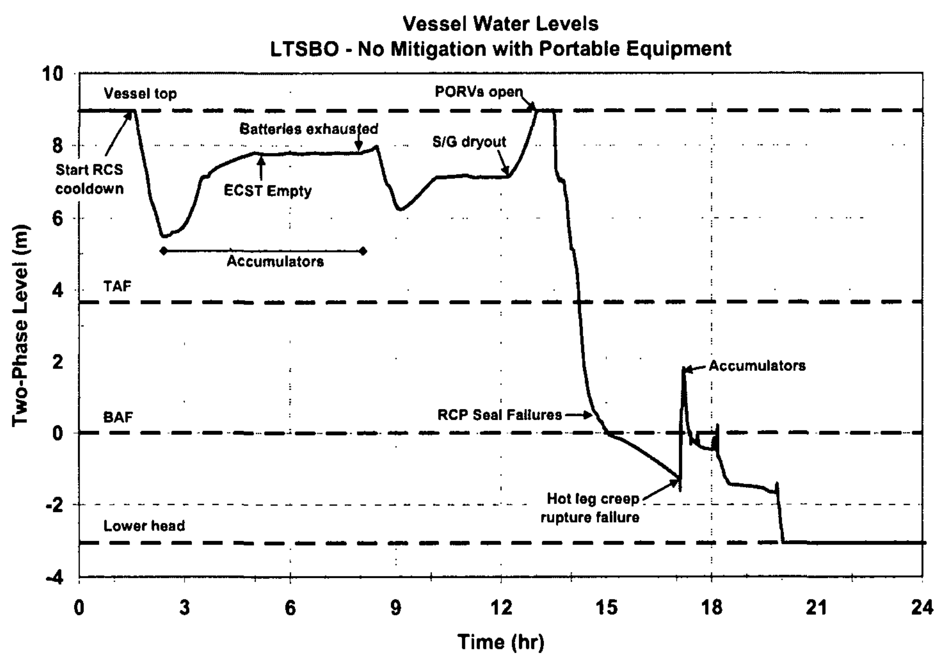

Figure 13 Unmitigated LTSBO vessel two-phase coolant level history.

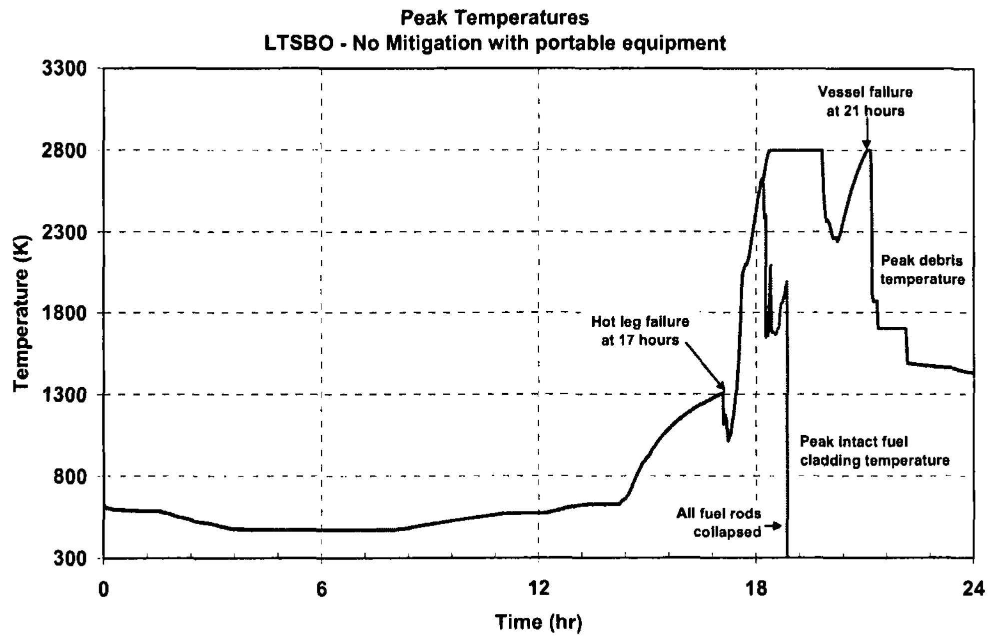

Figure 14 Unmitigated LTSBO core temperature history.

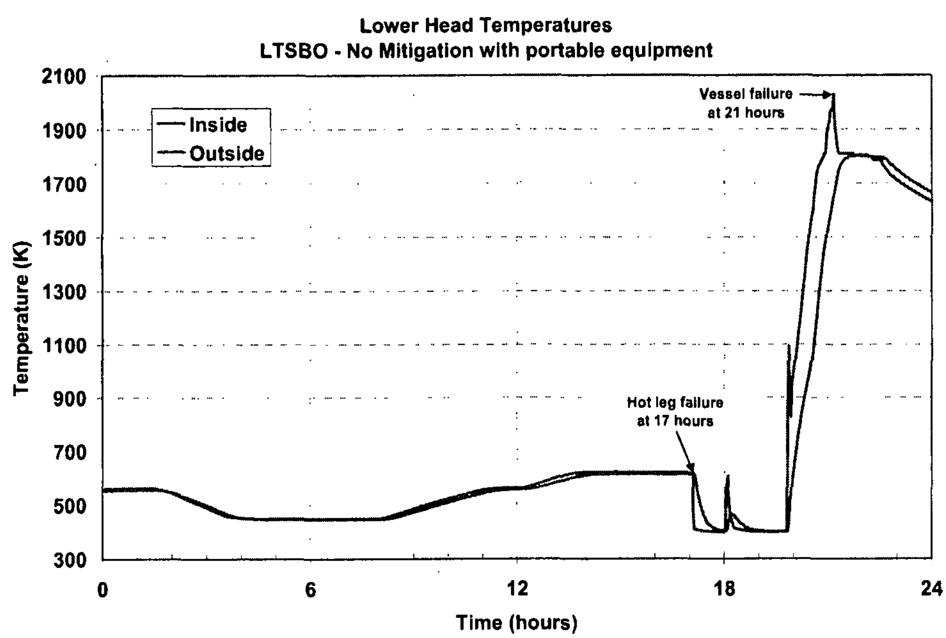

Figure 15 Unmitigated LTSBO lower head inner and outer temperature history.

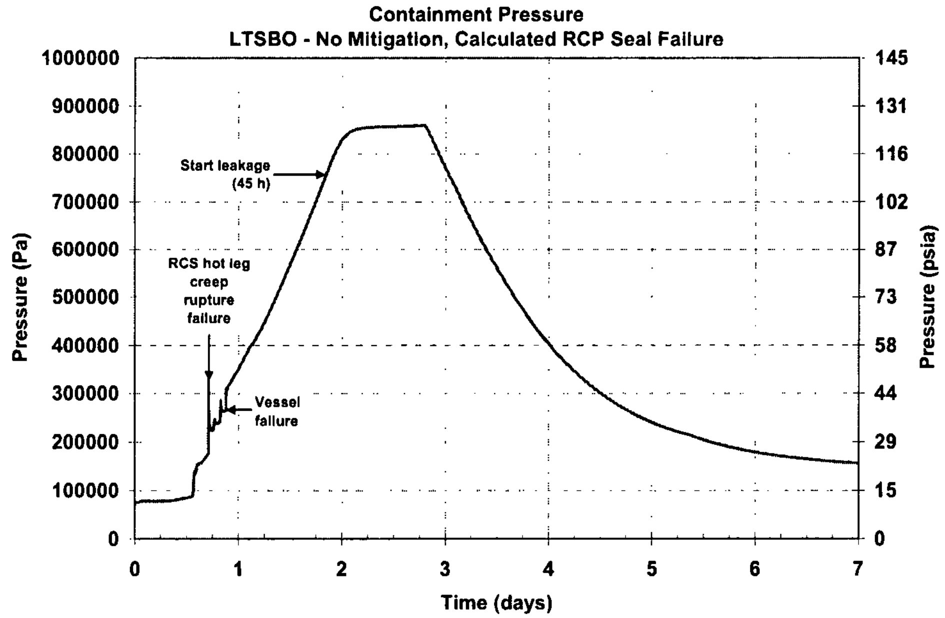

Figure 16 The unmitigated long-term station blackout containment pressure history.

5.1.1.2 Radionuclide Release

The fission product releases from the fuel started following the first thermo-mechanical failures of the fuel cladding in the hottest rods at 16 hr 4 min, or about 1 hr 40 min after the uncovery of the top of the fuel rods. The in-vessel fission product release phase continued through vessel failure at 21.1 hr. Initially, the fission product releases from the fuel circulated through the primary system as well as being released to the containment through the pressurizer safety relief valves. Since the pressurizer relief tank rupture disk had failed about 30 min before the start of the fission product releases, there was essentially no ex-vessel scrubbing in the PRT. Following vessel failure, the fission product releases continued from the ex-vessel fuel in the reactor cavity.

Figure 17 and Figure 18 show the fission product distributions of the iodine and cesium radionuclides that were released from the fuel, respectively. Approximately 99% of the iodine and cesium were released from the fuel prior to vessel failure while the remaining amount was released ex-vessel. At the time of the hot leg failure, only a small portion of the volatile radionuclides (4.3% of the noble gases and ~1% of the cesium and iodine) had been released from the fuel. The resultant blowdown of the vessel immediately discharged the majority of the release to the containment. Following the RCS blowdown after the hot leg nozzle failure, more radionuclides were released from the fuel as the core further degraded. At low pressure conditions, the fission products continued to circulate within the vessel and to the steam generators with a portion being depositing on the structural surfaces (i.e., 10% and 15% of the iodine and cesium are retained in the RCS). However, as shown in the figures, the majority of the released radionuclides went to the containment. Within 36 hr, most of the airborne fission products in the containment settled on surfaces. This was significant because the containment failure occurred at 45 hr 32 min. Consequently, there was little airborne mass that would be released to the environment.

The chemical form of the released iodine was cesium iodine, which was more volatile than the predominant form of the released cesium, which was cesium-molybdate (Cs_2MoO_4). As shown in the iodine history figure (see Figure 17), the in-vessel iodine mass was decreasing following vessel failure through 4 days. The decrease of mass represents a revaporization process of previously deposited radionuclides. The late in-vessel revaporization release was significant because it continued after containment failure and had a significant contribution to the environmental release. The primary thermal mechanisms for the revaporization of the iodine came from a natural circulation flow of hot gas. Very hot gases (i.e., from 990 K to >1616 K) flowed from the reactor cavity through the failed vessel lower head, through the reactor vessel, and out the failed hot leg nozzle. The combination of the decay heat and hot gases heated the deposited cesium iodine, which vaporized gaseous iodine and left behind the cesium that was chemisorbed to the stainless steel surfaces. The natural circulation flow pattern also vented the gaseous iodine from the RCS to the containment.

In contrast to the iodine response in Figure 17, the deposited cesium-molybdate was less volatile and remained deposited in the vessel. Except for inside the reactor cavity, the containment was cooler than vessel and well below conditions that would vaporize settled radionuclides. Consequently, none of the deposited radionuclides in the containment vaporized.

Finally, Figure 19 summarizes the releases of the radionuclides to the environment. At 4 days, 80% of the noble gases, 2.3% of the tellurium, 0.6% if the iodine, 0.75% of the radioactive cadmium, 0.08% of the cesium, 0.08% of the barium, and 0.04% of the radioactive tin had been released to the environment. All other releases were less than 0.02% of the initial inventory. As shown in the figure, there were some environmental releases prior to the containment failure at 45.5 hr due to nominal leakages (i.e., design specification of 0.1% vol/day at the design pressure). After the failure of the containment, the releases to the environment increased sharply. Over the first day after containment failure, 50% of the airborne noble gases in the containment were released. Over the next day, only 30% more was released.

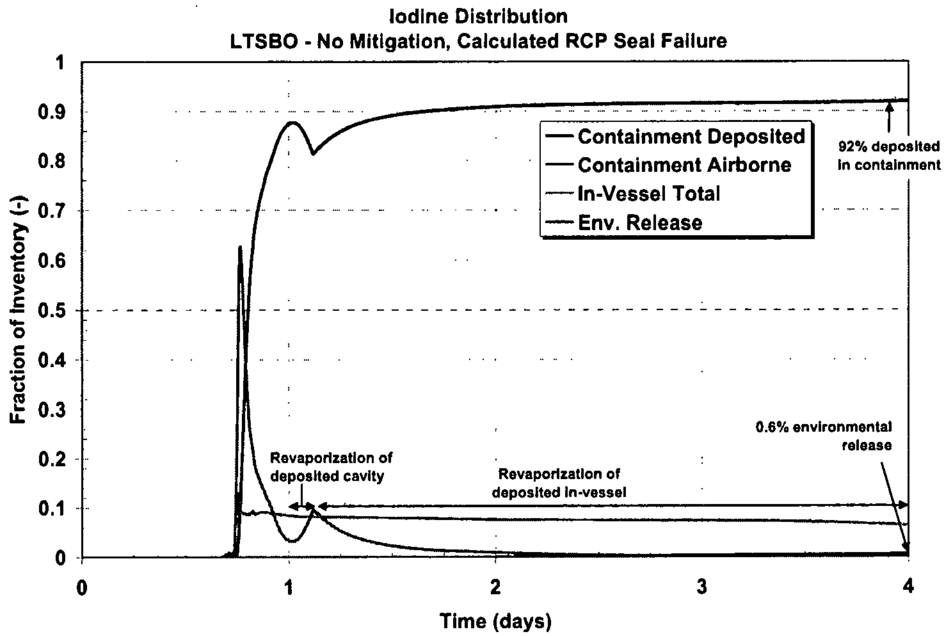

Figure 17 Unmitigated LTSBO iodine fission product distribution history.

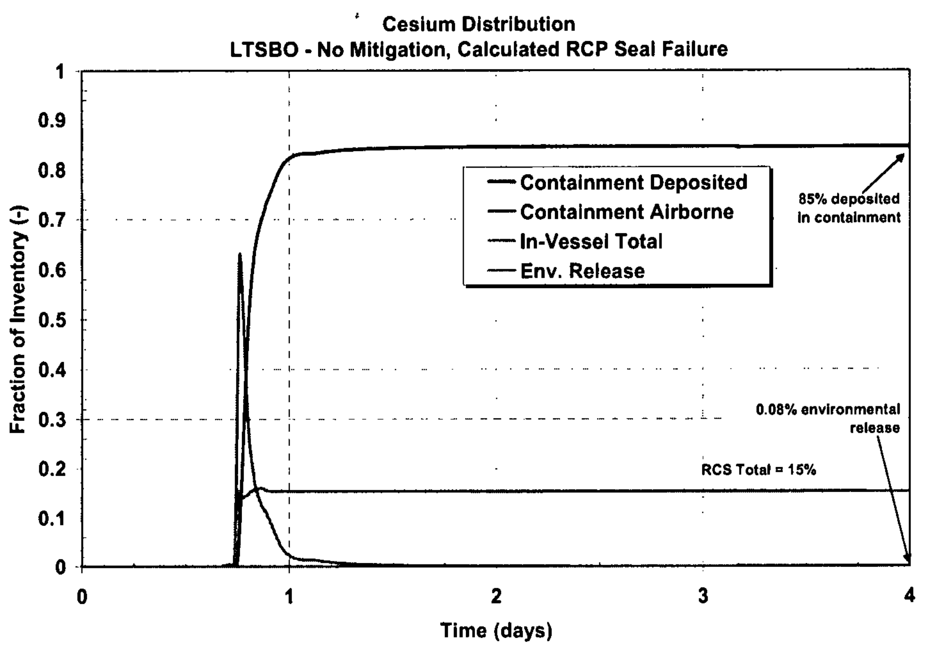

Figure 18 Unmitigated LTSBO cesium fission product distribution history.

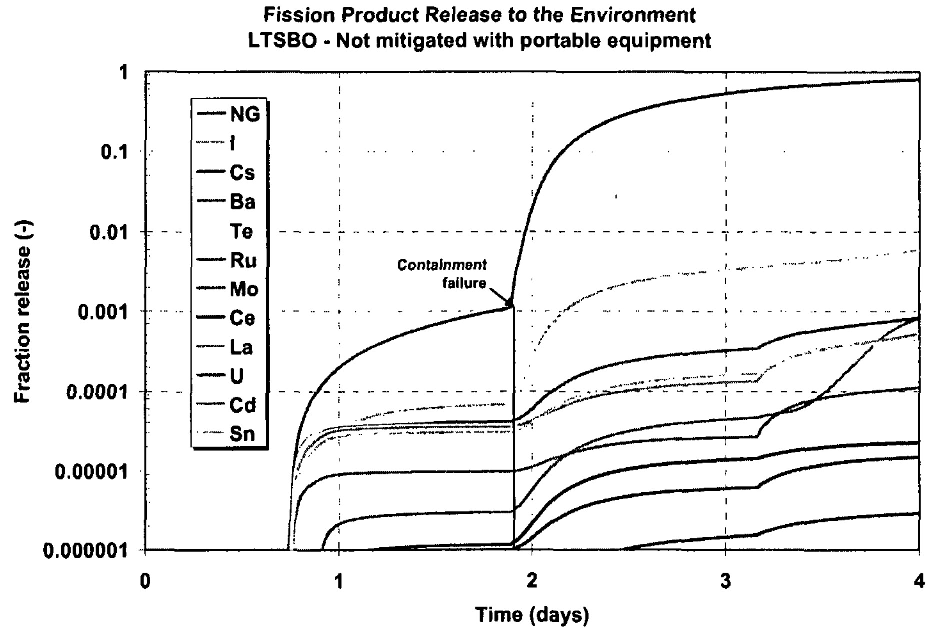

Figure 19 Unmitigated LTSBO environmental release history of all fission products.

5.1.2 LTSBO- Base Case

Table 2 summarizes the timing of the key events in the mitigated LTSBO. As described in Section 3.4.1, the accident scenario initiates with a complete loss of all onsite and offsite power. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. The timings of the key events are discussed further in Section 5.1.2.1. Unlike the unmitigated LTSBO described in Section 5.1.1, the mitigated LTSBO credits the successful connection of the portable, diesel-driven (Kerr) pump to three drain lines of the residual heat removal piping to the RCS.[13] The Kerr pump is a high-head pump with a design capacity of 100 gpm at 6.2 MPa (900 psi) and 500 gpm at 3.5 MPa (500 psi). The Kerr pump takes suction from the refueling water storage tank, which has a 350,000 gal capacity. The refueling water storage tank could be refilled as necessary. The sequence of events is identical to the unmitigated LTSBO until 3 hr 30 min when the Kerr pump starts operating. The Kerr pump operation starts prior to any core degradation (see Figure 76). The emergency Kerr pump is effective at maintaining the vessel water level above the top of the fuel for the duration of the sequence. In fact, the pump was throttled to a small fraction of its rated flow.

[13] The utility has a 3-way connection from the Kerr pump to the three drain lines and all connecting equipment near the residual heat removal piping.

Table 2 The timing of key events for mitigated long-term station blackout.

Event Description	Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC power	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump	00:00
TD-AFW auto initiates at full flow	00:01
Operators control TD-AFW to maintain level	00:15
Vessel water level drains into upper plenum	00:30
Operators initiate controlled cooldown of secondary at ~100°F/hr	01:30
Vessel water level drains into upper plenum	01:57
Minimum vessel water level	02:30
Accumulators begin injecting	02:25
150 gpm emergency high-head diesel injection to RCS	03:30
SG cooldown stopped at 120 psig to maintain TD-AFW flow	03:35
Pressurizer starts to refill	05:38
DC station batteries fail but operator actions continue to control the secondary pressure at 120 psi and maintain TD-AFW flow	08:00
Normal pressurizer level restored	24:00

5.1.2.1 Thermal-Hydraulic Response

The progression of events in the mitigated LTSBO is identical to the unmitigated LTSBO as described in Section 4.3.1 through the first 3 hr 30 min. In particular, the operators take actions to throttle the TD-AFW to maintain a normal level in the steam generators and perform a cool down of the RCS using the steam generator relief valves. Similar to the unmitigated case, the accumulators begin injecting at 2 hr 25 min in response to the decrease in the primary system pressure. It is estimated that the operators could begin RCS injection using the portable, diesel-driven Kerr pump by 3 hr 30 min.

At the time the emergency pump is ready for injection, the primary system pressure is 2.0 MPa (278 psig) or well within the pressure head capacity of the Kerr pump (see Figure 20). Similar to the unmitigated LTSBO, the secondary system is depressurized to 120 psi, or the lower limit of operability for the TD-AFW. Due to the RCP seal leakage and liquid volume shrinkage from the cool down, the vessel level initially deceased but started to recover after 2 hr 25 min following the start of the accumulator injection (see Figure 21). The peak fuel cladding temperature and vessel lower head followed the primary system liquid temperature, which was steadily cooled down by the steam generators (see Figure 22 and Figure 23, respectively).

At 3 hr 30 min, the emergency injection begins and supplements the RCS inventory make-up with the accumulators. For the purposes of the calculation, a simple control system was constructed to ramp the Kerr pump flow based on the pressurizer level. Initially, the flow started at 150 gpm (see Figure 25). The make-up flow started refilling the RCS and caused cold water to enter the empty pressurizer shortly after 3.5 hr. At 4.1 hr, the level was restored in the pressurizer and the flow was throttled back to 7.5 gpm for the remainder of the simulation. As seen in Figure 25, the emergency injection mass flow was approximately equal to the RCP leakage flow. Meanwhile, the cold water in the pressurizer collapsed the steam bubble and filled the pressurizer full of water. Consequently, the control system maintained a constant minimum flow of 7.5 gpm for the remainder of the transient. Due to the small mismatch between the leakage flow and the emergency injection flow, the water level in the vessel fell very slowly (see Figure 21). This was not significant because the vessel water level remained at least 1.5 m above the top of the fuel. In reality, the operator may use other indications to control the make-up flow. For example, the vessel wide range shutdown instrumentation or reactor vessel level indication system (RVLIS) could have given additional information about the water level in the vessel. However, if the operator did not take steps to monitor the water inventory in the RCS and vessel, the entire primary system could become water solid and pressurize above the pump shut-off head.

Finally, since heat removal from the RCS was maintained for this sequence, there was not a significant challenge to the containment. As seen in Figure 24, the containment pressure only rose slightly due to heat losses from the RCS.

There were several lessons learned while investigating the mitigation of the long-term station blackout. First, the operator action to reduce the primary system pressure to the threshold of the TD-AFW operation allowed the maximum injection from the accumulators (i.e., two thirds of their liquid inventory). The accumulator flow was significant in the short-term restoration of the vessel liquid inventory. Hence, the depressurization to 120 psi maintained TD-AFW flow but allowed for significant accumulator flow. Second, the reduction of the primary system pressure reduced the RCP seal leakage flow from 21 gpm per pump to less than 3 gpm per pump. Furthermore, if a RCP seal should fail under these conditions, then the resulting leakage flow would be much lower than if the primary system pressure was not actively controlled to low pressure. Third, the emergency pumps have excess capacity to maintain long-term make-up. Consequently, it would be possible to fill the primary system with water and quickly pressurize above the shutoff head of the pumps. The pressurizer level alone may not be a good indication of conditions in the vessel if the steam bubble collapses.[14] Fourth, operator actions were required to replenish the water supply to the ECST for the TD-AFW (exhausted after 5.2 hours and required 545,000 gal for 4 days) but not the RWST for the emergency RCS injection (at 87% inventory after 4 days). Finally, if successful operation actions were taken to replenish the ECST water supply and maintain the steam generator pressure at 120 psi, then considerably more time is available to establish the RCS emergency injection.

[14] Two water separate levels were calculated in the pressurizer and the vessel. Using the RVLIS system, the pressurizer level, and the primary system pressure instrumentation, it should be possible to assess whether the system is water solid.

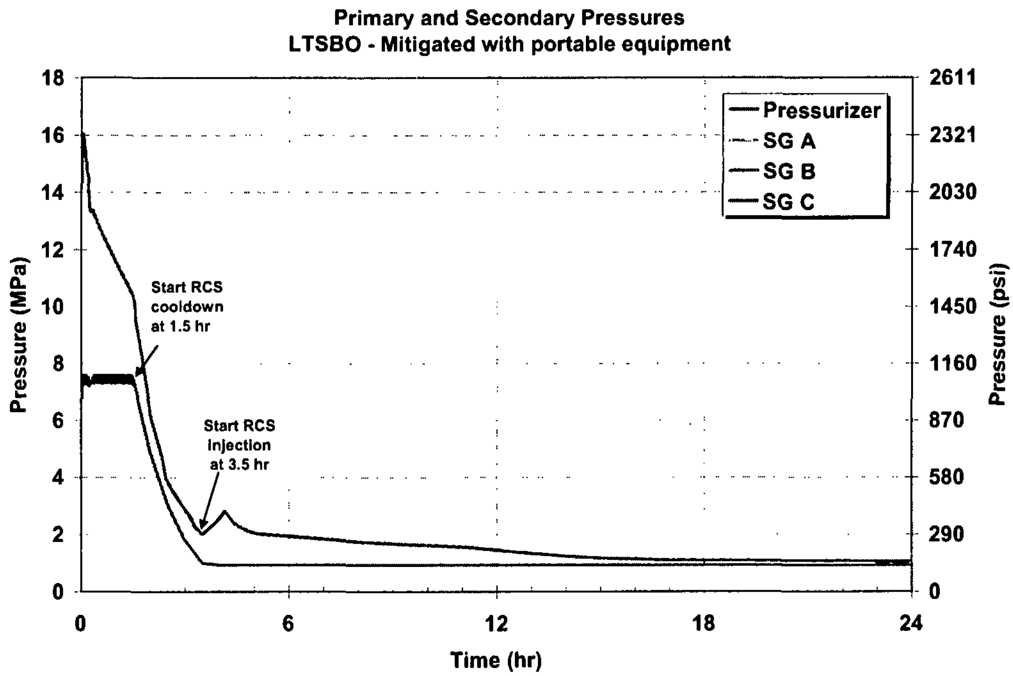

Figure 20 Mitigated LTSBO primary and secondary pressure history.

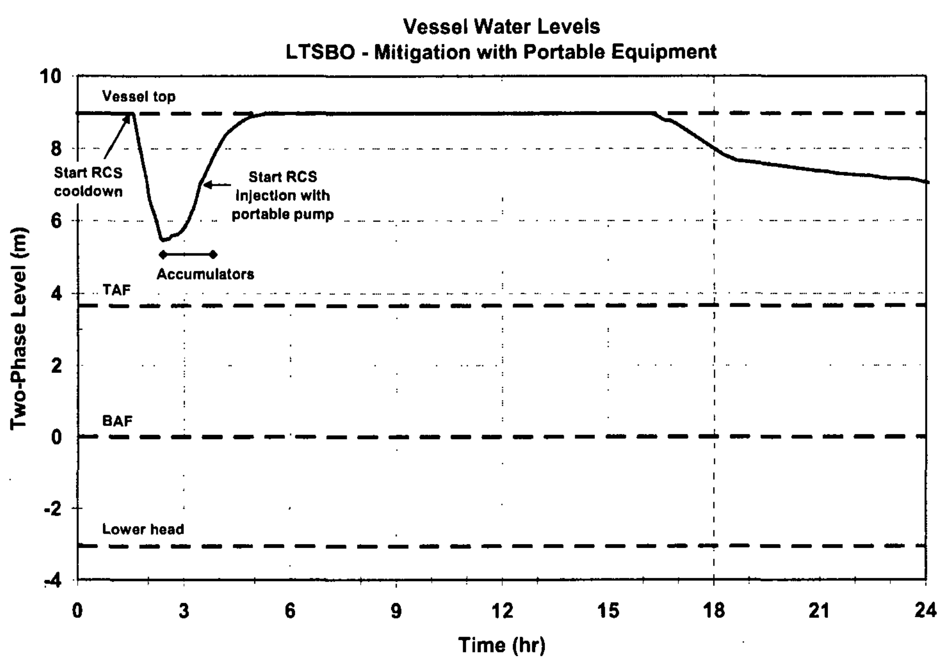

Figure 21 The mitigated long-term station blackout vessel two-phase coolant level.

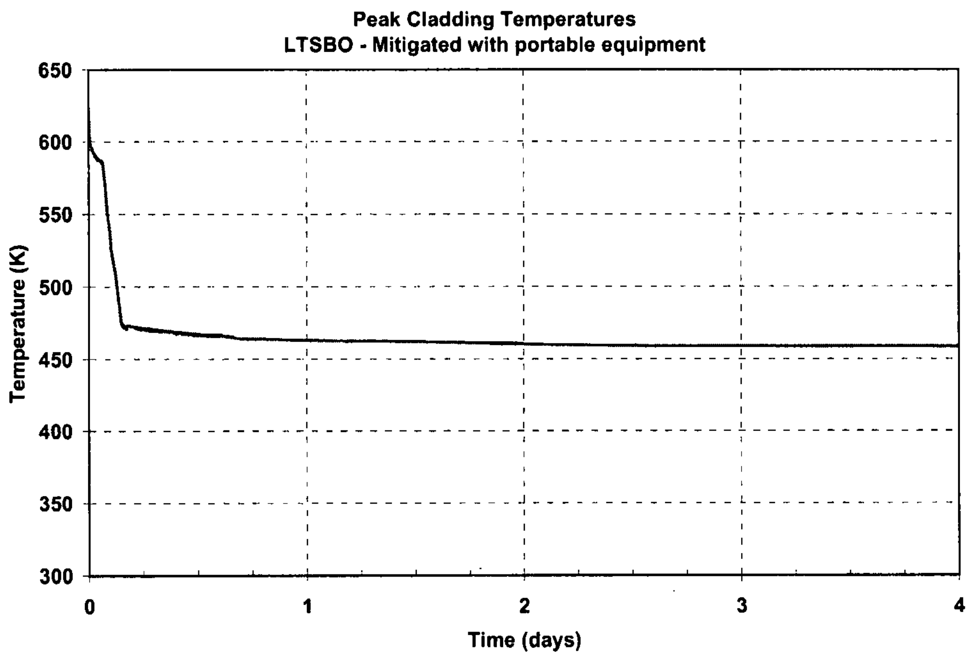

Figure 22 The mitigated long-term station blackout core temperature history.

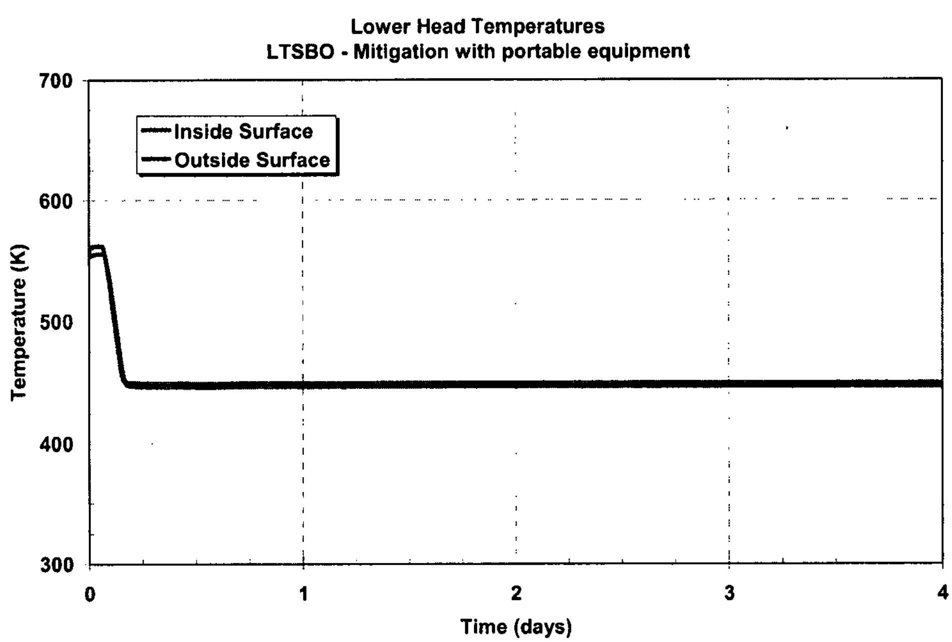

Figure 23 Mitigated LTSBO lower head inner and outer temperature history.

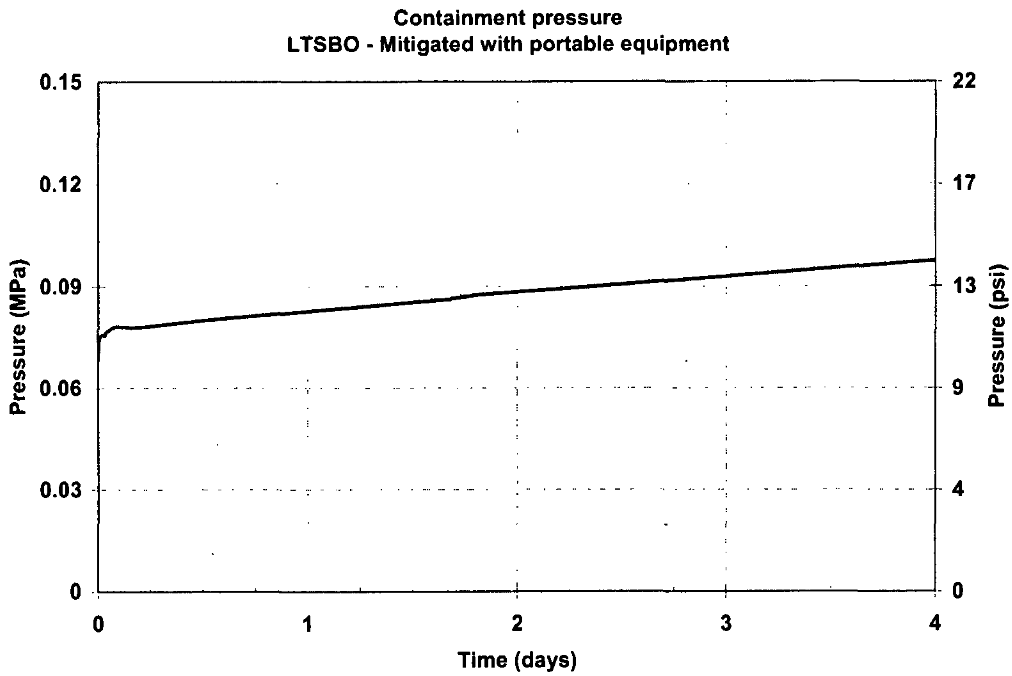

Figure 24 The mitigated long-term station blackout containment pressure history.

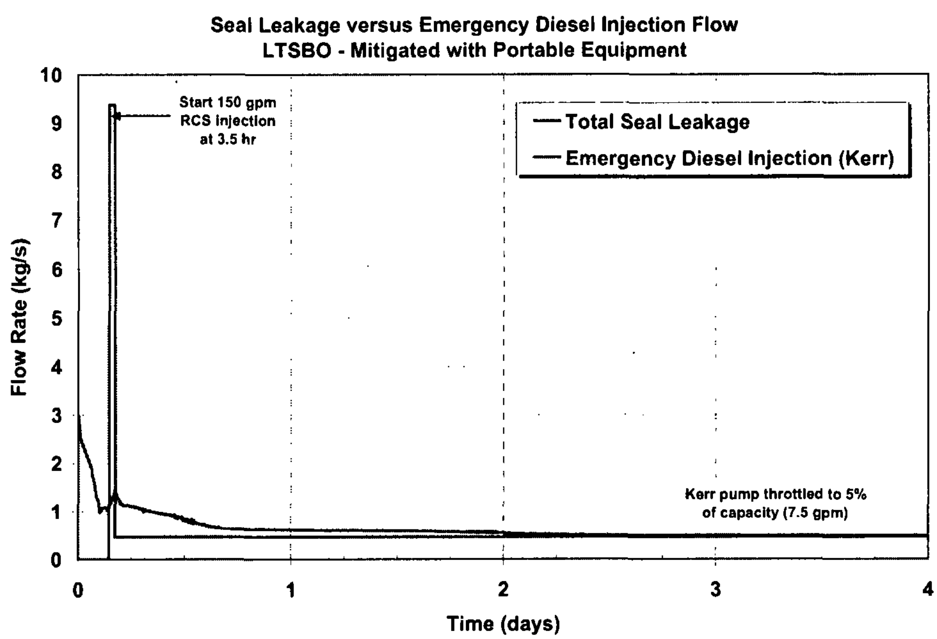

Figure 25 Mitigated LTSBO vessel emergency make up and pump seal leakage flows.

5.1.2.2 Radionuclide Release

There was no fission product release for the mitigated long-term station blackout scenario; thus, no figures were constructed.

5.1.3 Unmitigated Long-Term Station Blackout with Early RCP Seal Failure

Table 3 presents a comparison of the key events in the unmitigated LTSBO with and without an early RCP seal failure. The unmitigated LTSBO without an early RCP seal failure was previously discussed in Section 5.1.1. The accident scenario initiates with a complete loss of all onsite and offsite AC power but the DC station batteries are available. The reactor successfully scrams and the containment isolates but all powered safety systems are unavailable except the TD-AFW. At 13 minutes, the RCP cavity fills with hot water and the seals are specified to fail in all three loops. Although the core degradation and vessel failure progress more quickly in the early RCP seal failure case, the containment failure is later than the case with a late RCP seal failure. The timings of the key events are discussed further in Sections 5.1.3.1 and 5.1.3.2.

Table 3 Comparison of the timings of key events for unmitigated long-term station blackout with and without early RCP seal failures.

Event Description	Early Seal Failure Time (hh:mm)	Late Seal Failure Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC power	00:00	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump	00:00	00:00
TD-AFW auto initiates at full flow	00:01	00:01
First SG SRV opening	00:03	00:03
Assumed early RCP seal failures, 182 gpm/pump at full-pressure	00:13	n/a
Operators control TD-AFW to maintain level	00:15	00:15
Operators initiate controlled cooldown of secondary at ~100°F/hr	01:30	01:30
Vessel water level drains into upper plenum	00:33	01:57
Initial minimum vessel water level	n/a	02:30
Accumulators begin injecting	02:13	02:25
SG cooldown stopped at 120 psig to maintain TD-AFW flow	03:35	03:35
Emergency CST empty	05:29	05:08
DC Batteries Exhausted	08:00	08:00

Event Description	Early Seal Failure Time (hh:mm)	Late Seal Failure Time (hh:mm)
S/G PORVs reclose	08:00	08:00
Pressurizer SRV opens	n/a	13:06
PRT failure	n/a	13:40
Start of fuel heatup	09:38	14:16
RCP seal failures (calculated)	n/a	14:46
First fission product gap releases	11:03	16:04
Creep rupture failure of the C loop hot leg nozzle	12:29	17:06
Accumulator empty	12:29	17:06
Vessel lower head failure by creep rupture	14:26	21:08
Debris discharge to reactor cavity	14:26	21:08
Cavity dryout	14:36	21:16
Containment at design pressure (45 psig)	42:08	28:00
Start of increased leakage of containment (P/P_{design} = 2.18)	55:40	45:32

5.1.3.1 Thermal-hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 26. The system pressure responses through the first 10 hours are very similar to the unmitigated case without an early RCP seal failure (see Figure 12). However, as will be discussed below, the early RCP seal failure had an important impact on primary system pressure response after 10 hours. As shown in Figure 27, the early RCP seal failures occur at 13 minutes versus 14 hours 46 minutes in the late failure case. Consequently, the early leakage flowrate is much higher than the late failure case. The high RCP seal leakage causes a faster decrease in the vessel water inventory (see Figure 28), earlier core degradation and start of fission product release, and earlier hot leg creep rupture failure and vessel failure (see Table 3).

After the DC batteries fail, the relief valves on the secondary system close and the primary and secondary system pressures rise. In the early RCP seal failure case, the primary system pressure initially increased to the secondary pressure level. However, the primary system pressure subsequently decreased due to continued seal leakage (Figure 27) and reduced steam production in the core. The steam production decreased at this time in the sequence because of the core water level dropped below the bottom of the active fuel (compare the pressure response in Figure 26 after 11.3 hr to the water level in Figure 28). In contrast, the late RCP seal leakage case had vigorous boiling in the core after DC battery failure until ~15 hours. Consequently, the primary system pressurized above the secondary system pressure to the pressurizer relief valve pressure. Subsequently, rapid hydrogen generation maintained the high primary system near the pressurizer relief valve set point until the hot leg creep rupture failure at 17 hr 6 min.

A comparison of the containment pressure responses for the cases with and without early RCP seal failures is shown in Figure 29. Although vessel failure occurs earlier in the case with early RCP leakage, the subsequent containment pressurization is slower. The long-term containment pressurization is due to two components, (1) non-condensable gas production from core-concrete interactions and (2) vaporization of water in the containment. The non-condensable gas production was similar in the two cases. However, the case with late failure of the RCP seals had more water available on the containment floor for vaporization. Since all the water eventually vaporized due to high temperatures in the containment, the total steam production was higher in the late RCP seal failure case.

There was more water on the containment floor in the case with late failure of the RCP seals for two reasons. First, the primary system pressurizer relief valve cycled to relieve steam from the primary system after the DC battery failure in the case with late RCP seal failure, which eventually led to failure of the pressurizer relief tank (i.e., see event in Table 3 and primary system pressure response in Figure 12). Upon failure of the pressurizer relief tank, a large amount of water from the tank discharged onto the containment floor. As discussed above, the early RCP failure case did not have any pressurizer relief valve discharges after the DC batteries failed. Although the early RCP seal failure case initially had more water spilled to the containment floor from the seal leakage, the total amount of water on the containment floor was less than the late RCP seal failure case. The amount of water spilled from the pressurizer relief tank was large compared to the leakage from the failed seals (see Figure 30).

It is also interesting to look at similarities and differences in the events that affected the short-term containment pressurization. In the late RCP seal failure case, the containment pressurization had three events that caused step increases in the containment pressure. First, the failure of the pressurizer relief tank caused the first significant pressurization of the containment at 13 hr 40 min (0.57 days). Some hydrogen burns also cause short-term pressurizations following this event. However, the containment pressure returned to the pre-burn pressure after the burn. The early RCP seal leakage case did not have this event (i.e., failure of the pressurizer relief tank). Next, the hot leg failed by creep rupture and suddenly increased the pressure in the containment in both cases. Since the hot leg failure in the late RCP seal failure case occurred from 16.2 MPa versus 6.4 MPa in the early RCP seal failure case, significantly more energy was released to the containment as evidenced by the rapid pressure response rise following hot leg creep rupture (see Figure 29). Finally, both cases had similar pressurizations following vessel failure when the core debris rapidly evaporated the water in the reactor cavity. In summary, the first rapid containment pressurization event only occurred in the late RCP failure case, the second event was more severe in the late RCP failure case, and the third event was comparable in the two cases.

In an integral sense, a portion of the overall system energy generated in the late RCP seal failure case was released to the containment 4.5 hours after the early RCP failure case (i.e., the difference in the timings of the hot leg creep rupture failures, see Table 3)[15]. The additional energy storage in the primary system in the late seal failure case results in a higher pressure and

[15] The heat removal by the secondary system ended at the approximately same time in both cases. The energy released through the seal failure was lower in the late RCP seal failure until 14 hr 46 min, when the seal failed.

temperature of the gas in the primary system at the time of the leg creep rupture. In contrast, a portion of the early RCP seal failure decay energy generated over the time period between hot leg creep ruptures in the two cases (i.e., 12 hr 29 min to 17 hr 6 min) is absorbed into the concrete by core-concrete interactions in the early RCP seal failure case (i.e., from 14 hr 26 min to 17 hr 6 min). The result was a higher heat load to the containment atmosphere in the early RCP seal failure case, which resulted in a faster evaporation rate of the water on the containment floor.

In addition to the faster rate of evaporation, more water was available to evaporate in the late RCP seal failure case (see Figure 30). Consequently, the resulting pressurization was faster and longer. The timings of the containment failure were 55.7 and 45.5 hours for the early and late RCP seal failure cases, respectively. Due to the additional water for vaporization in the late seal failure case, the resulting containment pressurization was not only faster but also resulted in a higher pressure (see Figure 29). The importance of the water vaporization is demonstrated by comparing Figure 29 and Figure 30. The containment depressurization does not begin until all the water is evaporated. Hence, the water vaporization was significant component of the containment pressurization relative to the non-condensable CCI gas generation.

Finally, due to higher peak containment pressure in the late RCP seal failure case, the resultant containment leakage area was also slightly higher, 0.0053 m^2 (8.3 in^2) versus 0.0028 m^2 (4.4 in^2). Consequently, the long-term depressurization rate following the containment failure was faster in the late RCP seal failure case.

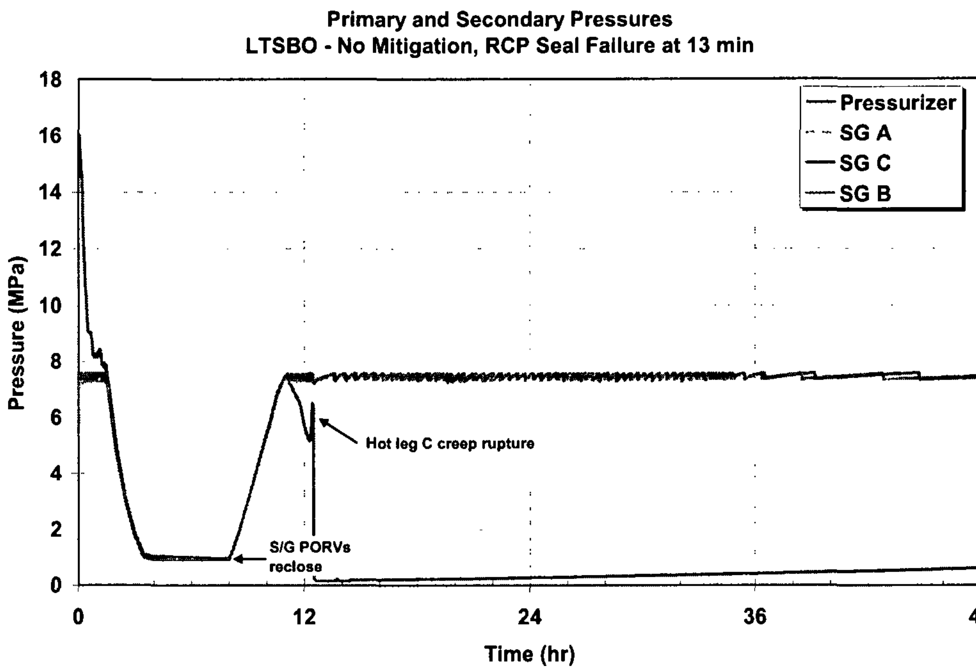

Figure 26 The unmitigated long-term station blackout primary and secondary pressure history.

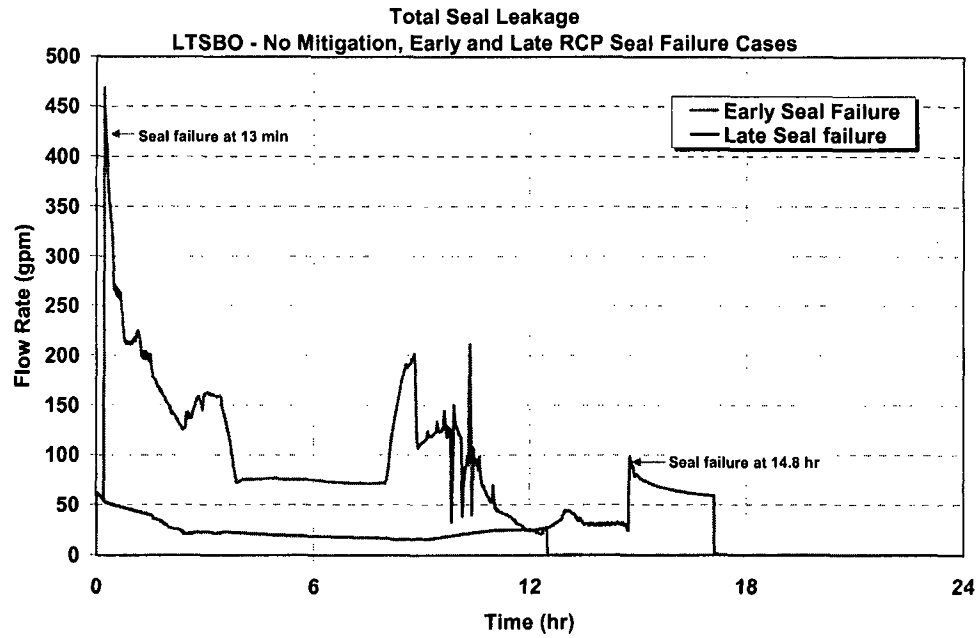

Figure 27 Comparison of the unmitigated long-term station blackout RCP seal leakages with and without early RCP seal failures.

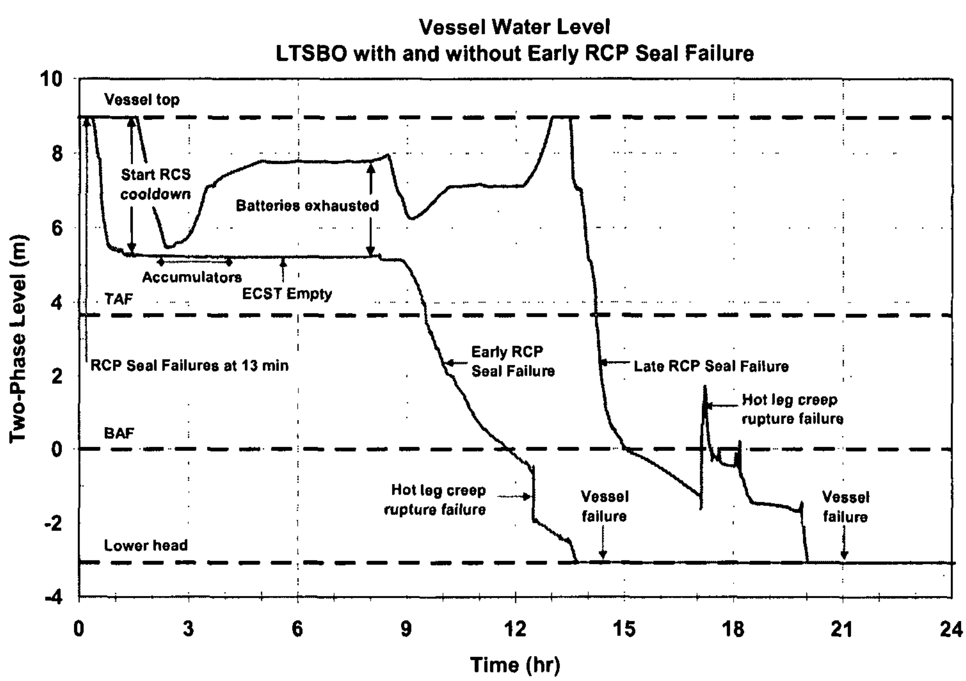

Figure 28 Comparison of the unmitigated long-term station blackout vessel level responses with and without early RCP seal failures.

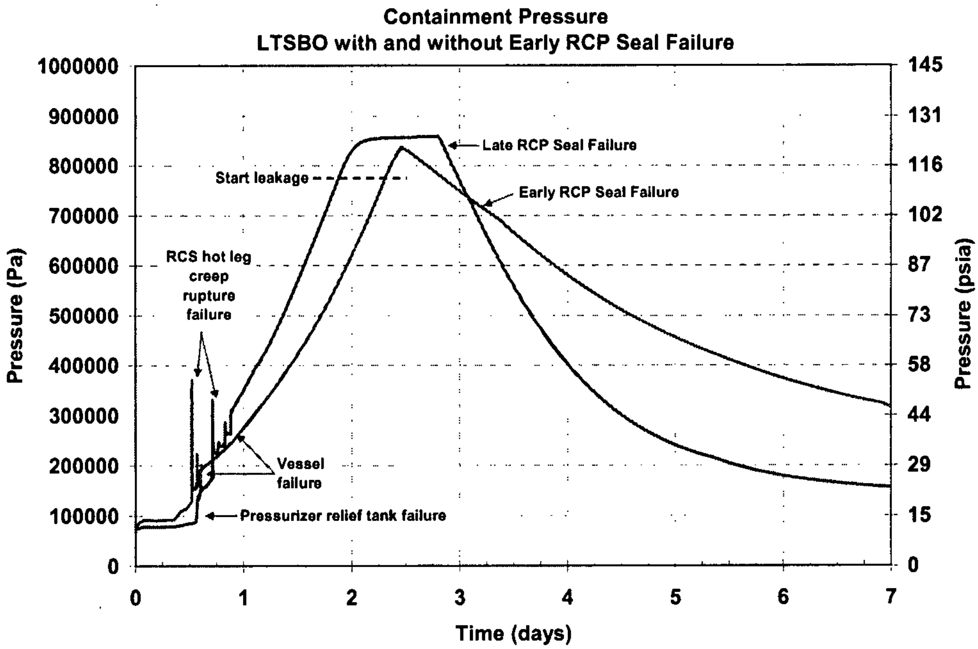

Figure 29 Comparison of the unmitigated long-term station blackout containment pressure responses with and without early RCP seal failures.

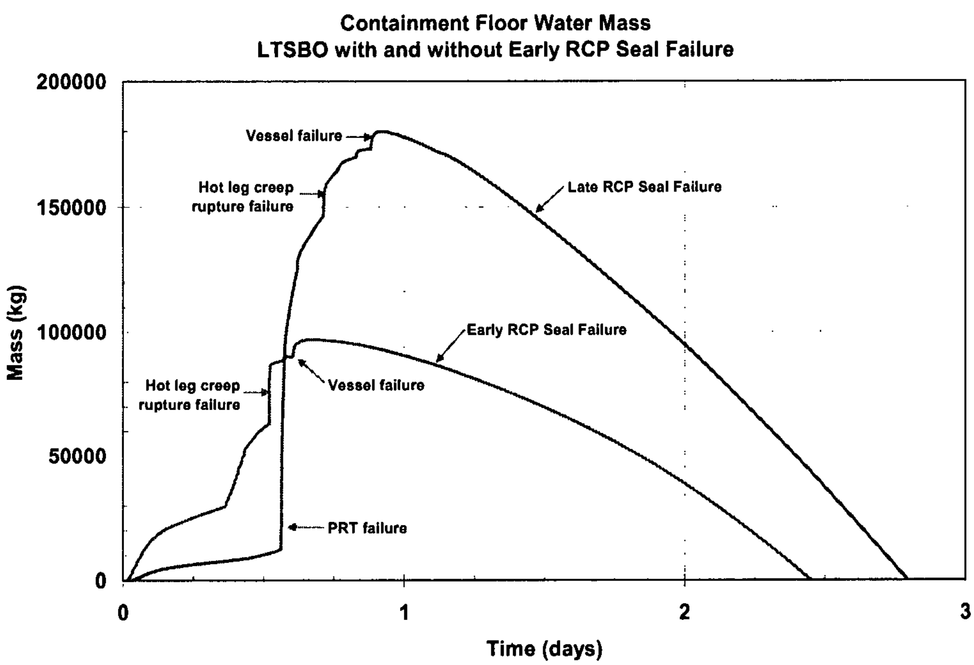

Figure 30 Comparison of the unmitigated long-term station blackout containment water pool masses with and without early RCP seal failures.

5.1.3.2 Radionuclide Release

The environmental releases for the noble gases, iodine, and cesium for the two cases are shown in Figure 31 through Figure 33, respectively. The radionuclide behavior was similar to the late RCP seal failure responses described in Section 5.1.1. However, the magnitude and timing of the releases from the late RCP seal failure case bounded the response from the early RCP seal failure. The early RCP failure releases were smaller and delayed because (a) the containment failure was later (2.3 days versus 1.9 days), (b) the containment failure area and associated depression rate were smaller (0.0028 m^2 versus 0.0053 m^2), and (c) the time for settling between hot leg failure and containment failure was longer (43 hours versus 28 hours). However, both cases had small environmental releases of iodine and cesium due to the late containment failure timing.

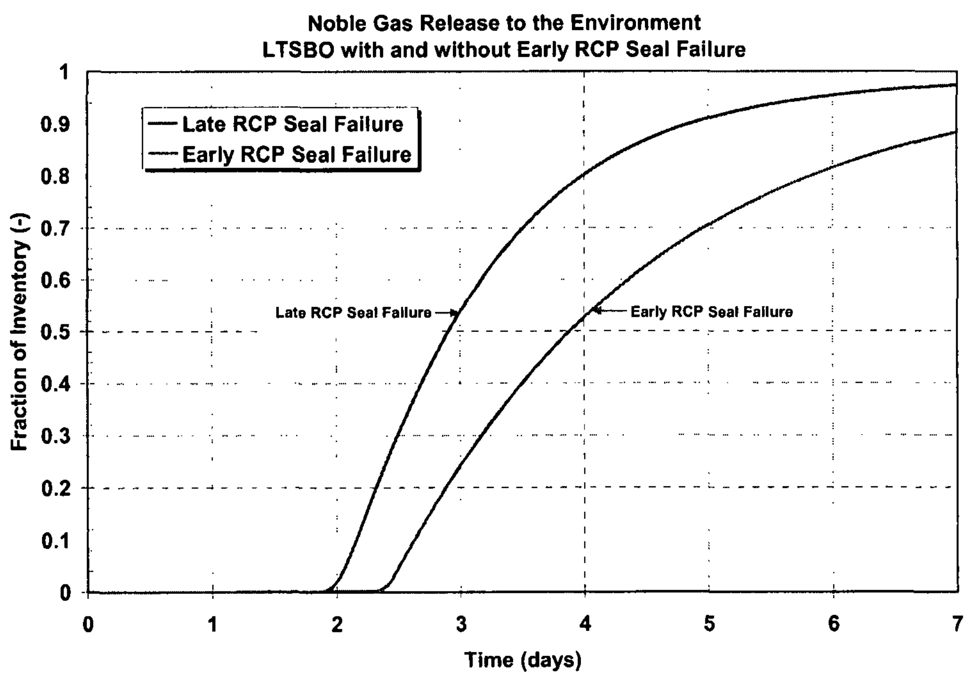

Figure 31 Comparison of the unmitigated long-term station blackout noble gas releases to the environment with and without early RCP seal failures.

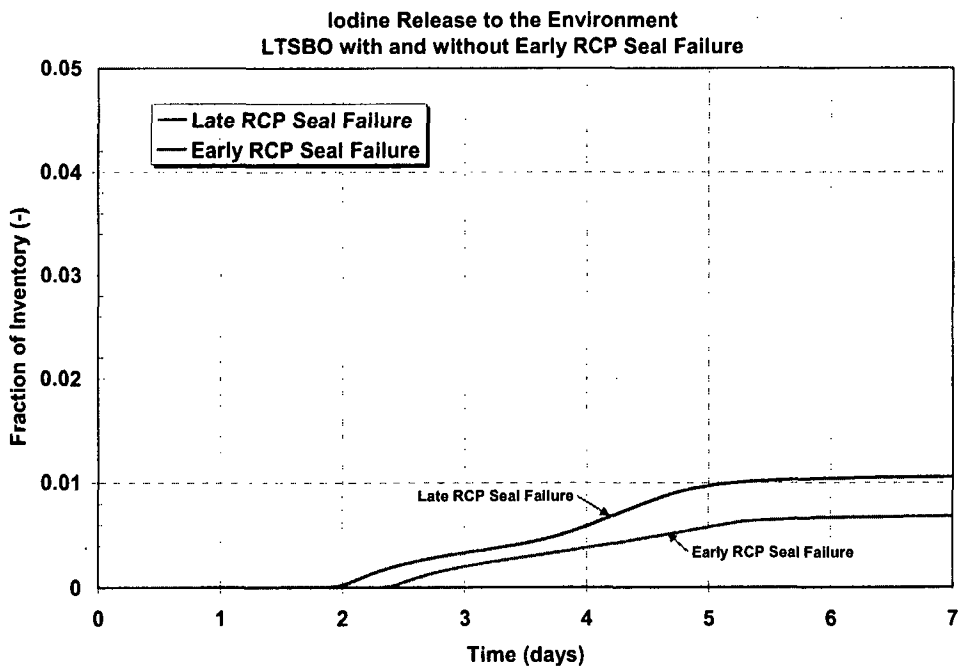

Figure 32 Comparison of the unmitigated long-term station blackout iodine releases to the environment with and without early RCP seal failures.

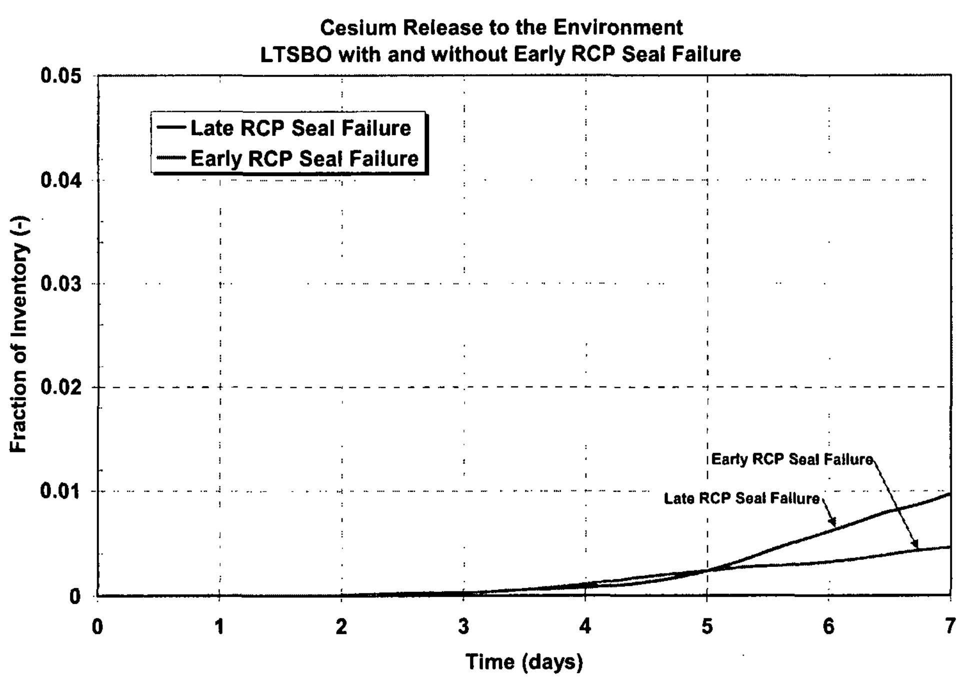

Figure 33 Comparison of the unmitigated long-term station blackout cesium releases to the environment with and without early RCP seal failures.

5.1.4 LTSBO with Early RCP Seal Failure- Base Case

Table 4 summarizes the timing of the key events in the mitigated LTSBO. As described in Section 3.1, the accident scenario initiates with a complete loss of all onsite and offsite power. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. At 13 minutes, the RCP seal cavity fills with hot water and the seals fail in all three loops. The mitigated LTSBO credits the successful connection of the portable, diesel-driven Kerr pump to three drain lines of the residual heat removal piping to the RCS. The Kerr pump is a high-head pump with a design capacity of 100 gpm at 6.2 MPa (900 psi) and 500 gpm at 3.5 MPa (500 psi). The Kerr pump takes suction from the refueling water storage tank, which has a 350,000 gal capacity and could be refilled as necessary. In addition, a portable power supply was available to maintain the secondary cooldown after the DC batteries fail.

The sequence of events is identical to the unmitigated LTSBO with early RCP seal failure until 3 hr 30 min when the Kerr pump starts operating. The Kerr pump operation starts prior to any core degradation (see Table 4) and is effective at maintaining the vessel water level above the top of the fuel for the duration of the sequence. In fact, the pump had excess capacity and was periodically throttled below the maximum flow rate. The timings of the key events are discussed further in Sections 5.1.4.1 and 5.1.4.2.

Table 4 Comparison of the timings of key events for mitigated long-term station blackout with and without early RCP seal failures.

Event Description	Early Seal Failure Time (hh:mm)	Late Seal Failure Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC power	00:00	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump	00:00	00:00
TD-AFW auto initiates at full flow	00:01	00:01
RCP pump seals fail, 182 gpm/pump at full-pressure	00:13	n/a
Operators control TD-AFW to maintain level	00:15	00:15
Vessel water level drains into upper plenum	00:30	00:30
Operators initiate controlled cooldown of secondary at ~100°F/hr	01:30	01:30
Vessel water level drains into upper plenum	00:30	01:57
Minimum vessel water level	02:20	02:20
Accumulators begin injecting	02:15	02:25
150 gpm emergency high-head diesel injection to RCS	03:30	03:30

Event Description	Early Seal Failure Time (hh:mm)	Late Seal Failure Time (hh:mm)
SG cooldown stopped at 120 psig to maintain TD-AFW flow	03:35	03:35
Pressurizer starts to refill	05:38	03:35
DC station batteries fail but operator actions continue to control the secondary pressure at 120 psi and maintain TD-AFW flow	08:00	08:00
Normal pressurizer level restored	24:00	24:00

5.1.4.1 Thermal-Hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 34. The system pressure responses are very similar to the mitigated case without an early RCP seal failure (see Figure 20). The impact of the early RCP seal failure did not have a significant impact on pressure response. More important was the successful operator actions to depressurize the primary system using the secondary system PORVs. As shown in Figure 35, the early RCP seal failures occur at 13 minutes. However, the total RCP seal leakage was less than 150 gpm after 2 hours because of the successful RCS depressurization. The RCP seals were not predicted to fail in the mitigated case without early failure due to adequate subcooling.

As discussed in Section 5.1.2, the operators take actions to throttle the TD-AFW to maintain a normal level in the steam generators and perform a cooldown of the RCS using the steam generator relief valves. The accumulators begin injecting at 2 hr 12 min following the decrease in the primary system pressure below the pressure of the accumulators. It was estimated that the operators could begin RCS injection using the portable, diesel-driven Kerr pump by 3 hr 30 min.

At the time the emergency pump is ready for injection, the primary system pressure is 2.0 MPa (278 psig) or well within the pressure head capacity of the Kerr pump (see Figure 34).
The secondary system is depressurized to 120 psi, or the lower limit of operability for the TD-AFW. As shown in Figure 36, the vessel level decrease in the early RCP failure case was more severe than the late RCP seal failure case. However, accumulator injection maintained the vessel level in the upper plenum until the emergency portable in injection started. Consequently, there was no fuel uncovery or fuel heatups.

At 3 hr 30 min, the emergency injection begins and supplements the RCS inventory make-up with the accumulators. For the purposes of the calculation, a simple control system was constructed to ramp the Kerr pump flow based on the pressurizer level. Initially, the flow started at the capacity of the pump through the flow restrictions in the LHSI drain lines (i.e., 150 gpm, see Figure 37). The make-up flow started refilling the RCS and started to fill the pressurizer after 5.6 hr. Once a level was established in the pressurizer, the flow was throttled back to 7.5 gpm. 7.5 gpm was insufficient to maintain the pressurizer level. Hence, the flow was periodically increased to capacity of the pump until the level was restored in the pressurizer.

The lessons learned that were described in Section 5.1.2 are also applicable in the mitigated early RCP failure case. There are the following quantitative and timing differences for the early RCP seal failure case versus the late RCP seal failure case values that were presented in Section 5.1.2. First, the reduction of the primary system pressure reduced the total RCP seal leakage flow from a nominal value of 182 gpm per pump to less than pumping capacity of the high-head emergency diesel pump (i.e., 150 gpm) after 2.2 hours. The reduction in RCP leakage delays core damage and increases the likelihood that the operators will be able to prevent core damage. The depressurization also allows use of a lower head pump to provide RCS make-up. Second, operator actions were required to replenish the water supply to the ECST for the TD-AFW (exhausted after 5.8 hours and required 420,000 gal for 72 hours) and the RWST for the emergency RCS injection (exhausted after 57 hours). These actions were necessary to maintain primary and secondary coolant injection.

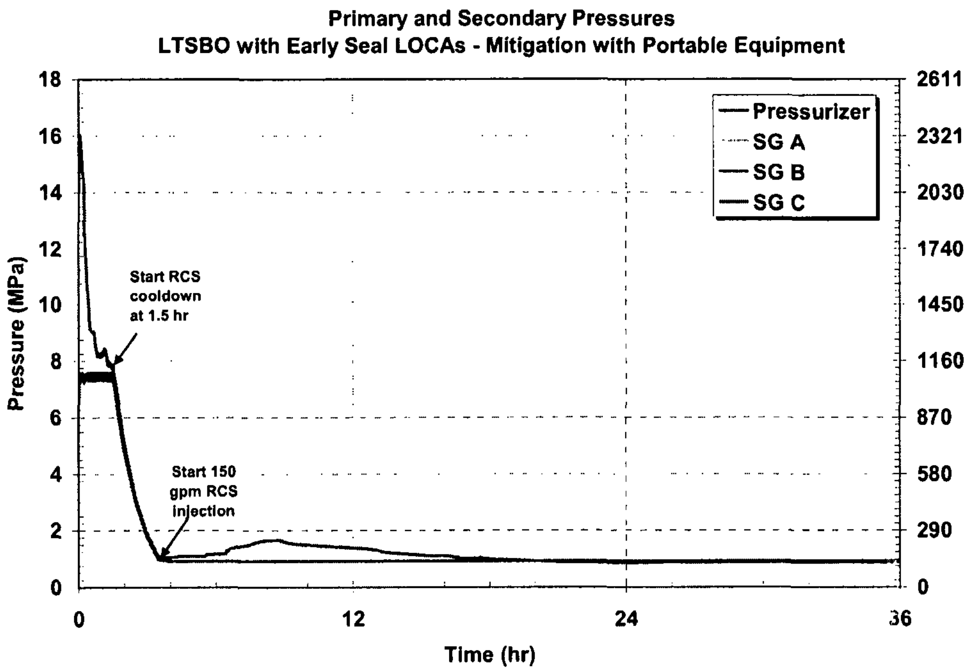

Figure 34 **The primary and secondary pressure responses for the mitigated long-term station blackout with early RCP seal failure.**

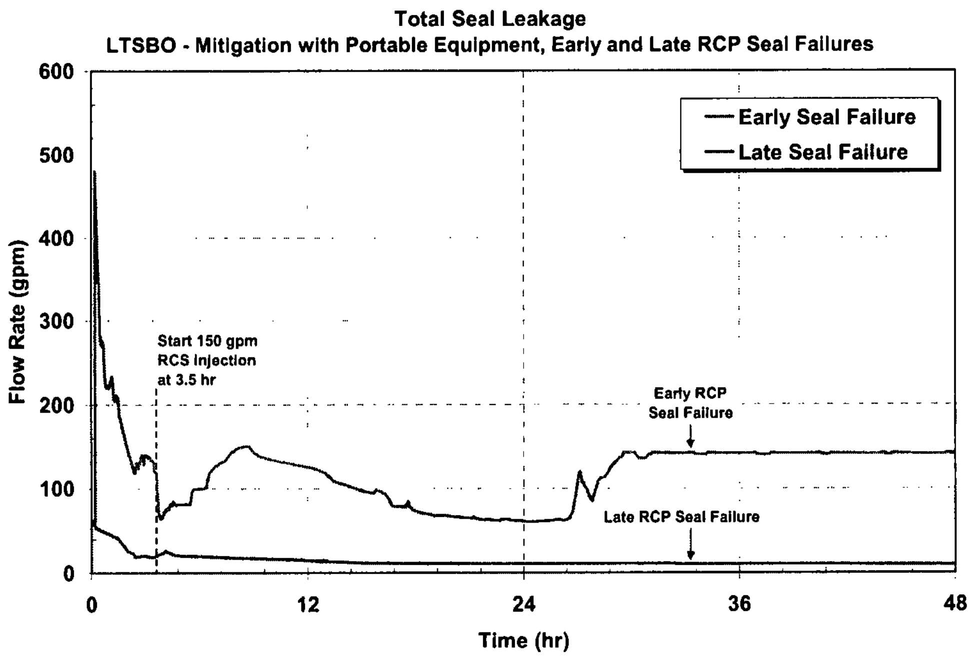

Figure 35 Comparison of the mitigated long-term station blackout RCP seal leakages with and without early RCP seal failures.

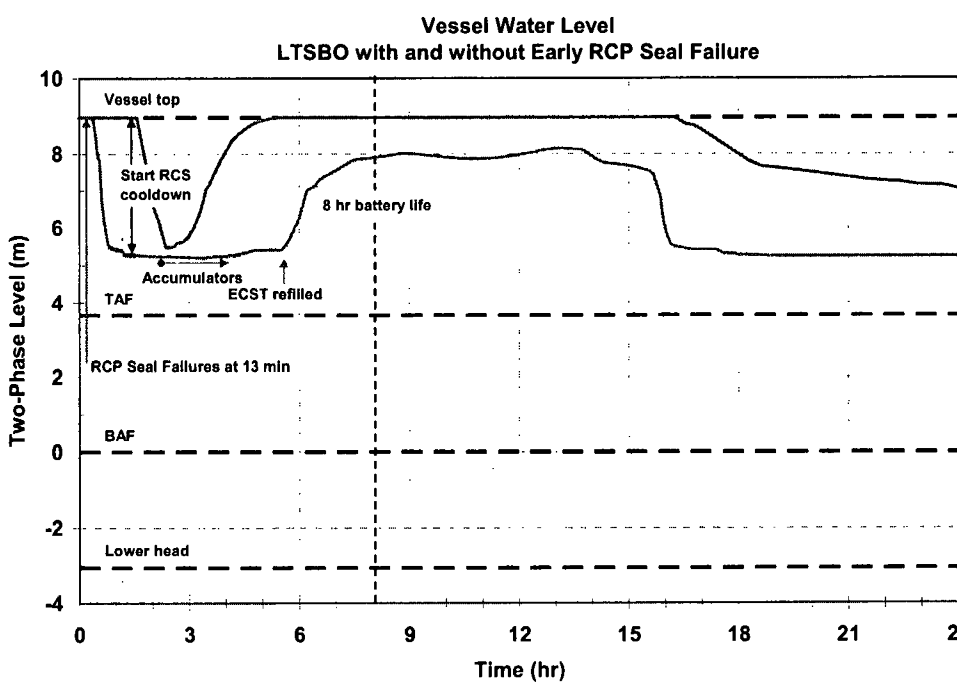

Figure 36 Comparison of the mitigated long-term station blackout vessel level with and without early RCP seal failures.

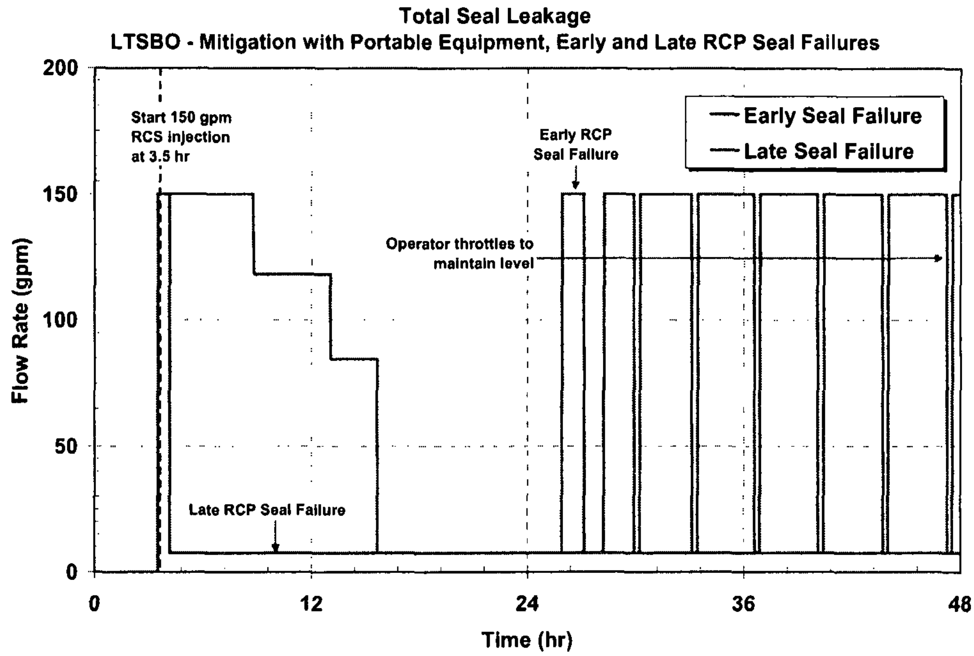

Figure 37 Comparison of the mitigated long-term station blackout portable pump injection rates with and without early RCP seal failures.

5.1.4.2 Radionuclide Release

There was no fission product release for the mitigated long-term station blackout scenario with early RCP failures; thus, no figures were constructed.

5.2 Short-Term Station Blackout

The short-term station blackout is assumed to be initiated by a seismic event. Section 5.2.1 presents the results of an unmitigated scenario with no successful operator actions. For the mitigated scenario in Section 5.2.2, a portable emergency pump is connected to the containment spray system at 8 hours and available to inject 1,000,000 gallons.

5.2.1 Unmitigated Short-Term Station Blackout

Table 5 summarizes the timing of the key events in the unmitigated STSBO. As described in Section 3.2.1, the accident scenario initiates with a complete loss of all onsite and offsite power and failure of the ECST. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. The timings of the key events are discussed further in Sections 5.2.1.1 and 5.2.1.2. However, it is worth initiating noting that fission product releases from the fuel do not begin until 2 hr 57 min and significant fission product releases to the environment do not begin until 25 hr 32 min. Section 5.2.1.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.2.1.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 5 The timing of key events for unmitigated short-term station blackout.

Event Description	Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC and DC power	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump TD-AFW fails	00:00
First SG SRV opening	00:03
SG dryout	01:16
Pressurizer SRV opens	01:27
Pressurizer relief tank rupture disk opens	01:46
Start of fuel heatup	02:19
RCP seal failures	02:45
First fission product gap releases	02:57
Creep rupture failure of the A loop hot leg nozzle	03:45
Accumulators start discharging	03:45
Accumulators are empty	03:45
Vessel lower head failure by creep rupture	07:16
Debris discharge to reactor cavity	07:16
Cavity dryout	07:27
Containment at design pressure (45 psig)	11:00
Start of increased leakage of containment (P/P_{design} = 2.18)	25:32
Containment pressure increase slows	32:00
Containment pressure stops decreasing	44:14
End of calculation	48:00

5.2.1.1 Thermal-Hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 38. At the start of the accident sequence, the reactor successfully scrams in response to the loss of power. The main steam line isolation and containment isolation valves close in response due to the loss of power. The reactor coolant and main feedwater pumps also trip to the loss of power. Once the main steam lines close, the normal mechanism of heat removal from the primary system is unavailable. Consequently, both the primary and secondary system pressures rise.

The secondary system quickly pressurizes to the safety relief valve opening pressure, which results in the safety relief valves to open and then subsequently close when the closing pressure criterion is achieved. The relief flow through the SG SRVs is the principle primary system energy removal mechanism in the first hour. There is also energy removal through the RCP seal leakage, but the energy flow is small relative to the SG SRV flow.

The water inventory in the steam generators was completely boiled away by 1 hr 16 min. Although the steam generators relief valves continue to cycle and release steam, the associated heat removal is inadequate and the primary system sharply increases to the pressurizer safety relief valve opening pressure. The safety valves on the pressurizer begin opening and closing to remove excess energy. However, the pressurizer relief valve flow causes a steady decrease in the primary system coolant inventory (see Figure 39). The fuel starts to uncover at 2 hr 19 min (see Figure 40). The fuel cladding fails at 2 hr 59 min, which starts the release of fission products from the fuel. The fuel rods starts to degrade above 2400 K as the molten zirconium breaks through the oxidized shell of the cladding on the fuel rods and eventually collapse due a thermo-mechanical weakening of the remaining oxide shell at high temperature. As shown in Figure 40, the peak fuel-debris temperature reaches the fuel-zirconium oxide eutectic melting temperature of 2800 K.

Following the uncovery of the fuel, an in-vessel natural circulation flow develops between the hot fuel in the core and the cooler structures in the upper plenum. Hot gases rise out of the center of the core rise into the upper plenum and return down the cooler peripheral sections of the core. Simultaneously, a natural circulation circuit develops between the hot gases in the vessel and the steam generator [2]. Hot gases from inside the vessel flow along the top the hot leg and into the steam generator. The hot gases flow through the steam generator in approximately half the tubes and return through the remaining tubes. The large masses of the hot leg nozzle, hot leg piping, and the steam generator tubes absorb the heat from the gases exiting the vessel. The cooler gases leaving the steam generator return to the vessel along the bottom of the hot leg. Due to its close proximity to the hot gases exiting in the vessel, the hot leg nozzle at the carbon steel interface region to the stainless steel piping was predicted to fail by creep rupture at 3 hr 45 min.[16]

Upon creep failure of the hot leg nozzle, a large hole opened that rapidly depressurized the RCS (i.e., like a large break loss-of-coolant accident). The RCS depressurization permitted a complete accumulator injection at low-pressure (see water level rise at 3 hr 45 min on Figure 39). Although the water filled above the core region, the hottest fuel in the core remained in film boiling and continued to collapse and degrade (see Figure 40). The lower temperature regions on the periphery of the core quenched but subsequently reheated once the water level decreased into the core.

[16] Alternate failure locations could include the pressurizer surge line and the steam generator tubes. There was some residual water in the pressurizer that cooled the surge line. Due to the relatively high pressure in the steam generator secondary side, the resultant thermo-mechanical stresses across the steam generator tubes were less severe than the hot leg nozzle. Consequently, the most vulnerable location was calculated to be the hot leg nozzle. The initial failure location for this scenario is also part of an on-going investigation of another research program in the NRC.

Following the accumulator injection at 3 hr 45 minute, the decay heat from the fuel boiled away the injected water. By 4.3 hr, a large debris bed had formed in the center of the core. The debris continued to expand until 5.8 hr when all the fuel had collapsed and was resting on the core plate. The hot debris failed the core support plate and fell onto the lower core support plate, which failed at 6.6 hr. Following the lower core support plate failure, the debris bed relocated onto the lower head. The small amount of remaining water in the lower head was quickly boiled away. As shown in Figure 41, the hot debris quickly heated the lower head to above the melting temperature of stainless steel (i.e., 1700 K) on the inside surface. As the heat transferred through the lower head, it eventually failed at 7 hr 16 min due to the creep rupture failure criterion (i.e., primarily due the thermal stress component due to the low differential pressure).

By 7.5 hr, nearly all the hot debris relocated from the vessel into the reactor cavity in the containment under the reactor vessel. The hot debris boiled away the water in the reactor cavity started to ablate the concrete. The ex-vessel core-concrete interactions (CCI) continued for the remainder of the calculation, which generated non-condensable gases. In addition, the hot gases exiting the reactor cavity and the radioactive heating from airborne and settled fission products steadily evaporated the water on the containment floor outside the reactor cavity from 7.3 hr to 44 hr. The resultant non-condensable gas and steam generation pressurized the containment (see Figure 42). At 25.5 hr, the containment failed due to liner tearing near the containment equipment hatch at mid-height in the cylindrical region of the containment. The containment continues to pressurize until the leakage flow balanced the steam and non-condensable gas generation. By 44 hr, all the water on the floor has evaporated. The containment depressurized thereafter due to only a smaller gas loading from the non-condensable gas generation.

The containment failure location was the around the equipment hatch, which is located on the side of the containment without a surrounding building (e.g., not adjacent to the auxiliary or safeguards buildings). Consequently, all released fission products are released directly to the environment.

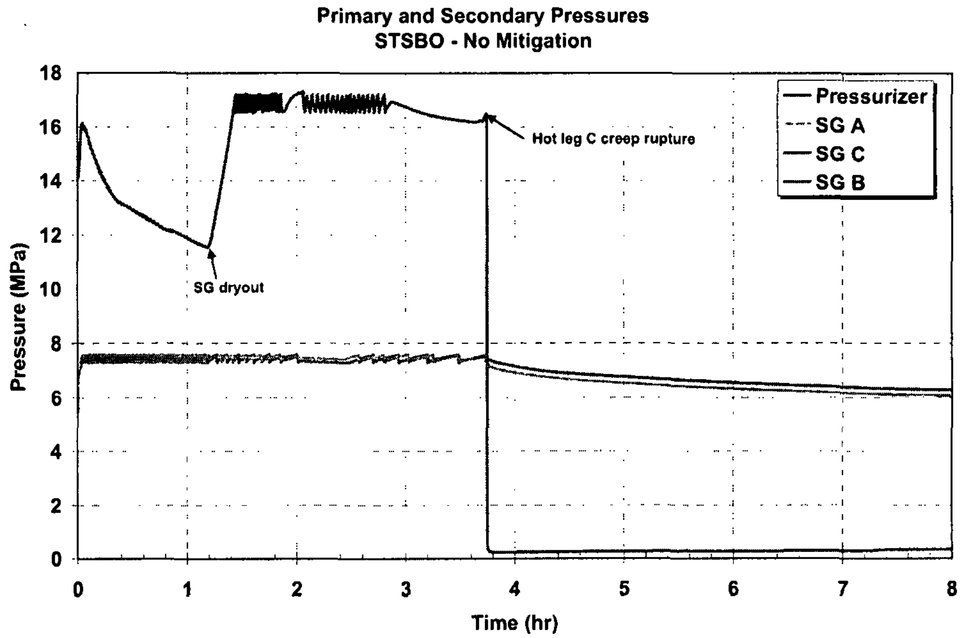

Figure 38 Unmitigated STSBO primary and secondary pressures history.

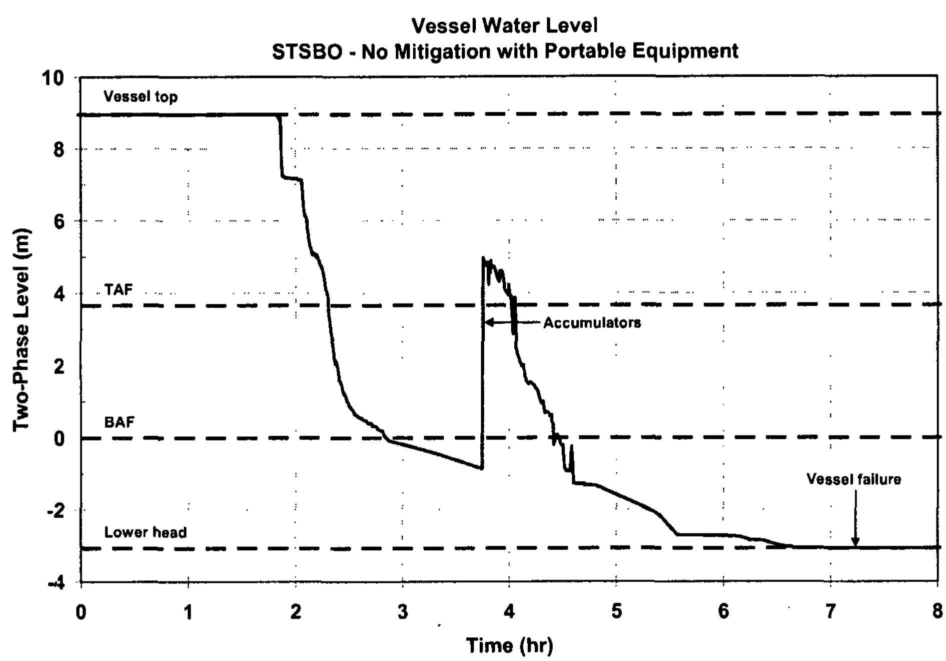

Figure 39 Unmitigated short-term station blackout vessel two-phase coolant level.

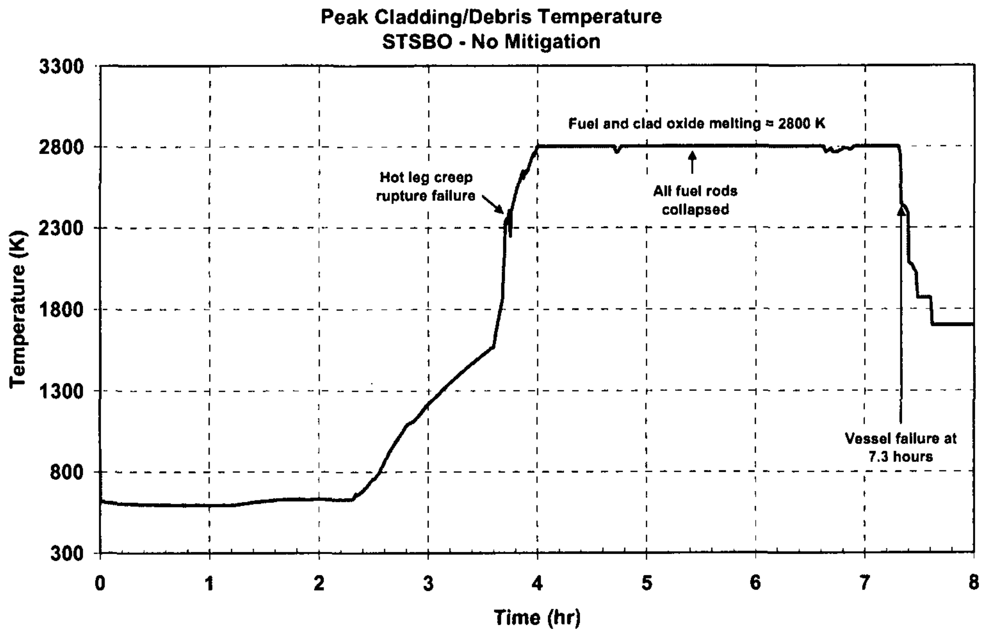

Figure 40 The unmitigated short-term station blackout core temperature history.

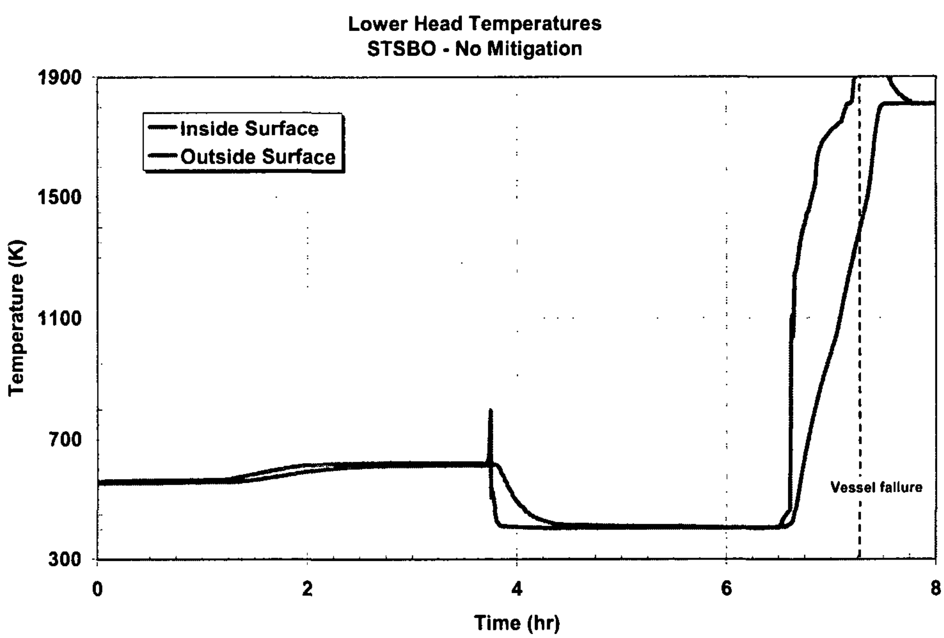

Figure 41 Unmitigated STSBO lower head inner and outer temperature history.

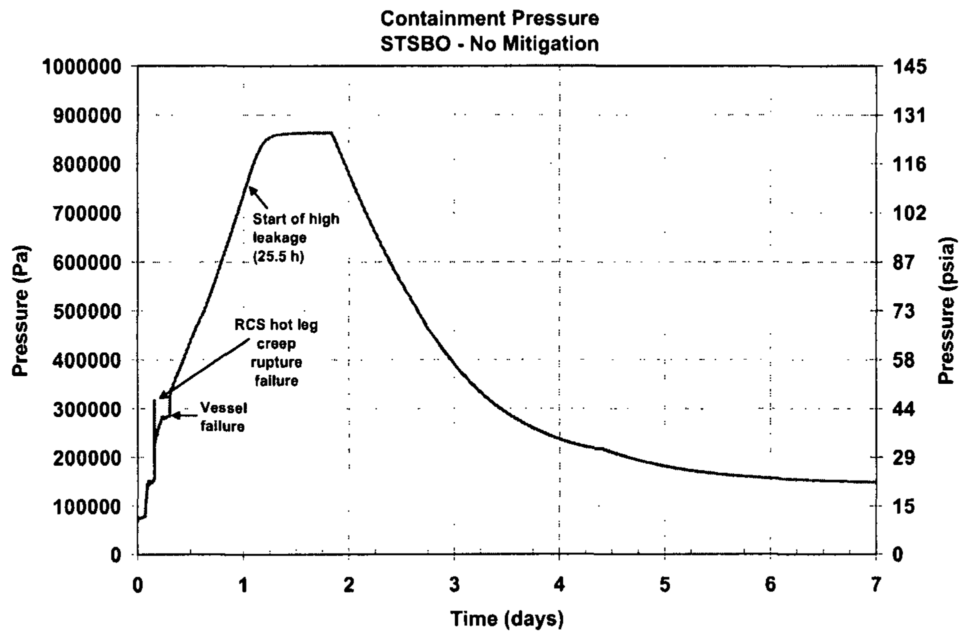

Figure 42 Unmitigated short-term station blackout containment pressure history.

5.2.1.2 Radionuclide Release

The fission product releases from the fuel started following the first thermo-mechanical failures of the fuel cladding in the hottest rods at 2 hr 57 min, or about 38 min after the uncovery of the top of the fuel rods. The in-vessel fission product release phase continued through vessel failure at 7.3 hr. Initially, the fission product releases from the fuel circulated through the primary system as well as being released to the containment through the pressurizer safety relief valves. Since the pressurizer relief tank (PRT) had failed about 1 hour before the start of the start of the fission product releases, there was essentially no ex-vessel scrubbing in the PRT. Following vessel failure, the fission product releases continued from the ex-vessel fuel in the reactor cavity.

Figure 43 and Figure 44 show the fission product distributions of the iodine and cesium radionuclides that were released from the fuel, respectively (see Appendix B.2 for a detailed radionuclide core inventory). Approximately 97% and 98% of the iodine and cesium, respectively, were released from the fuel prior to vessel failure while the remaining amount was released ex-vessel. At the time of the hot leg failure, approximately 40% of these volatile radionuclides had been released. The resultant blowdown of the vessel immediately discharged the airborne fission products to the containment. However, about 6% of the iodine and 5% of the cesium remained in the vessel. Most of the radionuclides retained in the RCS were deposited in the steam generators during the natural circulation phase of the accident. Following the RCS blowdown after the hot leg nozzle failure, more radionuclides were released from the fuel as the core further degraded. At low pressure conditions, the fission products continued to circulate within the vessel and to the steam generators with a substantial portion being depositing on the structural surfaces. However, as shown in the figures, the majority of the released radionuclides were transported to the containment. Within the first day, most of the airborne fission products in the containment settled on surfaces. This was significant because the containment failure occurred at 25 hr 32 min. Consequently, there was little airborne mass that could be released to the environment.

The chemical form of the released iodine was cesium-iodine, which was more volatile than the predominant form of the released cesium, which was cesium-molybdate (Cs_2MoO_4). As shown in the iodine history figure (see Figure 43), the in-vessel iodine mass was decreasing following vessel failure until approximately 2.9 days. The decrease of mass represents a vaporization process of previously deposited radionuclides. The late in-vessel vaporization release was significant because it continued after containment failure and had a significant contribution to the overall environmental release. The thermal mechanisms for the vaporization of the iodine were from two sources. First, the fission product decay of the settled radionuclides heated the structures. Second, a natural circulation flow of very hot gases (i.e., from 1050 K to >1600 K) flowed from the reactor cavity, through the failed vessel lower head, through the reactor vessel, and out the failed hot leg nozzle. As the deposited cesium-iodine heated, gaseous iodine was released and the cesium remained chemisorbed to the stainless steel surfaces. The natural circulation flow pattern also effectively vented the gaseous iodine from the RCS to the containment.

In contrast to the iodine response in Figure 43, the deposited cesium molybdate was less volatile and remained deposited in the vessel (see Figure 44). Except for inside the reactor cavity, the containment was cooler than vessel and well below conditions that would vaporize settled radionuclides. Consequently, none of the deposited radionuclides (i.e., including iodine) in the containment vaporized.

Finally, Figure 45 summarizes the releases of the radionuclides to the environment. At 4 days, 92 % of the noble gases, 1.7% of the tellurium, 1.0% if the iodine, 0.75% of the radioactive cadmium, 0.4% of the cesium, and 0.1% of the barium and radioactive tin had been released to the environment. All other releases were less than 0.1% of the initial inventory. There were some environmental releases prior to the containment failure at 25.5 hr due to nominal leakages (i.e., design specification of 0.1% vol/day at the design pressure). The releases to the environment increased sharply after the failure of the containment. Between 25.5 hr, 48 hr, 53% of the airborne noble gases in the containment were released. Over the next 2 days, ~40% more was released.

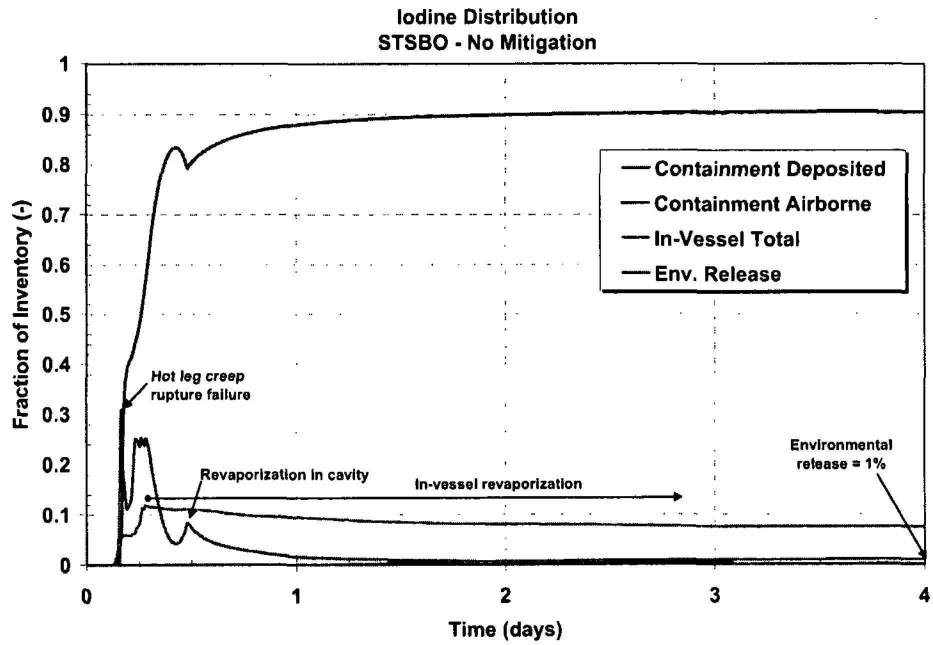

Figure 43 Unmitigated STSBO iodine fission product distribution history.

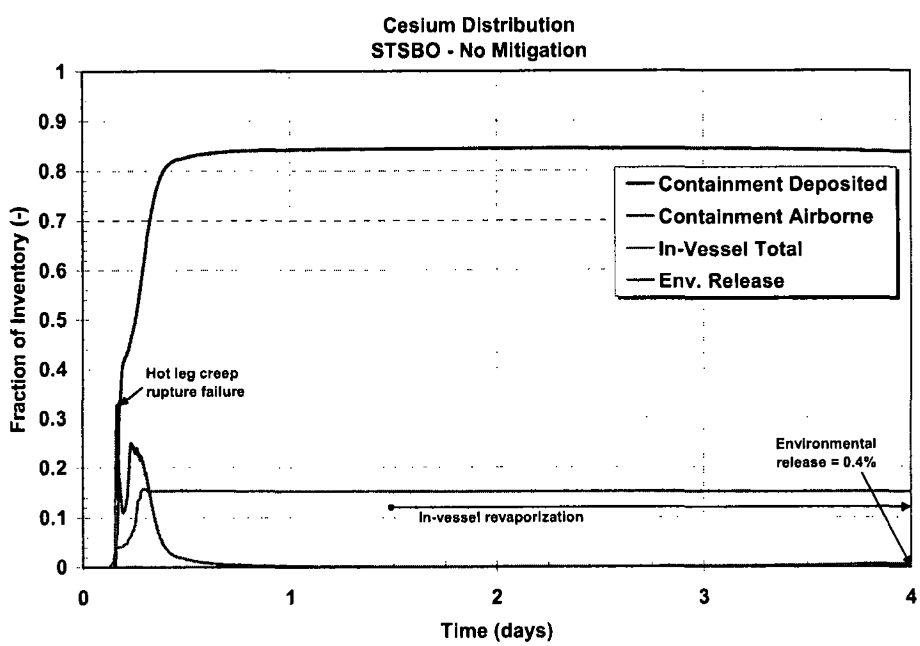

Figure 44 The unmitigated STSBO cesium fission product distribution history.

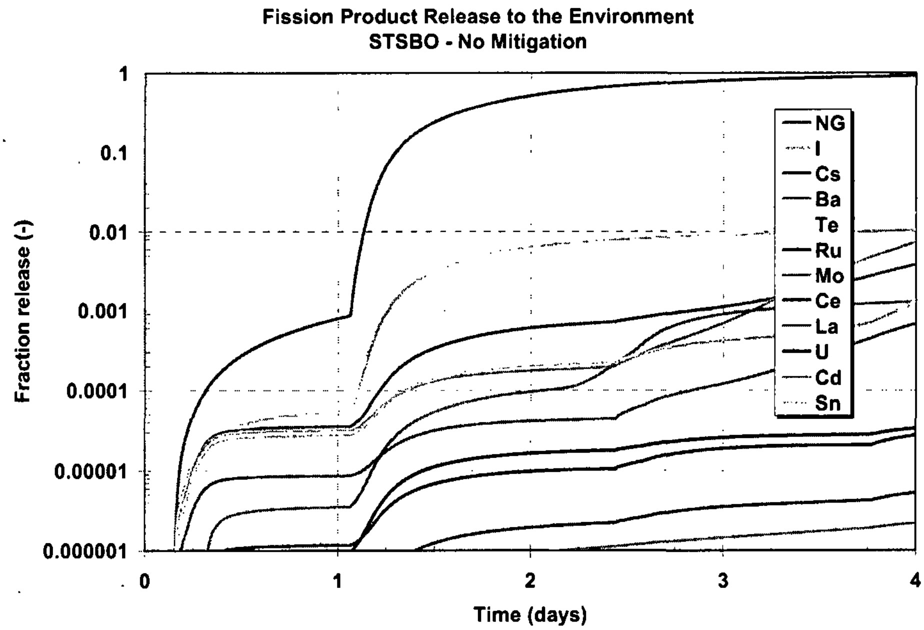

Figure 45 Unmitigated STSBO environmental release history of all fission products.

5.2.2 Mitigated Short-Term Station Blackout

Table 6 summarizes the timing of the key events in the mitigated STSBO. As described in 3.2, the accident scenario initiates with a complete loss of all onsite and offsite power. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. The timings of the key events are discussed further in Sections 5.2.2.1 and 5.2.2.2. Unlike the unmitigated STSBO described in Section 5.2.1, the mitigated STSBO credits the successful connection of the portable, low-pressure, diesel-driven (Godwin) pump to the containment spray system at 8 hr. The Godwin pump is a high-flow, low-head pump with a design capacity of 2000 gpm at 120 psi.[17] A reliable source of water is maintained while 1,000,000 gallons is injected into the containment through the containment sprays. The sequence of events is identical to the unmitigated STSBO until 8 hr. In particular, the core has degraded and failed the vessel lower head prior to the spray actuation (see Table 6). The emergency containment sprays are effective at reducing the containment pressure and knocking down airborne fission products while they are operating. However, the containment subsequently pressurizes after the sprays are terminated to the failure pressure. While not investigated, intermittent operation of the sprays and deeper flooding could have further delayed failure of the containment.

Table 6 Timing of key events for mitigated short-term station blackout.

[17] The rated containment pump spray flowrate was 3200 gpm. It was judged that the portable Godwin pump would pressurize the system and develop the spray droplet flow pattern.

Event Description	Time (hh:mm)
Initiating event Station blackout – loss of all onsite and offsite AC and DC power	00:00
MSIVs close Reactor trip RCP seals initially leak at 21 gpm/pump TD-AFW fails	00:00
First SG SRV opening	00:03
SG dryout	01:16
Pressurizer SRV opens	01:27
Pressurizer relief tank rupture disk opens	01:46
Start of fuel heatup	02:19
RCP seal failures	02:45
First fission product gap releases	02:57
Creep rupture failure of the A loop hot leg nozzle	03:45
Accumulators start discharging	03:45
Accumulators are empty	03:45
Vessel lower head failure by creep rupture	07:16
Debris discharge to reactor cavity	07:16
Cavity dryout	07:27
Start of containment sprays	8:00
End of containment sprays (1,000,000 gal)	15:02
Containment at design pressure (45 psig)	40:00
Start of increased leakage of containment ($P/P_{design} = 2.18$)	66:30

5.2.2.1 Thermal-Hydraulic Response

The progression of events in the mitigated STSBO is identical to the unmitigated STSBO as described in Section 5.2.1 through the first 8 hr, which includes core degradation and vessel failure. Consequently, the system pressure and peak fuel temperature responses though the first 8 hr is identical between the two sequences (i.e., compare Figure 46 and Figure 38, Figure 48 and Figure 40, and Figure 49 and Figure 41). However, after 8 hr, there are some key differences. For example, the long-term reactor vessel level shows different behavior after 8 hr. Although the water in the vessel boils away by 6.6 hr when the core relocates onto the lower head, the vessel water level starts to recover at 13.7 hr as shown in Figure 47. After 5.7 hr of containment spray operation, the water level in the containment was calculated to fill above the bottom of the vessel.[18] At 15 hr when the containment spray was terminated, the containment water had flooded ~1.3 m into the vessel.

The reactor cavity of Unit 1 of the Surry containment connects to the surrounding lower regions of the containment through (a) a 12" hole in the reactor cavity wall at 2'-7" above the bottom of the floor, (b) a penetration to ring duct bus at 24'-3" above the floor, and (c) the holes in the cavity wall for the RCS piping (nearly 40' above the bottom of the floor). In the case of this scenario, the lower hole into the reactor cavity was flooded whereas the upper openings were well above the water level. Hence, there was no natural circulation of water from the containment basement into the reactor cavity and out the gaps at the RCS piping penetration. While the containment sprays were operating, a significant portion of the spray water drained into the reactor cavity from refueling pool. The resultant water flow through the reactor cavity removed the heat from the fuel debris. Once the spray flow stopped, the water in the cavity heated to saturation conditions and started to boil. The resulting steam load from the boiling pressurized the containment to the failure pressure (see Figure 50). Although there was 1,000,000 gallons of water in the containment, the core debris was only in thermal contact with ~40,000 gallons in the reactor cavity. Consequently, the containment pressurized to failure conditions much faster than if all 1,000,000 gallons were being heated. As stated in Section 4.2, intermittent spray operation and/or flooding above the RCS piping penetrations would have substantially delayed containment failure. Although the containment sprays did not prevent containment failure, they delayed containment failure by over 40 hr relative to the unmitigated case.

Although the exact conditions following a severe seismic event were not known, it was estimated that portable sprays could be started by 8 hr. Based on the pressure response of the unmitigated STSBO, the containment sprays must be started before 15.6 hr while the containment pressure was below the shutoff head of the portable pump. Once the containment sprays are initiated, they are effective in quickly reducing the containment pressure. Consequently, there was almost eight additional hours from the assumed starting time to establish containment sprays, or 15.6 hr after the start of the scenario.

The selection of the containment sprays as a mitigation technique for this scenario was particularly beneficial for several reasons. First, as will be shown in Section 5.2.1.2, the containment sprays were extremely effective in knocking down the airborne aerosols into the large containment pool. Second, the sprays delayed containment failure for an additional 41 hr. In contrast, the alternate strategies of containment flooding or vessel injection would not be expected to be as beneficial for this scenario (i.e., assuming an initiation time after vessel failure). The high-head portable pump used for vessel injection can only provide 150 gpm versus >2000 gpm for the portable containment spray pump. Consequently, the time to deep flood the containment would be significantly longer. More importantly, the water would merely fall out of the failed vessel and not reduce the containment pressure (i.e., versus the highly effective heat and mass transfer from the containment spray system). In fact, the small amount

[18] At the time of the calculation, the exact flooding water level characteristics of the Surry containment were not known. Subsequently, information was obtained from the plant that shows ~1,160,000 gal are needed to fill to the bottom of the vessel. Consequently, the calculated water level response of this scenario is actually consistent with a slightly higher integrated spray flow.

of water flooding onto the ex-vessel core debris would enhance the pressurization of the containment versus a dry reactor cavity. Furthermore, the injection flow would not directly knockdown the airborne aerosol radionuclides. Similarly, direct containment injection using the high flow pump would not depressurize the containment nor reduce any airborne fission products.

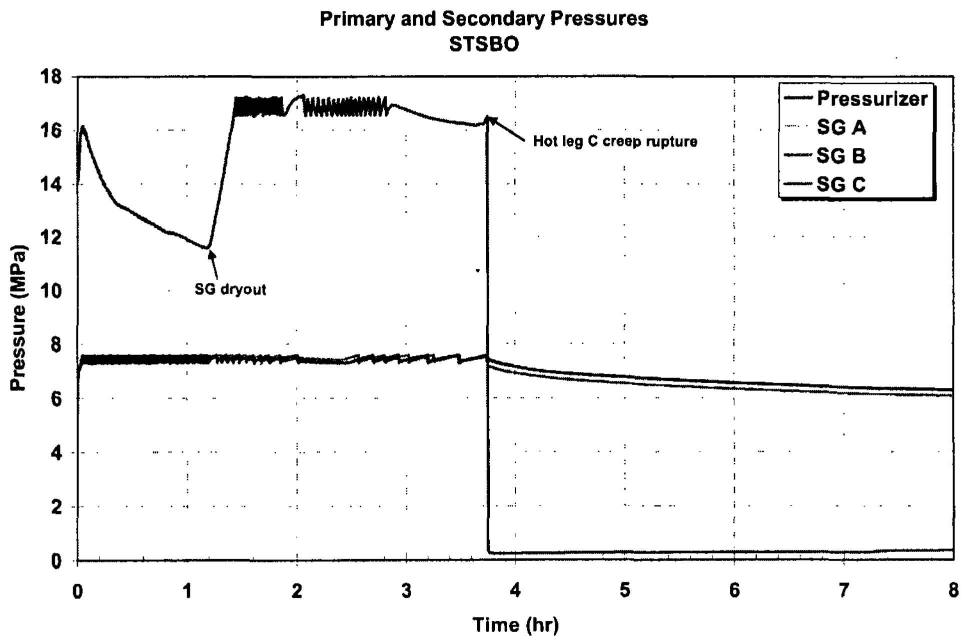

Figure 46 The mitigated STSBO primary and secondary pressure history.

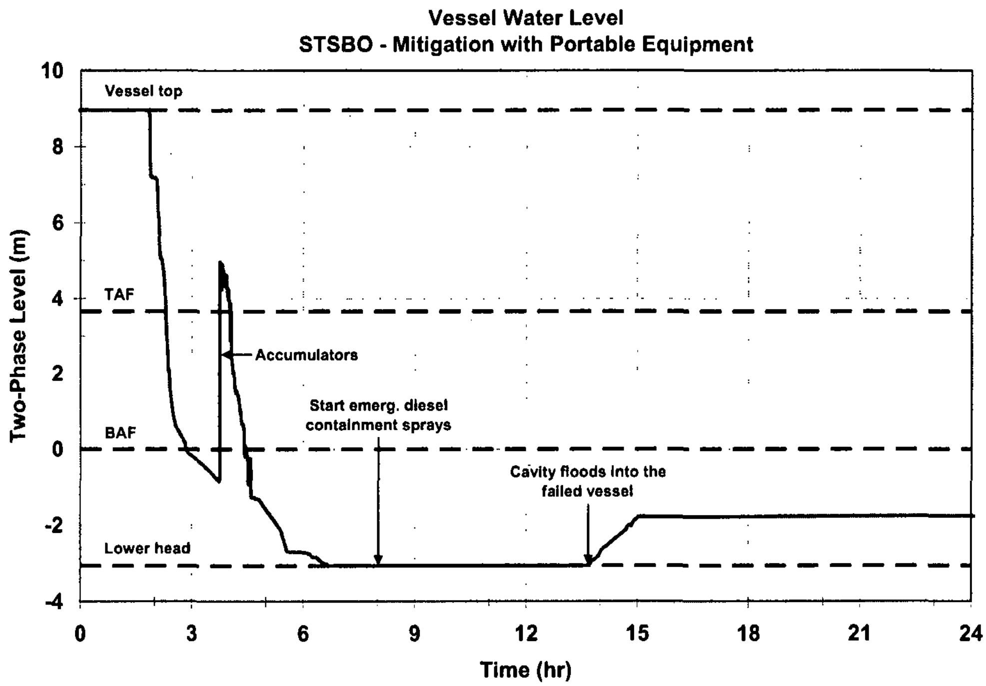

Figure 47 The mitigated short-term station blackout vessel two-phase coolant level.

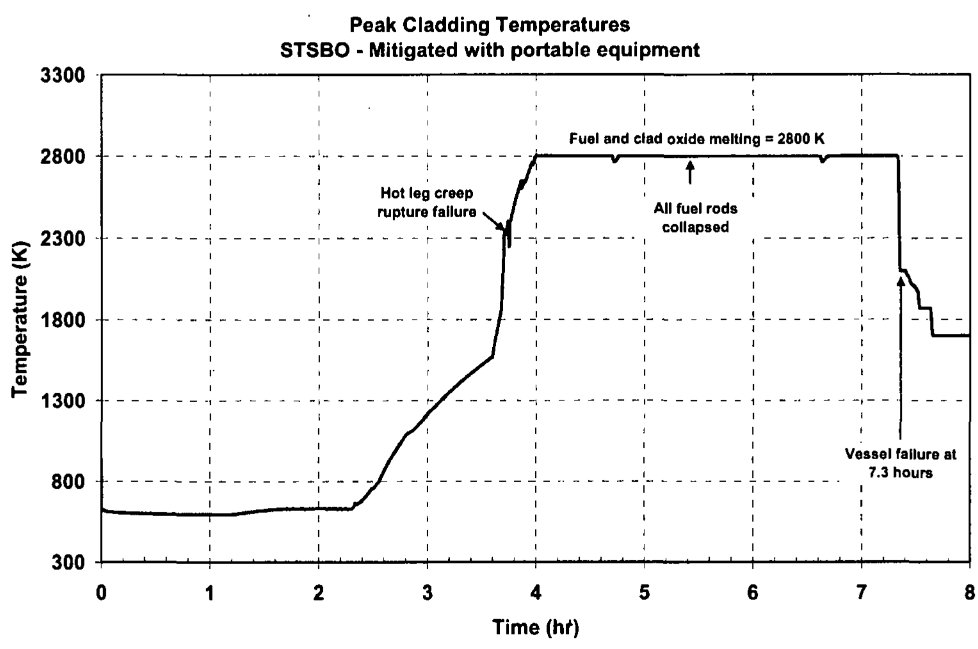

Figure 48 The mitigated short-term station blackout core temperature history.

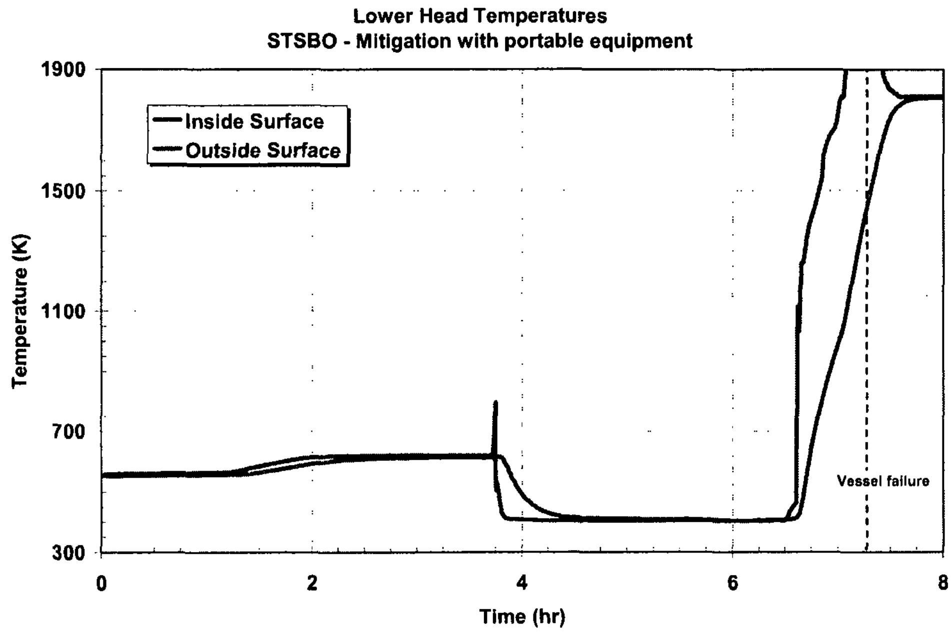

Figure 49 Mitigated STSBO lower head inner and outer temperature history.

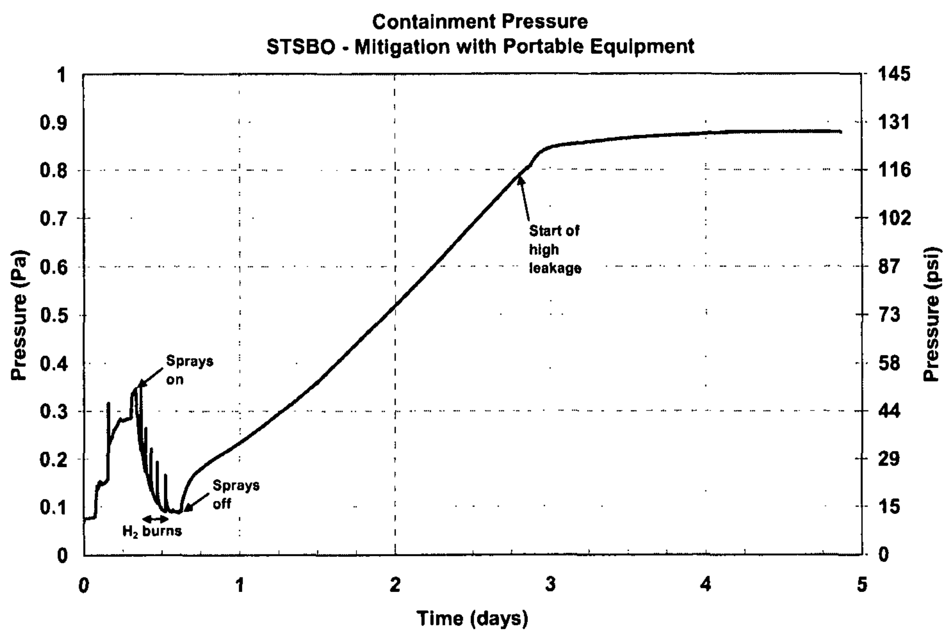

Figure 50 The mitigated short-term station blackout containment pressure history.

5.2.2.2 Radionuclide Release

The radionuclide response of the mitigated STSBO is identical to the unmitigated response described in Section 4.1.2 for the first 8 hr, or through vessel failure until the start of the containment sprays. Following the start of the emergency containment sprays at 8 hr (0.25 days), the airborne aerosols of iodine and cesium rapidly decease (see Figure 51 and Figure 52, respectively). By the time the sprays are terminated at 15 hr (0.63 days), almost all of the airborne aerosols have been captured in the pool on the containment floor. Since the containment failure was delayed until 66 hr 30 min (2.8 days), the amount of airborne mass available for release was insignificant. The environmental release of iodine and cesium was very small (i.e., 0.007% and 0.003%, respectively).

Due to the deep flooding in the reactor cavity by the spray operation, the bottom of the failed vessel lower head is submerged in water. Therefore, the natural hot circulation flow that promoted vaporization of deposited radionuclides in the unmitigated STSBO is not present. Instead, the water pool in the reactor cavity cools the bottom of the vessel. Due to the boiling in the cavity, relatively cool steam flows through the vessel and out the failed hot leg nozzle location, which also removes heat and inhibits vaporization of deposited radionuclides in the upper vessel and hot leg. Consequently, the vaporization of the in-vessel deposited fission products (i.e., especially cesium-iodine) that was seen in the unmitigated STSBO (i.e., characteristic of vaporization), was negligible in the mitigated case through 4 days.

Finally, Figure 53 summarizes the releases of the radionuclides to the environment. At 4 days, 60% of the noble gases, 0.0046% of the tellurium, 0.0065% of the iodine and cadmium, 0.0027% of the cesium had been released to the environment. Except for the noble gases, all the releases were less than 0.01% of the initial inventory. As shown in the figure, the initial releases to the environment were due to the nominal leakages (i.e., design specification of 0.1% vol/day at the design pressure) prior to the containment failure at 66 hr 30 min (2.8 days). Following containment failure at 2.8 days until 4 days, the noble gas release went from 0.24% to 60%, which represents a significant flushing of the containment gas space to the environment. There is some evidence of increased leakage of the other radionuclides after containment failure. However, the response is exaggerated in the figure due to the semi-logarithmic scale. The absolute magnitude of the releases was small relative to the unmitigated response.

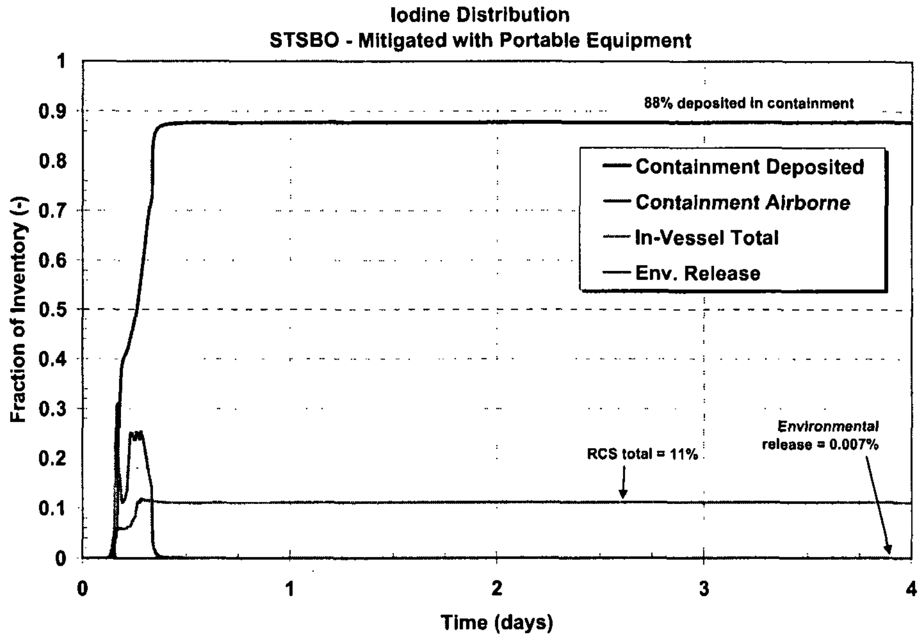

Figure 51 Mitigated short-term station blackout iodine fission product distribution history.

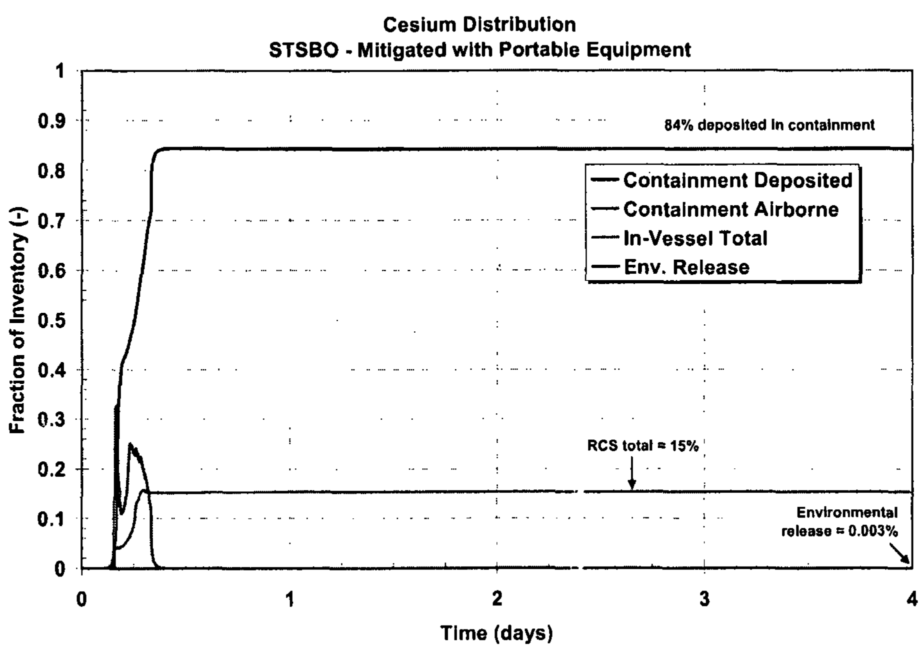

Figure 52 Mitigated STSBO cesium fission product distribution history.

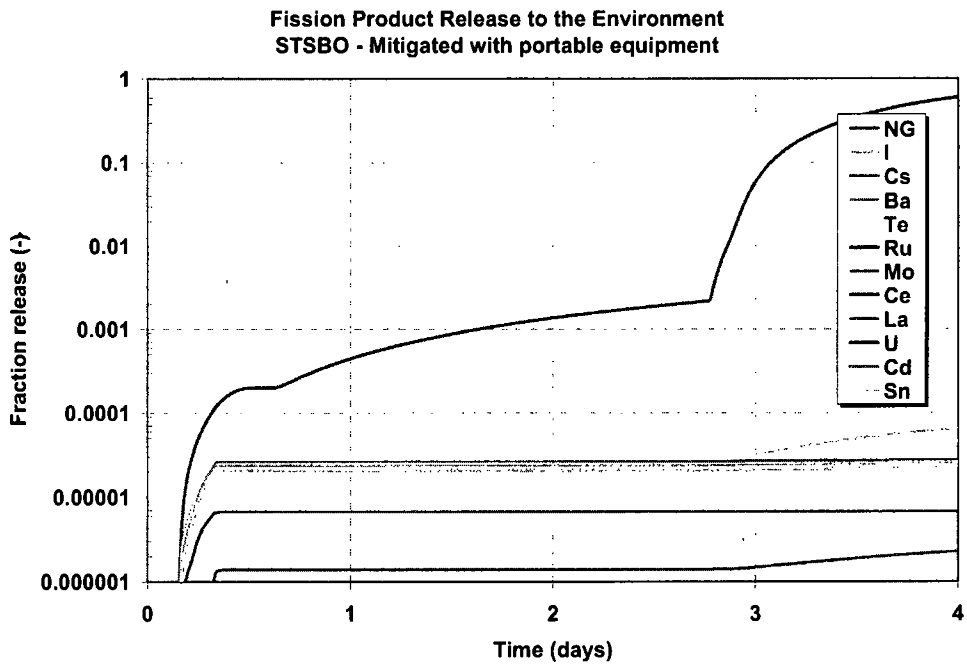

Figure 53 Mitigated STSBO environmental release history of all fission products.

5.2.3 Uncertainties in the Hydrogen Combustion in the Mitigated Short-Term Station Blackout

During the peer review of the mitigated short-term station blackout, there was concern about the undesirable effects of combustion following an emergency spray actuation in the containment. Although emergency sprays mitigated the accident, there was severe damage to the fuel with considerable hydrogen production. As shown in Figure 54, 200 kg of hydrogen was produced by the time of the hot leg failure (i.e., 3:45 hr). There was a hydrogen burn coincident with hot leg failure. Subsequently, the hydrogen production continued and more than 300 kg of hydrogen was produced by vessel failure (i.e., 7:16 hr).

In the unmitigated case, new steam production following vessel failure by residual water in the cavity kept the containment inerted.[19] Consequently, there were no further hydrogen burns. In contrast, the mitigated case had containment sprays, which condensed the steam in the containment atmosphere and reduced the containment pressure. The net effects of the emergency spray operation between 8 to 15 hours were high hydrogen and oxygen concentrations, good mixing, and a low steam concentration (i.e., conditions suitable for combustion). Hydrogen combustion was predicted whenever the gaseous concentrations reached the specified levels in MELCOR's default combustion model (i.e., $\mathcal{X}_{steam}$<55%,

[19] As described in Section 4.6, MELCOR's default hydrogen combustion model was used, which identifies steam concentrations greater than 55% to be steam-inerted and unable to support a hydrogen burn. Following vessel failure, the ex-vessel debris boiled away the water on the containment floor to maintain a high steam concentration.

$\chi_{hydrogen}$>10%, and χ_{oxygen} >5%). This resulted in several smaller burns as shown in Figure 54. Due the uncertainty of the ignition source, the SOARCA peer review group inquired about the consequences of a later but larger burn.

Several facets were investigated relative to the SOARCA peer review group comment. First, the potential ignition sources were reviewed. The most likely sources are the hot gases exiting the failed hot leg (i.e., a hot jet or hot pipe) and debris when the vessel fails. Two locations were examined, the containment cavity (see Figure 55) and the containment dome (i.e., see Figure 56, representative of the bulk conditions due to high mixing during spray operation).[20] Hot, hydrogen-rich gases discharge into the cavity following hot leg failure. The temperature of the gases is well above the auto-ignition temperature for a hydrogen jet (i.e., 950-110 K) as shown in see Figure 55. Hence, ignition is likely, which occurred in the base calculation. The specific conditions at vessel failure are summarized in the figure. However, due to the subsequent discharge of the accumulators and full depressurization of the primary system, the high jet temperature flow stopped shortly thereafter. Furthermore, steam from the vessel quickly inerted the cavity atmosphere to >90%. Following vessel failure, a hot debris ignition source existed in the cavity but the rapid steam production inerted the cavity atmosphere further. Although the hydrogen concentration was high, the steam concentration kept the containment inerted.

For the response of the regions outside the cavity (i.e., characterized by the dome region in Figure 56), the response was similar to the cavity at hot leg failure. A high temperature, hydrogen-rich gas jet exited into the dome for a short period of time until the accumulators discharged. Hence, ignition is likely in the dome within the jet. However, as shown in both the cavity and dome figures, the steam concentration rapidly increased to inerting conditions (i.e., χ_{steam}>55%) once the accumulators discharge. Subsequently, after vessel failure, the debris exiting the vessel could entrain cinders outside the cavity or a burn in the cavity could propagate into the surrounding regions. However, as indicated in Figure 56, the oxygen concentration is low (i.e., below the default ignition criterion) and the steam concentration is high (i.e., above the default ignition criterion). However, the hydrogen concentration is above the default ignition criterion when an ignition source is present (i.e., >7%).

In summary, ignition would be expected at vessel failure, which occurred. However, the amount of hydrogen available for combustion is limited at this phase of the accident and the pressurization during the burn was well below the pressure capacity of the containment. Ignition following hot leg failure (e.g., vessel failure, which is the next clear ignition source), is less certain due to high steam inerting throughout the containment. Consequently, the response in the unmitigated case was judged reasonable.

In the second facet of hydrogen uncertainty examined, the response of the mitigated case with emergency sprays introduces new phenomena after 8 hours, which warranted investigation. The spray operation led to conditions where hydrogen combustion was possible (i.e., see multiple burns during the spray operation in Figure 54). Consequently, a sensitivity case was run to investigate a delayed burn with a larger amount of hydrogen accumulation. In this case, two

[20] The unmitigated short-term station blackout is shown in Figure 55 and Figure 56, which had identical response to the mitigated case prior to the emergency containment spray actuation (i.e., 8 hours). Note, Figure 54 is the mitigated containment response, which includes emergency containment spray operation.

conservatisms were applied relative to the base case. First, all combustion in the containment was prevented until the emergency spray operation completed (i.e., ~15 hr). Second, an ignition source was activated simultaneously in all regions of the containment at 15 hr, which initiated burns without any propagation delay. The response in the sensitivity case is shown in Figure 57. The resulting pressure rise was 183 kPa (26.5 psi). However, the peak pressure was well below the failure pressure (779 kPa, 113 psi).

In the sensitivity calculation, MELCOR's default deflagration model was used that considers the relative gas concentrations and the overall geometry for a best-estimate calculation. For completeness, two conservative alternate combustion models were examined. First, the adiabatic, isochoric, complete combustion (AICC) pressure was calculated [24]. The complete combustion assumption refers to the participation of the reactants only. Some small fraction of the fuel will always exist in equilibrium with the combustion products. The AICC assumptions result in the highest possible equilibrium pressures. Inclusion of best-estimate heat transfer, volume expansion, and incomplete combustion will result in lower pressures (e.g., the default MELCOR model). If the AICC process assumptions are met, then at equilibrium, simple deflagrations, accelerated flames, and detonations reach the same final AICC pressure.[21]

Figure 58 shows the bulk hydrogen and oxygen concentrations in the containment. Due to the emergency spray operation, there was good mixing throughout the containment and the bulk minimum and maximum values were similar. At 15 hr, the peak hydrogen concentration was ~20% and the peak oxygen concentration was 12%. Following the large burn a 15 hours, the final hydrogen concentration was ~6% and the final oxygen concentration was essentially 0%. Hence, the hydrogen and carbon monoxide[22] combustion consumed all the oxygen. While physically impossible due to the amount of oxygen, the AICC pressure for a 20% hydrogen concentration is shown on Figure 59. The peak pressure was just slightly below the containment failure pressure. If the AICC pressure is adjusted for the amount of available oxygen, the peak pressure would be lower.[23]

Next, peak pressure from a detonation was estimated. The pressure of a freely propagating detonation exceeds the equilibrium AICC pressure. The detonation pressure can be estimated using the Chapman-Jouguet (CJ) model [25]. The CJ model is derived from conservation of mass, momentum, and energy across a one-dimensional flow discontinuity. The shock wave is assumed to be sonic. It is known that gaseous detonation waves are three-dimensional and are not discontinuously thin. However, the CJ model predicts the measured detonation pressure within about 15%. The CJ pressure represents the one-dimensional average of the actual pressure in a detonation wave. At 15 hours, the CJ pressure was calculated to be 12.5, or slightly less than twice the AICC pressure. Consequently, a detonation would momentarily exceed the pressure capabilities of the containment. As stated above, the final equilibrium pressure is below the AICC pressure.

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[21] The difference between deflagrations and detonations in a confined volume is the transient pressure-time histories between ignition and final equilibrium.

[22] For simplicity, the carbon monoxide gas concentration is not included in the figure. Carbon monoxide is a byproduct of the ex-vessel core-concrete interactions. The carbon monoxide concentration was 14-15% before the burn and ~5% thereafter.

[23] These calculations do not include the influence of carbon-monoxide, which has a lower heat of combustion than hydrogen.

To assess the impact of the containment failure on an earlier containment failing combustion event, the timing of the burn and the benefits of the emergency spray airborne radionuclide scrubbing should be simulateously considered. Figure 60 shows the airborne concentration of cesiom and iodine aerosols as a function of the containment hydrogen concentration. Following the actuation of the sprays, the airborne aerosol mass decreases rapidly. Before the hydrogen concentration reaches a detonable quantity, there is a negligible mass of airborne aerosols for release if a subsequent detonation should fail the containment.

In summary, emergency sprays are needed to achieve a combustible or detonable mixture. The best-estimate MELCOR models do not predict sufficiently high pressures to challenge the containment whether the default ignition model is used or ignition is delayed until the spray termination. The conservative AICC model predicts a pressure close to the containment failure critieron. However, there is insufficient oxygen for complete combustion of all the hydrogen used in the AICC evaluation. Consequently, the maximum possible containment pressure is below the AICC pressure. Finally, the CJ detonation pressure was calculated. In the event of a detonation, the peak pressure would momentarily exceed the containment failure pressure before decreasing to an equilibrium value at or below the AICC pressure If the detonation pressure wave fails the containment, then the release of fission products could occur as early as 9.3 hours. However, the sprays are effective at settling airborne aerosols before detonable quantities could be formed that could fail the containment. The resulting fission product release would consist of only noble gases and would not be expected to substantially increase the offsite health consequences. In summary, the conditions that potentially lead to severe combustion or detonable events (i.e., emergency spray operation) also include enhanced scrubbing of the airborne aerosols, which minimizes the impact of an early containment failure. Consequently, the best-estimate response reported in Section 5.2.2 is a reasonable representation of the source term.

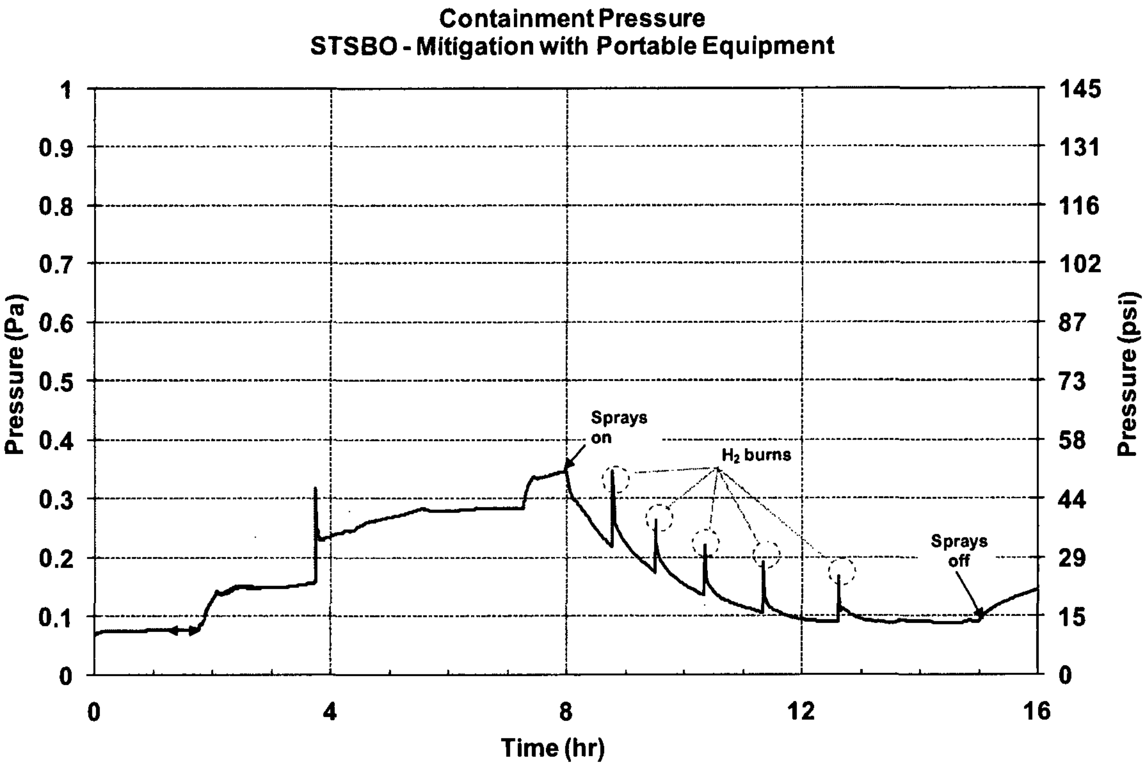

Figure 54 Comparison of the mitigated short-term station blackout containment pressure history versus the dome hydrogen mass and total hydrogen production.

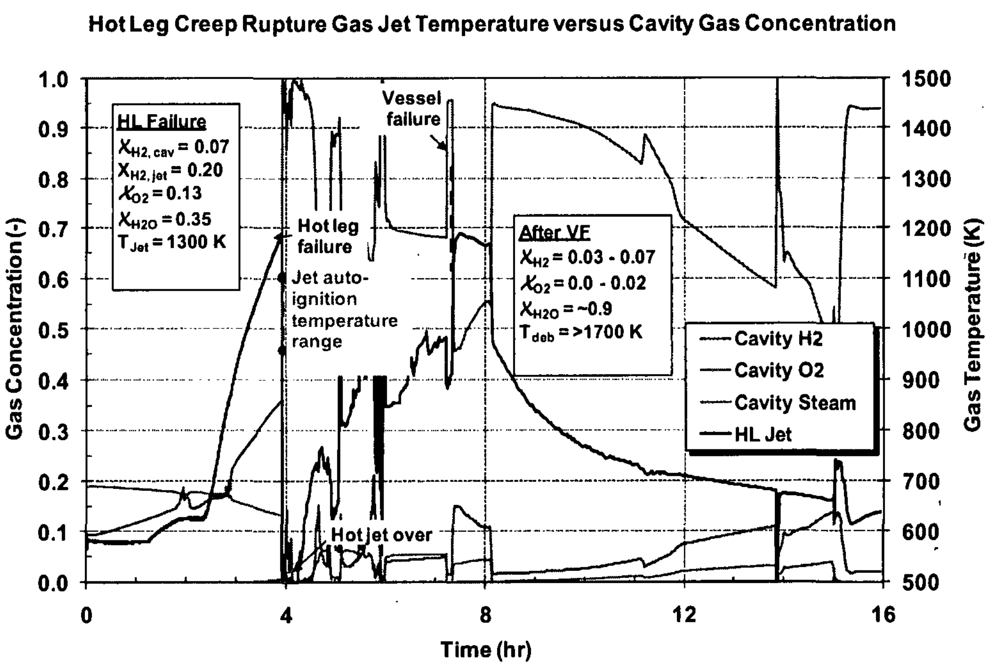

Figure 55 Comparison of the unmitigated short-term station blackout containment cavity gas concentration history and potential ignition source temperatures

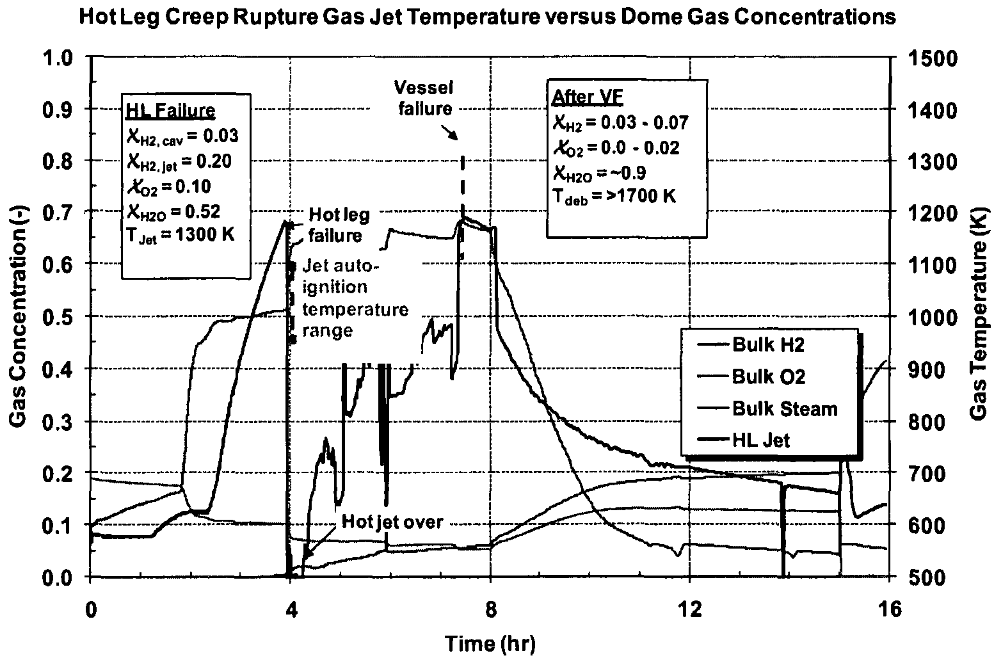

Figure 56 Comparison of the unmitigated short-term station blackout containment dome gas concentration history and potential ignition source temperature.

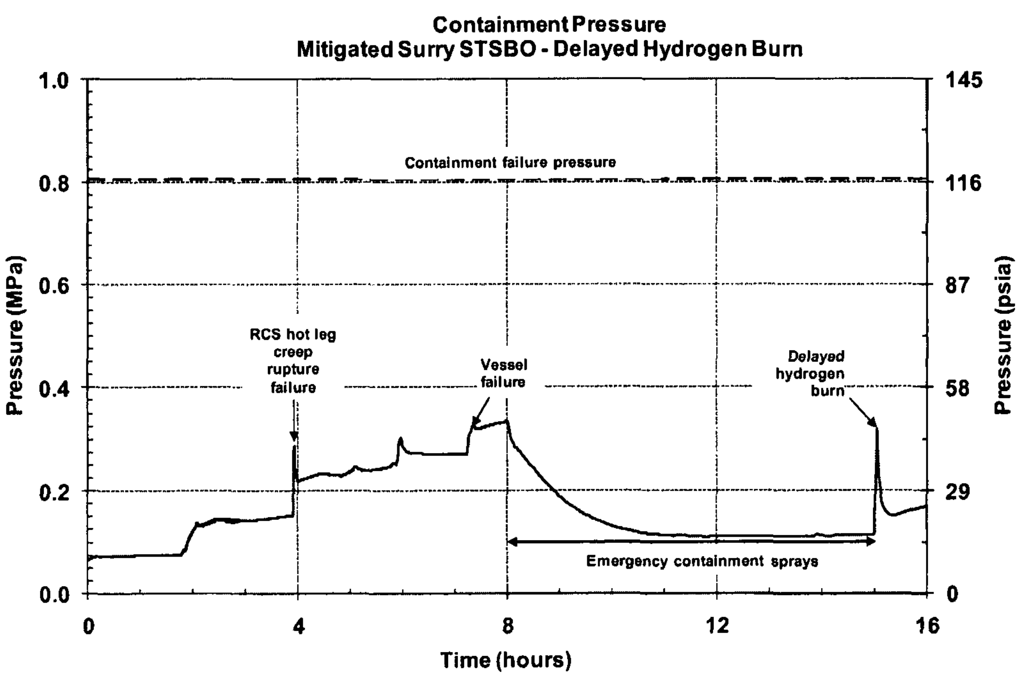

Figure 57 Mitigated short-term station blackout containment pressure history for the sensitivity calculation with delayed ignition.

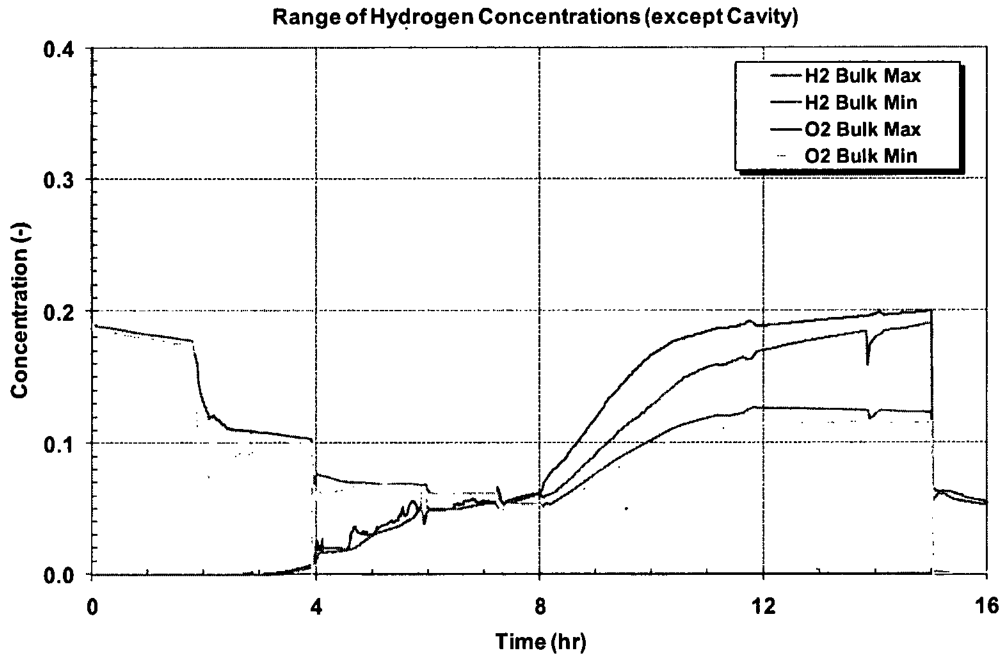

Figure 58 Mitigated short-term station blackout containment gas concentration history for the sensitivity calculation with delayed ignition.

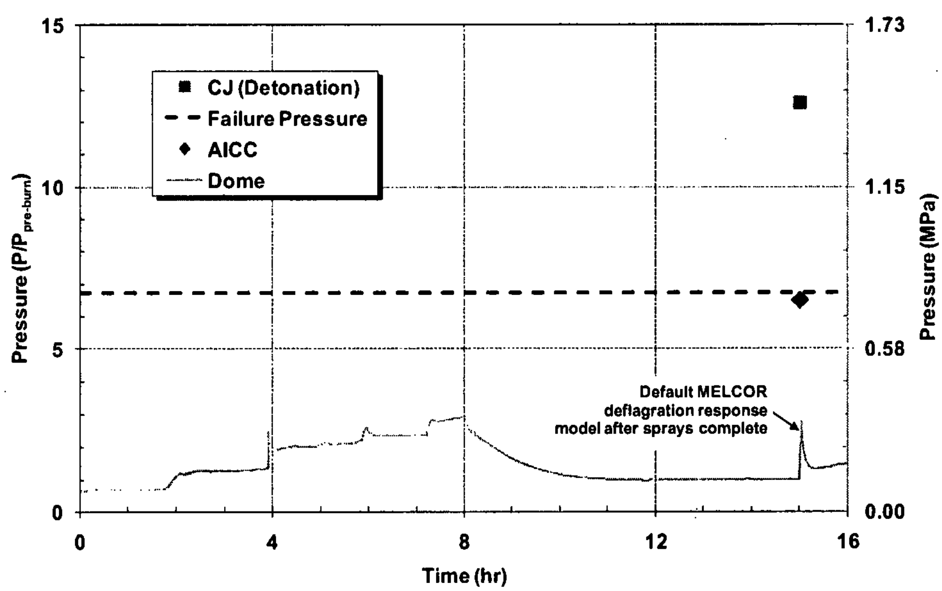

Figure 59 Mitigated short-term station blackout containment pressure history for the sensitivity calculation with delayed ignition.

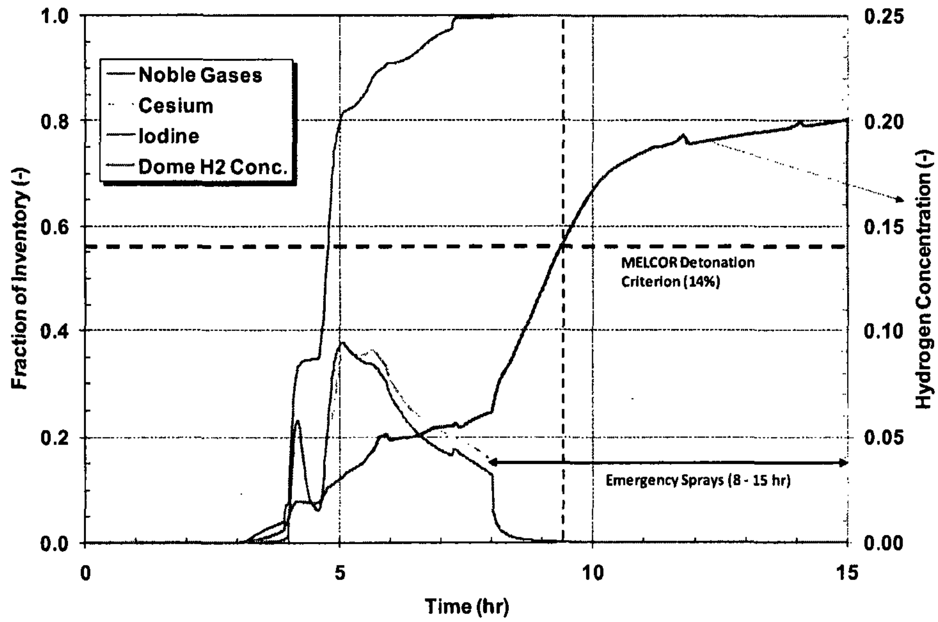

Figure 60 Mitigated short-term station blackout containment airborne radionuclide history versus hydrogen concentrations for the sensitivity calculation with delayed ignition.

5.3 Short-Term Station Blackout with Thermally-Induced SGTR

The short-term station blackout with thermally-induced SGTR scenario is assumed to be initiated by a large seismic event. Section 5.3.1 presents the results of an unmitigated scenario with no successful operator actions. For the mitigated scenario in Section 5.2.2, a portable emergency pump is connected to the containment spray system at 8 hours and available to inject 1,000,000 gallons.

5.3.1 Unmitigated Short-Term Station Blackout with Thermally-Induced Steam Generator Tube Rupture

Table 7 summarizes the timings of the key events for the unmitigated STSBO with a thermally-induced steam generator tube rupture scenarios. Unlike the unmitigated STSBO described in Section 5.2.1, either one (i.e., equivalent of 100% flow area) or two (i.e., equivalent of 200% flow area) steam generator tubes fail prior to any other RCS creep rupture failures along with a stuck open secondary safety relief valve. Consequently, there is a containment bypass pathway for fission products once the steam generator tubes fail. As described in Section 3.2, the accident scenario initiates with a complete loss of all onsite and offsite power. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. The timings of the key events are discussed further in Sections 5.3.1.1 and 5.3.1.2. Similar to the unmitigated STSBO, the fission product releases from the fuel do not begin until 2 hr 57 min. However, since a steam generator PORV sticks open at 3 hr and the tubes fail at 3 hr 33 min, fission product releases to the environment can begin earlier than the unmitigated STSBO described in Section 5.2.1 (i.e., 3 hr 33 min versus 25 hr 32 min). Two cases were performed to examine the sensitivity of the tube failure size to the magnitude of the fission product release to the environment and the potential for preventing hot leg creep rupture failure. Section 5.3.1.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.3.1.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 7 The timing of key events for unmitigated STSBO TI-SGTR.

Event Description	100% TI-SGTR Time (hh:mm)	200% TI-SGTR Time (hh:mm)
Station blackout – loss of all onsite and offsite AC and DC power MSIVs close Reactor trip RCP seal leak at 21 gpm/pump TD-AFW fails	00:00	00:00
First SG SRV opening	00:03	00:03
SG dryout	01:14	01:14
Pressurizer SRV opens	01:27	01:27
PRT failure	01:47	01:47
Start of fuel heatup	02:19	02:19
RCP seal failures	02:46	02:46
First fission product gap releases	02:57	02:57
Stuck open SG PORV	03:00	03:00
SGTR	03:33	03:33
Creep rupture failure of the Loop C hot leg nozzle	03:47	03:49
Accumulator discharges	03:47	03:49
Accumulator empty	03:47	03:49
Vessel lower head failure by creep rupture	07:30	06:51
Debris discharge to reactor cavity	07:30	06:51
Cavity dryout	07:54	07:21
Containment at design pressure (45 psig)	12:34	13:36
Start of increased leakage of containment ($P/P_{design} = 2.18$)	27.54	30:14
Containment pressure stops decreasing	40:18	40:20

5.3.1.1 Thermal-hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 38 for the 100% and 200% TI-SGTR cases. The initial response through 3 hr is identical to the unmitigated STSBO (see Section 5.2.1). At 3 hr, a safety valve on steam generator C (SG-C) fails open.[24] SG-C subsequently depressurizes to near atmospheric conditions and creates a large differential pressure across the steam generator tubes. During the core damage phase, hot gases circulate through the steam generator and increase the thermal stress across the tubes. The equivalent of a 100% or 200% tube area failure occurs at 3 hr and 33 min (see Figure 62), or about 12 min before the previously predicted creep rupture failure of the hot leg in the unmitigated STSBO (see Section 5.2.1). The combination of the TI-SGTR and the leakage through the failed RCP seals (2 hr and 45 min) causes a slow depressurization of the primary system. At 3 hr 45 min and 3 hr 47 min, respectively for the 100% and 200% TI-SGTR cases, the hot leg nozzle also fails due to a thermally-induced creep rupture. The failure of the hot leg nozzle leads to a rapid depressurization of the primary system and injection of the accumulator water. Following the depressurization of the RCS, the TI-SGTR flowrate drops to <0.2 kg/s through vessel failure at 7 hr 30 min and 6 hr 61 min, for the 100% and 200% TI-SGTR cases respectively.

There were some differences in the timing of events for the 100% versus the 200% TI-SGTR cases following the opening of the TI-SGTR. As shown in Figure 62, the flowrate through the 200% tube rupture case was approximately twice as large as the flow through the 100% tube rupture case. The net effect was (a) increased heat removal from the core (see Figure 63), (b) a higher flow of gas past the hot leg nozzle, (c) a reduction in the zirconium oxidation rate in the 200% case (see Figure 64), and (d) a slightly faster depressurization rate in the 200% case versus the 100% case. The first two effects increased the heat flow past hot leg nozzle while the second two effects reduced the core exit temperature and the mechanical stress across the hot leg nozzle. The net effect was a slightly later hot leg creep rupture failure in the 200% case relative to the 100% case. Hence, the 100% case represented a condition that enhanced the core rate oxidation and accelerated core damage whereas the 200% case increased core cooling and decreased oxidation.

The vessel water level is shown in Figure 65. In response to flow out of the pressurizer safety relief valve, the pump seal leakage, and leakage flow through the TI-SGTR, the vessel water level drops into the core and uncovers the fuel. The fuel heatup leads to a natural circulation phase that fails a steam generator tube(s) and eventually a hot leg nozzle creep rupture failure. The RCS pressure drops rapidly once the hot leg nozzle fails and the accumulators dump to refill the core with water. As discussed above, the timing to the RCS hot leg failure is not significantly different between the 100% and 200% cases.

Prior to the quench of the fuel by the accumulator water, the 100% case had more oxidation than the 200% case. Hence, the oxide layer thickness and potential for further oxidation following the core accumulator reflood was lower in 100% case than the 200% case. As shown in Figure 66, the zirconium cladding in the 200% case oxidizes at a higher rate than the 100% following the hot leg failure. Due to higher oxidation power in the 200% case in the post-reflood phase, the fuel degradation, the debris relocation to the lower head, and the failure of the vessel occurred faster in the 200% case. The fuel relocated to the lower plenum at 6.5 hr and 6 hr in the 100%

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[24] The valve failure was a specified boundary condition to develop a high differential pressure drop across the steam generator tubes and a direct bypass flow path to the environment. The valve failure occurred after the majority of the safety valve cycles but before the predicted time of the hot leg failure (i.e., to promote a steam generator tube failure).

and 200% cases, respectively. The vessel failure occurred at 7 hr 30 min and 6 hr 51 min in the 100% and 200% cases, respectively (see Figure 67).

Following vessel failure, the debris dropped into the reactor cavity under the reactor vessel. The hot debris immediately boiled away the water in the reactor cavity started to ablate the concrete. The ex-vessel core-concrete interactions (CCI) continued for the remainder of the calculation, which generated non-condensable gases. In addition, the hot gases exiting the reactor cavity and the radioactive heating from settled fission products steadily evaporated the water on the containment floor outside the reactor cavity from the time of vessel failure to 1.7 days (i.e., ~41 hr in both cases). The resultant non-condensable and steam production pressurized the containment (see Figure 68). However, due to the TI-SGTR, there was a leakage pathway from the containment through the vessel. The TI-SGTR slowed the pressurization of the containment relative to the unmitigated STSBO. Due to the larger TI-SGTR leakage area in the 200% case, the containment failure area in the 200% case was smaller than the 100% case (see relative leakage areas in Figure 69). In both cases, the containment continued to pressurize until the leakage flow balanced the steam and non-condensable gas generation. Hence, the containment failure leakage area increased in each case until the sum of the TI-SGTR and containment failure leakage areas balanced the gas generation. By 44 hr, all the water on the floor was evaporated and the steam generation stopped. The containment depressurized thereafter without any steam generation.

The predicted containment failure location was the around the equipment hatch, which is located on the side of the containment without a surrounding building (e.g., the auxiliary or safeguards buildings). Consequently, the fission products that leaked from the containment are released directly to the environment.

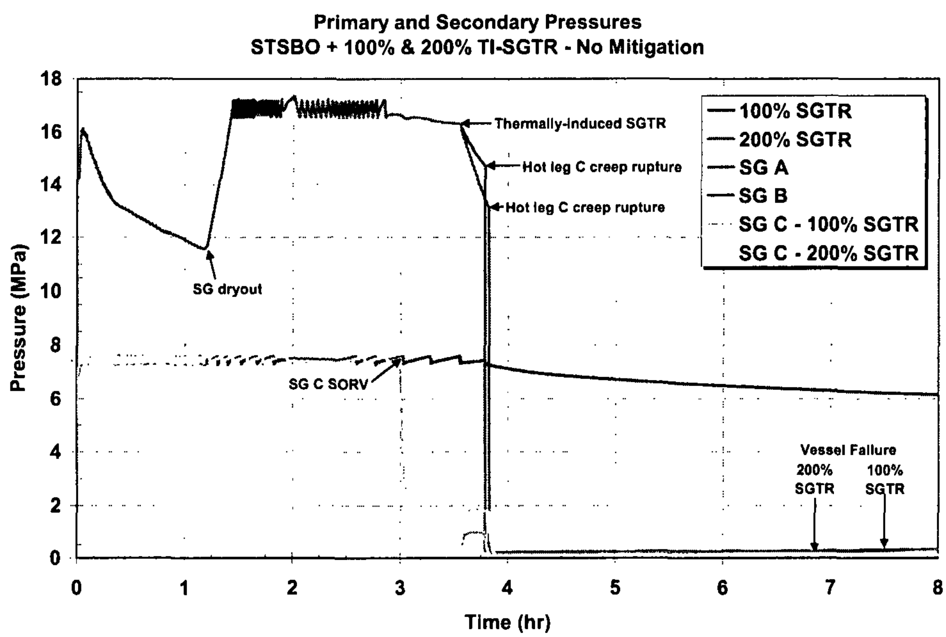

Figure 61 Unmitigated 100% and 200% TI-SGTR STSBO primary and secondary pressures histories.

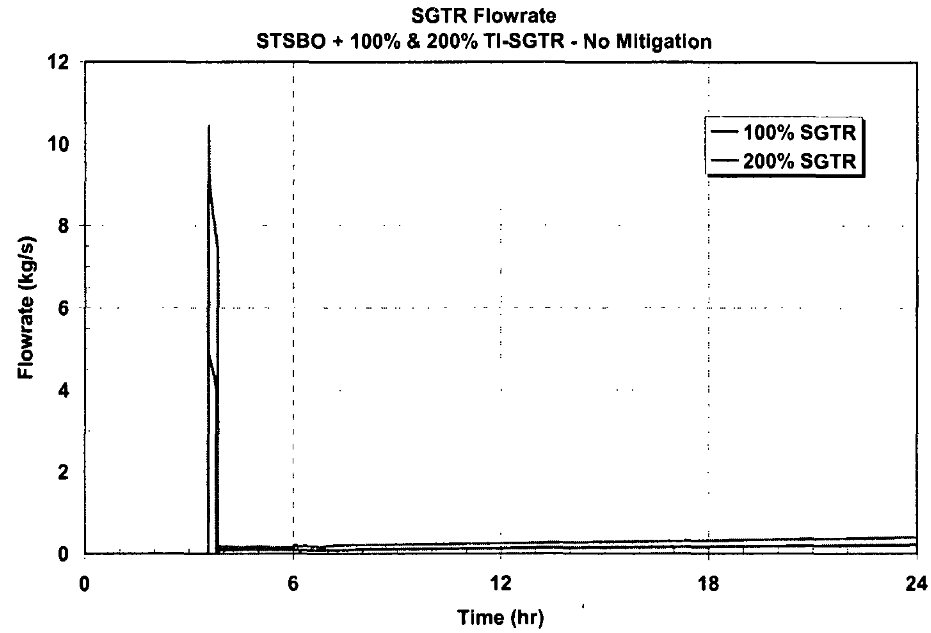

Figure 62 Unmitigated 100% and 200% TI-SGTR STSBO primary and TI-SGTR flowrate histories.

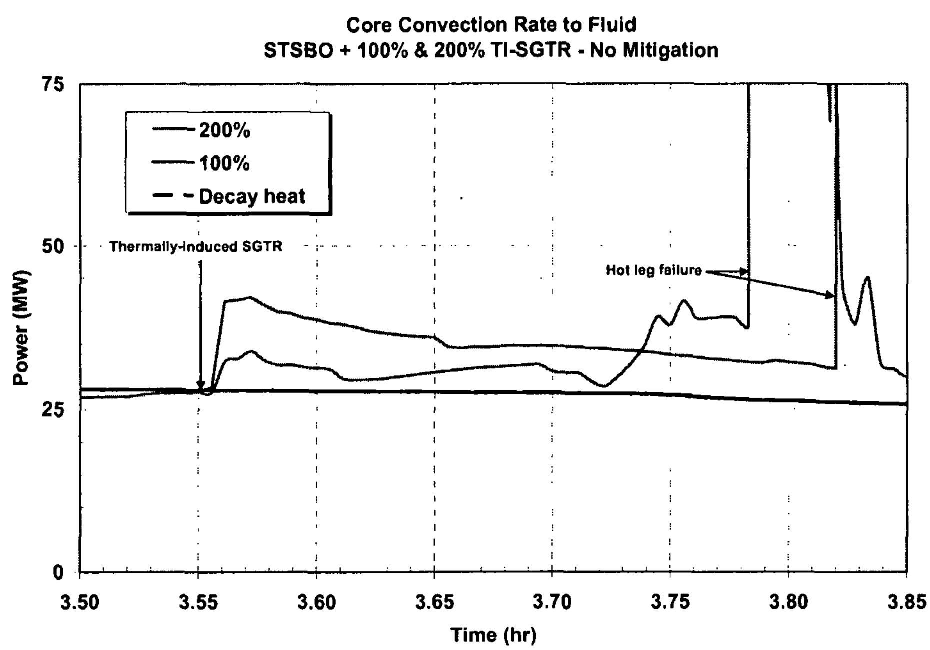

Figure 63 The unmitigated 100% and 200% TI-SGTR STSBO vessel convective heat removal rate from the fuel.

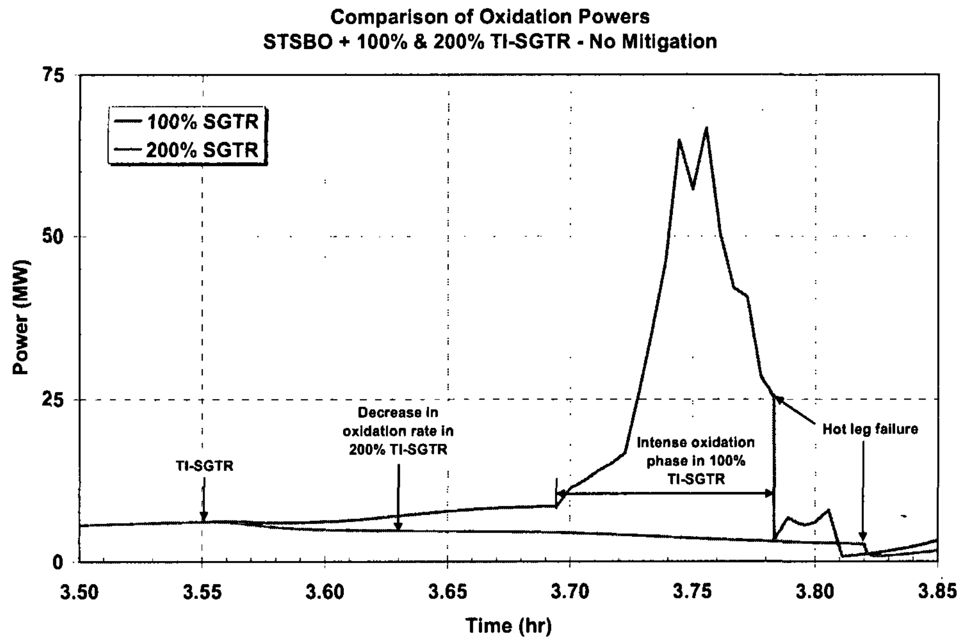

Figure 64 Unmitigated 100% and 200% TI-SGTR STSBO fuel oxidation power before the RCS hot leg failure.

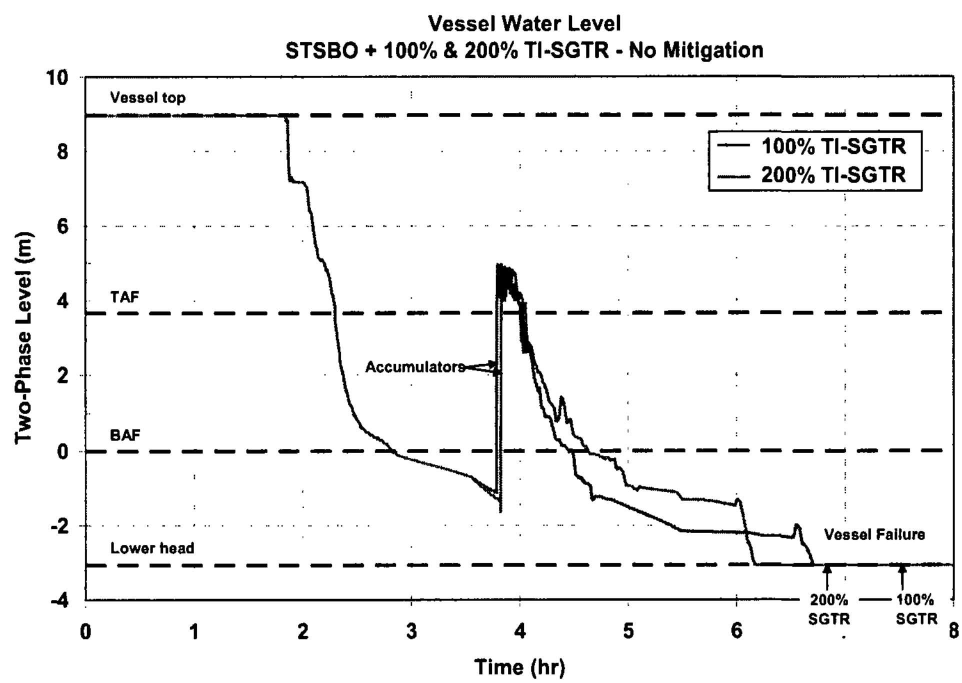

Figure 65 The unmitigated 100% and 200% TI-SGTR short-term station blackout vessel two-phase coolant level.

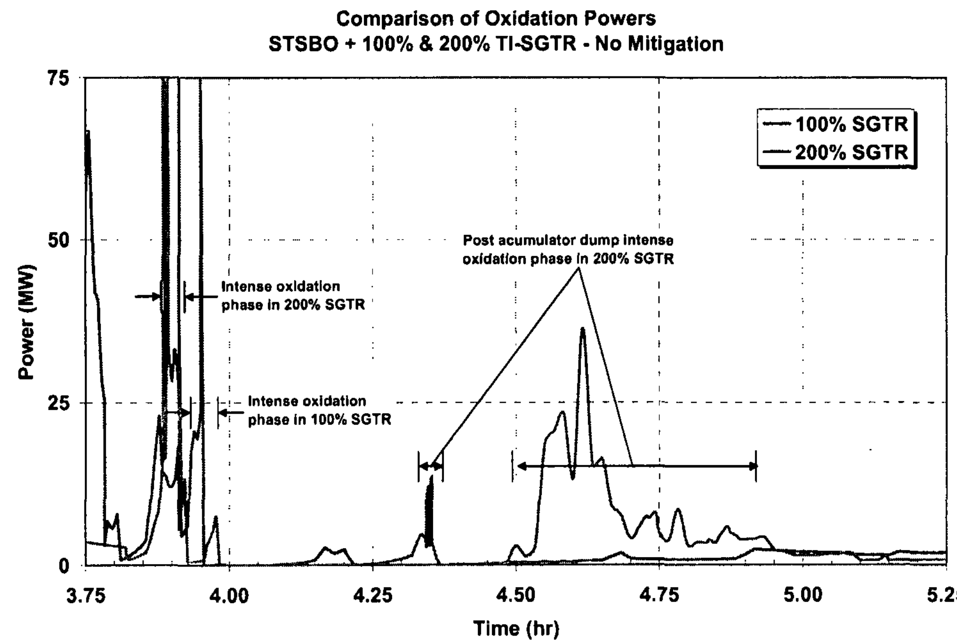

Figure 66 The unmitigated 100% and 200% TI-SGTR short-term station blackout fuel oxidation power after the RCS hot leg failure.

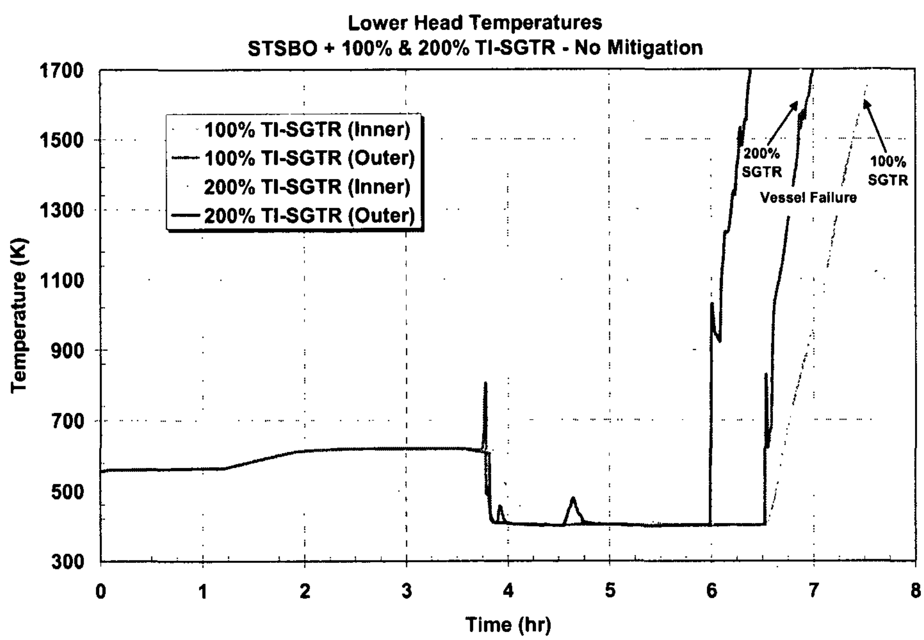

Figure 67 The unmitigated 100% and 200% TI-SGTR short-term station blackout lower head inner and outer temperature histories.

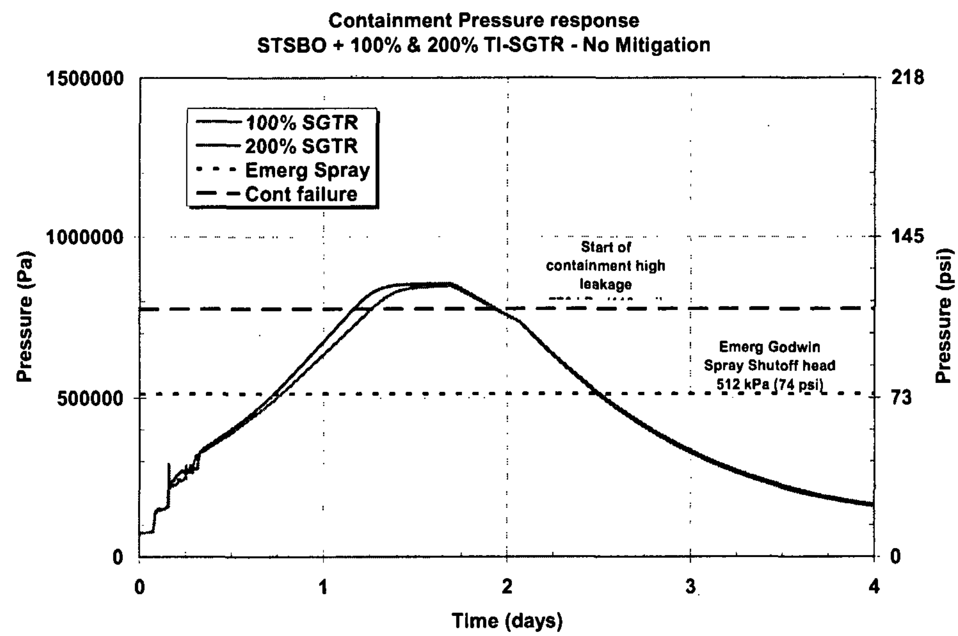

Figure 68 The unmitigated 100% and 200% TI-SGTR short-term station blackout containment pressure histories.

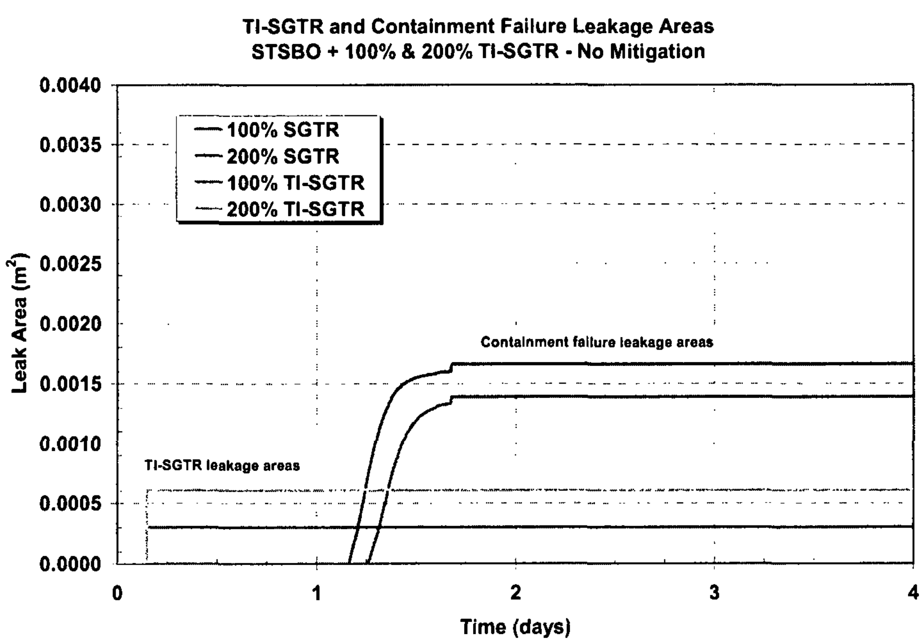

Figure 69 The unmitigated 100% and 200% TI-SGTR short-term station blackout containment and TI-SGTR leakage areas.

5.3.1.2 Radionuclide Release

The fission product releases from the fuel started following the first thermo-mechanical failures of the fuel cladding in the hottest rods at 2 hr 57 min, or about 38 min after the uncovery of the top of the fuel rods. At 3 hr, the secondary safety relief valve on SG-C sticks open and allows the steam generator to depressurize to near atmospheric conditions. Prior to the TI-SGTR, any fission products leaving the RCS would flow out the pressurizer safety relief valve to the pressurizer relief tank in the containment. However, the PRT over-pressurized and failed prior to the start of the fission product releases. Hence, any fission products vented to the containment were not scrubbed in the PRT.

The steam generators water inventory was completely boiled away by 1 hr 16 min. Consequently, there is no water on the secondary side of the steam generator after the TI-SGTR at 3 hr 33 min. Furthermore, since the steam generator relief valve stuck open at 3 hr, the released fission products can flow directly out the failed generator tube and through the stuck open relief valve to the environment. The flow of fission products through the tube rupture into the steam generator is very complicated and beyond the current modeling capabilities in MELCOR. Several decontamination mechanisms such as (a) impaction, vena contracta effects at the tube rupture, (b) deposition in bends, and (c) capture by the secondary side tube grid spacers are not addressed by the MELCOR aerosol deposition models. It was estimated from ARTIST tests that the steam generator aerosol decontamination in a full-scale steam generator would be between 4.7 and 9 [26]. The normal aerosol capture and settling models were disabled in MELCOR and the secondary side decontamination factor was prescribed to be 7 (i.e., approximately the average of 4.7 and 9).

Figure 70 and Figure 71 show the fission product distributions of the iodine radionuclides for the 100% and 200% TI-SGTR cases, respectively. The basic trends of the two cases were similar. The resultant distribution of iodine was partitioned between the RCS (i.e., including the vessel and the primary side of the steam generator tubes), the secondary side of the steam generators, the containment, and the environment. During the high release phase of the accident, the iodine is simultaneously released to the containment via the pressurizer safety relief valve, the secondary side of the steam generator and the environment via the TI-SGTR, or retained in the RCS. At the time of the TI-SGTR at 3 hr 33 min (0.15 days), only 8.4% of the iodine had been released from the fuel. About 1% was discharged to the containment via the pressurizer safety relief valve with the 7.4% retained in the RCS.

Between the timing of the 100% TI-SGTR and vessel failure, 98% of the iodine was released with 80% in the containment, 15% retained in the RCS, 3.1% in the SG secondary, and 0.5% in the environment. The overall steam generator and steam line decontamination factor was ~7 (i.e., the specified value). Eighty percent of the iodine transported to the containment versus only 3.6% in the steam generator secondary or the environment. The numbers were similar for the 200% TI-SGTR case with 97% released, 80% in the containment, 10% in the RCS, 5.3% in the steam generator secondary, and 0.8% in the environment. Due to the larger leak rate through the TI-SGTR, the 200% case had about twice the environmental release by vessel failure. The trends are similar for cesium, which are shown in Figure 72 and Figure 73.

The flow rate through the TI-SGTR decreased rapidly following hot leg failure at ~3.8 hr (see Figure 62), which slowed the release of the fission products to the faulted steam generator. Subsequently, the fission products moved from the reactor coolant system via the TI-SGTR rupture and the failed hot leg piping via natural circulation processes until the vessel lower head failure. As shown in Figure 70 through Figure 73, the releases to the containment or retention in the RCS increased most rapidly following hot leg failure. Prior to vessel failure, the fission product releases to the environment through the failed SGTR tube was roughly proportional to the size of the TI-SGTR leakage hole for the two cases (see Figure 74 and Figure 75).

After the lower head vessel failure, the releases to the environment for the 100% TI-SGTR were faster than the 200% case. By 4 days, the iodine releases to the environment were almost identical between the two cases (see Figure 74) and the cesium releases were much closer than at vessel failure (Figure 75). As shown in Figure 69, the TI-SGTR leakage areas were smaller than the containment leakage areas. However, the 100% TI-SGTR case needed a larger containment failure area to remove energy than the 200% case. Consequently, there was more leakage from the containment in the 100% TI-SGTR case than the 200% TI-SGTR case. Most of the releases through the TI-SGTR rupture were retained in the secondary side of the steam generator (i.e., a DF~7). In contrast, the fission products released through the containment failure went directly to the environment without any local retention. Since the 100% TI-SGTR case had more flow out the containment failure, the releases to the environment after the containment failure in the 100% case were higher than the 200% case. This non-intuitive trend eventually led to similar environmental releases for the two cases, which is evident in Figure 74 and Figure 75.

Finally, Figure 76 and Figure 77 summarize the releases of the radionuclides to the environment for the 100% and 200% cases, respectively. At 4 days, 95% of the noble gases, 4.2% of the molybdenum, 1.5% of the iodine, 0.7-0.8% of the cesium, 2.7% (100%) and 1.5% (200%) of the tellurium, and 0.2% of the barium had been released to the environment.

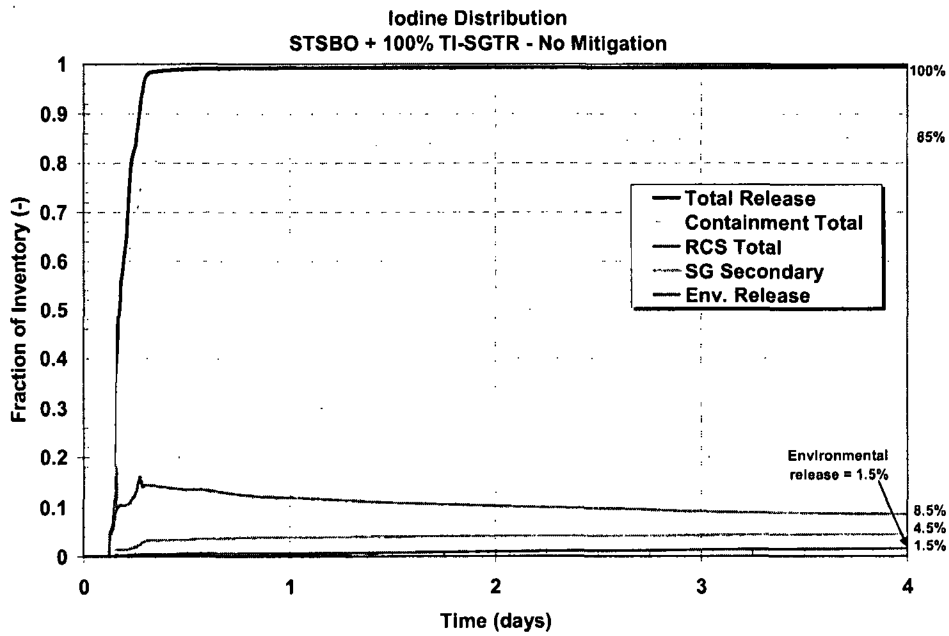

Figure 70 Unmitigated 100% TI-SGTR STSBO iodine fission product distribution history.

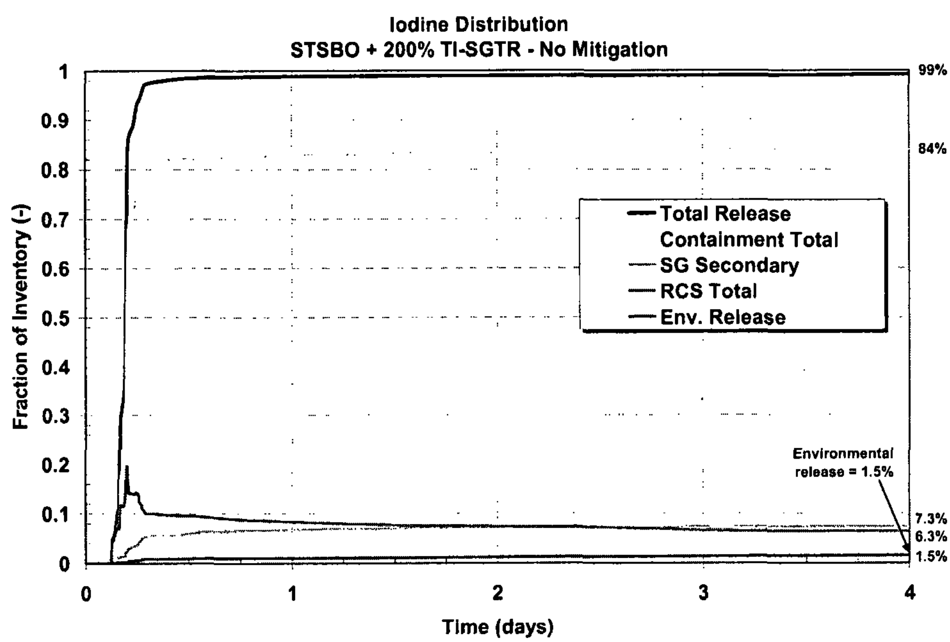

Figure 71 The unmitigated 200% TI-SGTR short-term station blackout iodine fission product distribution history.

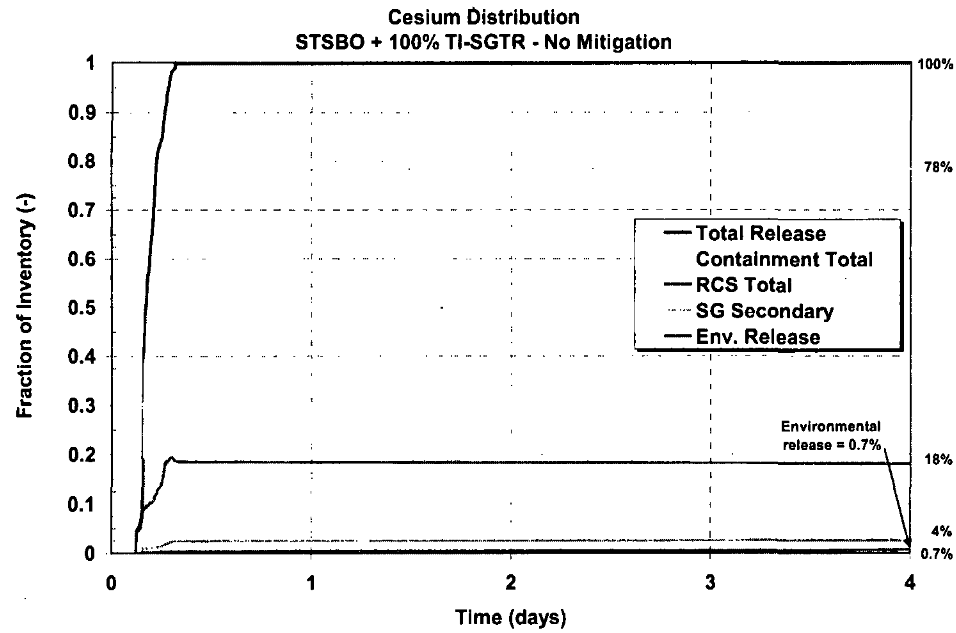

Figure 72 The unmitigated 100% TI-SGTR short-term station blackout cesium fission product distribution history.

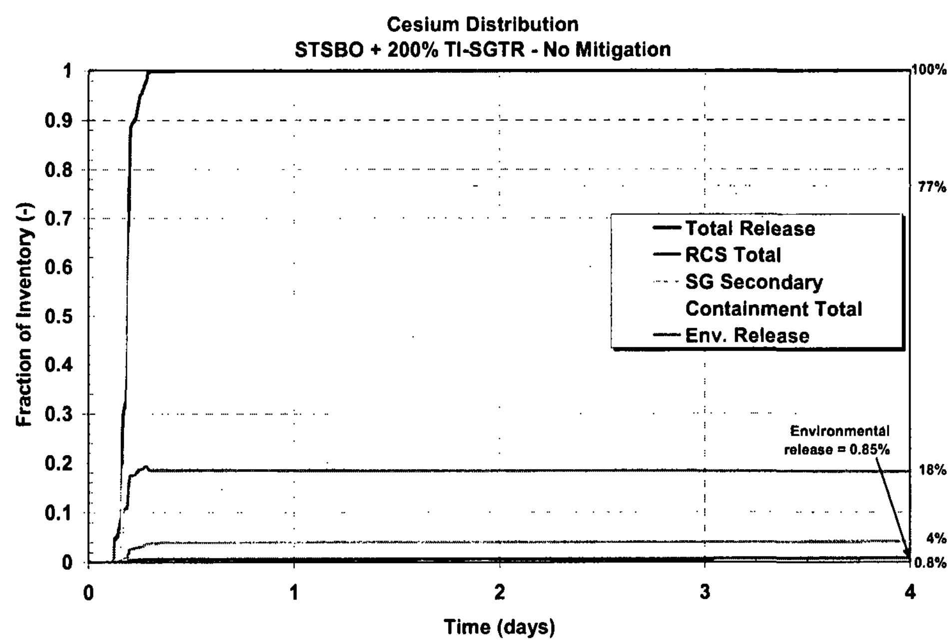

Figure 73 The unmitigated 200% TI-SGTR short-term station blackout cesium fission product distribution history.

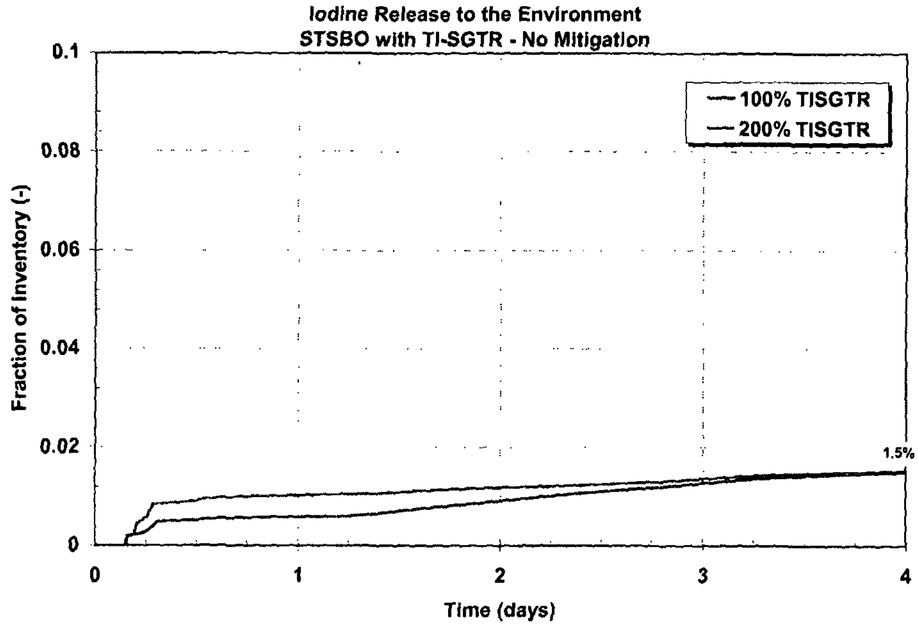

Figure 74 The unmitigated 100% and 200% TI-SGTR short-term station blackout iodine fission product distribution history.

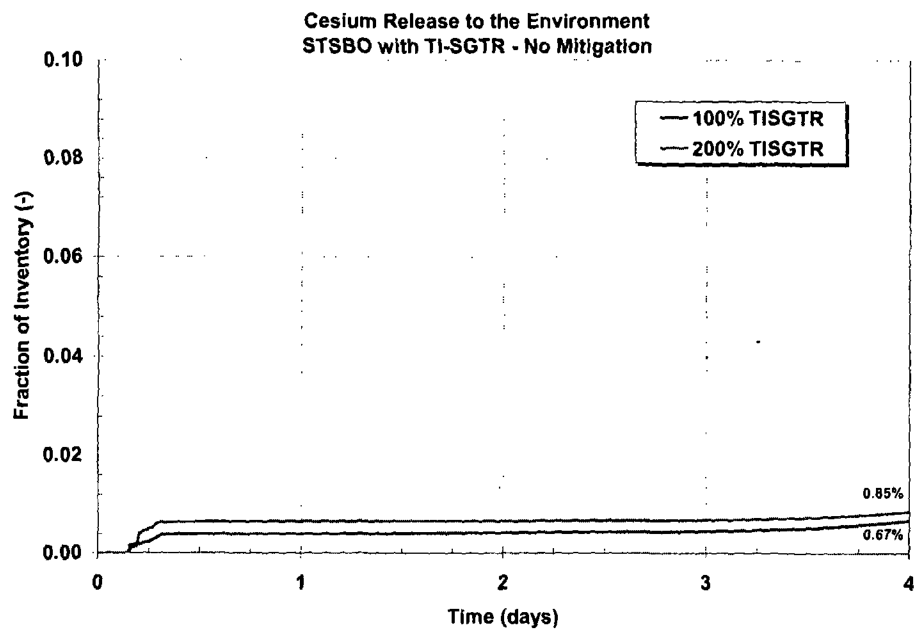

Figure 75 The unmitigated 100% and 200% TI-SGTR short-term station blackout cesium fission product distribution history.

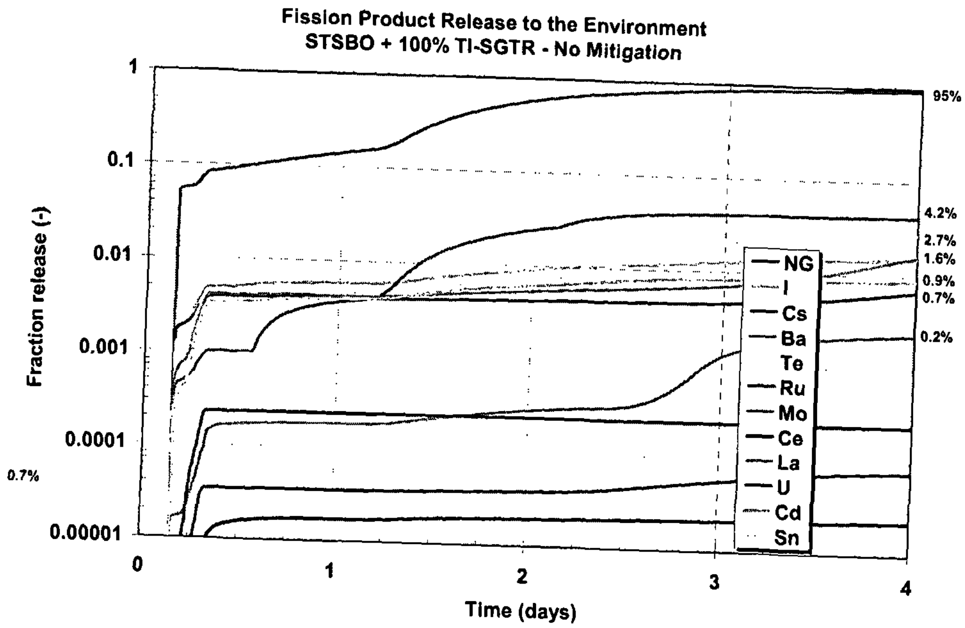

Figure 76 The unmitigated 100% TI-SGTR short-term station blackout environmental release history of all fission products.

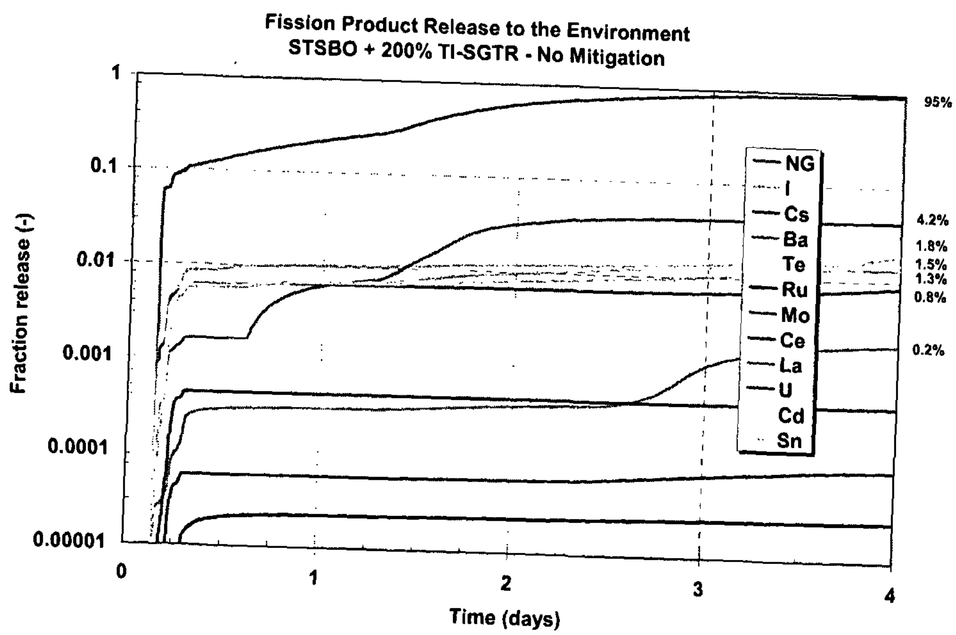

Figure 77 The unmitigated 200% TI-SGTR short-term station blackout environmental release history of all fission products.

5.3.2 Mitigated Short-Term Station Blackout with Thermally-Induced Tube Rupture

Table 8 summarizes the timings of the key events in the mitigated STSBO with a thermally-induced steam generator tube rupture. One (i.e., equivalent of 100% flow area) steam generator tube failed prior to any other RCS creep rupture failures along with a stuck open secondary safety relief valve. Consequently, there is a containment bypass pathway for fission products once the steam generator tube fails. As described in Section 3.2, the accident scenario initiates with a complete loss of all onsite and offsite power. The reactor successfully trips and the containment isolates but all powered safety systems are unavailable. The mitigated STSBO credits the successful connection of the portable, low-pressure, diesel-driven (Godwin) pump to the containment spray system at 8 hr. The Godwin pump is a high-flow, low-head pump with a design capacity of 2000 gpm at 120 psi. A reliable source of water is maintained while 1,000,000 gallons is injected into the containment through the containment sprays. The sequence of events is identical to the unmitigated STSBO with a thermally-induced steam generator tube rupture until 8 hr. In particular, the core has degraded and failed the vessel lower head prior to the spray actuation (see Table 8). The emergency containment sprays are effective at reducing the containment pressure and knocking down airborne fission products while they are operating. However, the containment subsequently pressurizes after the sprays are terminated to the failure pressure. While not investigated, intermittent operation of the sprays and deeper flooding could have further delayed failure of the containment. Section 5.3.2.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.3.2.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 8 The timing of key events for mitigated short-term station blackout with thermally-induced tube rupture.

Event Description	Time (hh:mm)
Station blackout – loss of all onsite and offsite AC and DC power MSIVs close Reactor trip RCP seal leak at 21 gpm/pump TD-AFW fails	00:00
First SG SRV opening	00:03
SG dryout	01:14
Pressurizer SRV opens	01:27
PRT failure	01:47
Start of fuel heatup	02:19
RCP seal failures	02:46
First fission product gap releases	02:57
Stuck open SG PORV	03:00

Event Description	Time (hh:mm)
SGTR	03:33
Creep rupture failure of the Loop C hot leg nozzle	03:47
Accumulator discharges	03:47
Accumulator empty	03:47
Vessel lower head failure by creep rupture	07:30
Debris discharge to reactor cavity	07:30
Cavity dryout (temporary)	07:54
Start of containment sprays	8:00
End of containment sprays (1,000,000 gal)	15:02
Containment at design pressure (45 psig)	44:10
Start of increased leakage of containment (P/P_{design} = 2.18)	74:48

5.3.2.1 Thermal-Hydraulic Response

The progression of events in the mitigated STSBO is identical to the unmitigated STSBO as described in Section 5.3.1 through the first 8 hr, which includes core degradation and vessel failure (e.g., compare the system pressure from Figure 78 and the 100% case in Figure 61 or the vessel level from Figure 79 and the 100% case in Figure 65). The portable emergency pump was connected to the containment spray system at 8 hours and begins injection. By 15 hours, 1,000,000 gallons were sprayed into the containment and the emergency injection was terminated. At the time of the analysis, there were no procedures for spray operation or termination, so the 1,000,000 gallons amount was somewhat arbitrarily selected.

After the containment sprays terminated at 15 hours, the containment water was flooded to ~0.1 m below the bottom of the vessel (see Figure 79).[25] The water levels in the reactor cavity and the containment basement were approximately equal due to the hydraulic connection through the 12" hole in the reactor cavity wall at 2'-7" above the bottom of the floor. The reactor cavity also connects to the containment basement via a penetration to ring duct bus at 24'-3" above the floor and the holes in the cavity wall for the RCS piping (nearly 40' above the bottom of the floor). Similar to response in mitigated short-term station blackout (Section 5.2.2), the water level was too low to allow natural circulation from the containment basement into the reactor cavity and out the gaps at the RCS piping penetrations. Since the reactor cavity contains the fuel debris from the failed reactor vessel, the water heated to boiling once the sprays terminated. As stated in Section 5.3.2, intermittent spray operation and/or flooding above the RCS piping

[25] At the time of the calculation, the exact flooding characteristics of the Surry containment was updated. Approximately 1,160,000 gal are needed to fill to the bottom of the vessel. The calculated containment water level was below the bottom of the vessel. In Section 5.2.2, an older, less accurate containment flooding model was used and the water level rose into the failed vessel. Both calculations assumed 1,000,000 gal of emergency spray injection. The containment water inventory in both calculations also includes pump seal leakage water, leakage from the pressurizer relief tank, and some RCS water following vessel failure.

penetrations would have substantially delayed containment failure. Although the containment sprays did not prevent containment failure, they delayed containment failure by over ~46 hr relative to the unmitigated case.

The containment sprays are effective in quickly reducing the containment pressure. As shown in Figure 70, the containment pressure would reach the shutoff head of the emergency portable pump by 17.5 hours. Based on the containment pressurization rate, it is estimated that there would be considerable additional time to connect the spray system. However, without additional spray flow above the initial 1,000,000 gal, the containment will pressurize above the emergency pump shutoff head by 2.2 days (52 hours) and to failure conditions by 3.1 days (74.8 hours). See the long-term containment pressure response in Figure 81.

The selection of the containment sprays as a mitigation technique for this scenario was particularly beneficial for several reasons, as previously discussed in Section 5.2.2.1. These benefits included aerosol knockdown in the containment, delaying containment failure by almost 2 days, and deep flooding and cooling the ex-vessel debris. The spray operation reduced the flow out the failed steam generator tube to the environment (i.e., a containment bypass leakage path prior to containment failure). The impact of these benefits on the source term is discussed in Section 5.3.2.2.

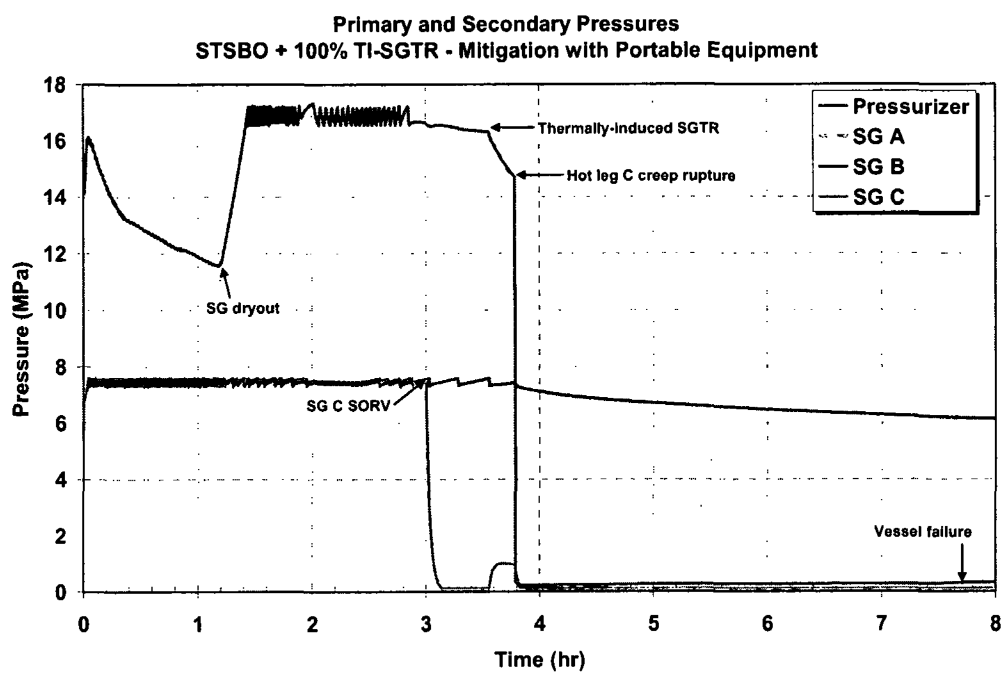

Figure 78 The mitigated STSBO primary and secondary pressure history.

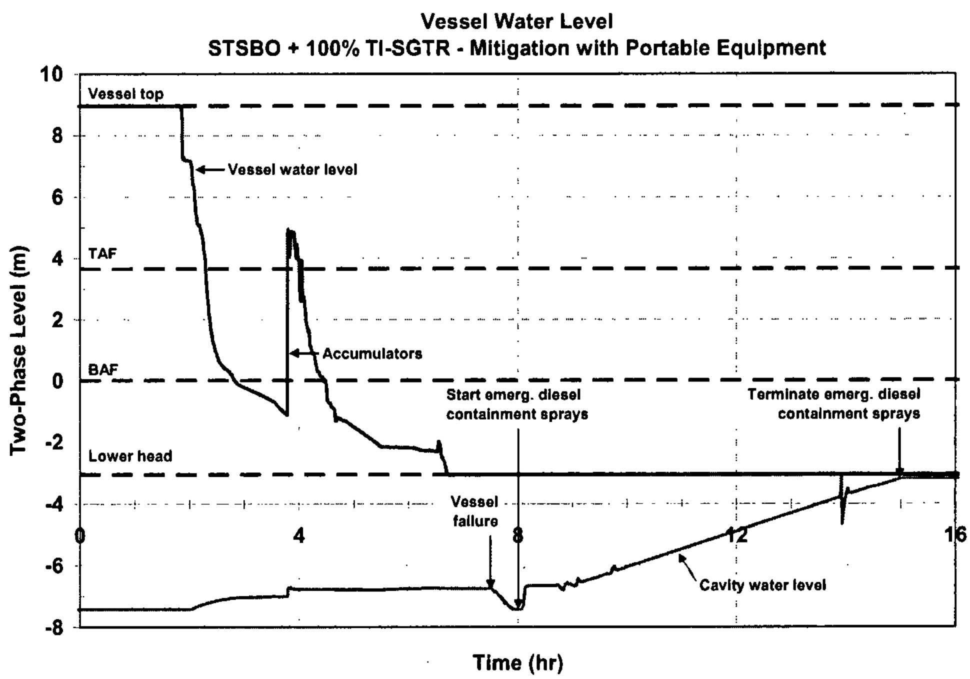

Figure 79 The mitigated short-term station blackout vessel two-phase coolant level.

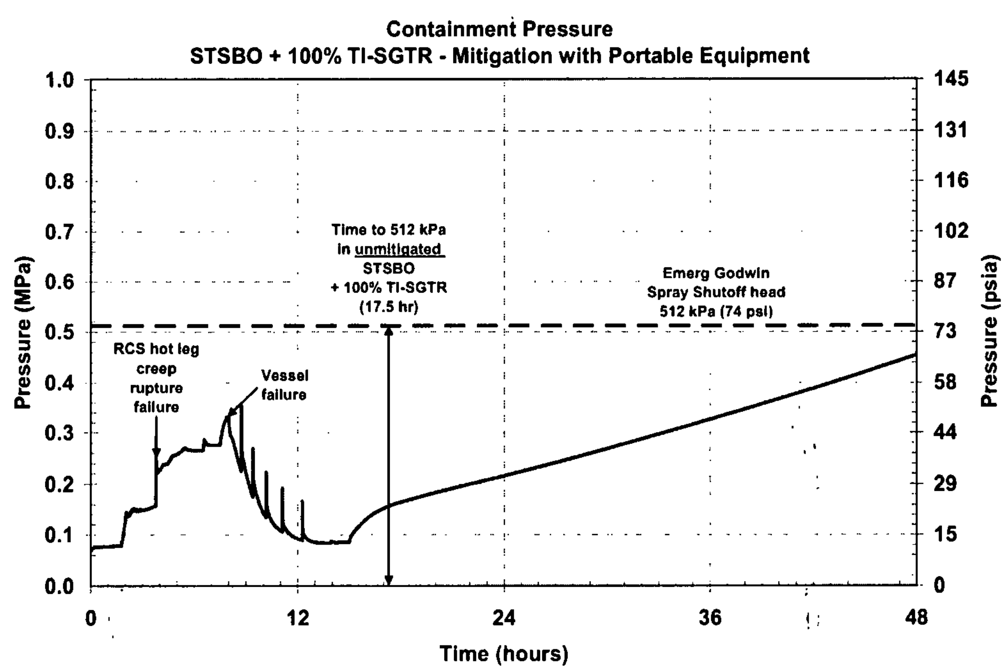

Figure 80 The mitigated short-term station blackout containment pressure history.

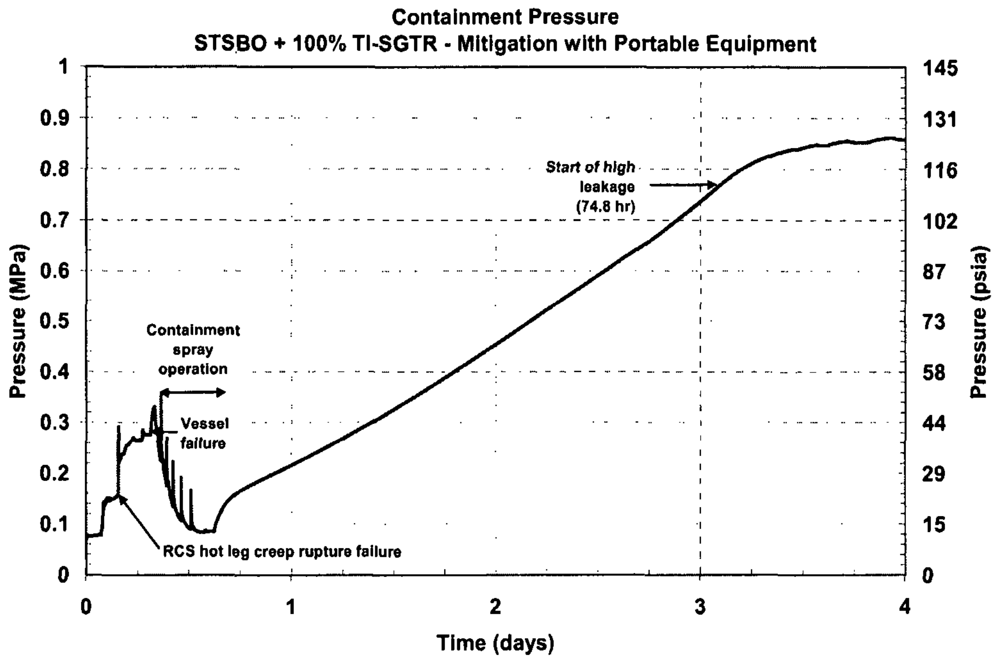

Figure 81 **The mitigated short-term station blackout containment pressure history.**

5.3.2.2 Radionuclide Release

The radionuclide response of the mitigated STSBO with a thermally-induced SGTR is identical to the unmitigated response described in Section 5.3.1.2 for the first 8 hr, or through vessel failure until the start of the containment sprays. Following the start of the emergency containment sprays at 8 hr (0.25 days), the airborne aerosols of iodine and cesium rapidly decease (see Figure 82 and Figure 83, respectively). By the time the sprays are terminated at 15 hr (0.63 days), almost all of the airborne aerosols have been captured in the pool on the containment floor. Since the containment failure was delayed until 74 hr 48 min (3.1 days), natural settling of the airborne mass in the containment was also significant, which is reflected in the small environmental release of iodine and cesium (i.e., 0.5% and 0.4%, respectively). However, natural settling was also effective in the unmitigated case, which does not occur until 27 hr 54 min. As will be discussed next, the spray water was important in preventing revaporization. This was the most significant difference between the mitigated and unmitigated cases.

Due to the deep flooding in the reactor cavity by the spray operation, the bottom of the failed vessel lower head is at the top of the water level in the reactor cavity.[26] Therefore, the natural

[26] Although the level is 0.1 m below the inside of reactor vessel, the level is covers the bottom of the outside surface of the lower head, which is 0.13 m thick. Hence, the water blocks the flow of air into the reactor vessel through the lower head failure hole.

hot circulation flow that promoted revaporization in the unmitigated STSBO is not present. Instead, the water pool in the reactor cavity cools the bottom of the vessel. Due to some boiling in the cavity, a relatively cool flow of steam passes through the vessel and out the failed hot leg nozzle location, which also removes heat and inhibits revaporization of deposited radionuclides in the upper vessel and hot leg. Consequently, the revaporization of the in-vessel deposited fission products (i.e., especially cesium-iodine) that was seen in the unmitigated STSBO with a thermally-induced SGTR is characteristic of revaporization) was negligible in the mitigated case through 4 days (see Figure 84 and Figure 85).

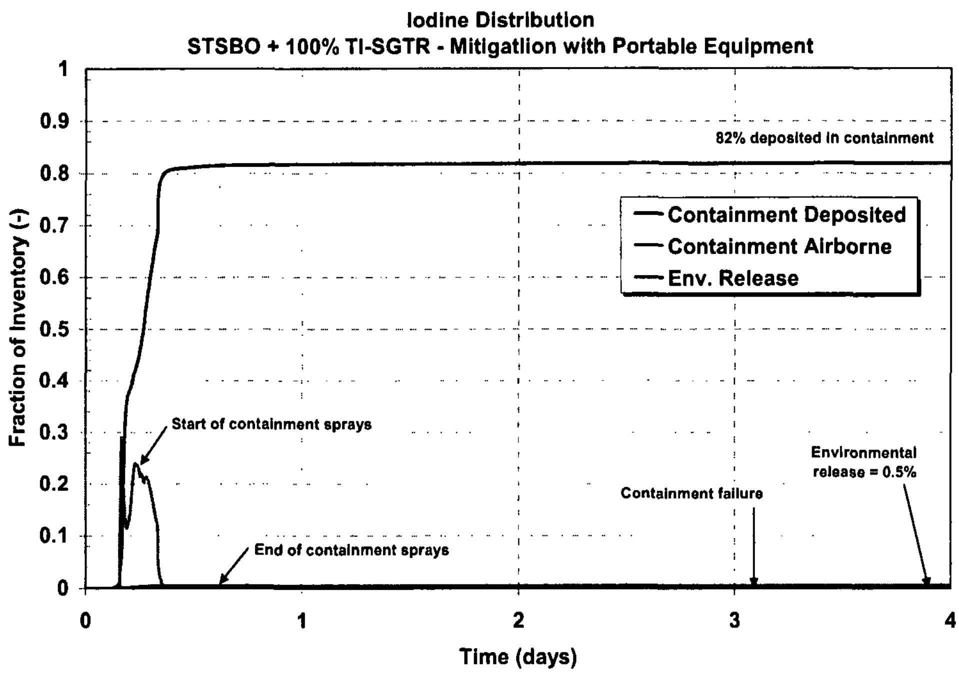

Figure 82 The iodine distribution in the containment for short-term station blackout with a 100% thermally-induced SGTR with spray mitigation.

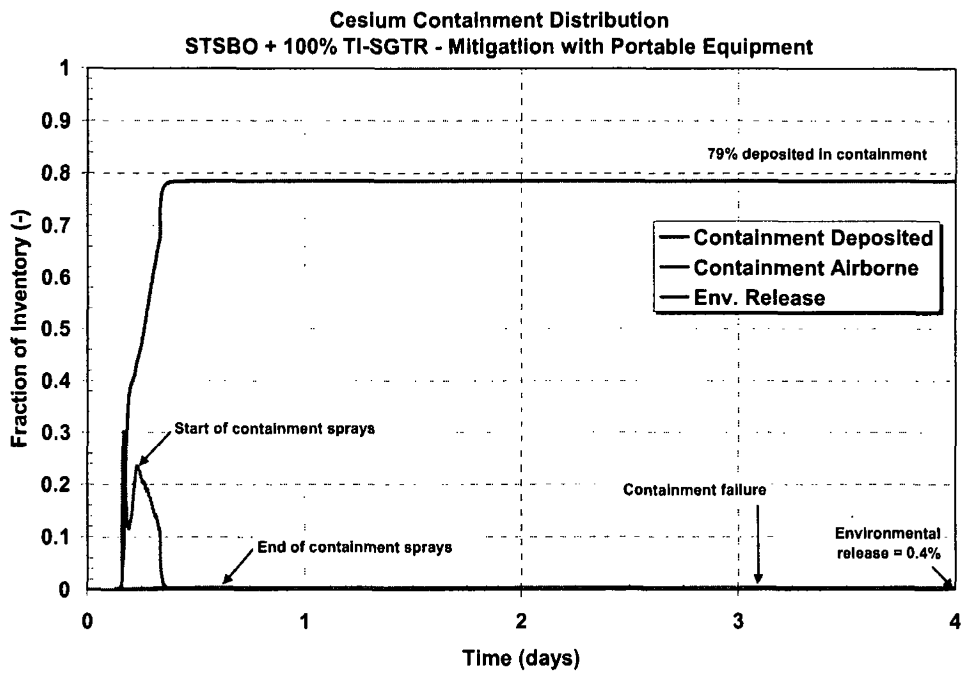

Figure 83 The cesium distribution in the containment for short-term station blackout with a 100% thermally-induced SGTR with spray mitigation.

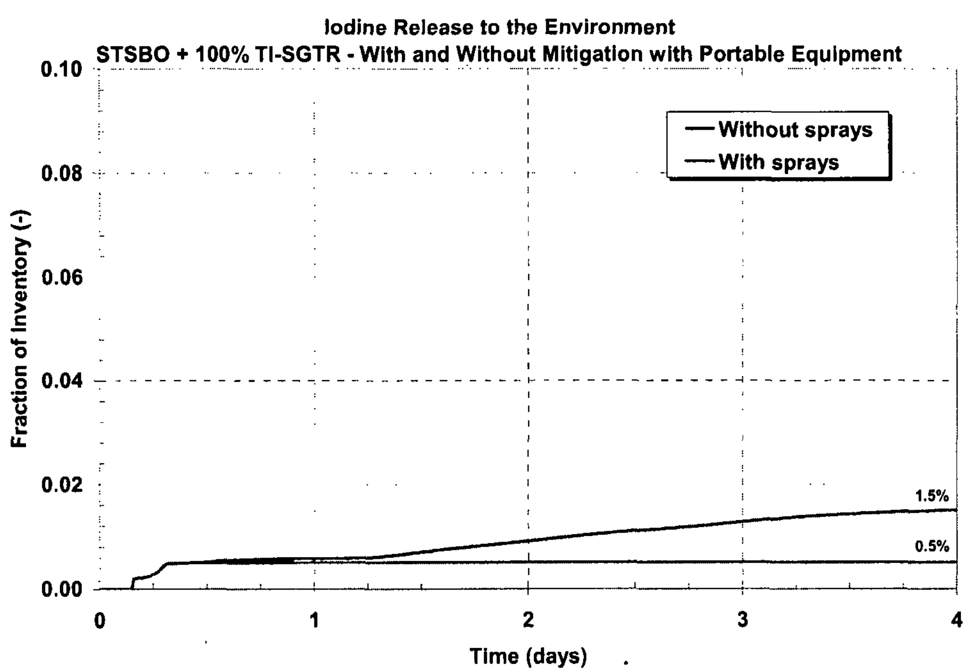

Figure 84 The short-term station blackout with a 100% thermally-induced SGTR with and without spray mitigation iodine environmental release.

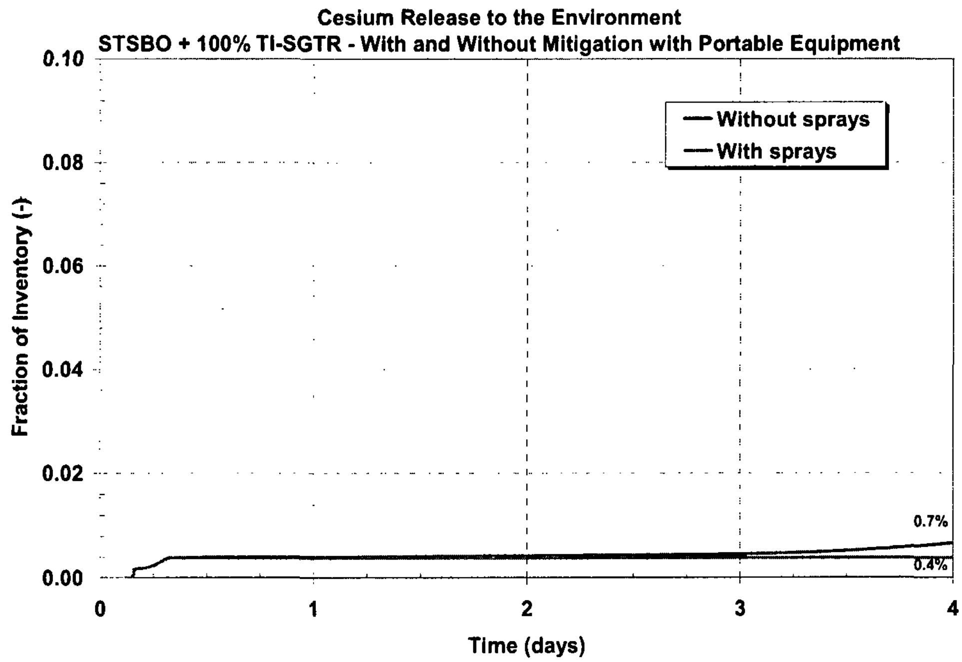

Figure 85 The short-term station blackout with a 100% thermally-induced SGTR with and without spray mitigation cesium environmental release.

5.3.3 Uncertainties in the Failure of the Thermally-Induced Steam Generator Tube versus the Hot Leg

During the peer review of the unmitigated short-term station blackout with a thermally-induced tube rupture, there were questions about the competing events of a thermally-induced steam generator tube versus the hot leg creep rupture failure. The probablility of a thermally-induced steam generator tube rupture has previously been assessed to be 0.25 [27].[27] Consequently, calculations were performed in Section 0 with thermally-induced steam generator tube rupture to supplement the calculations described in Section 5.2 without tube failures. More recent research has investigated the relative timing of the thermally-induced steam generator tube rupture relative to creep rupture failure of the hot leg with mechanistic simulations of natural circulation flow patterns[12][13][29][30]. The results of the research show comparable timings for hot leg creep rupture failure and thermally-induced steam generator tube failure, for a flawed tube at maximum thermal stress conditions, with the former slightly preceeding the later for most conditions.

[27]Recently, Liao and Guentay estimated the probability of TI-SGTR to be 0.025 [28]. They attributed the lower value to the following, "The major reason for a lower probability predicted in the current work is that the new generation steam generator tubing materials treated herein have a better operating performance since the number of flaws caused by inservice degradation has been significantly reduced."

MELCOR also predicts failure of a hot leg prior to any steam generator tubes (i.e., potential failures are monitored at both locations). Consequently, the calculations presented previously in Section 0 increased the mechanical stress across the tubes by prescribing a stuck-open safety relief valve and an increase in the thermal stress by inducing tube failure at a lower criterion than the default model. Subsequent to the failure of the steam generator tube, the hot leg failed and mitigated the magnitude of the potential release of radionuclides that bypass the containment.

To investigate the relative vulnerability of the hot leg to a thermally-induced steam generator tube rupture, a sensitivity calculation was performed with MELCOR where the failure of the hot leg was prevented. Figure 86 shows the creep rupture damage index of the hot leg. The steam generator tube failed at 3.55 hr. Hot leg failure failure was predicted 14 min later at 3.8 hr when the failure index reached a lifetime value of 1. Vessel failure was calculated to occur at 5.3 hr in the sensitivity calculation. Between 3.8 and 5.3 hr, the damage index increased from 1 to greater than four orders of magnitude larger. The creep index is highly sensitive to the thermal response of the hot leg nozzle as very hot gases continue to flow from the core (see hot leg temperature responses in Figure 87).

Figure 88 includes the iodine release to the environment for the failure and no failure case. As discussed in Section 5.3.1.2, there is a direct pathway for radionuclide releases to the environment through the failed steam generator tube prior to hot leg failure. However, the iodine release to the environment essentially stopped once the hot leg failed. Between 3.8 hours and 4 hours, the hot leg creep failure index in the no hot leg failure sensitivity case increased more than an order of magnitude (i.e., a factor of 18) above the best-estimate failure value. The iodine release to the environment increased by a factor of three during this time period to a 0.6% release. Consequently, the release of iodine to the environment is very sensitive to the timing of creep rupture in the hot leg.

In summary, it is not credible that the hot leg would not fail by creep crupture in the examined scenarios. The conditions that lead to the TI-SGTR are the same conditions that promote hot leg failure. As discussed in Section 5.3.1.1, the TI-SGTR increased heat removal from the core and the heat flow past the hot leg nozzle. The best-estimate creep rupture damage index is rapidly increasing near the time of the TI-SGTR. Within 10 minutes after the best-estimate failure time of the hot leg nozzle, the creep rupture damage index has increased by an order of magnitude due to the strong dependence of the nozzle strength to temperature. There is a factor of 3 increase in the iodine release to the environment while the creep rupture index increases to an order of magnitude larger. However, the release of iodine to the environment was only 0.6% at the order-of-magnitude higher damage value.

Three sensitivity calculations were also performed using the SCDAP/RELAP5 code and associated natural circulation severe accident model [12][13][29]. The best-estimate parameters in the SCDAP/RELAP5 calculation were based on the latest FLUENT CFD research [30]. Unlike the MELCOR calculation, which used a specified criterion to create the TI-SGTR (i.e., specified to occur ~10 min prior hot leg failure timing from the STSBO in Section 5.2.1), the SCDAP/RELAP5 simulation tied the TI-SGTR to stress enhancing vulnerabilities due to flaws developed during inservice operation. The three SCDAP/RELAP5 cases examined (1) a TI-SGTR in the hottest portion of the natural circulation plume and a stress multiplier of 2, (2) a

TI-SGTR in the hottest portion of the natural circulation plume and a stress multiplier of 3, and (3) multiple tube failures with a stress multiplier of 2. The results of the SCDAP/RELAP5 study (i.e., shown in Table 9) confirmed that (a) TI-SGTR will not preclude hot leg creep rupture failure and (b) hot leg creep rupture failure occurs within minutes of the TI-SGTR for a range of tube stress conditions.

Table 9 The timing of hot leg failure for SCDAP/RELAP5 simulations with thermally-induced tube rupture.

Case	Delay of Hot Leg Failure after TI-SGTR (min)
1. Steam generator tube stress multipler of 2	1.2
2. Steam generator tube stress multipler of 3	8.8
3. Multiple steam generator tubes w/stress multipler of 2	1.3

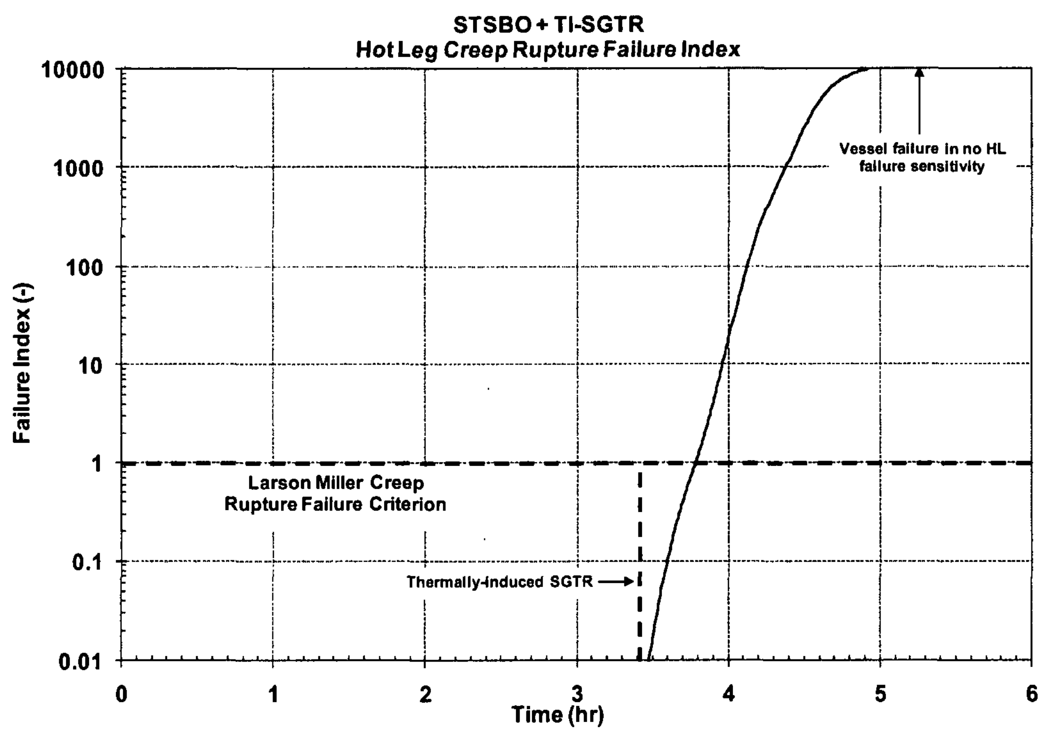

Figure 86 The hot leg creep rupture failure index in the short-term station blackout sensitivity case with a 100% thermally-induced SGTR and no hot leg failure.

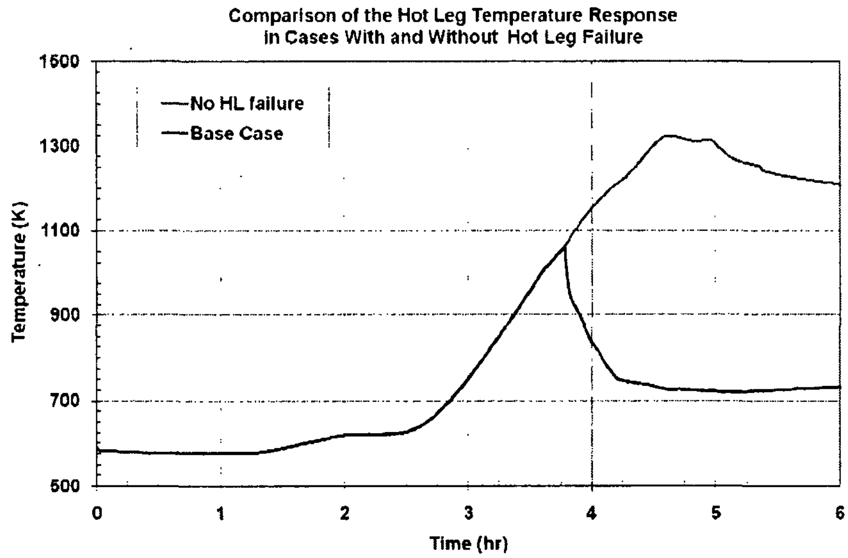

Figure 87 The hot leg temperature response in the thermally-induced steam generator tube rupture cases with and without hot leg failure.

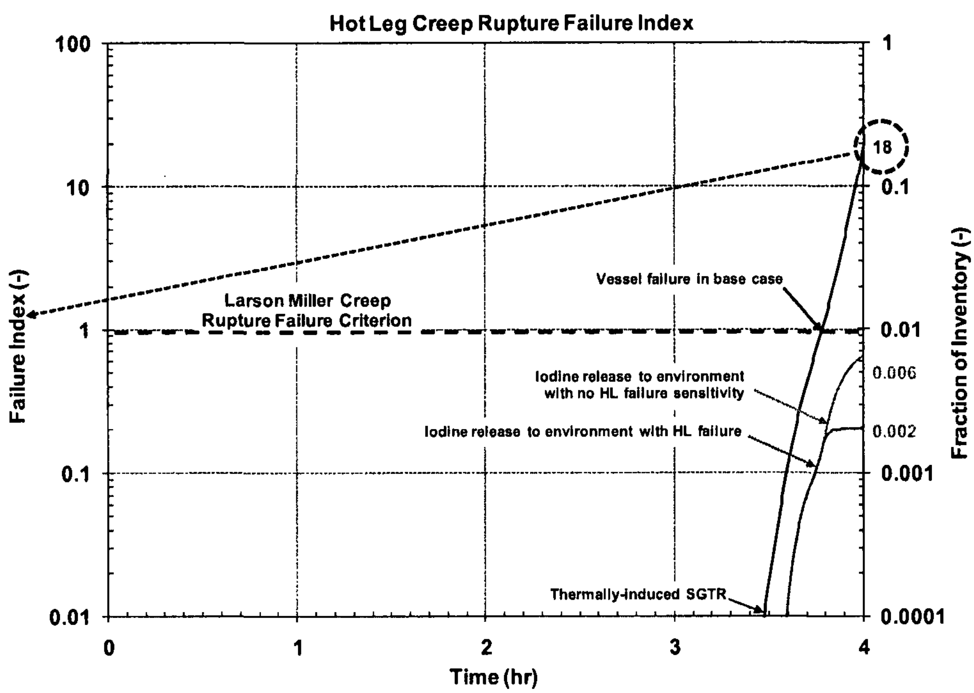

Figure 88 The hot leg creep rupture failure index and iodine release to environment for the thermally-induced steam generator tube rupture cases with and without hot leg failure.

5.4 Spontaneous SGTR

The spontaneous SGTR sequence is a double-ended-guillotine rupture of a single steam generator tube occurring while the reactor system is operating at normal conditions. For the unmitigated scenario in Section 5.4.2, the operator fails to isolate the faulted steam generator or cooldown the RCS using the two intact generators. Finally, the unmitigated scenario in Section 5.4.3 has the same failed operator actions as the previous unmitigated scenario. In addition, the relief valve on the faulted generator is assumed to fail open when water from the SGTR fills the steam generator and flows through the valve.

5.4.1 Spontaneous SGTR- Base Case

Table 10 summarizes the timing of the key events in the mitigated spontaneous steam generator tube rupture. As described in Section 3.3.1, the accident scenario initiates with a spontaneous failure of one steam generator tube. After about two minutes, the reactor successfully trips, the containment isolates, and all powered safety systems are available. The operator actions are successful to isolate the faulted steam generator and cooldown the system to permit operation of the residual heat removal (RHR) system. Section 5.4.1.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.4.1.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 10 Timing of key events for the Spontaneous SGTR with Expected Operator Action.

Event Description	Time (hh:mm)
Spontaneous SGTR	00:00
Reactor scram	00:03
Turbine stop valves close	00:03
Steam dump valves open and modulate	Not accomplished*
Steam dump valves close (RCS temperature < 547 °F)	Not accomplished*
HHSI initiated (3 pumps)	00:03
First AFW delivery	00:03
Operators take control of AFW	00:15
AFW delivery to faulted steam generator secured	00:15
1 of 3 HHSI pumps secured	00:15
Faulted steam generator flooded	00:20
TDAFW fails (turbine floods)	00:20
Faulted steam generator PORV 1st lifts	00:23
Faulted steam generator isolated	02:30

Event Description	Time (hh:mm)
HHSI secured	02:30
Leakage through faulted steam generator PORV stopped	02:30
100 °F/hr cool-down initiated	02:30
RHR entry pressure (400 – 450 psig) achieved	03:18 (450 psig)
RHR entry temperature (350°F) achieved	03:43

* The automatic operation of steam dump valves was not represented in the Surry MELCOR model. The thermal-hydraulic signature in the subject calculation suggests that the valves might be active for the first 6 min following scram but at no other time. Valve action would reduce RCS temperature by ~25 °F for the first few min and by a few °F in the next few min. The differences are thought to be inconsequential.

5.4.1.1 Thermal-hydraulic Response

Figure 89 through Figure 94 present the thermal hydraulic response for a spontaneous SGTR where reactor systems operate as designed and after a 2.5 hour delay, reactor operators respond as expected per training and procedure. The tube rupture quickly leads to a reactor scram, turbine stop valve closure, HHSI injection, and AFW injection. The flow of primary system coolant through the tube rupture into the secondary side of the faulted steam generator results in a sustained leak to the environment through the relief valve exhaust pipe beginning at 23 min when the steam generator relief valve first lifts.

Once water level is restored in the steam generators (i.e., shortly after the operators taking control of the TD-AFW), a feed and bleed decay heat energy removal process is established using the HHSI and the leakage through the SGTR. After 15 minutes, one HHSI pump is stopped, which leaves two HHSI pumps running. The two HHSI pumps keep the RCS and faulted steam generator full of water, the primary system pressure at ~15.5 MPa, and the faulted steam generator relief valve cycling to relieve steam and water. The feed (HHSI) and bleed (SGTR) process removes the core decay heat until the operator (a) stops the remaining two HHSI pumps, (b) starts a RCS cooldown using the intact generators, and (c) isolates the faulted steam generator.

The operator terminates HHSI at 2.5 hrs, which allows the primary system and faulted steam generator pressures to decrease immediately. The secondary side PORV on the faulted generator closes, which ends the SGTR leakage to the environment. The decay heat removal by the feed and bleed process is replaced by the 100°F/hr cool-down. The 100°F/hr cool-down brings the temperature of the RCS down to RHR entry temperature (i.e., <350°F or <450 K) within 1.5 hours of starting the cool down. The RHR entry pressure (i.e., <450 psia or <3.1 MPa) is achieved almost immediately after terminating the HHSI, which was pressurizing the RCS to >15 MPa (2180 psi).[28]

[28] The calculated timing to achieve the RHR pressure entry may be accelerated given that no active pressure control was represented in the MELCOR calculation, i.e., no pressurizer heater operation was modeled. However, the 100°F/hr cool-down was accomplished realistically in that the intact steam generators were vented in a controlled fashion while AFW was delivered as needed to maintain level.

The results of the SGTR with expected operator actions show that RHR entry conditions would be achieved without challenging the RWST inventory and without substantially draining the ECST. No uncovering or overheating of the reactor core would occur and no damage to the core would result.

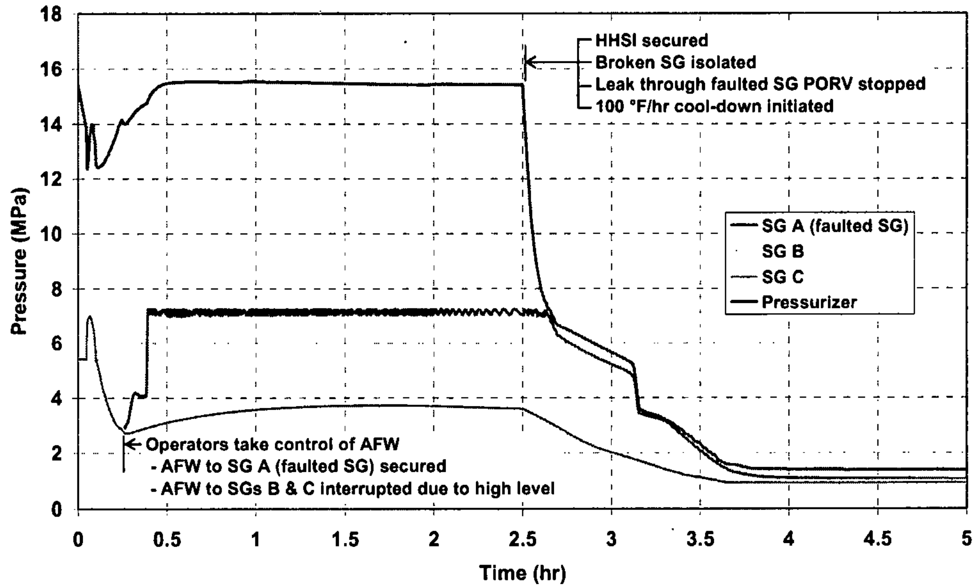

Figure 89 The SGTR with Operator Action – System Pressures.

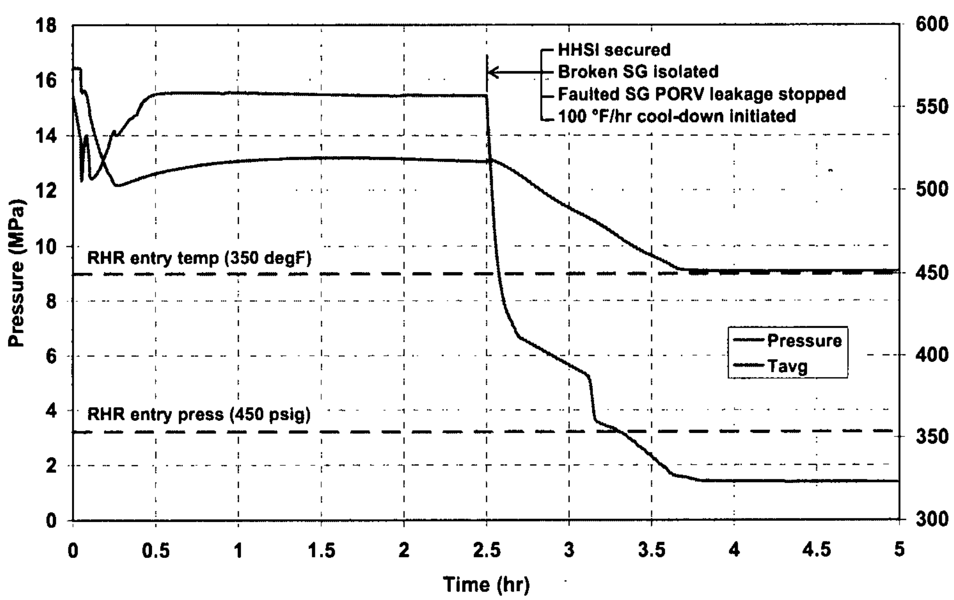

Figure 90 The SGTR with Operator Action – RCS Conditions Relative to RHR Entry Conditions.

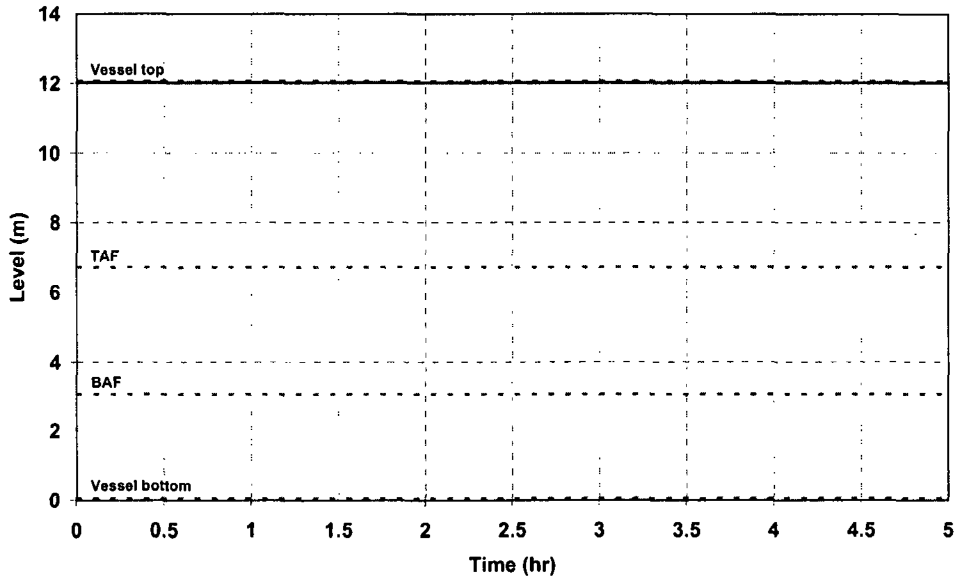

Figure 91 The SGTR with Operator Action – RPV Level.

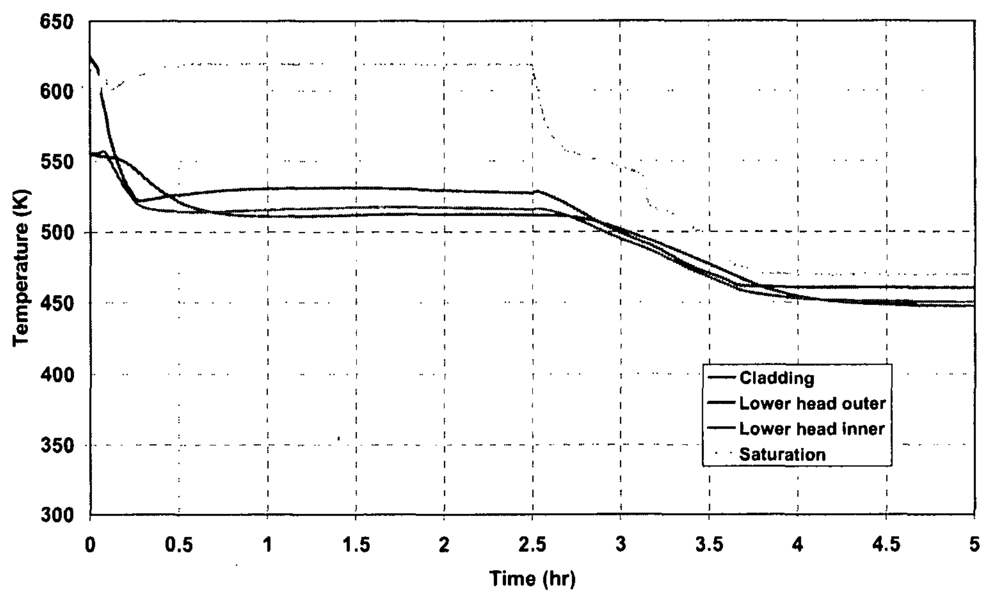

Figure 92 The SGTR with Operator Action – Maximum Cladding and Lower Head Temperatures.

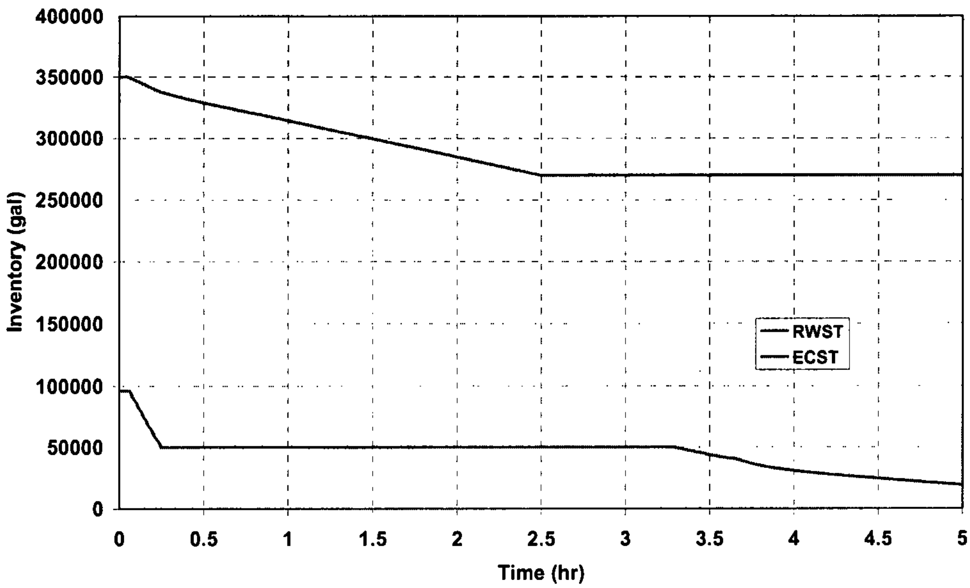

Figure 93 The SGTR with Operator Action – RWST and ECST Inventories.

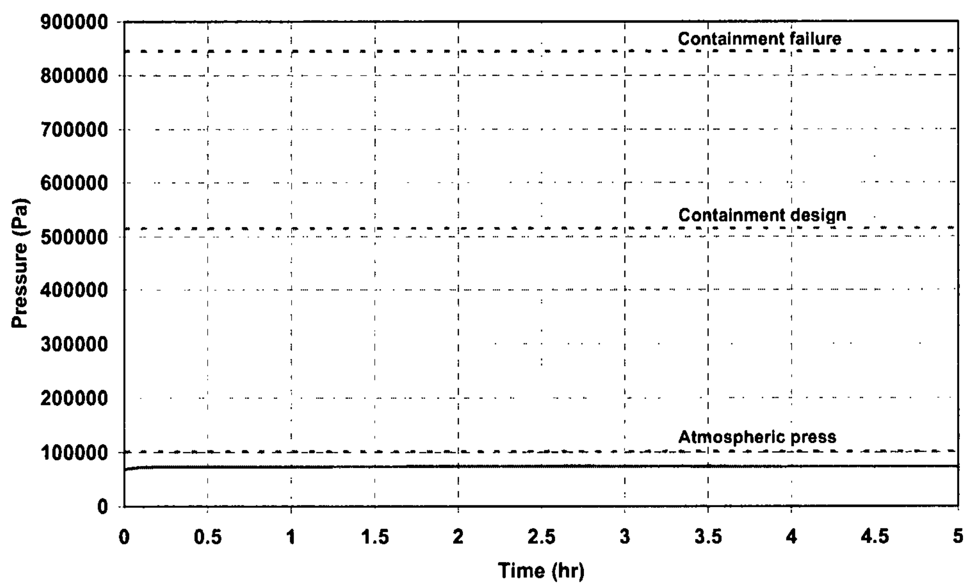

Figure 94 The SGTR with Operator Action – Containment Pressure.

5.4.1.2 Radionuclide Release

No fission product releases from the reactor core occurred in the spontaneous SGTR with expected operator actions.

5.4.2 Unmitigated - Spontaneous SGTR with Failed Operator Action

Table 11 summarizes the timing of the key events in the spontaneous steam generator tube rupture with some failed operator actions. As described in Section 3.3, the accident scenario initiates with a spontaneous failure of one steam generator tube. After about two minutes, the reactor successfully trips, the containment isolates, and all powered safety systems are available. The timings of the key events are discussed further in Sections 5.4.2.1 and 5.4.2.2. The operator actions are not successful to isolate the faulted steam generator or cooldown the RCS using the two intact generators. Eventually, all the water in the refueling water storage tank is exhausted and core damage starts. Due to the long amount of time until the RWST empties (11 hr) and the emergency condensate storage tank for the auxiliary feed water empties, the start of the core uncovery is delayed until 43 hr. It was judged unlikely that the operators would not correct missed actions (i.e., failure to isolate the faulted SG, failure to cool down and depressurize, and fail to refill RWST or connect to unaffected unit's RWST) to perform a safe shutdown. Section 5.4.2.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.1.4.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 11 Timing of key events for the Spontaneous SGTR with Failed Operator Action

Event Description	Time (hh:mm)
Spontaneous SGTR	00:00
Reactor scram	00:03
Turbine stop valves close	00:03
Steam dump valves open and modulate	Not accomplished*
Steam dump valves close (RCS temperature < 547 °F)	Not accomplished*
HHSI initiated (3 pumps)	00:03
First AFW delivery	00:03
Operators take control of AFW	10:00**
AFW delivery to faulted steam generator secured	00:12***
1 of 3 HHSI pumps secured	00:15
Faulted steam generator PORV 1st lifts	00:32
Faulted steam generator flooded	00:42
TDAFW fails (turbine floods)	00:42
Faulted steam generator isolated	Not accomplished by operators

Event Description	Time (hh:mm)
HHSI secured	Not accomplished by operators
Leakage through faulted steam generator PORV stopped	Not accomplished by operators
100 °F/hr cool-down initiated	Not accomplished by operators
RWST exhausted (safety injection ends)	11:03
RCPs trip	18:22
Steam Generator C PORV fails open (due to excessive cycling)	31:00
First accumulator discharge	31:16
ECST exhausted (AFW delivery ends)	33:29
Steam Generator B PORV fails open (due to excessive cycling)	38:20
Core uncovering begins	43:48
First fission product gap release	45:46

* The automatic operation of steam dump valves was not represented in the Surry MELCOR model. The thermal-hydraulic signature in the subject calculation suggests that the valves might be active for the first 6 min following scram but at no other time. Valve action would reduce RCS temperature by ~25 °F for the first few min and by a few °F in the next few min. The differences are thought to be inconsequential.

** Best-estimate timing for operators assuming manual control of AFW is 15 min. A discrepancy in the MELCOR input initiated level control of AFW at 10 min.

*** Best-estimate timing for operators securing AFW delivery to the faulted steam generator is 15 min. A discrepancy in the MELCOR input interrupted AFW to the steam generator at 12 min and 30 sec.

5.4.2.1 Thermal-hydraulic Response

Figure 95 through Figure 99 present the thermal hydraulic response for a spontaneous SGTR where the reactor systems operate as designed but the reactor operators fail to accomplish key actions per training and procedure. Specifically, the operators fail to depressurize and cool the RCS. The tube rupture quickly leads to a reactor scram, turbine stop valve closure, HHSI injection, and AFW injection. The flow of the primary system coolant through the tube rupture into the secondary side of the faulted steam generator results in a sustained leak to the environment through the relief valve exhaust piping beginning at 32 min when the steam generator PORV opens.

Once water level is restored in the steam generators (i.e., shortly after the operators taking control of the TD-AFW), a feed and bleed decay heat removal process is established using the HHSI and the leakage through the SGTR. After 15 minutes, one HHSI pump is stopped, which leaves two HHSI pumps running. The two HHSI pumps keep the RCS and faulted steam generator full of water, the primary system pressure at ~15.5 MPa, and the faulted steam generator relief valve cycling to relieve steam and water. The feed (HHSI) and bleed (SGTR) process removes the core decay heat until the RWST is drained at 11 hr 3 min. After the RWST empties, all vessel injection is unavailable and HHSI injection stops. The recirculation mode of the vessel injection is unavailable because all the RWST water leaked outside of the containment through the SGTR and out the steam generator relief valve.

After the HHSI stops, the RCS heats to saturation over the course of several hours and an extended boiloff of RCS inventory begins. The RCPs are stopped at the first occurrence of void in the RCS simulating the pumps tripping on their own or the operators shutting them down on account of erratic performance. As the RCS heats to saturation, the intact steam generators pressurize up to the setpoint on the PORVs. The PORVs remove heat from the RCS until the ECST is exhausted. The steam generators continue to remove heat from the RCS but the liquid level drops without any make-up. (Note that once the RCS begins to void and the RCPs stop, the heat rejection to the steam generators is by reflux cooling.)

The PORVs on the intact generators fail open due to excessive cycling (> 256 cycles) at 31 hr (SG-C) and 38 hr 20 min (SG-B). There was a pressure transient following each successive intact steam generator PORV failure. The stuck open PORV failure caused SG-C to depressurize and preferentially remove the heat from the primary system. However, the ECST empties at 33 hr 29 min, which the stops AFW make-up flow. For example, SG-C continues to remove the primary system heat until 36 hr until the water inventory is gone. Thereafter, the primary system and SG-C pressurize until the SG-B PORV starts to cycle to remove heat. However, the SG-B PORV fails shortly thereafter and the process repeats with Steam Generator B. Once both intact generators have dried out, the primary system and the faulted generator pressurize and the faulted generator's PORV starts to cycle, which continues to vent water via the SGTR. The water level in the vessel initially swells after the Steam Generator B dryout as steam and water are vented out of the primary system via the SGTR. However, the loss of primary system inventory leads to a sharp water level decrease after ~42 hours (see Figure 96). The core starts to uncover at 43 hr 48 min. The first release of fission products from a fuel/cladding gap occurs at 45 hr 46 min. The calculation was stopped at the start of the fission product releases due to the high unlikelihood that operators would fail to depressurize and cool the reactor system for 43 hr.

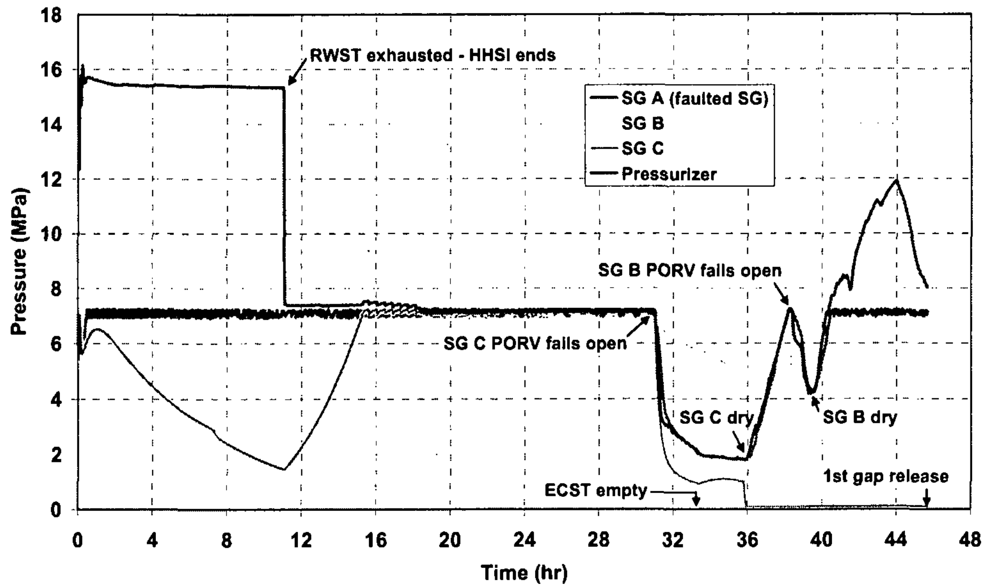

Figure 95 The SGTR without Operator Action – System Pressures.

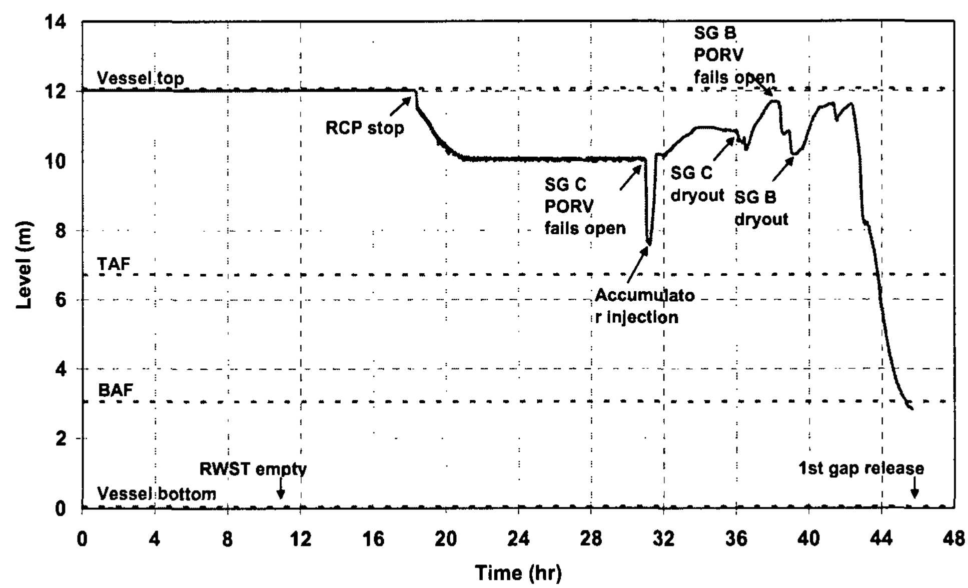

Figure 96 The SGTR without Operator Action – RPV Water Level.

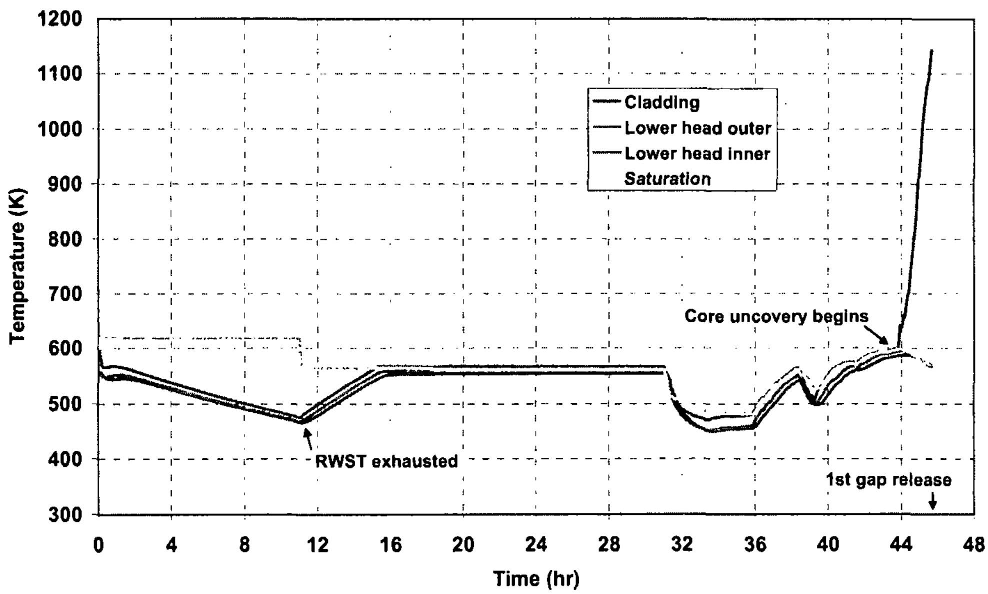

Figure 97 The SGTR without Operator Action – Maximum Cladding and Lower Head Temperatures.

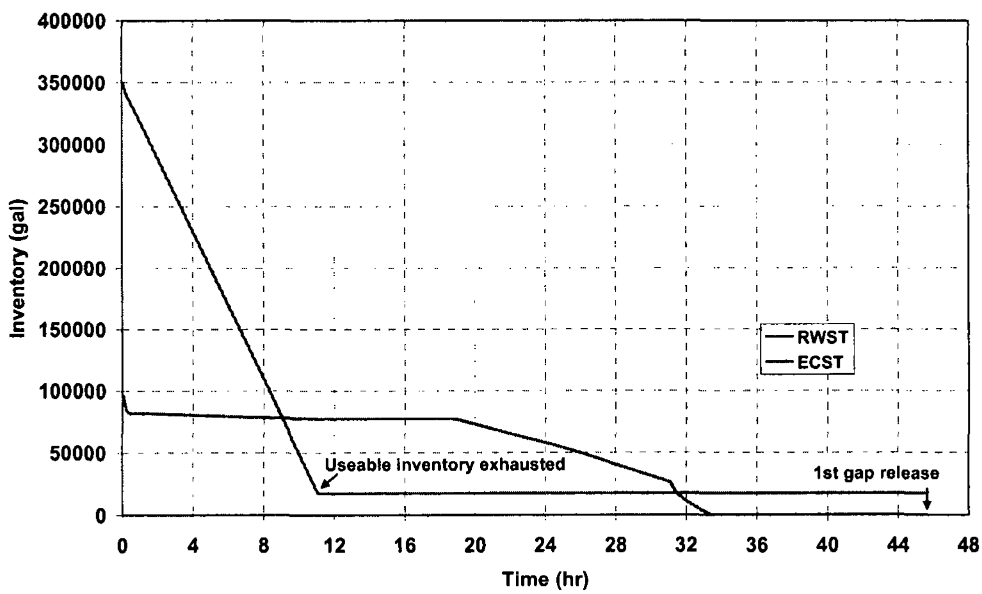

Figure 98 The SGTR without Operator Action – RWST and ECST Inventories.

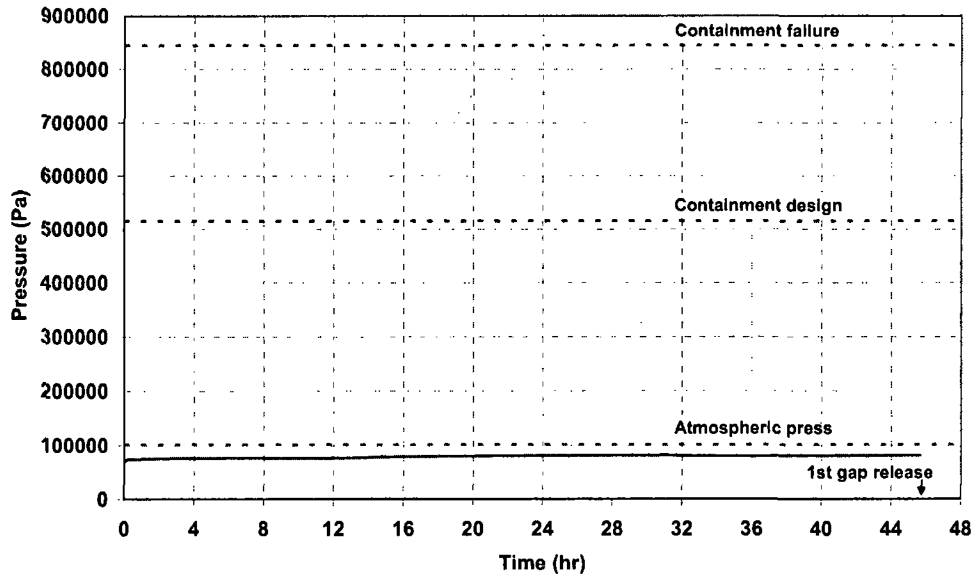

Figure 99 The SGTR without Operator Action – Containment Pressure.

5.4.2.2 Radionuclide Release

The radionuclide release analysis was not performed for this scenario due to the low likelihood that operators would fail to depressurize and cool the reactor system for the 43 hr (i.e., the elapsed time before the reactor core would begin to uncover).

5.4.3 Unmitigated - Spontaneous SGTR with Failed Operator Action and Faulted Steam Generator SORV

Table 12 summarizes the timing of the key events in the spontaneous steam generator tube rupture with some failed operator actions. As described in Section 3.3, the accident scenario initiates with a spontaneous failure of one steam generator tube. After about two minutes, the reactor successfully trips; the containment isolates; and all powered safety systems are available. The timings of the key events are discussed further in Sections 5.4.3.1 and 5.4.3.2. The operator actions are not successful to isolate the faulted steam generator or cooldown the RCS using the two intact generators. Eventually, all the water in the refueling water storage tank is exhausted and core damage proceeds. Due to the long amount of time until the RWST empties (11 hr) and the emergency condensate storage tank for the auxiliary feed water empties, the start of core uncovery is delayed until almost 23 hr. It was judged unlikely that the operators would not correct missed actions to perform a safe shutdown. Section 5.4.3.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.4.3.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 12 The timing of key events for the Spontaneous SGTR with Failed Operator Action and with Faulted Steam Generator SORV.

Event Description	Time (hh:mm)
Spontaneous SGTR	00:00
Reactor scram	00:03
Turbine stop valves close	00:03
Steam dump valves open and modulate	Not accomplished*
Steam dump valves close (RCS temperature < 547 °F)	Not accomplished*
HHSI initiated (3 pumps)	00:03
First AFW delivery	00:03
Operators take control of AFW	10:00**
AFW delivery to faulted steam generator secured	00:12***
1 of 3 HHSI pumps secured	00:15
Faulted steam generator PORV 1st lifts	00:32
Faulted steam generator flooded	00:42
TDAFW fails (turbine floods)	00:42
Faulted steam generator PORV fails open (1st liquid flow through valve)	00:44
Faulted steam generator isolated	Not accomplished by operators
HHSI secured	Not accomplished by operators
Leakage through faulted steam generator PORV stopped	Not accomplished by operators
100 °F/hr cool-down initiated	Not accomplished by operators
RWST exhausted (safety injection ends)	08:43
First accumulator discharge	08:53
RCPs trip	12:43
Core uncovering begins	22:48
First fission product gap release	26:44

* The automatic operation of steam dump valves was not represented in the Surry MELCOR model. The thermal-hydraulic signature in the subject calculation suggests that the valves might be active for the first 6 min following scram but at no other time. Valve action would reduce RCS temperature by ~25 °F for the first few min and by a few °F in the next few min. The differences are thought to be inconsequential.

** Best-estimate timing for operators assuming manual control of AFW is 15 min. A discrepancy in the MELCOR modeling initiated level control of AFW at 10 min.

*** Best-estimate timing for operators securing AFW delivery to the faulted steam generator is 15 min. A discrepancy in the MELCOR modeling interrupted AFW to the steam generator at 12 min and 30 sec.

5.4.3.1 Thermal-hydraulic Response

Figure 100 through Figure 104 present the thermal hydraulic response for a spontaneous SGTR with failed operator action with the same initiating event, system availabilities, and mitigative actions as Section 5.4.2 with one distinction. In this scenario, the PORV serving the faulted steam generator fails open when liquid first flows through it. Similar to the scenario described in Section 5.4.2, the SGTR quickly leads to a reactor scram, turbine stop valve closure, HHSI injection, and AFW injection. The flow of the primary system coolant through the tube rupture pressurizes the secondary side of the faulted steam generator. The steam generator PORV first opens at 32 min. The faulted steam generator floods at 42 min and starts flowing water through the PORV at 44.5 min. The liquid flow is assumed to cause the PORV to fail open, which creates a direct pathway to the environment that bypasses the containment. Unlike the scenario in Section 5.4.2, the pathway to the environment through the failed PORV is always open (i.e., the PORV does not close when the pressure decreases below the closing setpoint).

Once the water level is restored in the steam generators (i.e., shortly after the operators taking control of the TD-AFW), a feed and bleed decay heat removal process is established using the HHSI and the leakage through the SGTR. After 15 minutes, one HHSI pump is stopped, which leaves two HHSI pumps running. The two HHSI pumps keep the RCS and faulted steam generator full of water, the primary system pressure at ~15.5 MPa, and the faulted steam generator relief valve cycling to relieve steam and water. The faulted steam generator depressurizes once the PORV fails open at 42 min and becomes the prime mechanism for heat removal. The heat removal through the faulted generator depressurizes the primary system and the two intact generators. The feed (HHSI) and bleed (SGTR) process removes the core decay heat until the RWST is drained at 8 hr 43 min. After the RWST empties, all vessel injection is unavailable and HHSI injection stops. The recirculation mode of the vessel injection is unavailable because all the RWST water leaked outside of the containment through the SGTR and out the steam generator relief valve.

After the HHSI stops, the faulted generator continues to remove the heat from the primary system until the water stops flowing through the tube rupture and the faulted generator dries out at ~17.5 hr. Subsequently, the intact generators begin to remove some energy from the primary system through reflux cooling (i.e., see intact steam generator pressurization after 17.5 hr in Figure 100). However, the steam flow out the tube rupture steadily decreases the vessel inventory. The core starts to uncover at 22 hr 48 min. The first release of fission products from a fuel/cladding gap occurs at 26 hr 44 min. The calculation was stopped at the start of the fission product releases due to the high unlikelihood that operators would fail to depressurize and cool the reactor system for 26 hr.

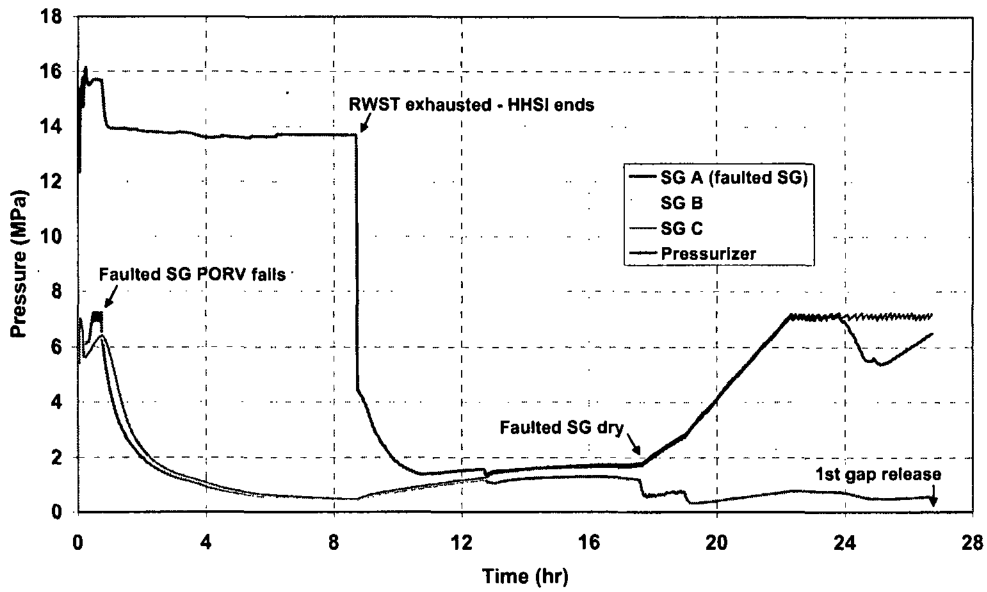

Figure 100 The SGTR without Operator Action with SORV – System Pressures.

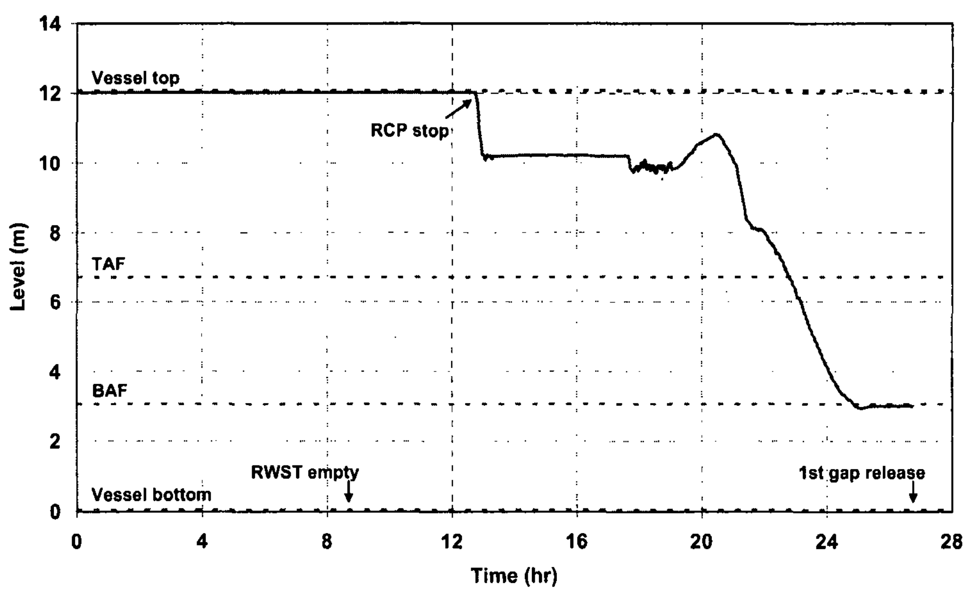

Figure 101 The SGTR without Operator Action with SORV – RPV Water Level.

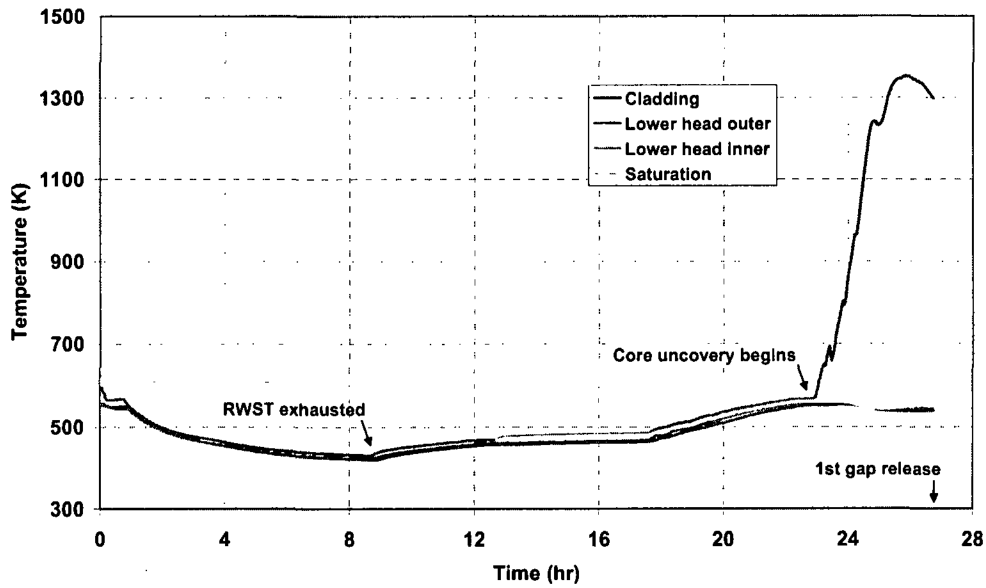

Figure 102 The SGTR without Operator Action with SORV – Max Clad and Lower Head Temperature.

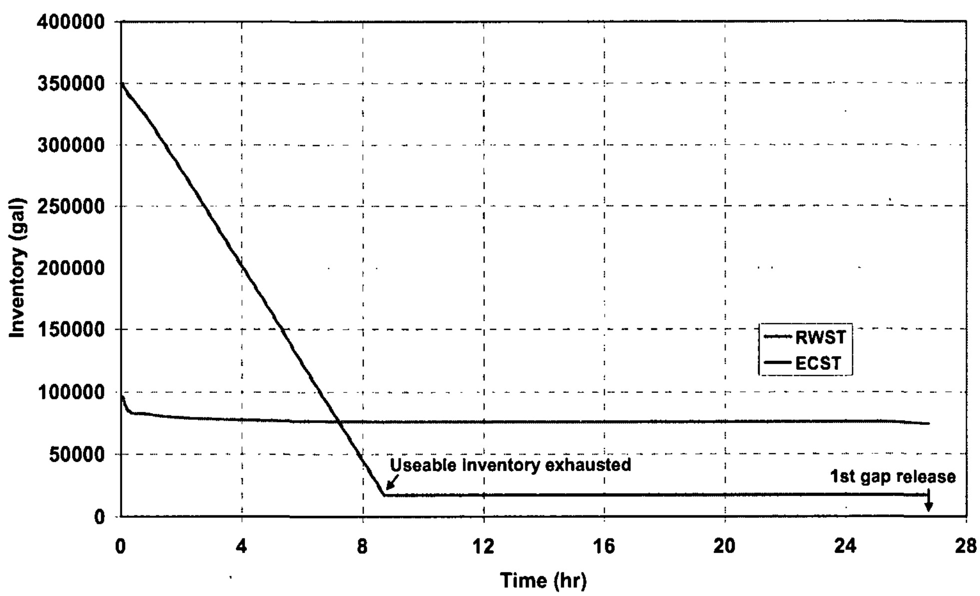

Figure 103 The SGTR without Operator Action with SORV – RWST and ECST Inventories.

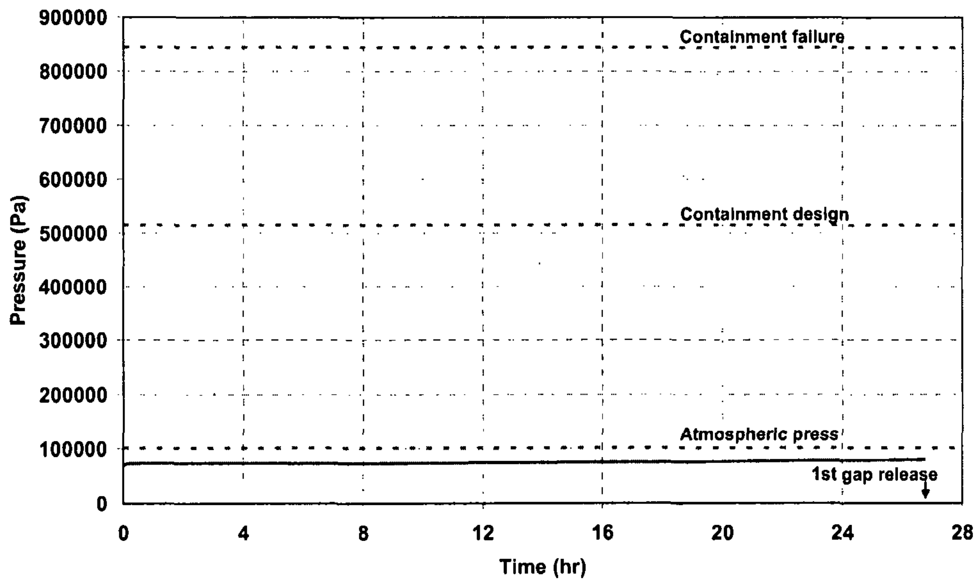

Figure 104 The SGTR without Operator Action with SORV – Containment Pressure.

5.4.3.2 Radionuclide Release

The radionuclide release analysis was not performed for this scenario due to the low likelihood that operators would fail to depressurize and cool the reactor system for the 22 hr (i.e., the elapsed time before the reactor core would begin to uncover).

5.5 Interfacing System Loss of Coolant Accident

The unmitigated interfacing systems loss-of-coolant accident initiates with a common mode failure of both of the inboard isolation check valve disks of the low-head safety injection piping. The downstream low-pressure piping outside of containment pressurizes to failure, which creates a loss-of-coolant accident in the safeguards building. Section 5.5.1 presents the results of an unmitigated scenario where the expected operator actions are successful but the injection systems fail when the supply water is exhausted. For the mitigated scenario in Section 5.5.2, the water supply from the unaffected unit is aligned to the unit with interfacing systems loss-of-coolant accident. Finally, the results of three separate sensitivity calculations based on the unmitigated interfacing systems loss-of-coolant accident are presented in Section 5.5.3 where the response of low-pressure injection piping is explicitly modeled and break location is varied.

5.5.1 Unmitigated Interfacing System Loss of Coolant Accident

Table 13 summarizes the timing of the key events in the unmitigated ISLOCA. As described in Section 3.4.1, the accident scenario initiates with a common mode disk failure of both low-head safety injection (LHSI) inboard isolation check valves. The resulting pressurization of the low pressure piping (design pressure 600 psi) between LHSI outboard isolation valve and the LHSI pump to normal RCS pressure (2200 psi) causes a bypass LOCA in the safeguards building (see schematic in Figure 105). The reactor successfully trips and the containment isolates. All powered safety systems are available. The timings of the key events are discussed further in Sections 5.5.1.1 and 5.5.1.2. However, it is worth initiating noting that fission product releases from the fuel do not begin until 9 hr 12 min and significant fission product releases to the environment (i.e., defined as a noble gas release >1% of inventory) do not begin until 9 hr 53 min. Section 5.5.1.1 summarizes the thermal-hydraulic response of the reactor and containment while Section 5.5.1.2 summarizes the associated radionuclide release from the fuel to the environment.

Table 13 Sequence of Events for the Unmitigated ISLOCA Sequence.

Event Description	Time (hh:mm)
Interfacing LOCA of LHSI line in Safeguards Bldg	00:00
Reactor SCRAM TCV begins closing MFW trips	< 00:01
ECCS signal LHSI initiates HHSI initiates	< 00:01
Safeguards building flooded LHSI fails LHSI gravity feed into Safeguards Bldg	00:02

Event Description	Time (hh:mm)
Accumulators starts discharging	00:13
Secure 1 of 3 HHSI pumps	00:15
Shift to hot leg injection	00:45
Start 100°F/h cooldown of primary	01:00
Auxiliary Building sump pumps flooded (depth = 2')	01:18
Secure 2 of 3 HHSI pumps	02:00
Accumulators are empty	02:24
RWST empty	03:20
Start of fission product gap releases	09:12
Vessel lower head failure by creep rupture	15:02
Debris discharge to reactor cavity	15:02

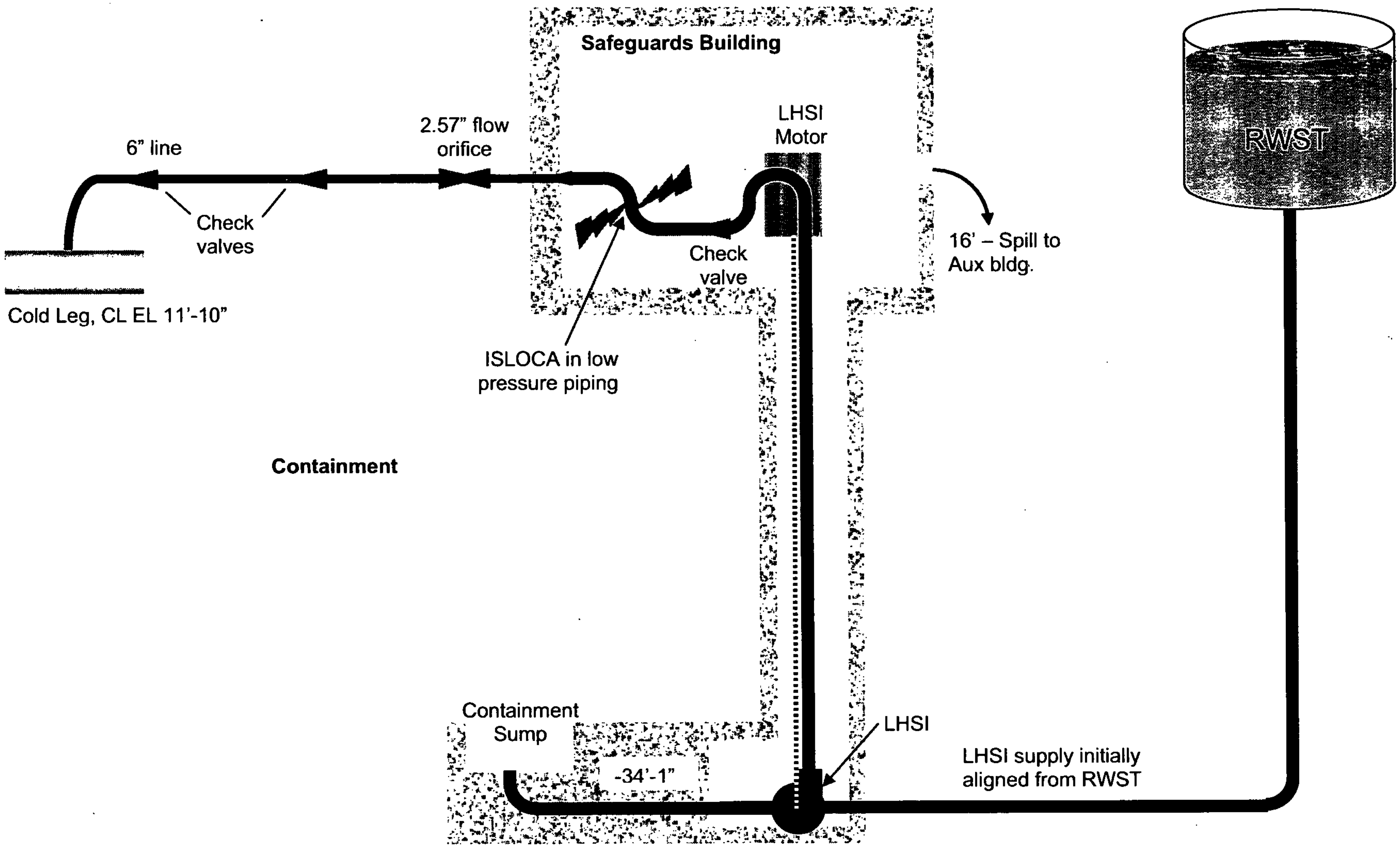

Figure 105: Schematic of the unmitigated ISLOCA piping layout (also see footnote 10 on page 35).

5.5.1.1 Thermal-hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 106. At the start of the accident sequence, the reactor pressure quickly falls following the pipe break in one of the low-head safety injection (LHSI) lines. The pipe break is a double-ended guillotine break of the LHSI line in the safeguards building at ~1 m below the eventual flooded water level. Approximately 20 seconds after the pipe break, the reactor successfully trips due to the rapidly decreasing reactor coolant system pressure and pressurizer level. Subsequently, the turbine control valves (TCVs) and the main feedwater pumps trip. However, the two motor-driven and one turbine-driven auxiliary feedwater pumps automatically start following the loss of the main feedwater flow. An emergency core cooling system (ECCS) safety injection signal is also generated due to the decreasing pressurizer pressure. The ECCS signal starts the three high-head and two low-head safety injection (HHSI and LHSI) pumps.

The secondary pressure initially rises in response to the TCV closure but cools down after the cold AFW starts to refill the steam generators. Once the primary system depressurizes to the steam generator pressure, the primary and secondary system pressures remain coupled for the first 20 min. However, the energy and inventory loss out the break eventually exceeds the thermal coupling through the steam generator and the primary system depressurizes more quickly than the secondary. After 15 min, the operator takes control of the AFW and shuts down one of the three HHSI pumps. At 1 hour, the operators begin a controlled 100°F/hr cooldown to reduce the primary system pressure and therefore reduce the magnitude of the break flow.

The ECCS flow is shown in Figure 107. The three HHSI and two LHSI pumps started in response to the EECS signal. Due to the double-end guillotine break in the LHSI line and the interconnectivity of the supply lines, all the LHSI flow went out the break into the safeguards building. By 1 min 39 sec, the two LHSI pumps flooded the safeguards building. Since the LHSI motors were in the safeguards building, they failed when they were flooded. Subsequently, the refueling water storage tank, which supplies water to the LHSI and HHSI pumps, started to gravity drain through the broken pipe at ~1200 gpm. As the RWST continued to drain, the gravity-driven flow decreased until the tank emptied at 3.3 hr (see Figure 108). Due to the connectivity through the ECCS piping, about one-third of the total HHSI back-flowed into the safeguards building through the broken LHSI line before entering the RCS.

Figure 109 shows the two-phase level response in the vessel. The vessel water level drops quickly following the pipe break but starts to recover after 16 min in response to the decreased break flow at lower pressure, the accumulator injection, and the ECCS flow. However, following the shift to hot leg ECCS injection at 45 min, the reactor pressure drops quickly in response to sudden condensation of steam in the hot leg. The sudden drop in primary system pressure immediately causes a discharge of accumulator water. After the condensation transient, the primary system pressure stabilizes but the two-phase level in the vessel temporarily dropped below the top of the core. Just after 2 hr, the level begins to recover as the primary system pressure begins to decrease with the secondary pressure. The accumulators inject from 2 to 3 hr as the primary system pressure continues to decreases. The ECCS injection continues to maintain the level until 3.3 hr when the RWST is empty. Subsequently, the water level decreases into the core region and a sustained fuel heatup begins. The fuel cladding fails in the innermost ring at 9 hr 12 min, which starts the release of fission products from the fuel. The fuel rods start to degrade above 2400 K as the molten zirconium breaks through the oxidized shell of the cladding on the fuel rods and eventually collapse due a thermo-mechanical weakening of the remaining oxide shell at high temperature. As shown in Figure 110, the peak fuel debris temperature reaches the fuel-zirconium oxide eutectic melting temperature of 2800 K.

By 10.4 hr, a large debris bed had formed in the center of the core. The debris bed continued to expand until 11.1 hr when all the fuel had collapsed and was resting on the core plate. The hot debris failed the core support plate and fell onto the lower core support plate, which failed at 13.8 hr. Following the lower core support plate failure, the debris bed relocated onto the lower head. The small amount of remaining water in the lower head was quickly boiled away. As shown in Figure 111, the hot debris quickly heated the lower head to above the melting temperature of stainless steel (i.e., 1700 K) on the inside surface. As the heat transferred through the lower head, it eventually failed at 15 hr 2 min due to the creep rupture failure criterion.

By 15.5 hr, nearly all the hot debris relocated from the vessel into the reactor cavity in the containment under the reactor vessel. The hot debris immediately boiled away the water in the reactor cavity started to ablate the concrete. The ex-vessel core-concrete interactions continued for the remainder of the calculation, which generated non-condensable gases. However, due to the leakage path through the failed LHSI line outside the containment, the containment did not pressurize significantly, as shown in Figure 112. The hot gases and fission products from the CCI vented through the vessel to the safeguards building.

The water volume in the auxiliary building is shown in Figure 113. The auxiliary building stopped filling once the RWST drained at 3.3 hours. Although the water in the primary system and RWST was discharged into the safeguards and auxiliary building, the total amount in the auxiliary building was below the amount required to flood the HHSI pumps (i.e., 530,000 gal). Consequently, if an additional water source was aligned to the HHSI pumps, the pumps were still operational.

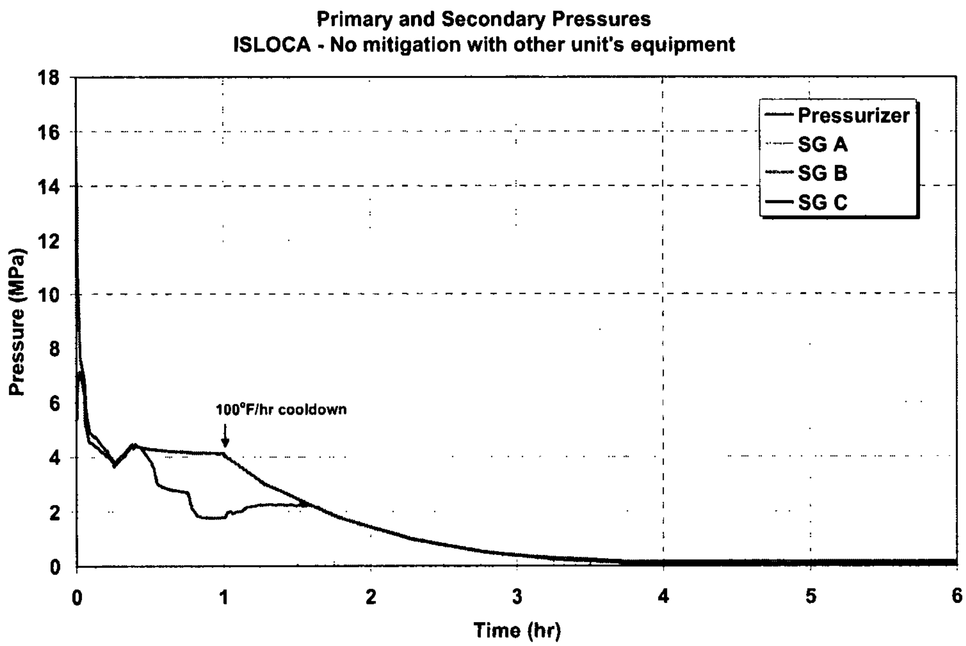

Figure 106 The Unmitigated ISLOCA Primary and Secondary Pressures History.

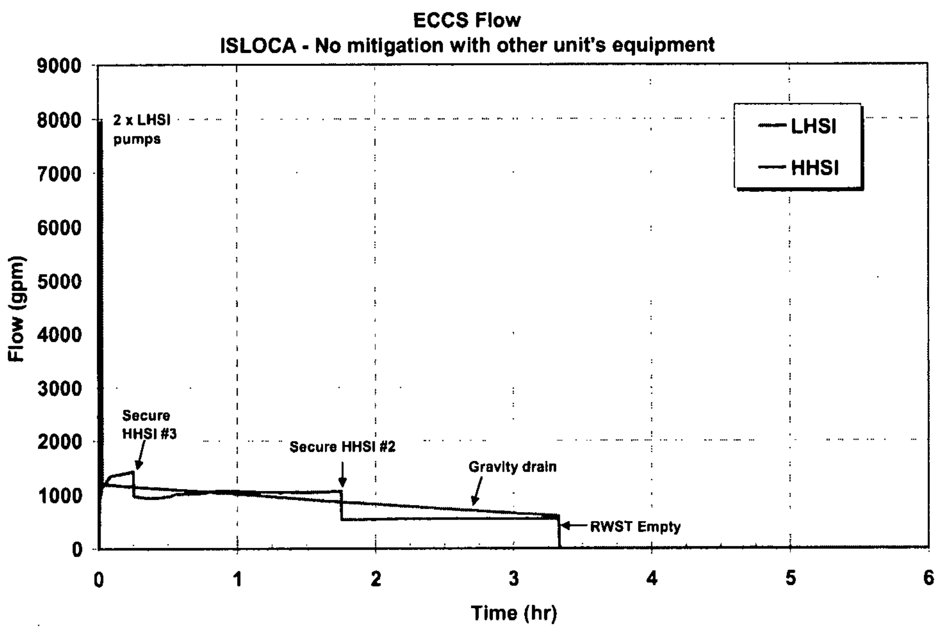

Figure 107 The Unmitigated ISLOCA ECCS Flow History.

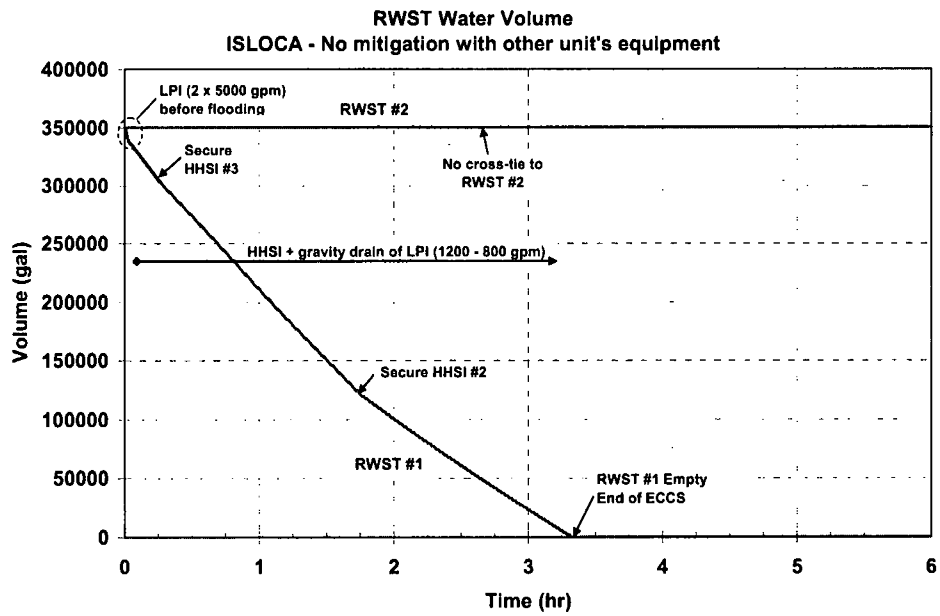

Figure 108 Unmitigated ISLOCA Affected and Unaffected Units RWST Water Volume History.

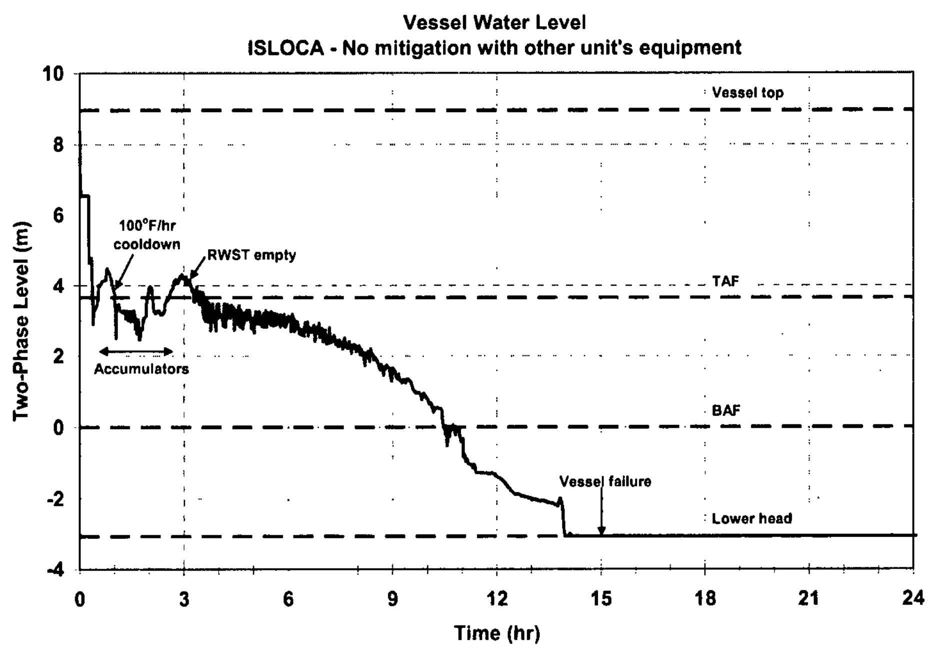

Figure 109 The unmitigated ISLOCA vessel two-phase coolant level.

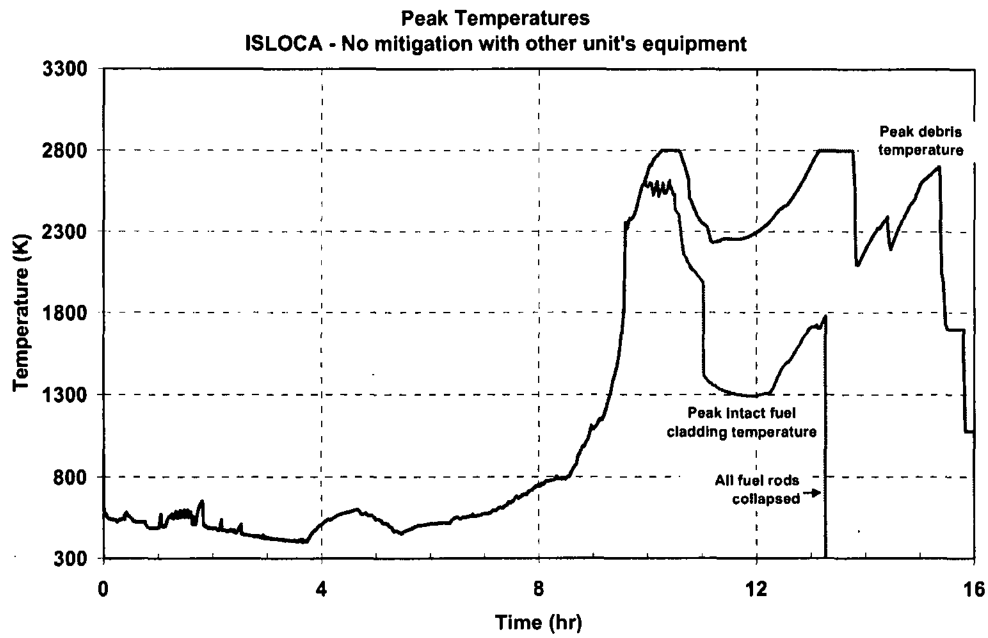

Figure 110 Unmitigated ISLOCA peak cladding temperature.

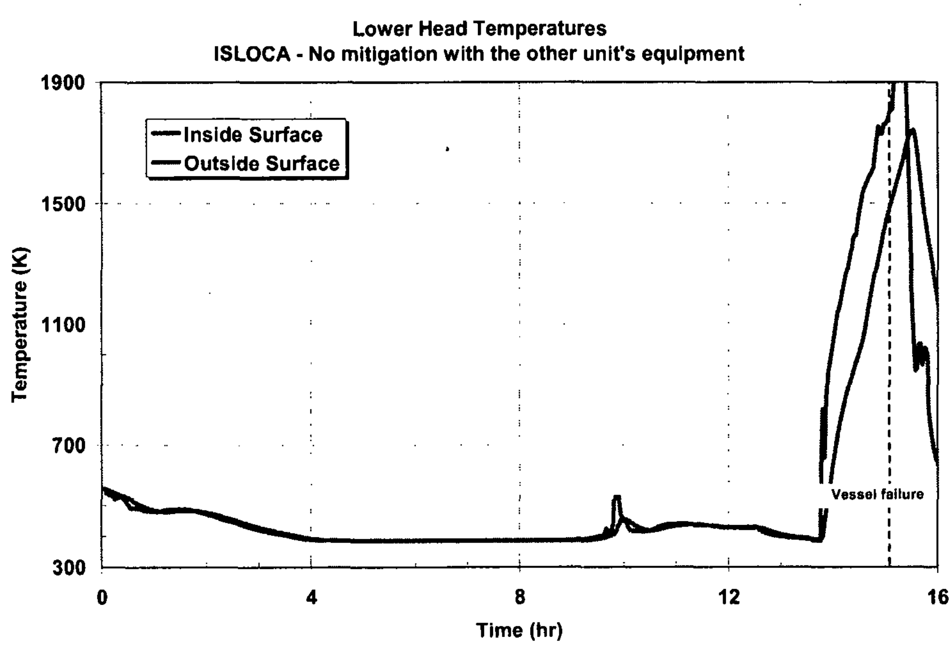

Figure 111 Unmitigated ISLOCA lower head inner and outer temperature history.

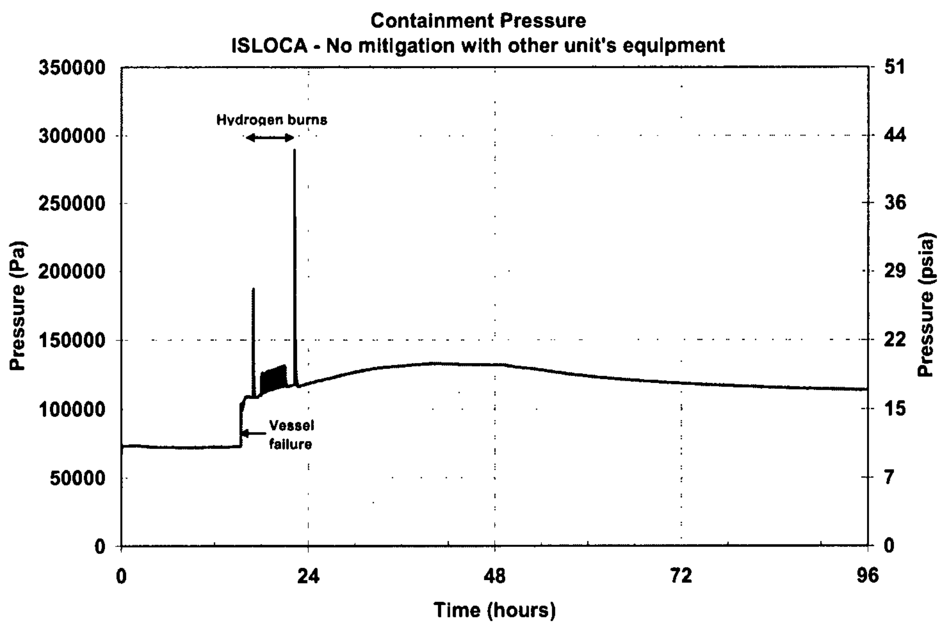

Figure 112 The unmitigated ISLOCA containment pressure history.

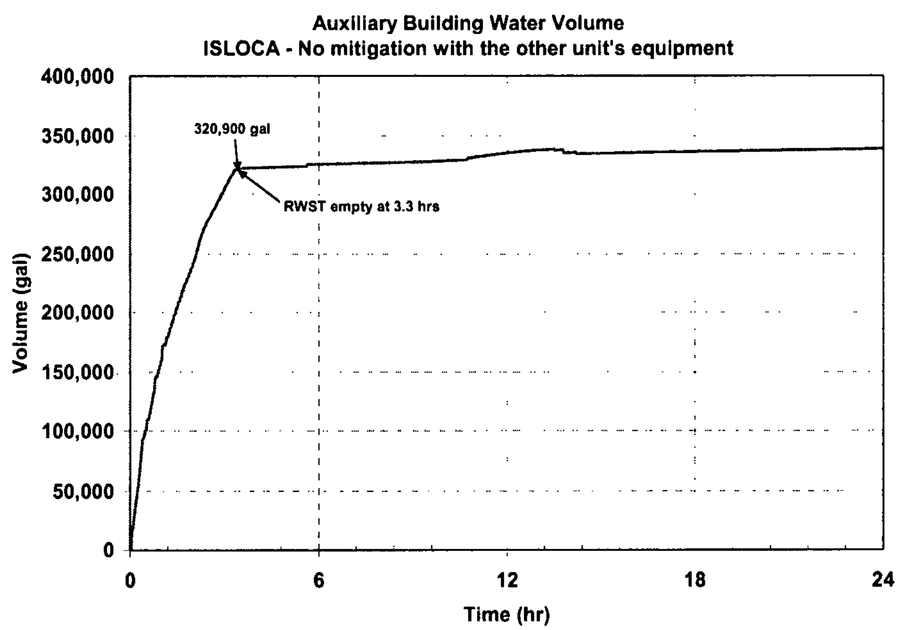

Figure 113 The unmitigated ISLOCA auxiliary building water volume.

5.5.1.2 Radionuclide Response

The fission product releases from the fuel started following the first thermo-mechanical failures of the fuel cladding in the hottest rods at 9 hr 12 min. The in-vessel fission product release phase continued through vessel failure at 15 hr. Due to the break in the cold leg piping, the fission products circulated around the RCS and out the failed LHSI line into the safeguards building. The break of the piping in the safeguards building was under water, which provided some scrubbing of the released fission products. However, (a) there was a high non-condensable hydrogen gas content in the carrier gas during the fission product releases; (b) the broken piping was large (10' diameter); and (c) the pool was near saturated conditions. Consequently, the water scrubbing and retention in the safeguards building was less than ~6.6 for aerosols, see Figure 114.

Figure 115 and Figure 116 show the fission product distributions of the iodine and cesium radionuclides that were released from the fuel, respectively. Approximately 96% and 98% of the iodine and cesium, respectively, were released from the fuel prior to vessel failure while the remaining amount was released ex-vessel. Once the fission products were released from the fuel, they circulated around the RCS through the steam generator to the cold leg with the failed LHSI line. The majority of the fission products were transported out the pipe break in the safeguards building. However, some of the fission products were retained in the vessel and the RCS. At 13.8 hr, the core relocated into the lower plenum and vaporized the remaining water, which resuspended aerosols that were settled in the pool. The resulting pressurization increased the flow of resuspended fission products into the auxiliary building. After vessel failure, the RCS pressure decreased.[29] As a result, some of the water in the safeguards building flowed back into the vessel and onto the containment floor.

Finally, Figure 117 summarizes the releases of the radionuclides to the environment. At 4 days, 98 % of the noble gases, 11% of the tellurium, 11% if the iodine, 10% of the cesium, 9% of the radioactive cadmium, and 6% of the molybdenum and radioactive tin had been released to the environment. All other releases were less than 0.1% of the initial inventory. As shown in the figure, most of the releases to the environment occurred during the in-vessel release phase from 10 to 11 hr. After the vessel failure at 15 hr, the environmental releases further diminished except for some late revaporization releases, which flow with the hot gases from the reactor cavity in the containment through the vessel to the break.

[29] The containment pressure was subatmospheric at vessel failure and caused a decrease in the primary system pressure. As the primary system pressure decreased, some water in the safeguards building back-flowed into the reactor and fell onto the containment floor.

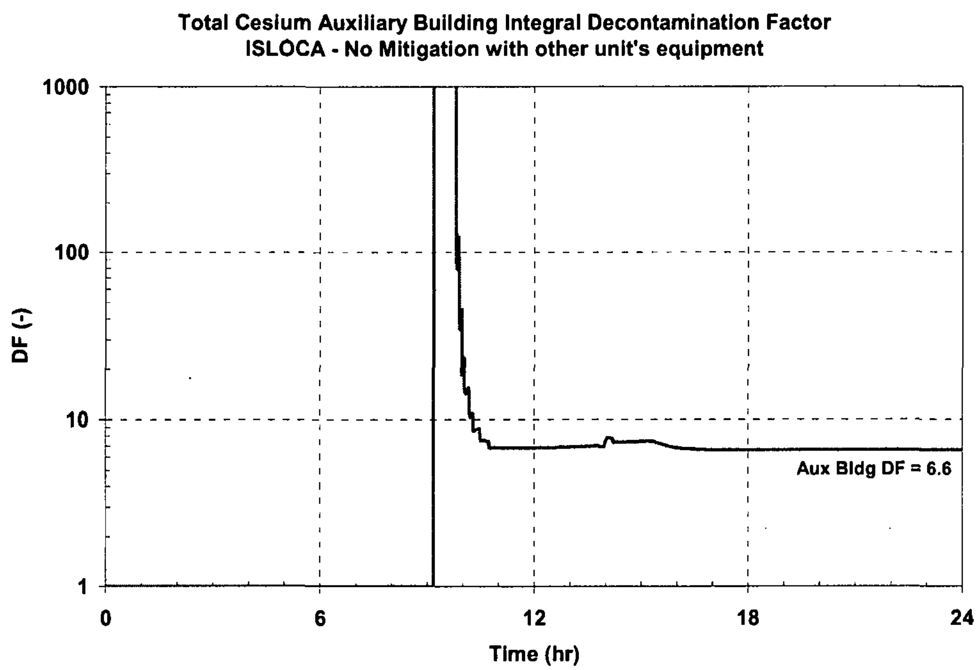

Figure 114 The unmitigated ISLOCA integral cesium aerosol decontamination factor.

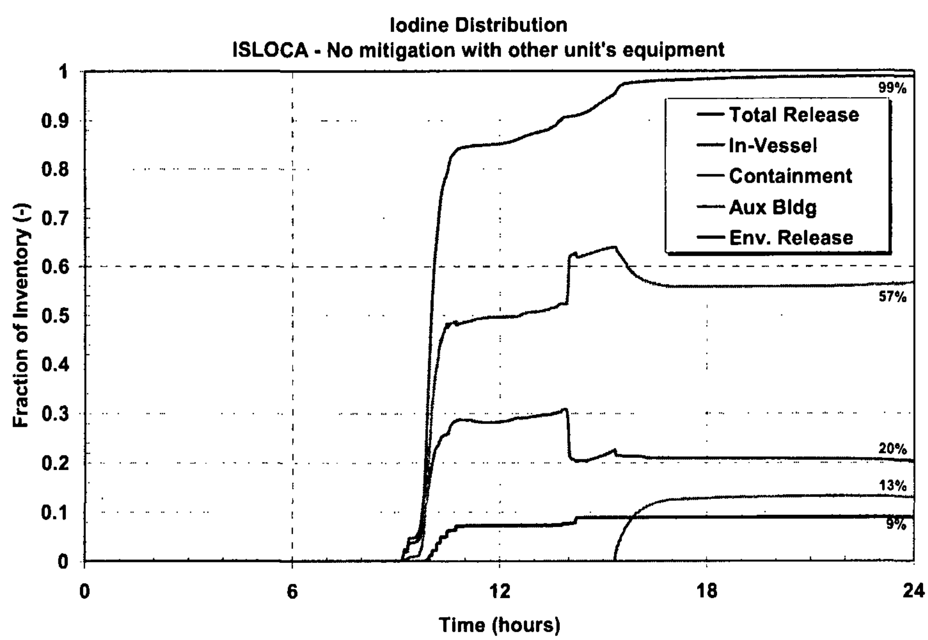

Figure 115 The unmitigated ISLOCA iodine fission product distribution history.

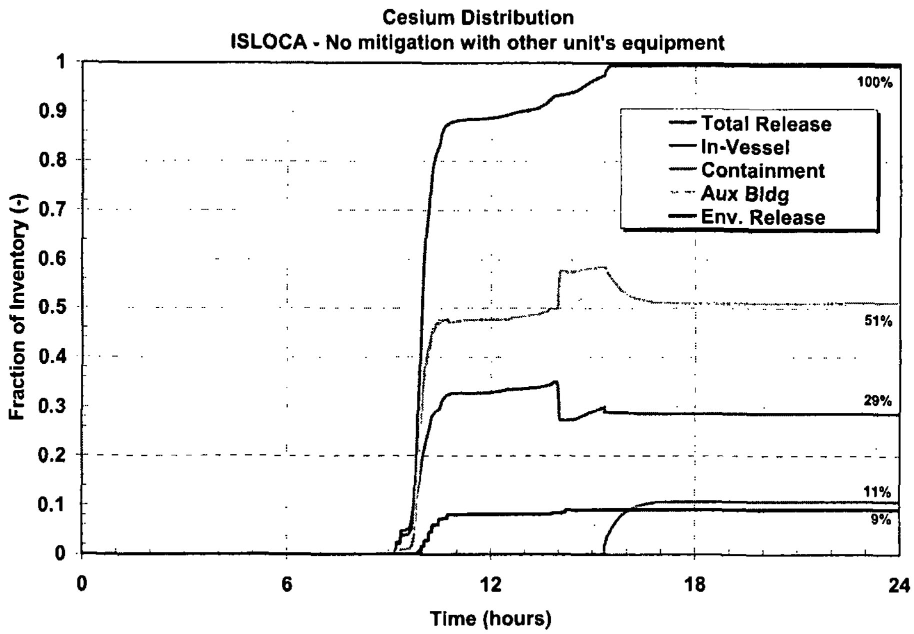

Figure 116 The unmitigated ISLOCA cesium fission product distribution history.

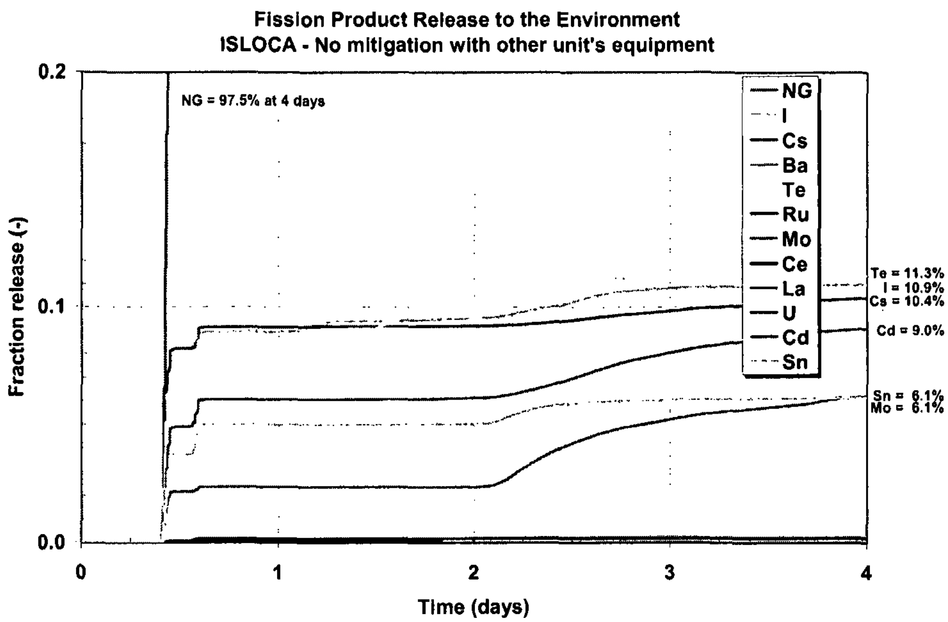

Figure 117 Unmitigated ISLOCA environmental release history of all fission products.

5.5.2 Mitigated Interfacing System Loss of Coolant Accident

Table 14 summarizes the timing of the key events in the mitigated ISLOCA. As described in Section 3.4, the accident scenario initiates with a common mode disk failure of both low-head safety injection (LHSI) inboard isolation check valves. The resulting pressurization of the low pressure piping (design pressure 600 psi) between LHSI outboard isolation valve and the LHSI pump to normal RCS pressure (2200 psi) causing a bypass LOCA in the safeguards building. The reactor successfully trips and the containment isolates. All powered safety systems are available. Several operator actions are credited that lead to mitigation of accident that are discussed further in Section 5.5.2.1. Since the accident was successfully mitigated without any radionuclide releases from the fuel, there is no discussion of the radionuclide response, as indicated in Section 5.5.2.2.

Table 14 Sequence of Events for the Mitigated ISLOCA Sequence.

Event Description	Time (hh:mm)
Initiating event Interfacing LOCA of LHSI line in Safeguards Bldg	00:00
Reactor SCRAM TCV begins closing MFW trips	< 00:01
ECCS signal LHSI initiates HHSI initiates	< 00:01
Safeguards building flooded LHSI fails LHSI gravity feed into Safeguards Bldg	<00:02
Accumulators starts discharging	00:13
Secure 1 of 3 HHSI pumps Operators manually control AFW to maintain the SG level	00:15
Shift to hot leg injection	00:45
Start 100°F/h cooldown of primary	01:00
Auxiliary sump pumps flooded (depth = 2')	01:18
Isolate RWST Shift injection to other unit's RWST	01:45
Initiate HHSI injection to hot leg Start 150 gpm make-up to other unit's RWST	01:45
Secure 2 of 3 HHSI pumps	02:00
Accumulators are empty	02:24
RHR initiated	06:00
Core is >10 K subcooled	06:15

Event Description	Time (hh:mm)
Break flow "stops"	9:12
HHSI throttled to maintain level (remains off)	10:00
Second Unit's RWST refilled	28:24
Calculation terminated	36:00

5.5.2.1 Thermal-Hydraulic Response

The responses of the primary and secondary pressure systems are shown in Figure 118. At the start of the accident sequence, the reactor pressure quickly falls following the pipe break in one of the low-head safety injection (LHSI) lines (see Figure 118). There is a double-ended guillotine break of the LHSI piping in the auxiliary building. Approximately 20 seconds after the pipe break, the reactor successfully trips due to the rapidly decreasing reactor coolant system pressure and pressurizer level. Subsequently, the turbine control valves (TCVs) close and the main feedwater pumps. However, the two motor-driven and one turbine-driven auxiliary feedwater pumps automatically start following the loss of the main feedwater flow. An emergency core cooling system (ECCS) safety injection signal is also generated due to the decreasing pressurizer pressure. The ECCS signal starts the three high-head and two low-head safety injection (HHSI and LHSI) pumps.

The secondary pressure initially rises following the TSV closure but cools down once the cold AFW starts to refill the steam generators. Once the primary system depressurizes to the steam generator pressure, the primary and secondary system pressures remain coupled for the first 20 min. However, the energy loss out the break eventually exceeds the thermal coupling through the steam generator, which allows the primary system to depressurize more quickly than the secondary. The operator takes control of the AFW after 15 min and shuts down one of the three HHSI pumps. At 1 hour, the operators begin a controlled 100°F/hr cooldown to reduce the primary system pressure and therefore reduce the magnitude of the break flow. At 1 hr 45 min, the HHSI injection from the unaffected reactor is aligned to the affected reactor, which extends the available injection. However, the accident is not terminated until the primary system pressure is low enough to terminate the break flow and the residual heat removal (RHR) system is operating (i.e., starting at 6 hr).

Figure 119 shows the two-phase level response in the vessel. The vessel water level drops quickly following the pipe break but starts to recover after 16 min in response to the decreased break flow at lower pressure, the accumulator injection, and the ECCS flow. However, following the shift to hot leg ECCS injection at 45 min, the reactor pressure drops quickly in response to sudden condensation of steam in the hot leg. The sudden drop in primary system pressure immediately causes a discharge of accumulator water. After the condensation transient, the primary system pressure stabilizes but the two-phase level temporarily drops below the top of the core. Just after 2 hr, the level begins to recover as the primary system pressure begins to decrease with the secondary pressure. The accumulators steadily inject from 2 to 3 hr as the primary system pressure decreases. After 2.7 hr, the water level remained near or above the top of the core. Then at 6 hr, the RHR system was initiated, which caused an immediate increase core level. The RHR system removes water out of the Loop A hot leg and returns it into the Loop B and C cold legs after cooling it in a heat exchanger. The RHR system is a relatively high flow system (~3700 gpm/pump versus 550 gpm/pump HHSI at runout conditions), which immediately flooded and subcooled the core. Since the RHR system is a closed recirculation system, the high injection flow rate did not pressurize the reactor. As shown in Figure 118, the activation of the RHR system further depressurized the reactor down to the pressure in the safeguards biulding, or approximately atmospheric conditions.

The peak cladding temperature response is shown in Figure 120. The fuel rods followed the liquid temperature for most of the transient but experienced some small, intermittent heatups between 1.5 and 2.5 hr when the two-phase water level dropped into the top quarter of the core. The heatups were relatively minor and not sustained due to steam cooling and an oscillating and frothing two-phase level. As expected, the lower head was cooled by the water in the vessel and was not challenged to fail (see Figure 121).

The ECCS flow is shown in Figure 122. The three HHSI and two LHSI pumps started in response to the EECS signal. Due to the double-end guillotine break in the LHSI line and the interconnectivity of the supply lines, all the LHSI flow went out the break into the safeguards building. By 1 minute 39 seconds, the two high flow pumps had flooded the safeguards building. Since the LHSI motors were in the safeguards building, they failed when they were flooded. Subsequently, the refueling water storage tank, which supplies water to the LHSI and HHSI pumps, started to gravity drain through the broken pipe at ~1200 gpm. As the RWST continued to drain, the flow slightly decreased until it was isolated at 1.75 hr. Due to the connectivity through the ECCS piping, about one-third of the total HHSI flow back-flowed into the safeguards building through the broken line before entering the RCS.

After the LHSI motor failure, the break flow from the RCS, one-third of the HHSI, and the LHSI gravity drain from the RWST were all simultaneously flooding the safeguards building. During their assessment of the accident, the operators would observe many things including (a) flooding in the safeguards and auxiliary buildings[30] (see Figure 123), (b) a sharp decrease in the RWST water volume (see Figure 124), (c) the failure of the LHSI motors, and (d) no water in the containment. To reduce the flooding rate in the auxiliary building, the operators shifted the HHSI from cold leg injection to hot leg injection at 45 min, which stopped the back-flow of one-third of the HHSI flow out the break. To slow the RWST drain rate, the operators secured HHSI pumps at 15 min (a standard procedure) and 1 hr 45 min (a credited action). However, due to the high flow from the gravity drain, a 50.54(x) decision was made at 1 hr 45 min to isolate the affected RWST and utilize the unaffected unit's RWST and HHSI pumps.[31] Simultaneous

[30] Once the safeguards building flooded at 1 minute 39 seconds, the water spilled into the auxiliary building basement. This was concerning because the HHSI pumps and motors are located in the auxiliary building basement, which was the sole source of ECCS after the LHSI motors failed. The flooding would be identified in the control room by the sump activation signal and perhaps local radiation monitors (i.e., from the activity in the water). If the auxiliary building flooded to 7'-9" above the floor, or 530,000 gal, the water would spill over the dams around the cubicles holding the HHSI pumps and motors.

[31] Due to the location of the break and the pumping connectivity, it was impossible to maintain HHSI and stop of the gravity drain through the pipe break. Consequently, an alternate source of injection and water supply was needed. The shift to the unaffected unit's HHSI and RWST is not in the normal emergency procedures but well known to the operators and a redundant design feature of the Surry ECCS. Since it would temporarily affect the other unit's

with the decision to shift to the unaffected unit's RWST, a 150 gpm make-up flow starts to refill the RWST. The unaffected unit's HHSI was used until 9 hr when the break flow had stopped[32] and the RHR was providing cooling. The unaffected unit's RWST was refilled by 28 hr 24 min.

Finally, Figure 125 shows the containment pressure response. Since the break was outside of the containment and the RCS was actively cooled, the pressure rise was very small and well below conditions for automatic containment spray actuation.

In summary, several key operator actions were performed to mitigate ISLOCA. First, the operators took active measures to depressurize the RCS as far as possible using a 100°F/hr cooldown via the steam generators. Next, the HHSI was shifted to hot leg injection to stop the 33% back-flow out the break. Third, HHSI pumps were secured at 15 min and 2 hr to reduce the drain down of the RWST. Fourth, the affected RWST was isolated and HHSI was established from the unaffected unit using the unaffected unit's RWST. A pump was aligned to the unaffected unit's RWST to refill it. Finally, the high flow RHR system was used to subcool the core, further depressurize the RCS, and refill the vessel to several meters above the core.

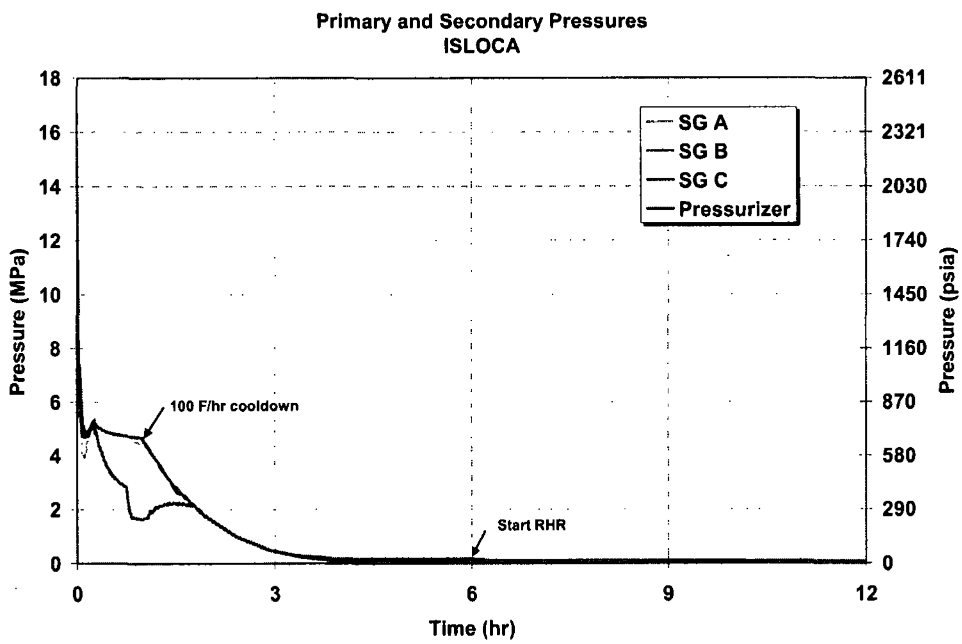

Figure 118 The mitigated ISLOCA primary and secondary pressures history.

resources, it is a 50.54(x) decision (i.e., a directive that the operators can go outside of their procedures if necessary to ensure the safety of the plant).

32 The most likely pipe break was approximately at the same elevation as the RCS cold leg. However, the safeguards building flooded well above the break elevation to an opening at the top of the room. Once the RCS had fully depressurized following RHR initiation, the water level in the safeguards room balanced the water level in the RCS to several meters above the core. The RHR system subcooled the core and stopped all boiling that might pressurize the RCS.

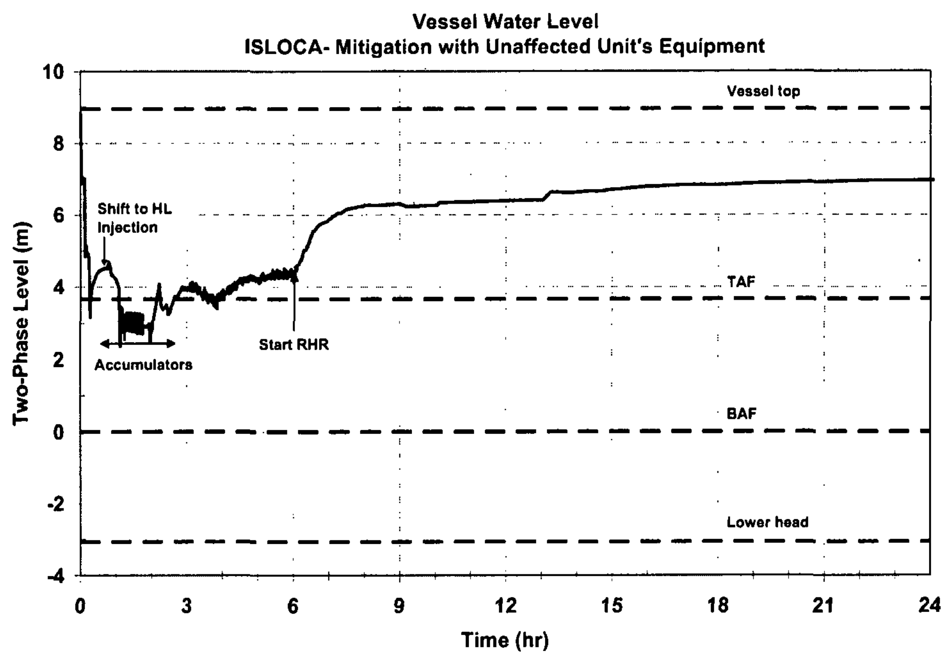

Figure 119 The mitigated ISLOCA vessel two-phase coolant level.

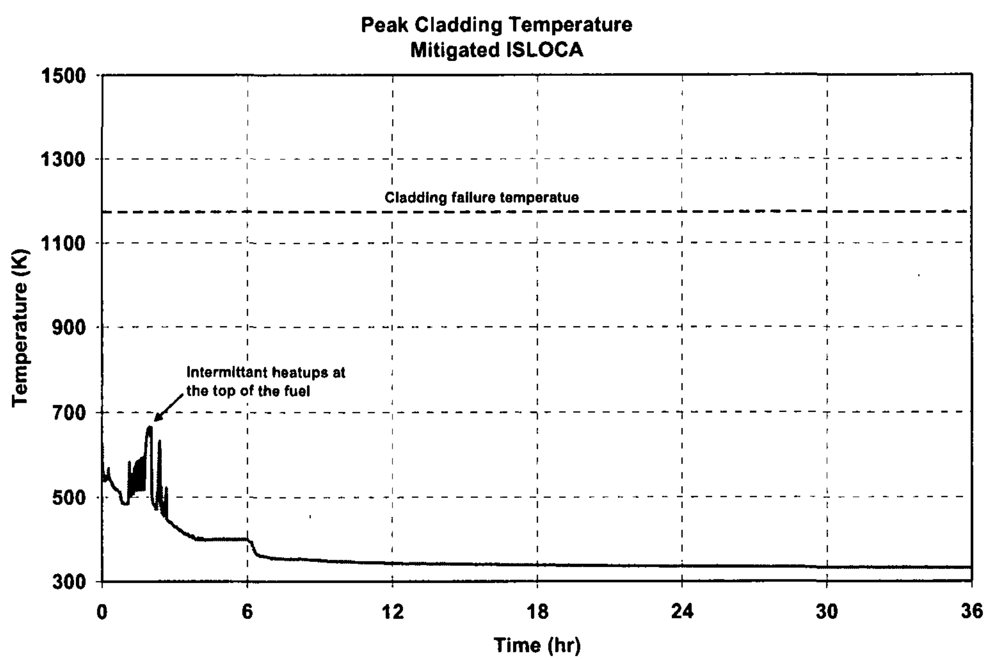

Figure 120 The mitigated ISLOCA peak cladding temperature.

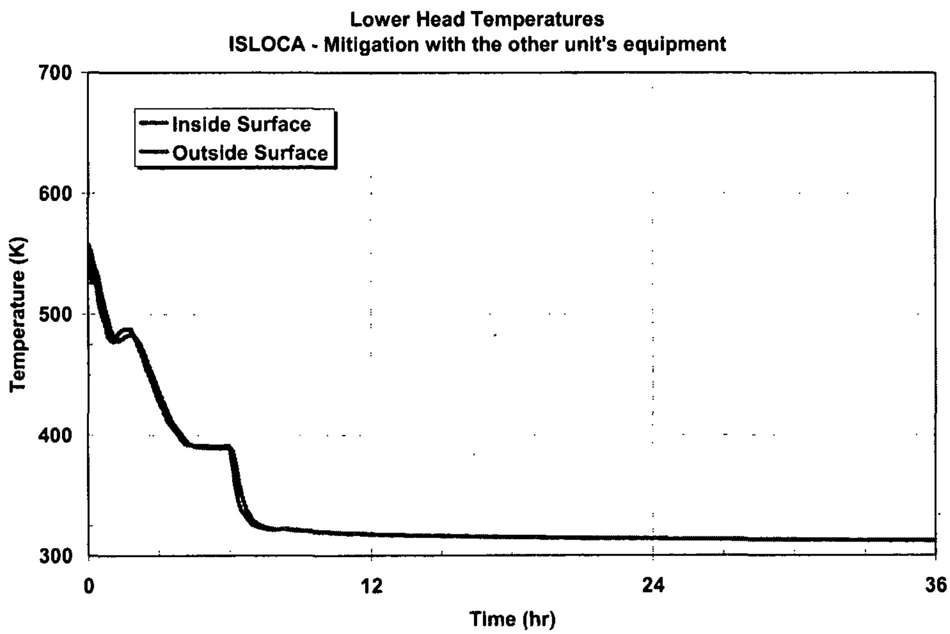

Figure 121 The mitigated ISLOCA lower head inner and outer temperature history.

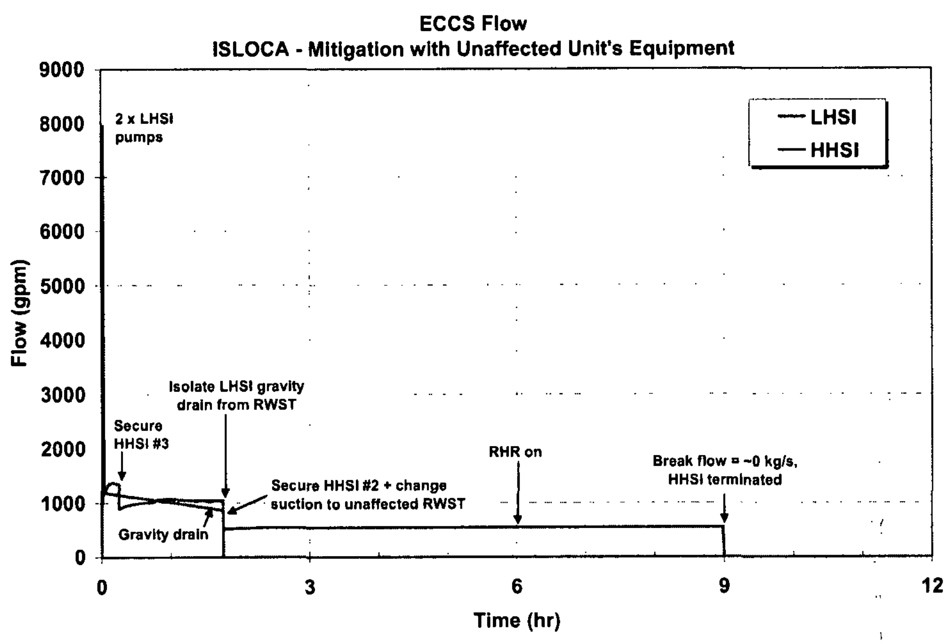

Figure 122 The mitigated ISLOCA ECCS flow history.

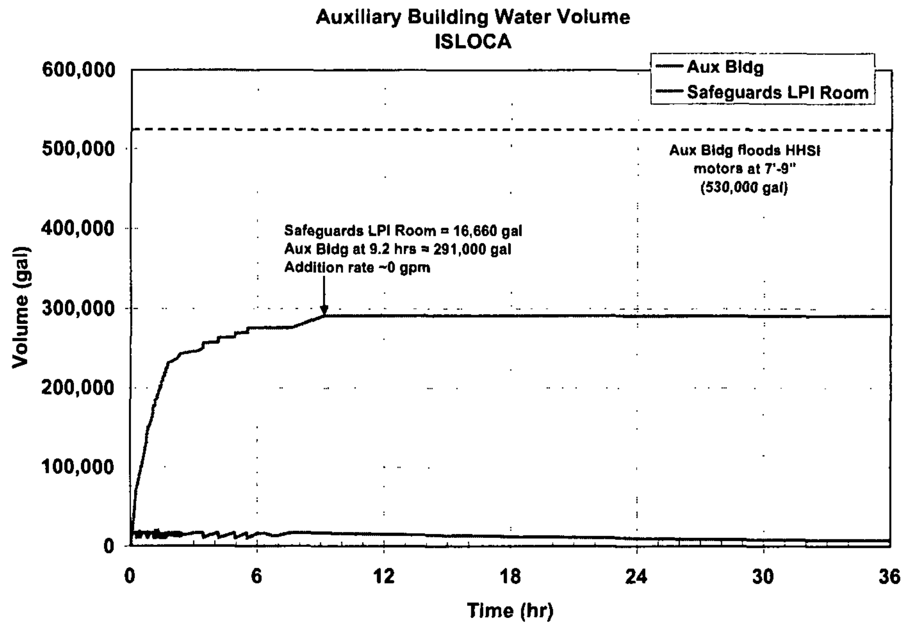

Figure 123 The mitigated ISLOCA auxiliary and safeguards building water volume history.

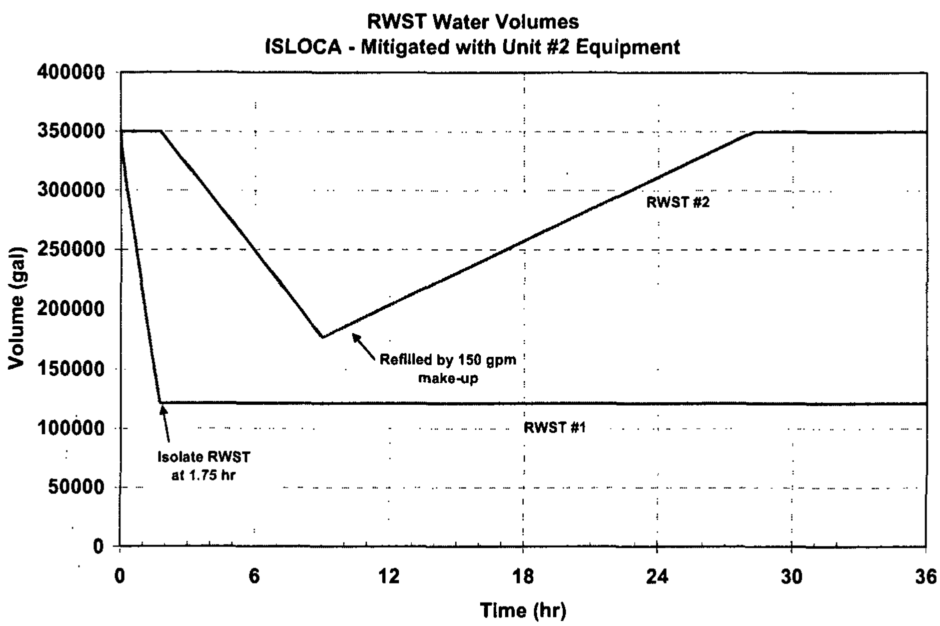

Figure 124 The mitigated ISLOCA affected and unaffected units RWST water volume history.

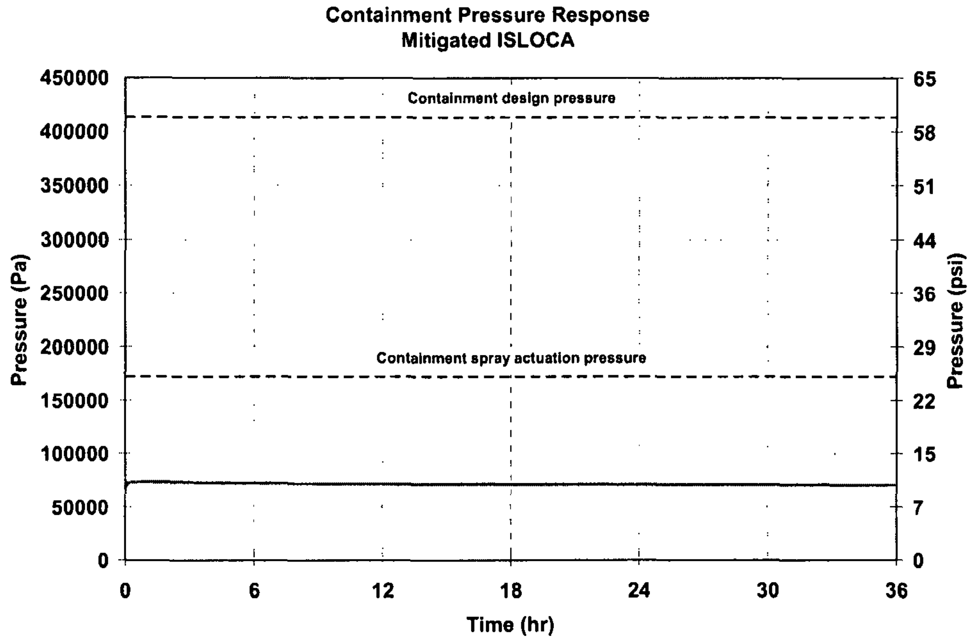

Figure 125 The mitigated ISLOCA containment pressure history.

5.5.2.2 Radionuclide Response

There was no fission product release for the mitigated ISLOCA scenario.

5.5.3 Unmitigated ISLOCA Accident Break Location Sensitivity Study

A sensitivity study was performed where the location of the piping break was varied relative to the water level in the safeguards building. The separate effects calculations used boundary conditions from the unmitigated ISLOCA (see Section 5.5.1). The separate effects model also included a more detailed representation of the LHSI line. Due to computational constraints associated with high flow through the explicit representation of the LHSI line[33], the scope of the separate effects model only included the LHSI piping, the safeguards building, and the auxiliary building and only simulated the in-vessel fission product release phase (i.e., 8 to 16 hours, see Table 15). Using identical initial and boundary conditions from the unmitigated ISLOCA, the break location was modeled (a) in the same location as Section 5.5.1 (i.e., ~1 m below the water level in the safeguards building) and (b) above the water level. However, there was no attempt to adjust the initial and boundary conditions for influences due to the break location. The calculations were started prior to the release of fission products but after the thermal-hydraulic

[33] In Sections 5.5.1 and 5.5.2, the LHSI line was represented as a flow path between the RCS cold leg piping to the safeguards building. The more detailed separate effects model representation used 4 control volumes and 5 flow paths with explicit representation of heat transfer to the LHSI piping and between the LHSI piping and water pool in the safeguards building. Due Courant limiting in the detailed nodalization, it was not practical to run the full RCS nodalization or the evolution of the sequence from the accident initiation through the fission product releases. Consequently, the scope of the model and transient calculation were limited to the LHSI line, safeguards building and auxiliary building during the primary in-vessel fission product release phase (i.e., 8 to 16 hours), respectively.

transient of the blowdown, the RCS cooldown, the HHSI injection phase, and the core uncovery phase. Table 15 summarizes the timings of the key events simulated in the separate effects calculations of the unmitigated ISLOCA. The description of the overall plant response for the boundary conditions used in the separate effects calculations is given in Section 5.5.1. The thermal hydraulic and radionuclide responses are described in Sections 5.5.3.1 and 5.5.3.2, respectively.

Table 15 Sequence of Events for the Unmitigated ISLOCA Sequence.

Event Description	Time (hh:mm)
Start of separate effects calculation	08:00
Start of fission product gap releases	08:04
Vessel lower head failure by creep rupture	15:02
Debris discharge to reactor cavity	15:02
End of the separate effects calculation	16:00

5.5.3.1 Thermal-Hydraulic Response

The thermal-hydraulic material and energy flows into the low-pressure injection line were extracted from the unmitigated ISLOCA calculation (see Section 5.5.1). Tabular inputs were developed that represented the time covering the in-vessel fission product release phase from 8 to 16 hours.[34] The water, steam, nitrogen, and hydrogen flows were added at the connection of the LHSI line to the RCS cold leg. The LHSI line penetration through the containment wall into the safeguards building is at the 16-ft elevation. The LHSI line immediately increases from 6" high-pressure piping to 10" low-pressure piping and drops down to the 12-ft elevation in the safeguards building (see Figure 105). The safeguards building has a deep pit where the LHSI pump is located. The motor for the LHSI pump is at the 12-ft elevation. Two pipe break locations were analyzed in the separate effects calculations: (1) at the same location as the unmitigated ISLOCA (~1 m below the water level), and (2) above the water level. The two cases (i.e., especially in the figures) will be referred to as the wet and the dry cases, respectively.

Both models had identical thermal-hydraulic boundary conditions, which resulted in identical flowrates into the LHSI line. However, after the hydraulic materials (i.e., steam, nitrogen, and hydrogen) flowed into the LHSI line, the break location affected the subsequent flow from the safeguards building to the auxiliary building, which had the release pathway to the environment. Figure 126 through Figure 128 show the integrated steam, hydrogen, and nitrogen flow from the safeguards building into the auxiliary building, respectively. All the non-condensable nitrogen and hydrogen that enters the LHSI line passes through the safeguards building to the auxiliary building. As shown on the figures, the separate effects model results were identical to the full model results. The source of the hydrogen is from the cladding oxidation during the fuel

[34] Preliminary attempts to run the entire calculation were unsuccessful due to the high computational time.

degradation. A small amount of residual nitrogen from the accumulator discharge also flowed into the LHSI during the initial rapid hydrogen generation phase (~9.5 to 10.5 hr, see Figure 127).

The steam flow (see Figure 126), however, can condense in the water pool in the safeguards building. Due to heat transfer to the water, more steam left the safeguards building than passed through the LHSI piping (i.e., boiling versus condensing). When the break location was 1 m below the surface of safeguards pool, the superheated gas exiting the broken LHSI line entered the pool and boiled away some of the water in the safeguards building pool. By 16 hours, the superheated gas exiting the wet break location had generated near 31% more steam than had entered the LHSI line. Similarly, the full model generated 38% more steam than enter the LHSI line, respectively. The break location in the dry case was approximately at the water surface. Therefore, only a portion of the steam leaving the break entered into the pool. Some additional steam was generated in the dry case (i.e., 7% more steam than enter the LHSI line) due to a small portion of the hot jet entering the pool and some mass transfer at the pool surface. As expected, the full model and the separate effects wet model generated approximately the same amount of additional steam. The differences between the two models were attributed to (a) additional heat transfer through the piping walls to the safeguards water in the separate effects models and (b) transient swelling effects beyond the detail of the supplied boundary conditions in separate effects model (e.g., a rapid steam generation event at 13.9 hours in the full model).

As will be shown in Section 5.5.3.2, the additional steam flow is important because it increases the motive force for fission product transport to the environment. While not characterized, the local steam concentration throughout the building affects the hydrogen burn potential and flame speed.

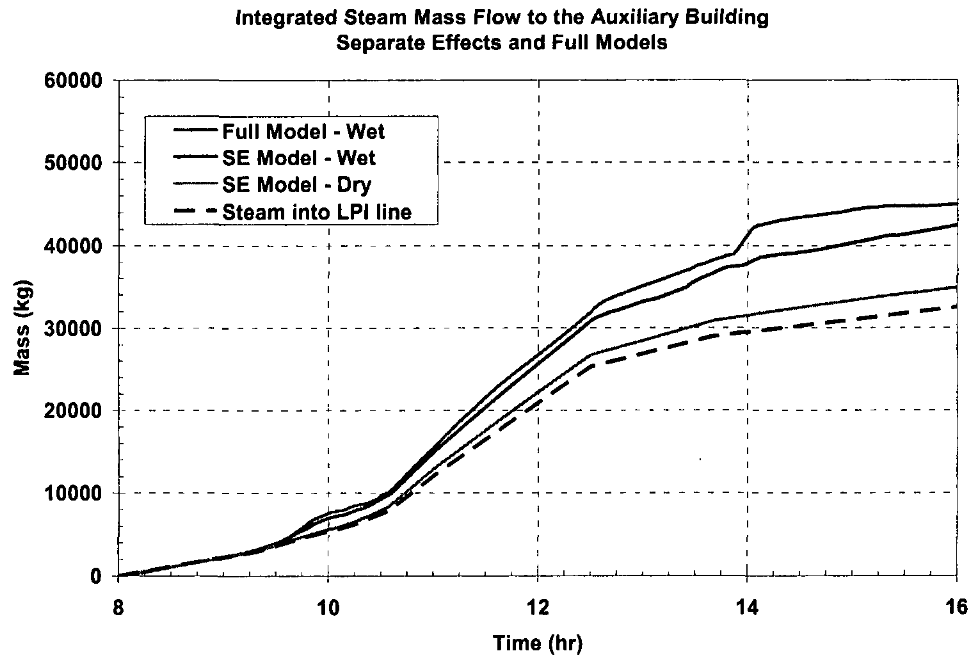

Figure 126 The integrated steam flow from the safeguards building to the auxiliary building in the full and separate effects unmitigated ISLOCA models.

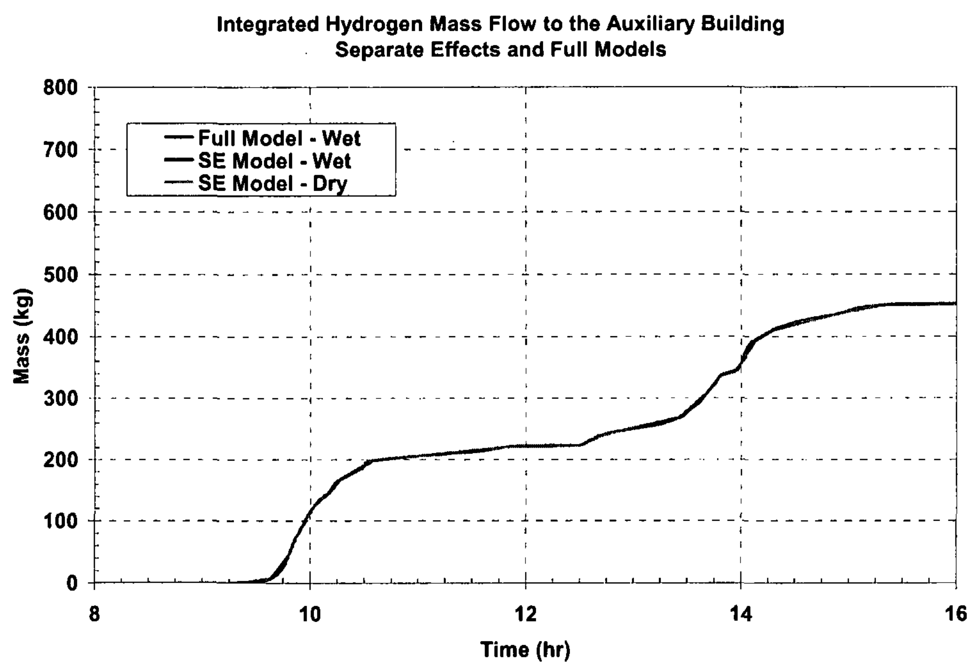

Figure 127 The integrated hydrogen flow from the safeguards building to the auxiliary building in the full and separate effects unmitigated ISLOCA models.

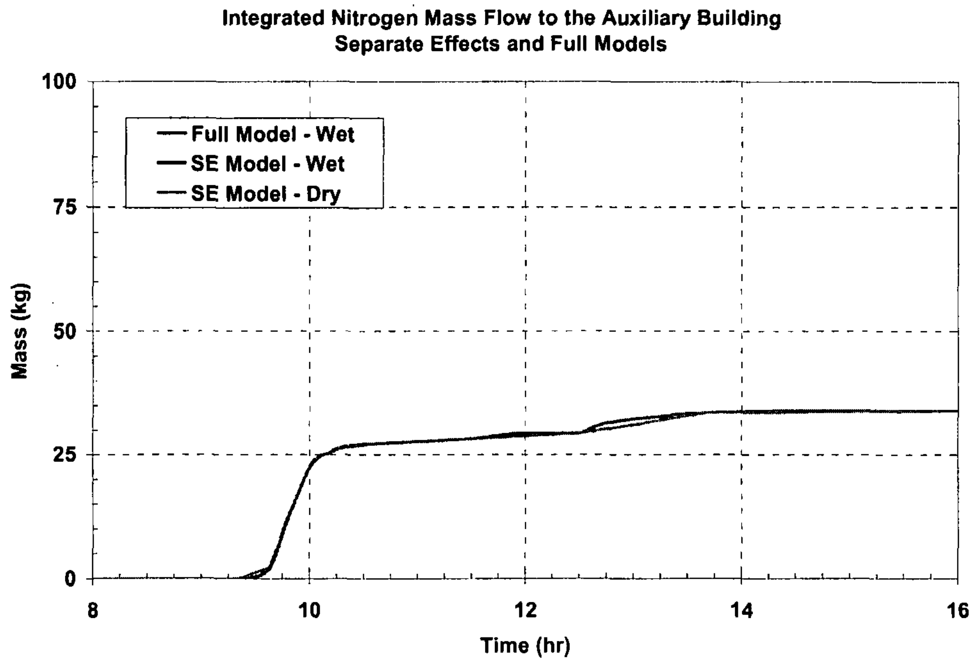

Figure 128 The integrated nitrogen flow from the safeguards building to the auxiliary building in the full and separate effects unmitigated ISLOCA models.

5.5.3.2 Radionuclide Response

Similar to the thermal-hydraulic sources, the radionuclides releases into the LHSI piping were extracted from the unmitigated ISLOCA calculation (see Section 5.5.1). Tabular inputs were developed that represented the time covering the in-vessel fission product release phase from 8 to 16 hours. The radionuclide aerosol and gas mass flowrates were released into the LHSI line with the thermal-hydraulic materials. The MELCOR decay heat package automatically adds the appropriate decay heat power based on the radionuclide mass in each control volume.

Although both separate effect models had identical thermal-hydraulic boundary conditions into the LHSI line, the location of the break affected the potential for water scrubbing. As described in Section 5.5.3.1, different amounts of steam were produced due to the break location, which affects the hygroscopic agglomeration of the airborne aerosols, the building leakage flow, and the hydrogen burn characteristics. Since the noble gases do not condense or deposit, their response gives a characterization of the building ventilation rate due to the ISLOCA.
Figure 129 compares the noble gas releases to the environment for the two separate effects models and the full model. The environmental release rates show the effects of the hydrogen burns as well as the non-condensable and steam sources into the auxiliary building. The overall responses are similar but timings and locations of the hydrogen burns affects the temporal release rates. However, the net noble gas releases just after the end of the in-vessel fission product release phase (i.e., vessel failure occurs at 15.02 hr and the separate effects calculations were terminated at 16 hours, see Table 15) were almost identical in all cases. As shown in Figure 130, the noble gas Auxilllary Building decontamination factor was 1.26 at 16 hours. A 1.26 noble gas

auxiliary building decontamination factor corresponds to a release of ~80% of the mass that entered the LHSI line to the environment.

Next, Figure 131 through Figure 134 show the iodine and cesium release to the environment and their auxiliary building decontamination factors. The iodine and cesium responses include the competing effects of aerosol deposition that was not present in the noble gas response. The response of the unmitigated ISLOCA using the full model (see Section 5.5.1) is included with the separate effect model responses. The separate effects models show smaller releases to the environment and higher retention in the auxiliary building than the full model. The detailed modeling of the LHSI line in the separate effects model allowed deposition along the piping wall.[35] There are also possibly some uncharacterized differences due to the approximation of the boundary conditions. Consequently, it is most meaningful to compare the relative differences between the separate effects wet case (i.e., same break location as the full model) and the separate effects dry case. However, the comparison of the wet separate effects model with the full model shows the expected trend of higher retention in the separate effects case (i.e., the full model did not include deposition in the LHSI line).

The overall building ventilation due to the thermal-hydraulic sources and hydrogen burns yields nearly identical noble gas environmental releases for all cases (see Figure 129). However, the iodine and cesium responses include capture in the safeguards pool as well as aerosol deposition mechanisms. Somewhat unexpectedly, the wet and dry separate effects models yielded approximately the same magnitude of iodine and cesium releases to the environment and therefore the same amount of retention in the safeguards and auxiliary buildings. Several factors contributed to the similar results. First, although the aerosol flow through the dry break location received essentially no scrubbing in the safeguards pool, the scrubbing in the pool for the wet case was marginal. The scrubbing in the pool for the wet case was not very efficient because (a) the pool was saturated and boiling (i.e., negligible condensation potential), (b) the bubble size from a 10" pipe break was large, and (c) there was a significant non-condensable fraction of hydrogen and nitrogen in the break discharge.

Second, as discussed in Section 5.5.3.1, the steam production was higher in the wet case. Consequently, the motive force for ventilation to the environment was greater than the dry case. Finally, the hydrogen burns had a first order effect on the release magnitude. All cases had the similar number, timings, and magnitudes of hydrogen burns that caused rapid discharges of fission products from the auxiliary building. Consequently, the residence times for gravitational settling were similar for all cases. The resultant iodine and cesium decontaminations building were about 9-11 for the wet and dry cases after 10 hours. The decontamination factors include capture in the safeguards building pool (i.e., if applicable) as well as the retention in the safeguards and auxiliary buildings.

In summary, the most important first order effect on aerosol retention was the hydrogen burns in the auxiliary building (i.e., see the rapid expulsions of fission products with each burn). Both cases had a similar number of hydrogen burns and therefore similar building retention. The

[35] MELCOR does not model inertial deposition, which was expected to be a significant capture mechanism in the LHSI line and valve components. However, other forms of deposition including gravitational settling, thermophoresis, and diffusiophoresis deposition processes are calculated and were important.

secondary effects were (a) a wet versus a dry break location (i.e., the shallow "wet" case benefitted from some scrubbing in the safeguards pool), (b) the magnitude of the steam production in the safeguards pool (i.e., this increased the ventilation rate to the environment for the "wet" case), and (c) the burn timing and location (i.e., the location where burn occurred and propagation characteristics were almost stochastic and led to minor variations in the temporal release rates). The results from the separate effects model show there were only minor differences between the full, wet, and dry cases (i.e., the same case modeled by the full model in Section 5.5.1). Hence, the full model results are representative of shallow "wet" and "dry" break locations.

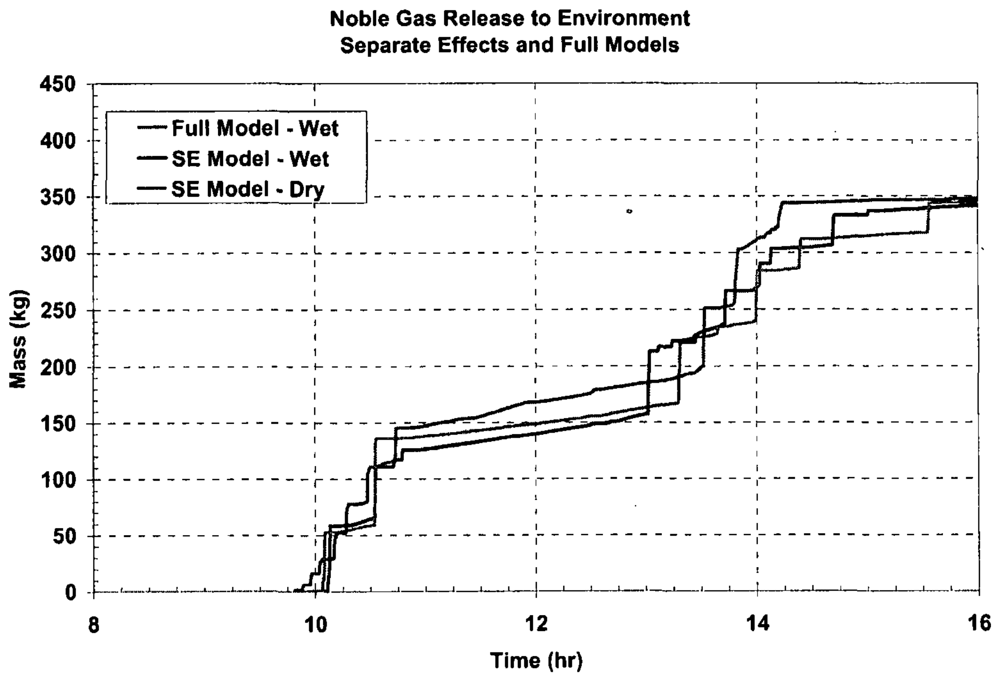

Figure 129 The noble gas release fraction to the environment in the full and separate effects unmitigated ISLOCA models.

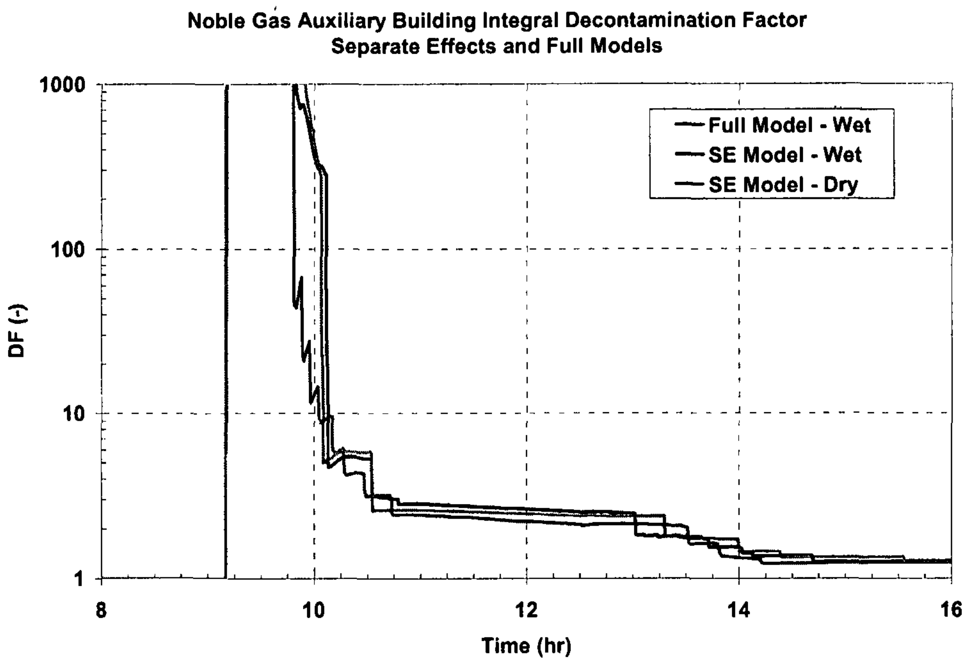

Figure 130 The auxiliary building decontamination factor for the noble gases in the full and separate effects unmitigated ISLOCA models.

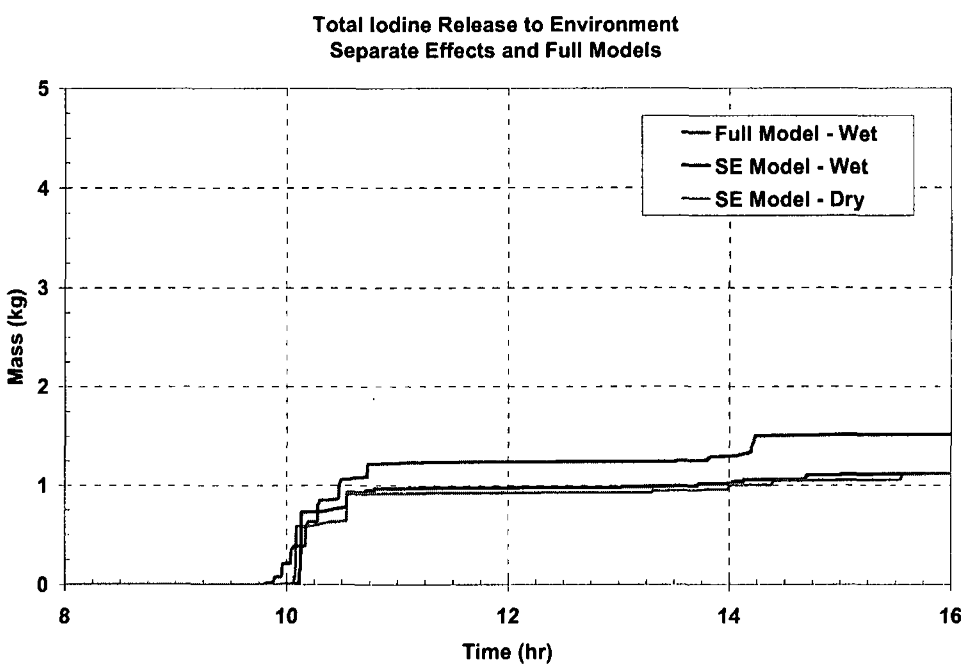

Figure 131 The iodine release fraction to the environment in the full and separate effects unmitigated ISLOCA models.

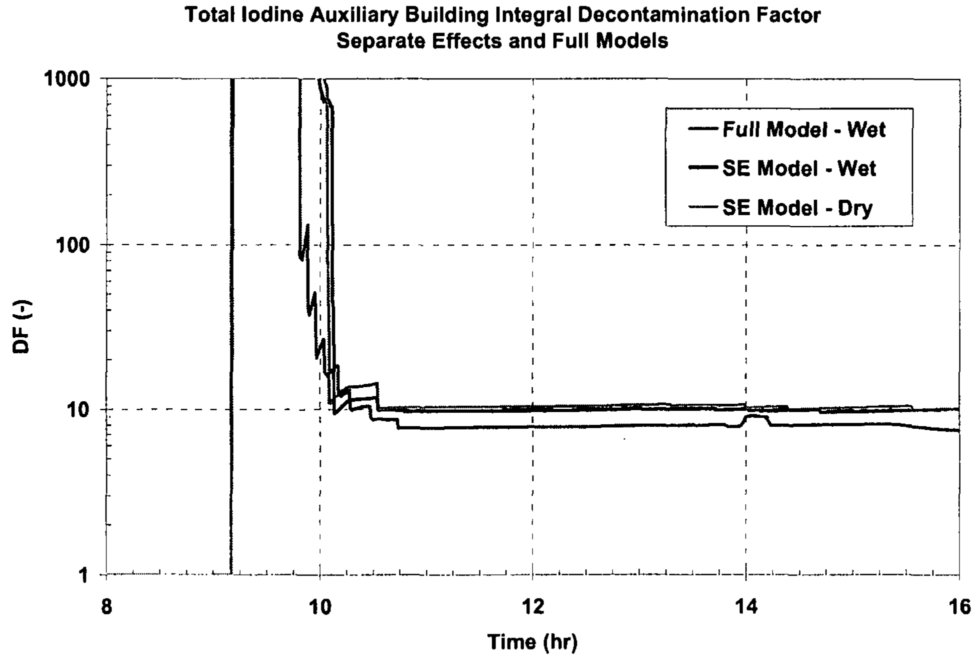

Figure 132 The auxiliary building decontamination factor for the iodine in the full and separate effects unmitigated ISLOCA models.

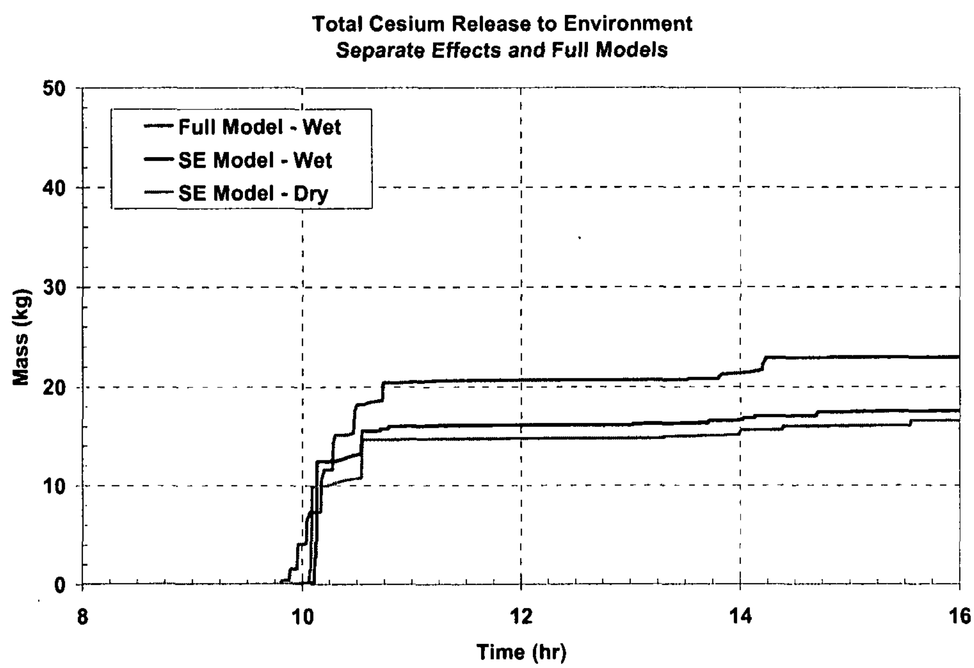

Figure 133 The cesium release fraction to the environment in the full and separate effects unmitigated ISLOCA models.

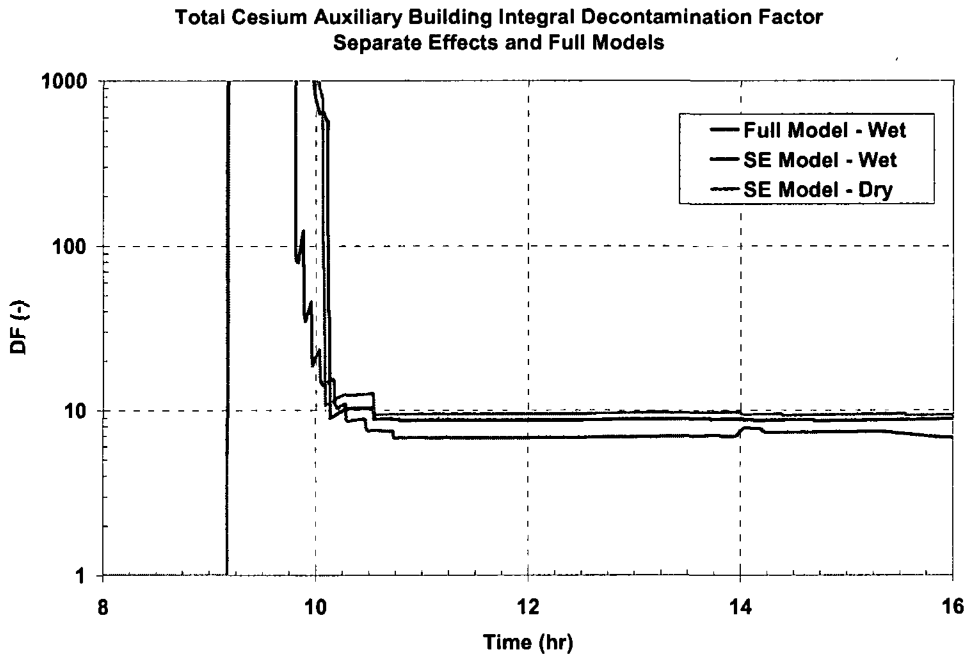

Figure 134 The auxiliary building decontamination factor for the cesium in the full and separate effects unmitigated ISLOCA models.

5.5.4 Uncertainties in the Aerosol Deposition Rate in the Interfacing Piping

During the peer review of the unmitigated interfacing loss-of-coolant accident, there were questions about the amount of deposition in the interfacing LOCA piping. In particular, turbulent deposition and impaction, which are not modeled in MELCOR, were identified as a mechanism that could increase retention in the piping. In addition, resuspension of deposited material was identified as a mechanism that could decrease retention in the piping.

Relative to the first uncertainty, the conditions in the piping were evaluated. Due to the nature of the magnitude of the critical flow through the piping during the volatile radionuclide release, the aerosol residence time in the piping was only 0.1-0.3 s. Powers has recently done a review of aerosol deposition models in leakage pathways [31]. Based on the review of turbulent deposition rate data and aerosol residence time, a best-estimate decontamination factor of 1.1 was determined, which is a negligible amount. There was insufficient geometric data to estimate the decontamination due to impaction, and there is significant uncertainty in resuspension under those conditions. Consequently, it was concluded to conservatively neglect turbulent deposition, impaction, and resuspension in the piping (i.e., the approach used to develop the unmitigated source term in Section 5.5.1 and 5.5.3).

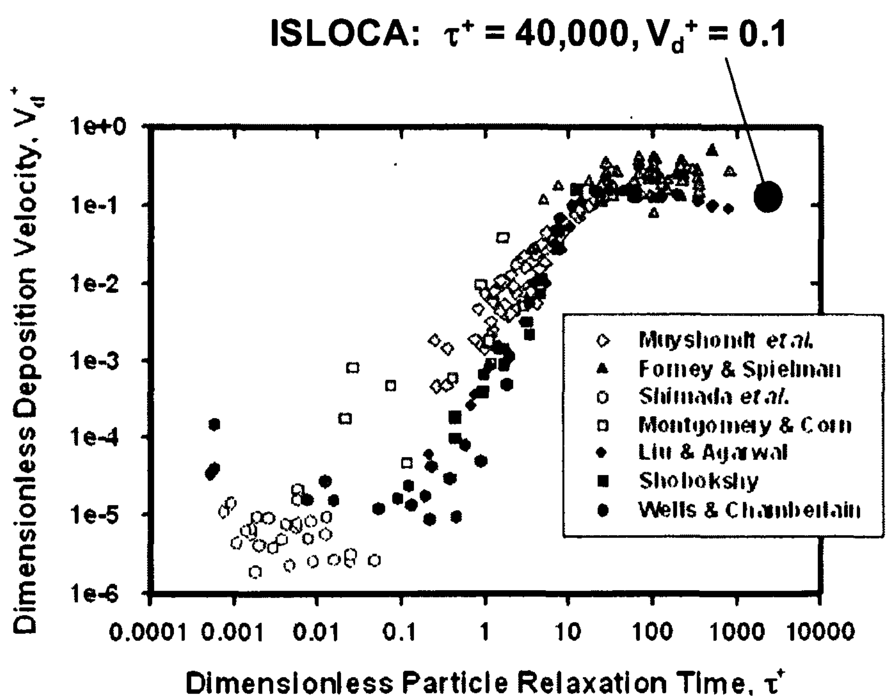

Figure 135 The dimensionless turbulent deposition velocity versus the dimensionless particle relaxation time from Powers with conditions from the ISLOCA.

5.6 Other Sensitivity Studies

During the peer review of the MELCOR calculations, several other more generic issues were identified relative to the ones already discussed in Sections 5.2.3, and 5.5.4. They include uncertainties in the chemical form of iodine, iodine spiking, uncertainties of the impact of air ingression into the vessel, and uncertainties in the aerosol deposition rate in containment.

5.6.1 Chemical Form of Iodine

The chemical forms and quantities of gaseous iodine are an active research topic as new information is still being evaluated in existing and planned tests. The SOARCA calculations did not include gaseous iodine. All iodine was assumed to be combined with cesium to form cesium iodine and remain in that chemical form. New data from Phebus suggests some iodine is released in elemental form yet undergoes complex chemical reactions in the containment to form organic compounds unless liberated by the chemosorption process. MELCOR does not include a model for surface chemistry with paint. Furthermore, MELCOR's ex-vessel iodine pool model is very slow running and not fully validated. Consequently, all the iodine was modeled as a cesium iodine compound and the pool iodine model was not used. It should be noted that the uncertainty study will impact of different fixed amounts of gaseous iodine (i.e., elemental and organic forms).

Relative to the current results, it is worth making some simple evaluations using recent interpretations of Phebus data [33]. Phebus Test FTP-1 shows that the concentration of iodine reaches a steady state in the containment that is independent of the pool pH and condensing or evaporating conditions. In particular, the protypical Phebus configuration shows a steady state exchange between the painted surfaces where the iodine is absorbed and released to maintain a steady concentration.

Two evaluations were performed to assess the impact of gaseous iodine on the source term using Phebus FTP1 data. In the first evaluation using the short-term station blackout, a range of gaseous iodine concentrations were considered with the calculated containment leakage rate to estimate the additional iodine source term. The measured Phebus gaseous iodine containment concentrations are shown in Figure 136 with the conversion to an iodine release fractions based on the containment release rate. The calculated iodine release magnitude was 0.65% in the unmitigated short-term station blackout at 48 hours. Assuming gasous iodine concentrations of 0.05%, 0.10%, and 0.15%, the additional source term would be less than 0.10%. Given the small absolute and relative magnitude of the iodine release, the impact of a 0.10% additional gaseous iodine release was judged as not significant.

The second evaluation examined the additional source term to the environment through the failed steam generator tube, which occurred earlier in the accident progression. The measured Phebus gaseous iodine containment concentrations are shown in Figure 137. The higher short-term values were used to estimate the additional gaseous release to the environment. Using the noble gas leakage rate into the environment through the steam generator secondary and the early, higher concentrations from Phebus containment, the gaseous iodine leak rate was calculated. The calculated iodine release rate was 0.6% in the first 24 hours when the dominant releases through the TI-SGTR occurred. Assuming gasous iodine concentrations of 0.10%, 0.15%, and 0.20%, the additional source term would be <<0.10%, respectively. Given the small absolute and relative magnitude of the iodine release, the impact of gaseous iodine on the source term was also judged small. Following vessel failure, any remaining gaseous iodine in the reactor vessel was discharged into the containment. All further releases through the failed TI-SGTR were diluted by the volume of the containment.

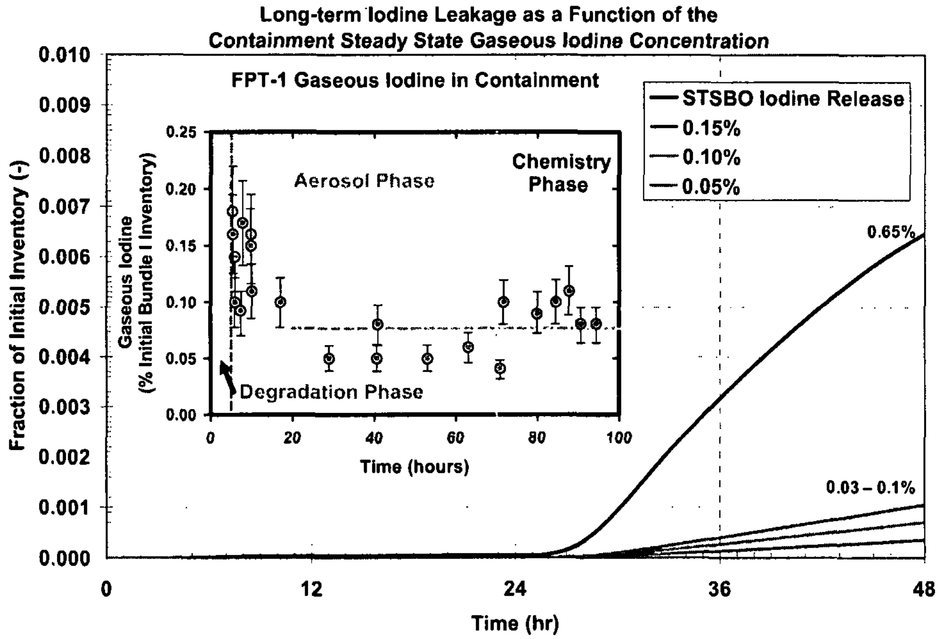

Figure 136 The additional gaseous iodine soure term using Phebus data is compared to the iodine source term for the unmitigated short-term station blackout.

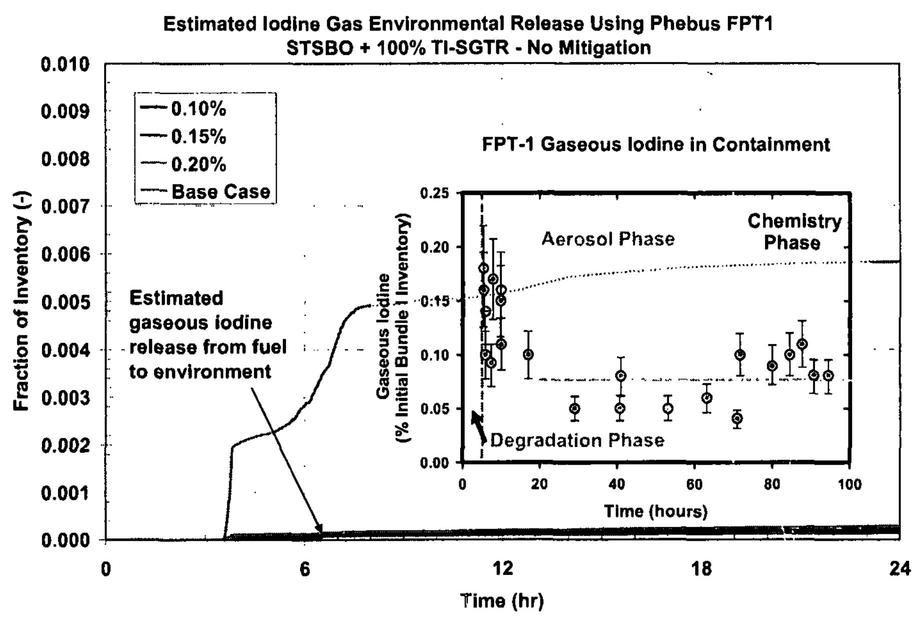

Figure 137 The additional gaseous iodine soure term using Phebus data is compared to the iodine source term for the unmitigated short-term station blackout with a thermally-induced steam generator tube rupture.

5.6.2 Additional Source Term from Iodine Spiking

Iodine spiking was identified by one of the review panelist as a possible alternate source of iodine to the environment for an early release in the spontaneous SGTR. Using the water leakage from the unmitigated SGTR, the maximum recorded iodine spike (18 μCi/g), and the recommended partition factor from Regulation Guide 1.83, the fractional iodine release was 10^{-6} [36]. While an iodine spike is an important operational concern, it is not significant relative to the magnitude of release fractions from the other considered severe accidents (see Table 16).

Table 16 Comparison of Iodine Spike Source Term to Iodine Source Terms from the Other Unmitigated Accidents

Unmitigated Scenario	Core fraction of iodine released to environment
Long-term SBO	0.003
Short-term SBO	0.006
Short-term SBO with thermally induced SGTR	0.009
ISLOCA	0.095
Spontaneous SGTR (Iodine Spike)	10^{-6}

5.6.3 Air Ingression into the Vessel

Air ingression into the vessel was identified by one of the review panel members as an important concern for enhanced air oxidation of metals and enhanced ruthenium releases. The failure mode of the reactor vessel in the unmitigated transients was due to gross creep failure of the vessel. Following failure of reactor vessel, the hot contents in the lower plenum poured into the reactor cavity. In the progression of events calculated in the unmitigated scenarios, all injection had terminated and the entire core had degraded and collapsed prior to vessel failure. Consequently, all the debris relocated to the lower plenum prior to any significant air-ingression (e.g., see Figure 138).

It is worth noting that there is an openinginto the reactor system prior to vessel failure while the fuel is degrading due to creep rupture of the hot leg or the pipe break in the ISLOCA scenario. Since there was a large decay heat source in the reactor vessel, all cases showed a slight pressurization of the reactor coolant system relative to the containment (or auxiliary building) that maintained a steady flow outward of the pipe breaks. Consequently, inward flow of air during this time period was not expected.

Finally, MELCOR includes models for both steam and air oxidation of metals in the core package. However, there are no models for automatically changing the ruthenium release model in an air oxidizing environment. Consequently, each calculation must be reviewed for the presence of high air concentration conditions, which was done.

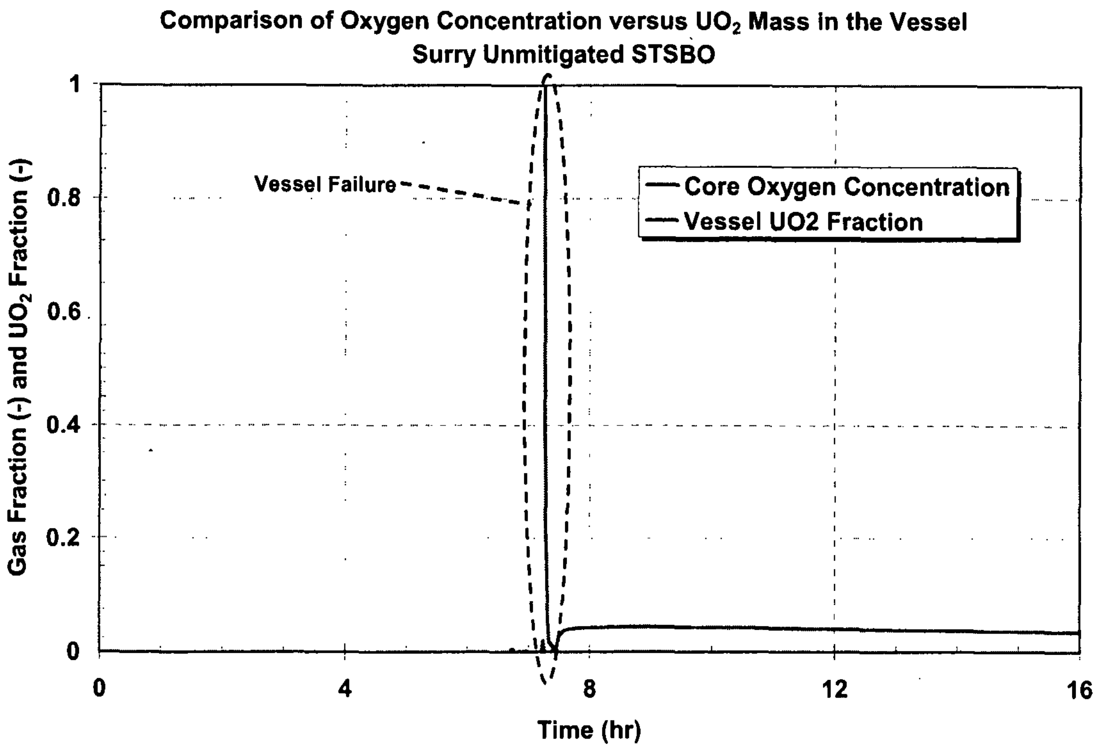

Figure 138 Comparison of the oxygen concentration and UO_2 mass in the vessel during the unmitigated STSBO.

5.6.4 Aerosol Settling Rate in the Containment

A review panel member thought the aerosol settling rate in the containment looked high (e.g., Figure 43) for the STSBO scenario without B.5.b mitigation. To address this issue, two time phases were investigated. The first time phase occurred after the hot leg failure. Following hot leg failure, codispersing and flashing water from accumulator injection with the aerosols immediately led to a very high mass median diameter of the airborne aerosols (>10 μm), which caused them to settle very quickly. The codispersed water and steam acted as a scrubbing mechanism by condensing steam on the airborne aerosols and creating fog droplets for enhanced aerosol agglomeration. Within one hour after the hot leg failure, over 50% of the airborne aerosols had settled.

The second phase occurred with the releases following prior to and at vessel failure. To analyze the settling rate, the mass of airborne aerosols in the STSBO at vessel failure were normalized to one (see Figure 139). Following vessel failure, the airborne aerosol concentration decreased steadily. The airborne decay constant, λ, was calculated and compared to Phebus FTP0 data [34]. The calculated decay constant is a strong function of the mass-median diameter of the airborne aerosols. However, the results in Figure 139 show the settling rate was comparable to the test data and actually slightly slower.

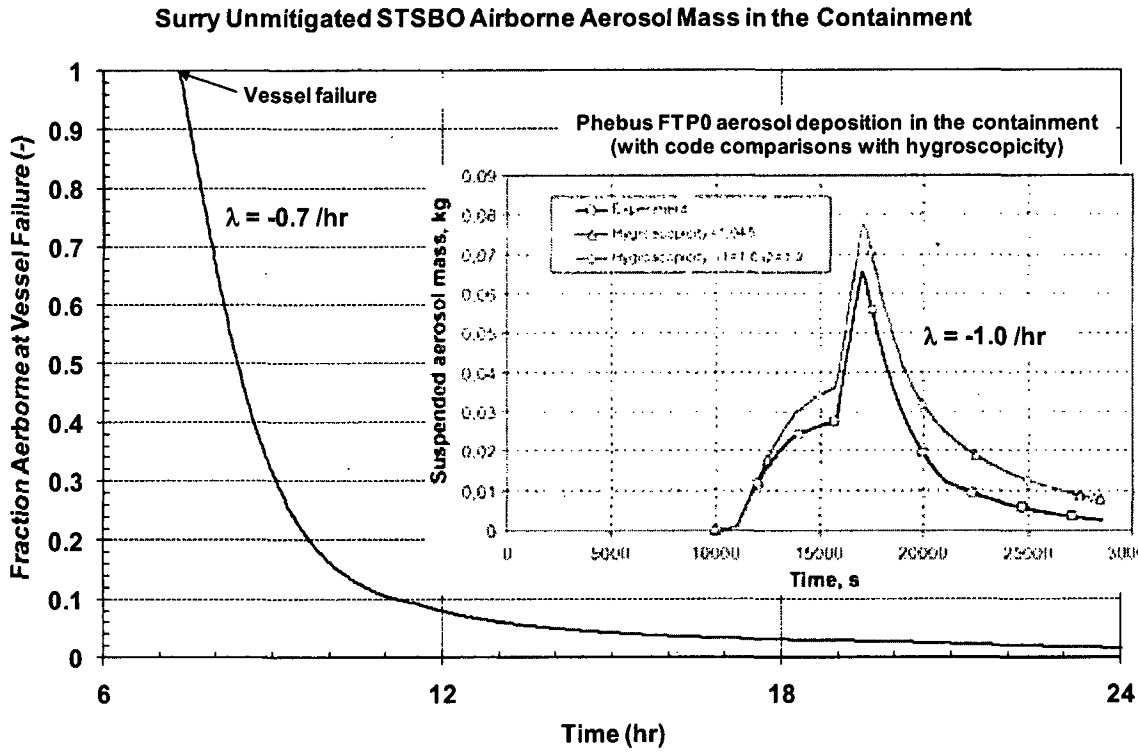

Figure 139 Unmitigated STSBO airborne aerosol mass in the containment following vessel failure.

6. EMERGENCY RESPONSE

Advancements in consequence modeling provide an opportunity to integrate realism in the implementation of protective action decisions applied for discrete population segments. To best utilize these advancements, detailed information was obtained from local sources and offsite response organizations (OROs). Through a user interface added to the consequence model, this detailed information was input to account for differences in the implementation of protective actions by various population segments. These advancements are significant because they now allow the modeling of response activities, timing of decisions, and implementation of protective actions across different population segments.

Emergency response programs for nuclear power plants (NPPs) are designed to protect public health and safety in the event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as defense in depth to respond in the unlikely event of an accident. To support a state of the art approach and integrate realism in the analyses, the modeling of the emergency response was based on the site-specific emergency planning documentation and on research of public response to non-nuclear emergencies. The information developed in this Emergency Response section was used to support the MACCS2 consequence analyses for the accident scenarios. For each accident scenario, evacuation of the plume exposure pathway emergency planning zone (EPZ) was assessed along with consideration of a shadow evacuation to a distance of 20 miles from the plant. Also, for each scenario, members of the public are relocated from any area where doses exceed established criteria.

Sensitivity analyses were completed for one accident scenario to assess evacuation distances of 16 miles and 20 miles from the plant. A sensitivity analysis was performed to assess the effect of a delay in the implementation of protective actions, as suggested by the peer review committee. An analysis was also conducted that included consideration of the effects on infrastructure, emergency response, and response of the public due to a seismic event. Figure 140 identifies the location of the Surry plant and radial distances of 10 and 20 miles from the plant.

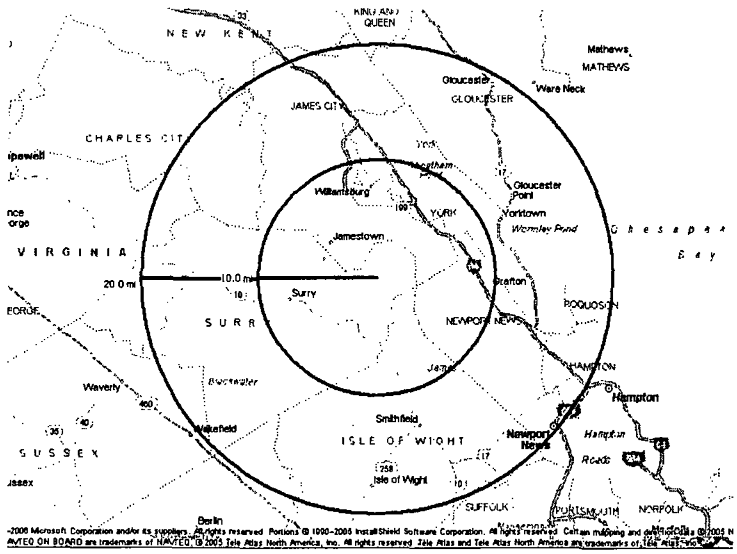

Figure 140 Surry 10 and 20 Mile Areas.

As required by 10 CFR 50, OROs develop emergency response plans for implementation in the event of an NPP accident. These plans are regularly drilled and inspected biennially through a demonstration exercise performed in conjunction with the licensee. In biennial exercises, ORO personnel demonstrate timely decision making and the ability to implement public protective actions. Emergency plans escalate response activities in accordance with a classification scheme based on emergency action levels (EALs). Preplanned actions are implemented at each classification level including Unusual Event, Alert, Site Area Emergency (SAE), and General Emergency (GE). Public protective actions are required at the GE level, but ORO plans commonly include precautionary protective actions at the SAE level and sometimes at an Alert.

The plume exposure pathway EPZ is identified in NUREG-0654 / FEMA – REP-1, Rev.1, [19] as the area around an NPP of about 10 miles. Within the EPZ, detailed emergency plans are in place to reduce the risk of public health consequences in the unlikely event of an accident . Emergency planning within the EPZ provides a substantial basis for expansion of response efforts should it be necessary. ORO personnel have repeatedly demonstrated the ability to implement protective actions within the EPZ during inspected biennial exercises. Modeling of expected protective action response is described in this section. Analyses were conducted for accident scenarios identified in Table 6-1.

Table 6-1 Scenarios Assessed for Emergency Response

Report Section#	Scenario
6.3.1	STSBO Unmitigated
6.3.2	Unmitigated STSBO with TI-SGTR
6.3.3	Mitigated STSBO with TI-SGTR

6.3.4	Unmitigated LTSBO
6.3.5	Unmitigated ISLOCA
6.4.1	Sensitivity 1 ISLOCA and evacuation to 16 miles
6.4.2	Sensitivity 2 ISLOCA and evacuation to 20 miles
6.4.3	Sensitivity 3 ISLOCA with a Delay in Implementation of Protective Actions
6.5.6	Seismic Analysis - STSBO with TI-SGTR

6.1 Population Attributes

SOARCA modeled the population near the Surry plant as several cohorts. A cohort is any population group that mobilizes or moves differently from other population groups. Modeling includes member of the public who evacuate early, evacuate late, and those who refuse to evacuate. The consequence model does not constrain the number of cohorts but there is no benefit to defining an excessive number of cohorts with little difference in characteristics. The following cohorts were established for SOARCA analyses:

Cohort 1: 0 to 10 Public. This cohort includes the public residing within the EPZ.

Cohort 2: 10 to 20 Shadow. This cohort includes the shadow evacuation from the 10 to 20 mile area beyond the EPZ. A shadow evacuation occurs when members of the public evacuate from areas that are not under official evacuation orders and generally begin when a large scale evacuation is ordered [45]. The size of a shadow evacuation varies depending upon many factors and for technological hazards is typically observed in areas immediately adjacent to evacuation areas. For SOARCA, a distance of 10 miles beyond the EPZ was selected because residents of this area would likely observe evacuees from the EPZ passing through. A shadow evacuation of 20 percent of the public was assumed based on the quantitative assessment of shadow evacuations completed by the NRC [49].

Cohort 3: 0 to 10 Schools. This cohort includes elementary, middle and high school populations within the EPZ. Schools receive early and direct warning from OROs and have response plans in place to support busing of students out of the EPZ.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities population includes residents of hospitals, nursing homes, assisted living communities and prisons. Special facility residents are assumed to reside in robust facilities such as hospitals, nursing homes, or similar structures which provide additional shielding. Shielding factors for this population group consider this fact. In an emergency, Special Facilities would be evacuated individually over a period of time based upon available transportation and the number of return trips needed.

Cohort 5: 0 to 10 Tail. The 0 to 10 Tail is defined as the last 10 percent of the public to evacuate from the 10 mile EPZ. The approach to modeling the Tail is an analysis simplification to support inclusion of this population group. In reality, this population group is performing multiple activities prior to the evacuation of this cohort. The Tail takes longer to evacuate for many valid reasons such as the need to return home from work to evacuate with the family, pick up children, shut down farming or manufacturing operations or performing other actions prior to evacuating as well as those who may miss the initial notification.

Cohort 6: Non-evacuating public. This cohort represents a portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population. Research of large scale evacuations has shown that a small percent of the public refuses to evacuate [45] and this cohort accounts for this potential group. It is important to note that emergency planning is in place to support evacuation of 100 percent of the public.

6.1.1 Population Distribution

The Surry 2001 evacuation time estimate (ETE) was used to develop the population estimates for the cohorts within the EPZ. The populations provided in the Surry 2001 ETE [20] present a detailed estimate of the population of the 0 to 10 mile region.

A separate estimate was developed for the permanent residents and special facilities population beyond the EPZ to support development of the shadow population cohort and sensitivity analyses. SECPOP2000 was used to estimate the population within 20 miles of the plant. The population was projected to 2005, which was the year the SOARCA project was initiated, using a multiplier of 1.0533 obtained from Census Bureau information. The population of the 10 to 20 mile area outside the EPZ was then calculated as the difference between the total estimated population within 20 miles and the EPZ population identified in the Surry 2001 ETE. School children are not a separate cohort in the 10 to 20 mile area because it is assumed there is ample time for schools to close and children to go home and evacuate with families; therefore they are included in the 10 to 20 public. Special facilities data for hospitals, nursing homes, and detention facilities in the 10 to 20 mile area was developed by researching available public information.

To establish the population distributions, the Shadow population was assessed first and defined as 20 percent of the total population within 10 to 20 miles from the plant. This value for the Shadow was then combined with non-evacuee, special facilities, and schools and then subtracted from the remaining total to establish the general public. Ten percent of the general public defines the evacuation tail, and the remainder was used as the total for the general public. The non-evacuating population is 0.5 percent of the total population in each region. Cohort populations are provided in Table 6-2.

Table 6-2 Surry Cohort Population Values.

Cohort	Description	Population
1	0 to 10 Public	101,125
2	10 to 20 Shadow	59,645
3	0 to 10 Schools	26,513
4	0 to 10 Special Facilities	969
5	0 to 10 Tail	8,181

6	0 to 10 Non-evacuating	687

6.1.2 Evacuation Time Estimates

As provided in 10 CFR 50.47 Appendix E, each licensee is required to estimate the time to evacuate the EPZ. Appendix 4 of [19] provides information on the requirements of ETEs, and [21] provides detailed guidance on development of ETEs. A typical ETE includes many scenarios to help identify the combination of events for normal and off-normal conditions and provides emergency planners with estimates of the time to evacuate the EPZ under varying conditions [21]. The ETE study provides information regarding population characteristics, mobilization of public, special facilities, transportation infrastructure, and other information used to estimate the time to evacuate the EPZ.

The SOARCA project used a normal weather weekday scenario that includes schools in session. This scenario was selected because it presents several challenges to timely protective action implementation including evacuating while residents are at work and mobilizing buses to evacuate children at school. The Surry 2001 ETE report [20] provides the following regarding evacuation of the general public:

East of the James River- The populates area thorough which Interstate 64 traverses.

- 100 percent evacuation: 13 hours; and
- 90 percent evacuation: 10 hours and 50 minutes (rounded to 11 hours).

West of the James River- The rural, low population area.

- 100 percent ETE: 3 hours 10 minutes; and
- 90 percent ETE: (not provided).

These values were used to derive the speeds for the cohorts used in the analysis. The Surry study [20] describes the ETE scenario used as a 'worst case' because it includes the high number of transients in the area and schools in session. This scenario can be considered the bounding ETE case for the analysis and alternative seasonal evaluations and time of day are not necessary.

For the evacuation scenarios, a speed is input into the consequence model. The evacuation speed is derived from the ETE and is primarily influenced by population density and roadway capacity. When using ETE information, it is important to understand the components of the time estimate. The ETE includes mobilization activities the public undertakes upon receiving the initial notification of the incident [24, 27]. These actions include receiving the warning, verifying information, gathering children, pets, belongings, etc., packing, securing the home, and other evacuation preparations. Thus, a 13 hour ETE does not indicate that all of the vehicles are en route for 13 hours, but is the end of a 13 hour period in which the public mobilizes and evacuates the area. MACCS2 cohort is modeled to begin moving at a specific time after notification. This requires the speed be developed as a single linear value based on distance divided by time (the ETE). This distance over ETE ratio provides a slower average speed than would be expected in an evacuation and adds some conservatism to the analysis. Expert judgment was used to balance the number of cohorts considering data uncertainties and model run time.

Evacuations can therefore be represented as a curve that is relatively steep at the beginning and tends to flatten as the last members of the public exit the area. Through review of more than 20 existing ETE studies, the point at which the curve tends to flatten occurs where approximately 90 percent of the population has evacuated. This is consistent with research that has shown that a small portion of the population that takes a longer time to evacuate than the rest of the general public and is the last to leave the evacuation area [19]. This last 10 percent of the population is identified as the evacuation tail. To best achieve the goal of protecting the public health and safety, it is not appropriate to utilize the total ETE in the analysis. For the analyses in this study, the 90 percent ETE value was used to derive evacuation speeds.

6.2 WinMACCS

WinMACCS is a user interface for the MACCS2 code and was used to generate input for MACCS2 model runs. WinMACCS has the ability to integrate the information descrived above into the consequence analysis. The entire evacuation area was mapped onto a radial sector grid network around the plant. The roadway network was reviewed against site-specific evacuation plans to determine likely evacuation direction in each grid element. The results of the ETE were reviewed to determine localized areas of congestion as well as areas where no congestion occurs. Using this information, speed adjustment factors were applied at the grid element level to speed up vehicles in the rural uncongested areas and to slow down vehicles in more urban settings.

6.2.1 Hotspot and Normal Relocation and Habitability

OROs generally do not develop detailed protective action plans for areas beyond the EPZ. However, in the unlikely case of a severe accident and radiological release, the population outside the EPZ could be relocated if their potential dose exceeds protective action criteria. OROs would base this determination on dose projections using state, utility, and Federal agency computer models as well as measurements taken in the field. Hotspot relocation and normal relocation models are included in the MACCS2 code to reflect this contingency. Total dose commitment pathways for the relocation models are cloudshine, groundshine, direct inhalation, and resuspension inhalation. Relocated individuals are removed from the calculation for the remainder of the emergency phase and receive no additional dose during that phase. The dose criteria are applied after plume arrival at the affected area.

Hotspot relocation of individuals beyond ten miles occurs 24 hours after plume arrival if the total lifetime dose commitment for the weeklong emergency phase exceeds 0.05 Sv (5 rem). Normal relocation of individuals occurs 36 hours after plume arrival if the total lifetime dose commitment exceeds 0.01 Sv (1 rem). The relocation times of 24 hours for hotspot and 36 hours for normal relocation were established based on review of the emergency response timelines which suggest that OROs would not likely be available earlier to assist with relocation due to higher priority tasks in the evacuation area. The hotspot value used in NUREG-1150 [1] was 0.5 Sv (50 rem) and the relocation value was 0.25 Sv (25 rem). The long term habitability criteria used in NUREG-1150 was 0.04 Sv (4 rem) over a 5 year period. The NUREG-1150 long term habitability criterion is the same as the site specific value used for the Surry analysis. It should be noted that the non-evacuating cohort is still subject to the Hotspot and Normal Relocation criterion. It is assumed these individuals will evacuate when they understand a release has in fact occurred and they are informed they are located in high dose areas.

6.2.2 Shielding Factors

Shielding factors vary by geographical region across the United States, and those used in the Surry analysis are shown in Table 6-3. The factors represent the fraction of dose that a person would be exposed to when performing normal activities, evacuating, or staying in a shelter in comparison to a person outside with full exposure and are applied to all cohorts except the Special Facilities. Special Facilities are typically larger and more robust structures than housing stock and therefore have better shielding factors as identified in the table. Special faciliteies have the same factor for normal and shelter indicationg these individuals are all indoors.

Table 17 Surry Shielding Factors.

	Ground Shine			Cloud Shine			Inhalation/Skin		
	Normal	**Evac.**	**Shelter**	**Normal**	**Evac.**	**Shelter**	**Normal**	**Evac.**	**Shelter**
Cohorts	0.26	0.50	0.20	0.68	1.00	0.60	0.46	0.98	0.33
Special Facilities	0.05	0.50	0.05	0.31	1.00	0.31	0.33	0.98	0.33

The shielding factors provided in Table 17 were obtained from a variety of sources. Where appropriate, site specific values for sheltering were obtained from NUREG-1150 [1]. An updated inhalation/skin evacuation shielding factor was obtained from NUREG/CR-6953, Vol. 1, [46]. The normal activity shielding factors have been adjusted to account for the understanding that people do not spend a great deal of time outdoors. The normal activity values are all weighted averages of indoor and outdoor values based on being indoors 81 percent of the time and outdoors 19 percent of the time [46]. Indoor values are assumed to be the same as sheltering.

6.2.3 Potassium Iodide

The State of Virginia implements a potassium iodide (KI) program. The Virginia Department of Health provides potassium iodide to people who live, work or visit within 10 miles of the Surry NPP. Potassium iodide also is available to the public for purchase without a prescription at pharmacies and from manufacturers.

The purpose of the KI is to saturate the thyroid gland with stable iodine so that further uptake of radioactive iodine by the thyroid is diminished. If taken at the right time and in the appropriate dosage, KI can nearly eliminate doses to the thyroid gland from inhaled radioiodine. Factors that contribute to effectiveness of KI include the availability of KI (i.e., whether residents can find their KI), the timing of ingestion, and the degree of pre-existing stable iodine saturation of the thyroid gland. It is considered that some residents will not remember where they have placed their KI or may not have it available and will therefore not take KI. It is also assumed some residents will not take their KI when directed (i.e., they may take it early or late which reduces the efficacy). To account for this, KI was turned on in the model for approximately 50 percent of the public, and the efficacy of the KI was set at 70 percent.

6.2.4 Adverse Weather

Adverse weather is typically defined as rain, ice, or snow that affects the response of the public during an emergency. The affect of adverse weather on the mobilization of the public is not directly considered in establishing emergency planning parameters for this project because such a consideration more approximates a worst case scenario. However, adverse weather was addressed in the movement of cohorts within the analysis. The ESPMUL parameter in WinMACCS is used to reduce travel speed when precipitation is occurring as indicated from the meteorological weather file. The ESPMUL factor was set at 0.7 which effectively slows down the evacuating public to 70 percent of the established travel speed when precipitation exists.

6.2.5 Modeling using Evacuation Time Estimates

The purpose of using the ETE as a parameter in consequence modeling is to better approximate the real time actions expected of the public. Although consequence modeling has evolved to allow use of many cohorts and can address many individual aspects of each cohort, the approach to modeling evacuations is not direct. As stated earlier, evacuations include mobilizing and evacuating the public over a period of time, which is best modeled as a distribution of data. To use WinMACCS, this distribution of data must be converted into discrete events. For instance, upon the sounding of the sirens and issuance of the Emergency Alert System messaging, it is assumed all members of the public shelter and one hour later all members of the public enter the roadway network at the same time and begin to evacuate. In research of existing evacuations for technological hazards, it is shown that members of the public would actually enter the roadway network over a period of about an hour. It is not realistic that all vehicles would load simultaneously; however, this treatment within the model is necessary due to the current modeling abilities of WinMACCS.

Although WinMACCS can accommodate more cohorts, expert judgment was used to balance the number of cohorts with model run time. The speeds for each cohort are derived from the ETE, and the elements that factor into the speeds include:

- Time to receive notification and prepare to evacuate (mobilization time);
- Time to evacuate; and
- Distance of travel.

The time to receive notification requires assurance that sirens sound when needed. In review of the Reactor Oversight Program data regarding sirens for Surry, the average siren performance indicator was 99.9 percent, indicating that sirens do perform when tested. With few exceptions, travel speeds were established as whole numbers.

A simple ratio of distance to time would show that evacuation of the 0 to 10 public from the 10 mile EPZ at Surry which has an ETE of 10 hours 50 minutes, would provide a speed of 0.92 mph. However, as indicated above, notification and preparation to evacuate are included in the ETE.

For the general public, a one hour delay to shelter is assigned to reflect the mobilization time where residents receive the warning and prepare to evacuate. If the one hour mobilization time is subtracted from the ETE (10:50 – 1 hour) there remains 9 hours and 50 minutes to travel a maximum of 10 miles. As observed in actual evacuations due to technological or other hazards, people perform these mobilization activities at varying times with some residents ready to evacuate quickly while others can take up to an hour or longer. While this cohort is sheltered, a greater shielding factor is applied, and while en route during the evacuation, a lower shielding factor is applied.

During the evacuation, roadway congestion occurs rather quickly and traffic exiting the EPZ begins to slow. In review of over 20 ETE studies, this congestion typically occurs in 1 to 2 hours depending upon the population density and roadway capacity of the EPZ and considering that the vehicles are loaded onto the roadway network as a distribution. In the SOARCA analysis the 0-10 public is sheltered and preparing to evacuate for one hour. The public is then loaded onto the roadway and congestion is assumed to occur within 15 minutes. This total time of 1 hour 15 minutes for congestion to occur was established to be consistent with ETE studies.

The calculation of the speed of evacuees includes the first 15 minutes to the point when congestion occurs. For this first 15 minutes, evacuees are assumed to travel at 5 mph. This appears slow, but considering stop signs, traffic signals, and the build up of congestion, the speed is comparable to ETE modeling results. In the first 15 minutes at 5 mph, a distance of 1.25 miles has been traveled. At that time congestion is heavy and speeds slow for the next 8.75 miles.

The ETE is 10 hours 50 minutes for this cohort. Having sheltered and prepared to evacuate for 1 hour and then traveled the first 15 minutes at 5 mph, the remaining time is 9 hours and 35 minutes (10:50 - 1 hour shelter - 15 minutes at 5 mph). To determine the speed of travel for the remaining 8.75 miles, the distance is divided by the time (8.75 miles / 9 hours and 35 minutes) which provides a speed of 0.9 mph. The calculated speed used in the analysis for this cohort was rounded to 1 mph for this cohort. The process of dividing the maximum distance by the ETE provides a conservative speed.

6.2.6 Establishing the Initial Cohort in the Calculation

The WinMACCS parameters for the cohorts are stored in multi-dimensional arrays, and the dimensions of the arrays are defined by geographical area for the analysis. WinMACCS requires the dimensions be established with the first cohort. All subsequent cohorts must be defined within these array dimensions, meaning they can extend from the origin to any distance equal to or less than the maximum distance established with the first cohort.

Cohort 1 was defined as the 0 to 10 mile public and has the same response characteristics as Cohort 2. The cohort that extends the greatest distance and defines the limits of the array is the Shadow Evacuation, which is Cohort 2. Thus, in the WinMACCS model, Cohorts 1 and 2 had to be redefined to meet the above requirement. The WinMACCS model input parameters for Cohort 1 were extended from the plant out to the maximum array distance of 20 miles, and Cohort 2 extends from the plant out to 10 miles. In the WinMACCS input file, Cohort 1 is input as 20 percent of the population from 0 to 20 miles. This captures the 20 percent of the population between 10 and 20 miles involved in the shadow evacuation beyond the EPZ. Cohort

2 is input as 35.5 percent of the population from 0 to 10 miles. The combination of Cohorts 1 and 2 from 0 to 10 miles in the WinMACCS model represent the Public (0 – 10) Cohort defined above. For the remaining cohorts, application of parameters in the WinMACCS model is direct, and the population fractions directly correspond to the cohort descriptions.

6.3 Accident Scenarios

An emergency response timeline was developed for each accident scenario using information from the MELCOR analyses, expected timing of Emergency Classification declarations, and information from the ETE. The timeline identifies points at which cohorts would receive instruction from OROs to implement protective actions. In practice, initial evacuation orders are based on the severity of the accident and in Virginia would likely include an evacuation of the 2 mile zone and the 5 mile downwind keyhole consistent with the guidance in Supplement 3 to [19]. However, WinMACCS does not readily support modeling a keyhole area; therefore, the SOARCA project modeled evacuation of the full EPZ and a shadow evacuation from the 10 to 20 mile area.

6.3.1 Unmitigated LTSBO

The emergency response timeline for the unmitigated LTSBO scenario is shown in Figure 141. For this scenario, EAL SS1.1 specifies that if all offsite power and all onsite AC power is lost for greater than 15 minutes an SAE is declared. If restoration of power is not likely within 4 hours, EAL SG1.1 establishes that a GE be declared. It is assumed the SAE is declared in about 15 minutes, and plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE. It is assumed that notification to OROs is timely and the logistics of preparing and sounding sirens and broadcast of EAS messages occurs approximately 45 minutes after declaration of GE. From the MELCOR analysis, core damage is evidenced by the first fission product gap release occurs 16 hours into the event with an increased radioactive release to the environment occurring 45.5 hours into the event. The duration of specific protective actions for each cohort are summarized in Figure 142.

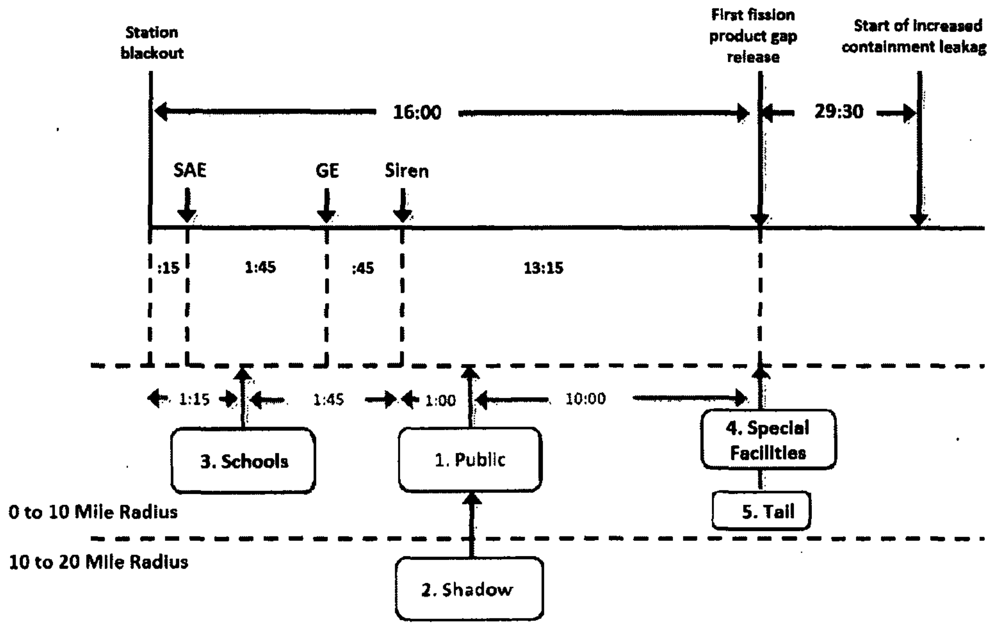

Figure 141 Unmitigated LTSBO Emergency Response Timeline

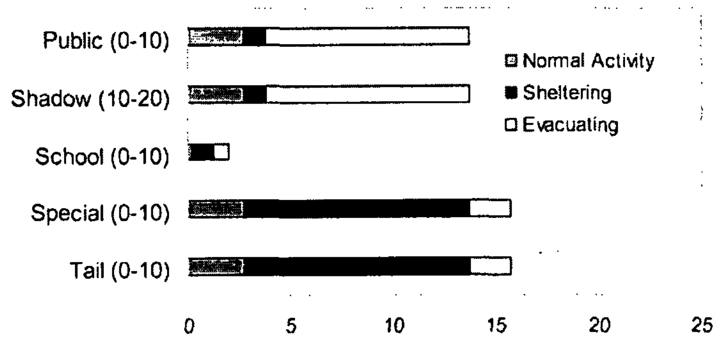

Figure 142 Duration of Protective Actions for Unmitigated LTSBO

The initiating event for the Unmitigated STSBO, Unmitigated STSBO with TI-SGTR, Mitigated STSBO with TI-SGTR, and Unmitigated LTSBO scenarios is a station blackout and EAL SS1.1 and SG1.1 are reached. Therefore, the cohort actions are identical for each of these scenarios.

Cohort 1: 0 to 10 Public. Cohort 1 is assumed to shelter when the sirens sound and the EAS message is broadcast. The time to receive the warning and prepare to mobilize is assumed to be 1 hour after notification of the public at which time this cohort begins to evacuate.

Cohort 2: 10 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway.

Cohort 3: 0 to 10 Schools. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the emergency response plan. Buses would be mobilized, and it is assumed schools begin evacuating 1 hour after notification.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail.

Cohort 5: 0 to 10 Tail. The Tail evacuates 11 hours after notification to evacuate.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 18 provides a summary of the evacuation timing actions for each cohort.

Table 18 Unmitigated LTSBO cohort timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 10 Public	2.75	1	0.25	9.75	5	1
10 to 20 Shadow	2.75	1	0.25	9.75	5	1
0 to 10 Schools	0.25	1	0.25	0.5	10	10
0 to 10 Special Facilities	2.75	11	1	1	1	10
0 to 10 Tail	2.75	11	1	1	1	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

Departure speeds and durations of the beginning and middle periods for the WinMACCS runs were derived from the Surry ETE study. Adjustments were made to individual elements of the WinMACCS grid to reflect differences in vehicle direction and speed of travel through the network. The timeline identifies the point at which it is assumed that cohorts begin to take action. The actions taken by each cohort last for a given period as indicated in Table 18.

6.3.2 Unmitigated STSBO

The emergency response timeline for the unmitigated short-term station blackout scenario is shown in Figure 143. The timing of emergency classification declarations was based on the EALs contained in site emergency plan implementing procedures. Protective actions were assumed to be recommended by OROs in accordance with approved emergency plans and procedures. Discussions were held with site representatives to help ensure proper understanding of EALs for each accident scenario and emergency response practices. Discussions with OROs confirmed that sirens are only sounded for a GE at Surry. Siren systems are tested routinely within all EPZs and the results of the Response Oversight Program indicate a 99.9 percent performance rating for sirens at Surry. Therefore, it is assumed that sirens do not fail and in the event one or two do fail, societal notification and route alerting by OROs would alert residents in these areas within the same mobilization time period as estimated for the EPZ. Figure 144 summarizes the duration of specific protective actions for each cohort.

For this scenario, EAL SS1.1 specifies that if all offsite power and all onsite AC power is lost for greater than 15 minutes an SAE is declared. If restoration of power is not likely within 4 hours, EAL SG1.1 establishes that a GE be declared. It is assumed the SAE is declared in about 15 minutes as shown in Figure 143 and that plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE. It is assumed that notification to OROs is timely and the logistics of preparing and sounding sirens and broadcasting the EAS message occurs approximately 45 minutes after declaration of GE. From the MELCOR analysis, core damage is evidenced by the first fission product gap release which occurs about three hours into the event with a significant radioactive release occurring 25.5 hours into the event.

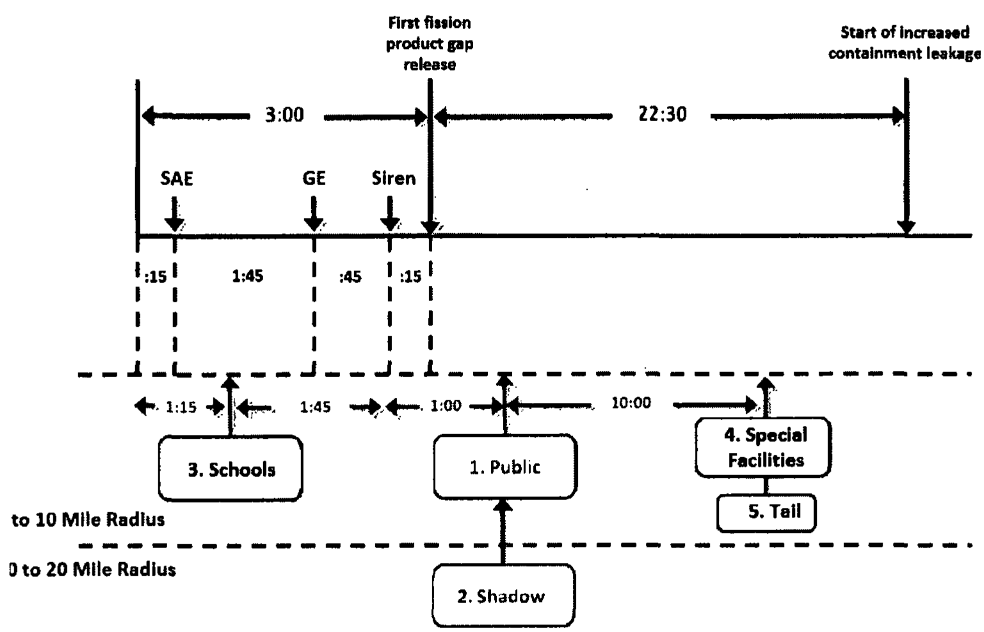

Figure 143 Unmitigated STSBO emergency response timeline.

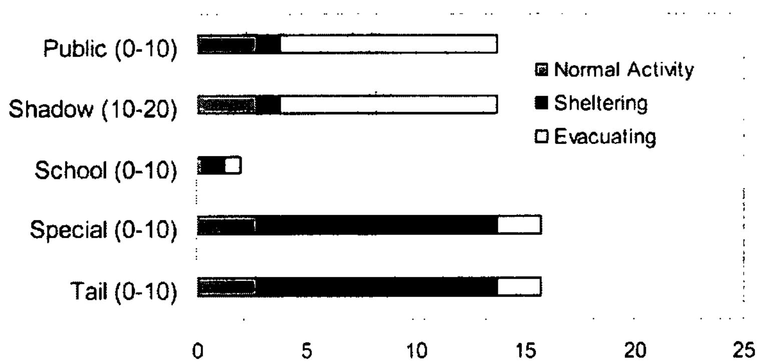

Figure 144 Duration of Protective Actions for the Surry Unmitigated STSBO

The Virginia Department of Emergency Management will directly communicate with schools upon receiving the declaration of SAE. This allows for the preparation and early response of schools, but the public is largely unaware at this time. It could be noted that there would be a societal communication process as members of the public become aware of the school evacuation. Sirens are sounded at an SAE in many states, but Virginia only sounds sirens in response to declaration of a GE. Although there could potentially be some shadow evacuation due to societal communication, it is assumed that there would be no significant movement of the general public. The initiating event for the Unmitigated STSBO, Unmitigated STSBO with TI-SGTR, Mitigated STSBO with TI-SGTR, and Unmitigated LTSBO scenarios is a station blackout and EAL SS1.1 and SG1.1 are reached. The cohorts are expected to respond to the EAS messages as described below.

Cohort 1: 0 to 10 Public. Following declaration of a GE, sirens are sounded and an evacuation order would be issued via an EAS message for affected areas within the EPZ. Cohort 1 is assumed to shelter when the sirens sound. The time to receive the warning and prepare to mobilize is assumed to be 1 hour after the siren. One hour is based on evacuation research which shows the pubic mobilizes over a period of time with some members of the public moving soon after hearing the sirens, while most take some time to prepare and then evacuate. One hour was selected as a reasonable centroid of an evacuation curve for this cohort which is consistent with empirical data from previous large scale evacuations [23].

Cohort 2: 10 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway. Residents in the 10 to 20 area begin seeing large numbers of people evacuating and initiate a shadow evacuation. There is no warning or notification for the public residing in this area which is not under an evacuation order.

Cohort 3: 0 to 10 Schools. Schools are the first to take action. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the offsite emergency response plan. It is assumed schools are notified at SAE and begin sheltering in about 15 minutes. Buses would be mobilized, and it is assumed schools begin evacuating 1 hour after the start of the incident. At this time in the event, roads are uncongested and school buses are able to exit the EPZ quickly.

Cohort 4: 0 to 10 Special Facilities. Special Facilities can take longer to evacuate than the general public because transportation resources, some of which are very specialized, must be mobilized. Special Facilities would be evacuated individually over a period of time based upon available transportation and the number of return trips needed. Special Facilities provide better shielding for the residents, thus while residents are in the facility, they are better protected than when they are evacuating. It was determined that the best representation of this cohort in the modeling is to evacuate with the tail and apply shielding factors consistent with the types of structures within which these residents reside. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail, although it is recognized this cohort would begin mobilization about the same time as the schools.

Cohort 5: 0 to 10 Tail. Using the evacuation data provided in the Surry ETE study [20], 90 percent of the evacuation of the EPZ is complete at approximately 11 hours into the evacuation, and this corresponds to the departure time for the 0 to 10 Tail..

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population. Any member of the public who does not evacuate is still subject to the Hotspot and Normal Relocation criterion discussed earlier.

The evacuation timing for each cohort is presented in Table 19. Selected input parameters for WinMACCS are provided to support detailed use of this study. More detailed information regarding modeling parameters is available in the MACCS2 User's Guide [42]. A brief description of the parameters is provided below.

- Delay to Shelter (DLTSHL) represents a delay from the time of the start of the accident until cohorts enter the shelter.
- Delay to Evacuation (DLTEVA) represents the length of the sheltering period from the time a cohort enters the shelter until the point at which they begin to evacuate.
- The speed (ESPEED) is assigned for each of the three phases used in WinMACCS including Early, Middle, and Late. Average evacuation speeds were derived from the Surry 2001 ETE report. Speed adjustment factors were then utilized in the WinMACCS application to represent free flow in rural areas and congested flow in urban areas.
- Duration of Beginning phase (DURBEG) is the duration assigned to the beginning phase of the evacuation and may be assigned uniquely for each cohort.
- Duration of Middle phase (DURMID) is the duration assigned to the middle phase of the evacuation and may also be assigned uniquely for each cohort.

For the 0 to 10 Public and the 0 to 10 Tail, the sum of the DLTEVA, DURBEG and DURMID is equal to the ETE.

Table 19 Unmitigated STSBO cohort timing.

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 10 Public	2.75	1	0.25	9.75	5	1
10 to 20 Shadow	2.75	1	0.25	9.75	5	1
0 to 10 Schools	0.25	1	0.25	0.5	10	10
0 to 10 Special Facilities	2.75	11	1	1	1	10
0 to 10 Tail	2.75	11	1	1	1	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

6.3.3 Unmitigated STSBO with TI-SGTR

The emergency response timeline for the unmitigated STSBO with TI-SGTR scenario is shown in Figure 145. For this scenario, EAL SS1.1 specifies that if all offsite power and all onsite AC power is lost for greater than 15 minutes an SAE is declared. If restoration of power is not likely within 4 hours, EAL SG1.1 establishes that a GE be declared. It is assumed the SAE is declared in about 15 minutes, and plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE. It is assumed that notification to OROs is timely and the logistics of preparing and sounding sirens and broadcast of an EAS message occurs approximately 45 minutes after declaration of GE. From the MELCOR analysis, core damage is evidenced by the first fission product gap release occurs about three hours into the event with the SGTR occurring 30 minutes later. The duration of specific protective actions for each cohort in this scenario is described in Figure 146.

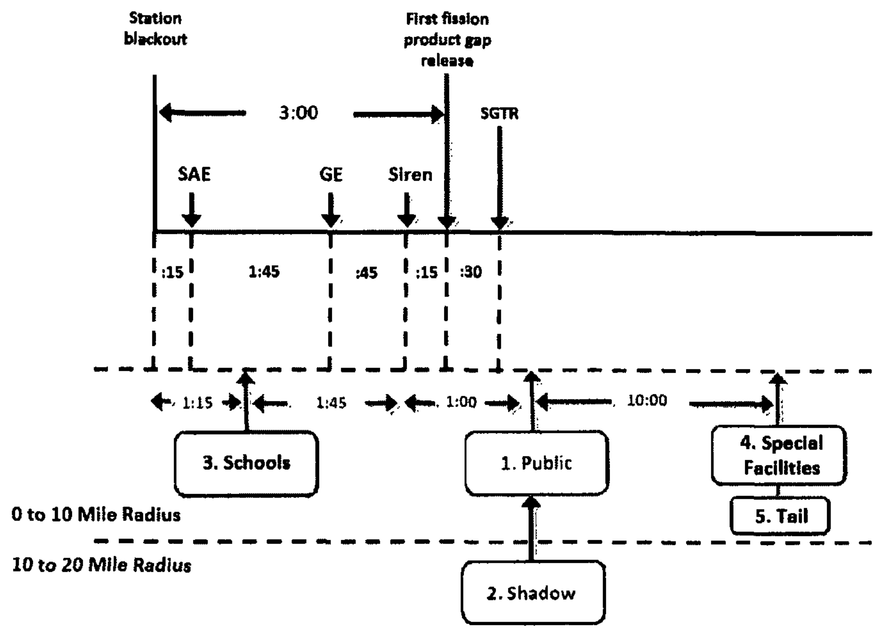

Figure 145 Unmitigated STSBO with TI-SGTR emergency response timeline

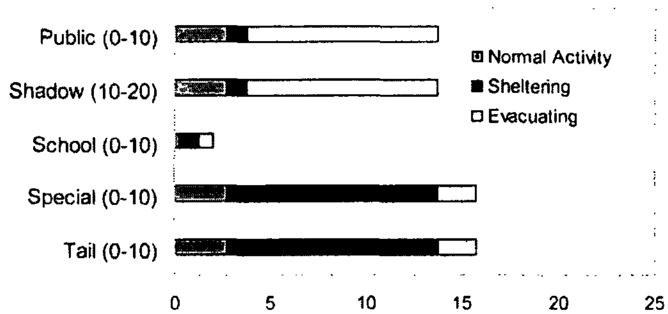

Figure 146 Duration of Protective Actions for the Surry Unmitigated STSBO with TI-SGTR

The initiating event for the Unmitigated STSBO, Unmitigated STSBO with TI-SGTR, Mitigated STSBO with TI-SGTR, and Unmitigated LTSBO scenarios is a station blackout and EAL SS1.1 and SG1.1 are reached. Therefore, the cohort actions are identical for each of these scenarios. The following description summarizes the actions taken by each cohort and Table 20 provides durations and evacuation speeds during the evacuation.

Cohort 1: 0 to 10 Public. Cohort 1 is assumed to shelter when the sirens sound and the EAS message is broadcast. The time to receive the warning and prepare to mobilize is assumed to be 1 hour after notification of the public at which time this cohort begins to evacuate.

Cohort 2: 10 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway.

Cohort 3: 0 to 10 Schools. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the emergency response plan. Buses would be mobilized, and it is assumed schools begin evacuating 1 hour after notification.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail.

Cohort 5: 0 to 10 Tail. The Tail evacuates 11 hours after notification to evacuate.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 20 Unmitigated STSBO with TI-SGTR cohort timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 10 Public	2.75	1	0.25	9.75	5	1
10 to 20 Shadow	2.75	1	0.25	9.75	5	1
0 to 10 Schools	0.25	1	0.25	0.5	10	10
0 to 10 Special Facilities	2.75	11	1	1	1	10
0 to 10 Tail	2.75	11	1	1	1	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

6.3.4 Mitigated STSBO with TI-SGTR

The accident scenario timeline for the Mitigated STSBO with TI-SGTR is identical to the unmitigated STSBO with TI-SGTR as shown in Figure 147 and Figure 148. The values identified Table 20 were used to support the consequence analyses for the Mitigated STSBO with TI-SGTR.

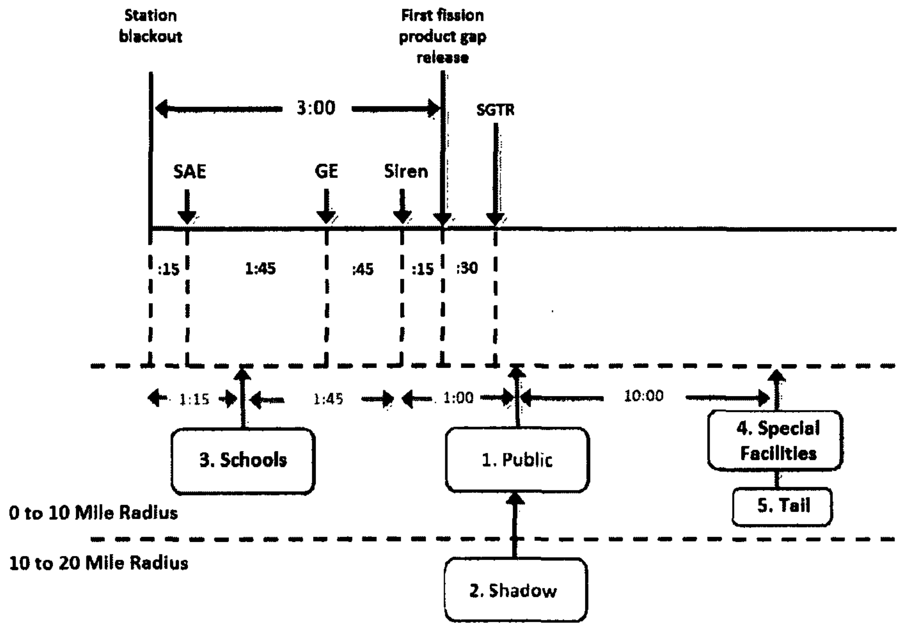

Figure 147 Mitigated STSBO with TI-SGTR emergency response timeline

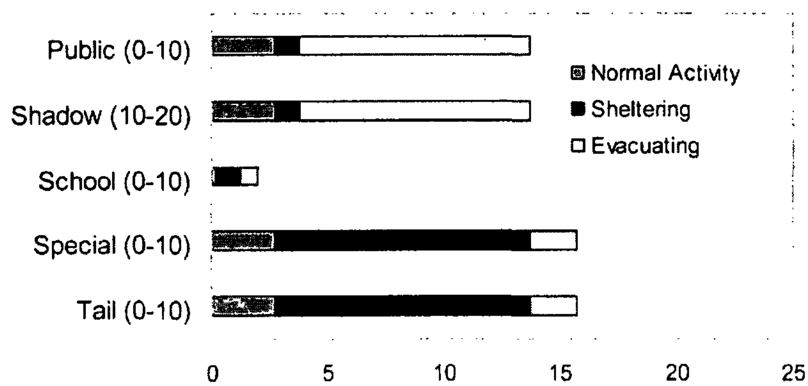

Figure 148 Duration of Protective Actions for the Mitigated STSBO with TI-SGTR

The initiating event for the Unmitigated STSBO, Unmitigated STSBO with TI-SGTR, Mitigated STSBO with TI-SGTR, and Unmitigated LTSBO scenarios is a station blackout and EAL SS1.1 and SG1.1 are reached. Therefore, the cohort actions are identical for each of these scenarios.

Cohort 1: 0 to 10 Public. Cohort 1 is assumed to shelter when the sirens sound and the EAS message is broadcast. The time to receive the warning and prepare to mobilize is assumed to be 1 hour after notification of the public at which time this cohort begins to evacuate.

Cohort 2: 10 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway.

Cohort 3: 0 to 10 Schools. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the emergency response plan. Buses would be mobilized, and it is assumed schools begin evacuating 1 hour after notification.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail.

Cohort 5: 0 to 10 Tail. The Tail evacuates 11 hours after notification to evacuate.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

6.3.5 Unmitigated ISLOCA

The emergency response timeline for the unmitigated ISLOCA scenario is shown in Figure 149. As shown in the figure, SAE is declared 15 minutes after the initiating event based on EAL FS1.1. A GE is declared approximately 1 hour and 15 minutes after the start of the event, based on EAL FG1.1, when water level in the RPV drops below top of active fuel (TAF). Sirens sound and EAS messages are broadcast approximately 45 minutes after declaration of GE. The timing used for SAE and GE declarations was based on review of the site specific emergency action levels from which corresponding protective action recommendations were derived. The durations of specific protective actions for each cohort are summarized in Figure 150.

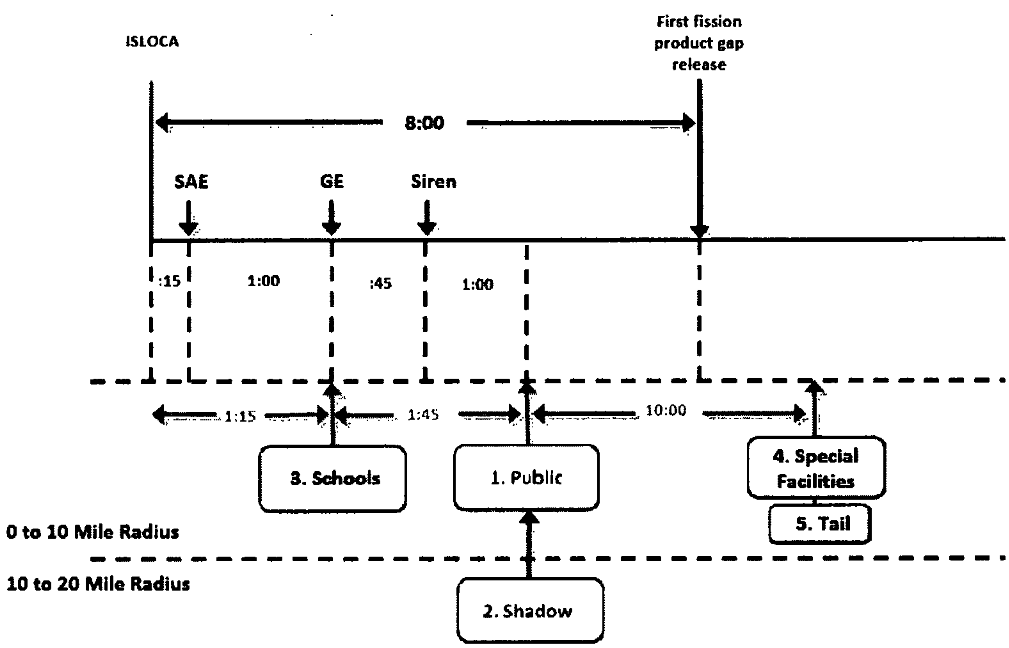

Figure 149 Unmitigated ISLOCA Emergency Response Timeline

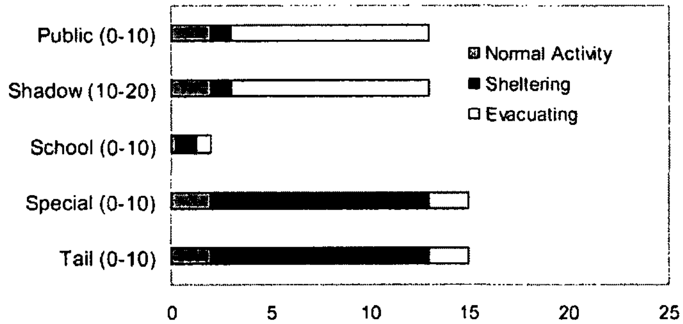

Figure 150 Duration nof Protective Actions for Unmitigated ISLOCA

The initiating event for the Unmitigated ISLOCA is an ISLOCA and EAL FS1.1 and FG1.1 are reached. The timing of these EALs and cohort actions are different than the scenarios where the initiating event is a station blackout.

Cohort 1: 0 to 10 Public. Cohort 1 is assumed to shelter when the sirens sound, and the time to receive the warning and prepare to mobilize is assumed to be 1 hour after the siren at which time this cohort begins to evacuate.

Cohort 2: 10 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway.

Cohort 3: 0 to 10 Schools. Schools are the first to take action. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the emergency response plan. Buses would be mobilized, and it is assumed schools begin evacuating 1 hour after receipt of notification at SAE.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail.

Cohort 5: 0 to 10 Tail. The Tail evacuates 11 hours after notification to evacuate.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 21provides a summary of the evacuation timing actions for each cohort.

Table 21 Unmitigated ISLOCA cohort timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 10 Public	2	1	0.25	9.75	5	1
10 to 20	2	1	0.25	9.75	5	1

Shadow						
0 to 10 Schools	0.25	1	0.25	0.5	10	10
0 to 10 Special Facilities	2	11	1	1	1	10
0 to 10 Tail	2	11	1	1	1	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

Departure speeds and durations of the beginning and middle periods for the WinMACCS runs were derived from the Surry ETE study. Adjustments made to individual elements of the WinMACCS grid to reflect differences in vehicle direction and speed of travel through the network remained the same for this scenario. The timeline identifies the point at which it is assumed that cohorts begin to take action. The actions taken by each cohort last for a given period as indicated in the timing table.

6.4 Sensitivity Studies

Three additional calculations were performed to assess variations of protective actions. Each of the sensitivity studies was conducted using the ISLOCA accident scenario.

Sensitivity 1 - Evacuation of a 16 mile area including a shadow evacuation from within the 16 to 20 mile area.

Sensitivity 2 – Evacuation of the 0 to 20 mile area.

Sensitivity 3 – Delay in implementation of protective actions for the public within the EPZ.

Protective actions beyond the EPZ are not planned in detail but utility, state and federal emergency response organizations will perform dose projection calculations and take field measurements. If assessment activities show that protective action guides would be exceeded beyond the EPZ, protective actions would be implemented in an ad hoc manner [19]. Ad hoc protective action decision timing and extent was based on reasonable estimates and judgment for modeling of population movement.

A full scale evacuation model was developed to assess the sensitivity of consequences to changes in protective action strategies. Although the modeling of the area beyond the EPZ includes a full scale evacuation for the sensitivity analysis, this does not reflect likely protective action recommendations. To support the assessment of implementing protective actions outside of the EPZ, data was obtained for the 10 to 20 mile area around the NPP. Evacuation speeds for the area 10 to 20 miles beyond the EPZ were developed using OREMS Version 2.6. OREMS is a Windows-based application used to simulate traffic flow and was designed specifically for emergency evacuation modeling [22]. The main features of OREMS utilized in the analysis include:

- Determining the length of time associated with complete or partial evacuation of the population at risk within an emergency zone, or for specific sections of highway network or sub-zones;
- Determining potential congestion areas in terms of traffic operations within the emergency zone.

The OREMS model considers special conditions, which may be imposed during an emergency evacuation. For example, intersections that normally having pre-timed controllers are assumed to be manned by emergency personnel to facilitate traffic flow [22]. This function is consistent with the emergency response actions that may be expected during an evacuation. Detail for roadway networks was obtained from aerial mapping and was input into OREMS using the standard intersection functions available in the model. Judgment and experience were necessary in determining the number of nodes that are established for the model. OREMS can manage hundreds of nodes, but there is a point at which the addition of nodes and links provides little change in the results. The nodal network established for the Surry Plant is a moderately populated network for this code.

The population values for the 10 to 20 mile area were developed using SECPOP 2000. A total of 171,182 vehicles were loaded onto 47 nodes distributed over 5 one-hour time periods. Vehicle data from the Surry 2001 ETE was also loaded onto the 10 to 20 mile area evacuation network consistent with the Surry ETE. The following evacuation times were produced from the OREMS calculation:

- 100 percent evacuation: 17 hours and 30 minutes; and
- 90 percent evacuation: 13 hours and 15 minutes.

These times were used to derive the evacuation speeds for input into the WinMACCS model. The evacuation modeling conducted for the Surry plant was developed consistent with the characteristics observed in prior evacuations conducted for non-nuclear incidents. Most notably, the analysis includes the common phenomenon of evacuations in which travelers who depart the threat zone the earliest experience lower amounts of delay. This occurs because the routes have yet to become fully utilized during the emergency and the traffic volume and corresponding route congestion is generally lower. Evacuees who depart during the middle part of the evacuation, when the greatest number of people are seeking to depart, generally experience the highest amount of congestion and delay. This is because the demand on the roadway network is at its greatest, exceeding the available capacity in many areas. Evacuees who depart the hazard zone later, while potentially putting themselves at greater risk, enter the transportation network as the demand is near or even less than the roadway capacity. This means that this group is able to generally avoid the longer delays associated with the peak evacuation demand period. The OREMS output evacuation curve for the 10 to 20 mile area is provided in Figure 151.

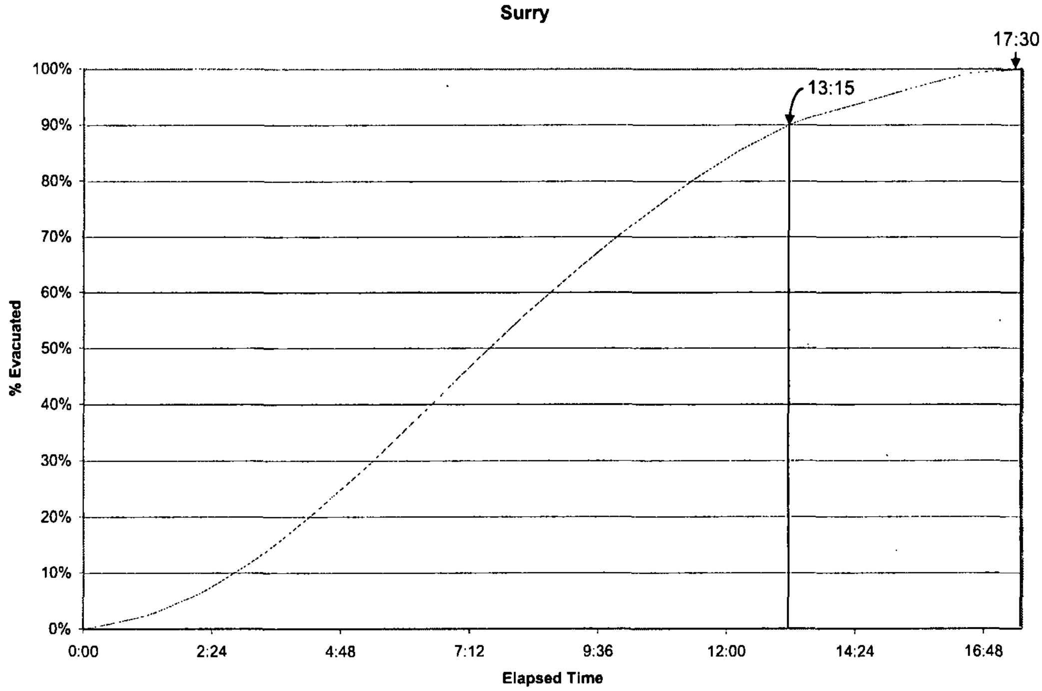

Figure 151 Evacuation Timeline from Surry for the 10 to 20 Mile Region

The initial accident scenarios were evaluated for protective actions within the EPZ. Expanding the protective actions to distances beyond the EPZ is not readily accommodated using the modeling approach selected for these analyses. Therefore, although OROs may request the 10 to 20 population shelter, this population group is treated within the modeling as performing normal activities throughout the emergency. The normal activity shielding factors are weighted averages of indoor and outdoor values based on being indoors 81 percent of the time and outdoors 19 percent of the time [46]. The hotspot and normal relocation model within MACCS2 will move affected individuals out of the area if the dose criteria apply.

6.4.1 Sensitivity 1 ISLOCA

For sensitivity 1, evacuation of a 16 mile area around the NPP is assessed. In addition, a shadow evacuation occurs from within the 16 to 20 mile area, and the remaining members of the public in the 16 to 20 mile area were assumed to shelter. Figure 152 and Figure 153 summarize the cohort timing for sensitivity 1.

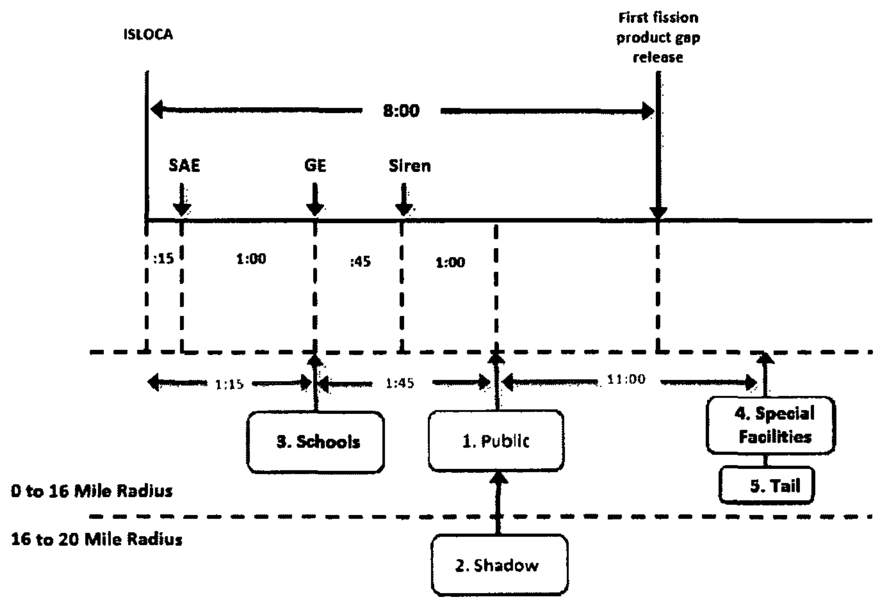

Figure 152 ISLOCA Timeline for Sensitivity 1

The 16 mile distance was selected as approximately half the distance between 10 and 20 miles beyond the plant. WinMACCS establishes control rings within the model to support the calculations and these rings are at even numbered distances (e.g., 2,4,6, miles). Because 15 miles was not an option, 16 miles was selected.

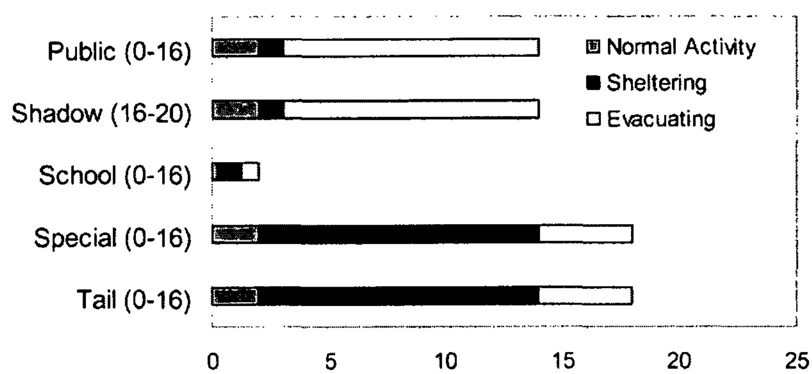

Figure 153 Duration of Protective Actions for ISLOCA Sensitivity 1

Cohort 1: 0 to 16 Public. Following declaration of a GE, sirens are sounded and an EASmessage is broadcast that includes an evacuation order for affected areas within the EPZ.The public is assumed to shelter upon receipt of the EAS message, and the time to receivethe warning and prepare to mobilize is assumed to be 1 hour. An assumption in this sensitivity analysis is the 16 to 20 public would be notified at the same time as the EPZ via EAS messaging and route alerting. The ETE for the public was estimated as a linear projection between the Surry 2001 ETE Study and the 10 to 20 mile ETE developed for the sensitivity 2 analysis.

Cohort 2: 16 to 20 Shadow. This cohort is assumed to begin movement at the same time as the 0 to 16 Public after sirens have sounded within the EPZ, EAS messages are broadcast, and when widespread media broadcasts are underway. Residents in the 16 to 20 area begin seeing large numbers of people evacuating and initiate a shadow evacuation.

Cohort 3: 0 to 16 Schools. Schools are the first to take action. Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools within the EPZ in accordance with the emergency response plan. For this scenario, it is assumed schools within the 10 mile to 16 mile would evacuate beginning 1 hour after receipt of notification. There are no sirens in the 10 to 16 mile area and no preplanned EAS messages; therefore, notification is assumed to be media broadcasts to residents in this area.

Cohort 4: 0 to 16 Special Facilities. All Special Facilities are required to have evacuation plans, and in this scenario, it is assumed the facilities within the 0 to 16 mile area would evacuate.

Cohort 5: 0 to 16 Tail. An estimate of the departure for the evacuation tail is established as a linear projection between the Surry 2001 ETE Study and the OREMS 10 to 20 mile ETE developed for evacuation to a distance of 20 miles from the plant.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the 0 to 16 mile public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 22 Sensitivity Case 1 Cohort Timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 16 Public	2	1	0.25	10.75	5	1.5
16 to 20 Shadow	2	1	0.25	10.75	5	1.5
0 to 16 Schools	0.25	1	0.25	0.5	10	10
0 to 16 Special Facilities	2	12	2	2	1.5	10
0 to 16 Tail	2	12	2	2	1.5	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

The timeline identifies the point at which it is assumed that cohorts begin to take action. The actions taken by each cohort last for a given period as indicated in the timing table.

6.4.2 Sensitivity 2 ISLOCA

For Sensitivity Case 2, evacuation of a 20 mile area around the NPP is assessed. It is not expected that evacuation would be required beyond the EPZ; however, this sensitivity analysis considers the possibility. Because the limit of the evacuation in this sensitivity analysis extends a considerable distance away from the plant, it was determined that adding a shadow evacuation beyond 20 miles would provide an overly conservative assumption. Therefore, no shadow evacuation is assumed in this calculation.

The WinMACCS model structure requires the first cohort to extend to the limits of the calculation, and for earlier calculations this was the limit of the shadow evacuation. In this sensitivity analysis, because there is no shadow evacuation, the limit of the first cohort is 20 miles and represents the 0 to 20 public.

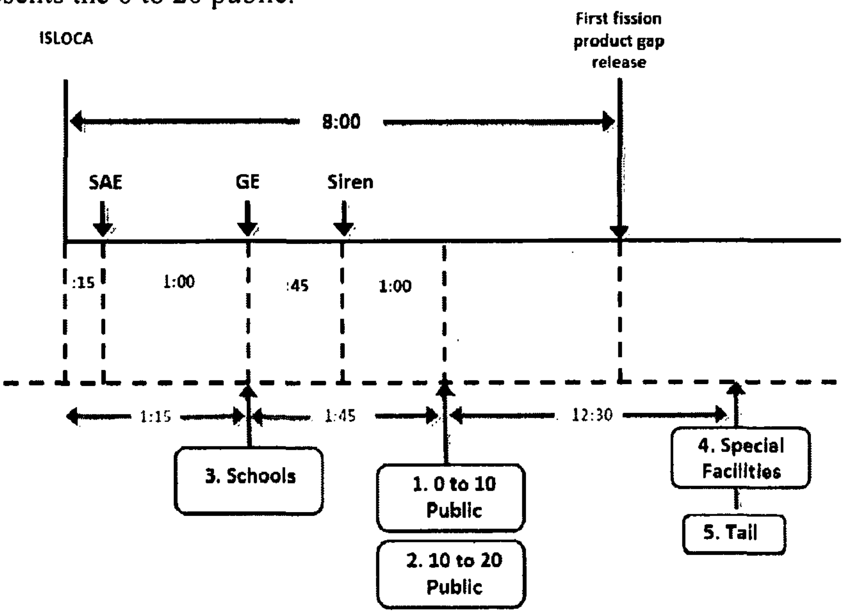

Figure 154 ISLOCA Timeline for Sensitivity 2

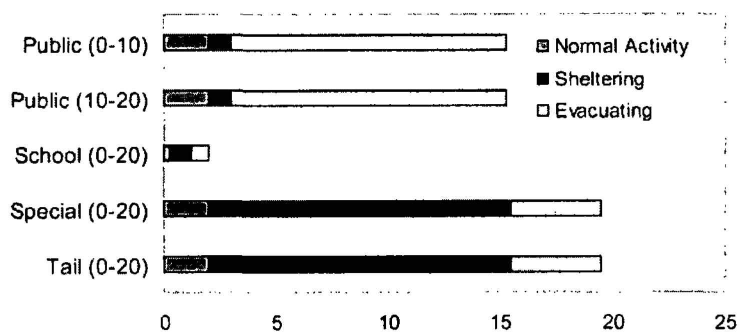

Figure 155 Protective Action Durations for ISLOCA Sensitivity 2

Cohort 1: 0 to 10 Public. Cohort 1 is assumed to shelter when the sirens sound and the EASmessage is broadcast. The time to receive the warning and prepare to mobilize is assumed to be 1 hour after the siren at which time this cohort begins to evacuate.

Cohort 2: 10 to 20 Public. Following declaration of a GE, sirens are sounded within the EPZ and an evacuation order would be issued for affected areas within the EPZ. There are no sirens in the 10 to 20 mile area and no preplanned EAS messages; therefore, notification is assumed to be media broadcasts to residents in this area. The time to receive the warning and prepare to mobilize is still assumed to be 1 hour after the initial notification. The ETE for the 10 to 20 public was estimated using OREMS.

Cohort 3: 0 to 20 Schools. Upon receipt of the declaration of SAE by the site, the VirginiaDepartment of Emergency Management would notify the schools within the EPZ in accordance with the emergency response plan. For this sensitivity study, it is assumed schools beyond the EPZ would decide, based upon media information that it is prudent to evacuate or close schools immediately.

Cohort 4: 0 to 20 Special Facilities. For this sensitivity study, is assumed Special Facilities beyond the EPZ would decide, based upon media information that it is prudent to evacuate. Special Facilities can take longer to evacuate than the general public because transportation resources must be mobilized, some of which are very specialized; therefore, the Special Facilities cohort is assumed to depart at the same time as the evacuation tail.

Cohort 5: 0 to 20 Tail. The ETE for the evacuation tail was estimated based on the OREMS analysis.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the 0 to 20 mile public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 23 Sensitivity Case 2 Cohort Timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph
0 to 10 Public	2	1	0.25	12	5	1.6
10 to 20 Public	2	1	0.25	12	5	1.6
0 to 20 Schools	0.25	1	0.25	0.5	10	10
0 to 20 Special Facilities	2	13.5	2	2	1.6	10
0 to 20 Tail	2	13.5	2	2	1.6	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

The timeline identifies the point at which it is assumed that cohorts begin to take action. The actions taken by each cohort last for a given period as indicated in the timing table.

6.4.3 Sensitivity 3 for ISLOCA with a Delay of Implementation of Protective Actions

There is a high level of confidence regarding the actions expected from control room operators in the event of accident scenarios identified for analysis in the SOARCA project. The initiating conditions provide clear indication to these operators and the response actions of the control room are prescribed. The Peer Review of the response timelines identified that a delay of the implementation of protective actions should be considered. Such a delay could be due to delay in control room declaration of an incident, delay in the decision process of OROs, or delay in communication to the public regarding implementation of protective actions. To address the potential for delay, an additional protective action timeline has been developed for the ISLOCA. This timeline reflects a delay in the implementation of protective actions by the public within the EPZ. Because protocols and procedures are in place, exercised and tested frequently, it is assumed that a delay of 30 minutes is adequate for this sensitivity study. Figure 156 and Figure 157 summarize cohort timing for sensitivity 3.

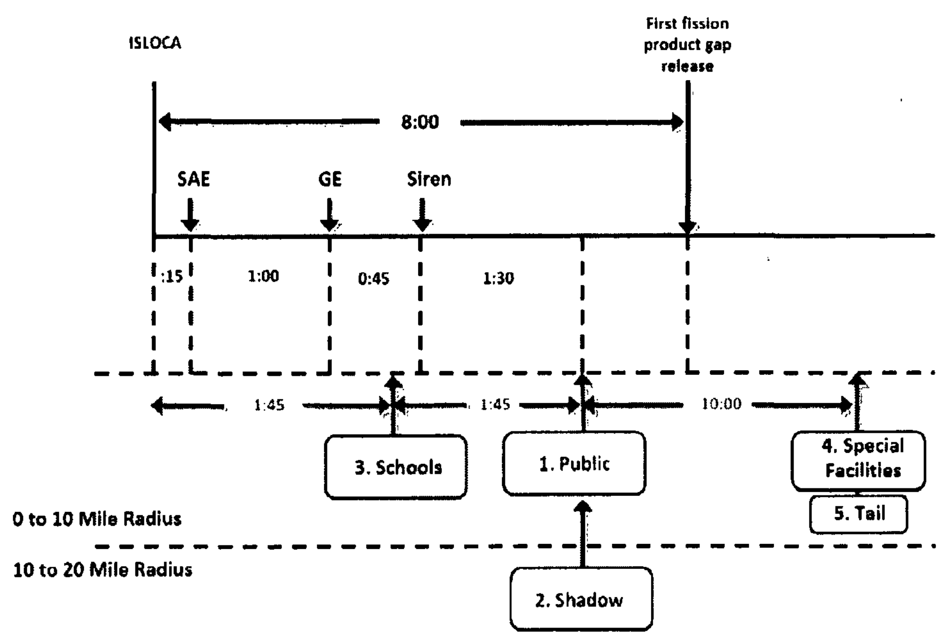

Figure 156 ISLOCA Timing for Sensitivity 3

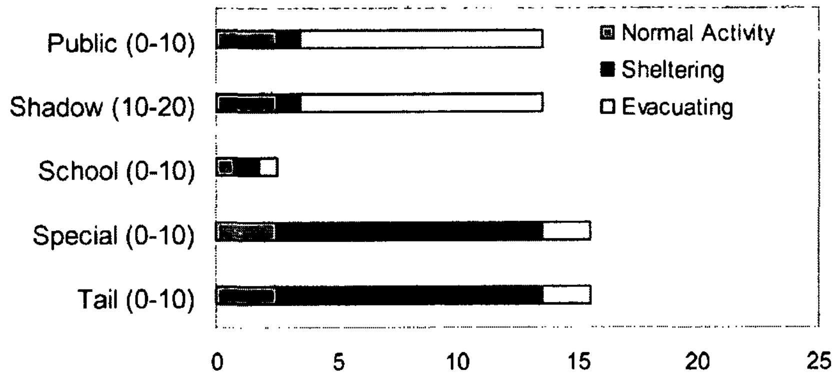

Figure 157 Protective Action Durations for ISLOCA Sensitivity 3

The initiating event for the Unmitigated ISLOCA is an ISLOCA and EAL FS1.1 and FG1.1 are reached. The timing of these EALs and cohort actions are different than the scenarios where the initiating event is a station blackout. This sensitivity study includes a delay of 30 minutes in the implementation of protective actions. This delay is reflected in the cohort descriptions below. **Cohort 1**: **0 to 10 Public.** Cohort 1 is assumed to shelter when the sirens sound and the initial EAS message is broadcast. The time to receive the warning and prepare to mobilize is assumed to be 1.5 hours after the receipt of the EAS message at which time this cohort begins to evacuate.

Cohort 2: **10 to 20 Shadow.** This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway.

Cohort 3: **0 to 10 Schools.** Upon receipt of the declaration of SAE by the site, the Virginia Department of Emergency Management would notify the schools in accordance with the emergency response plan. Buses would be mobilized, and it is assumed schools begin evacuating 1.5 hours after notification.

Cohort 4: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail. Both of these start times were delayed an additional 30 minutes for this sensitivity study.

Cohort 5: 0 to 10 Tail. The Tail evacuates 11.5 hours after the public has been notified to evacuate.

Cohort 6: Non-evacuating public. This cohort group represents the portion of the public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 24 Sensitivity Case 3 Cohort Timing

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED† (early) mph	ESPEED† (mid) mph

0 to 10 Public	2.5	1	0.25	9.75	5	1
10 to 20 Public	2.5	1	0.25	9.75	5	1
0 to 10 Schools	0.75	1	0.25	0.5	10	10
0 to 10 Special Facilities	2.5	11	1	1	1	10
0 to 10 Tail	2.5	11	1	1	1	10
Non-Evac	NA	NA	NA	NA	0	0

† - Values represent speeds east of the James River. Speeds West of the river are increased through use of multipliers in the WinMACCS model.

6.5 Analysis of Earthquake Impact

A seismic analysis was developed to assess the potential effects on local infrastructure, communications, and emergency response in the event of a large scale earthquake. The accident used in the earthquake analysis is the STSBO with TI-SGTR. Integrating the effects of the earthquake into the analysis required assessing the damage potential of the earthquake, identification of parameters that would be affected, and determining the new values for affected parameters.

The potential for an earthquake is largely identified by the occurrence of previous earthquakes in the region. Understanding of where earthquake faults exist in the eastern United States is not robust; whereas, in the west geological fault lines can be identified on the surface. Faults in the east are usually buried below layers of soil and rock and are not identifiable making prediction of earthquake location and magnitude difficult. The earthquakes hypothesized in SOARCA are assumed to be close to the plant site, and it may be assumed that severe damage is generally localized. Housing stock would generally survive the earthquake, with some damage. The local electrical grid is assumed out of service due to the failure of lines, switch yard equipment, or other impacts. There is backup power system for the sirens at Surry, and it is expected sirens would function. Under these postulated conditions, the potential for such an earthquake to affect emergency response and public evacuation is considered.

6.5.1 Soils Review

To approximate the extent of damage, an evaluation of the potential failure of infrastructure was conducted by NRC seismic experts to determine which, if any, roadways or bridges may fail under the postulated earthquake conditions. The assessment was performed using readily available information and professional judgment. Existing information on basic bedrock geology of the region was developed from reports and papers from the United States Geological Service (USGS). Soils of this region are formed from unconsolidated sediments deposited when the ocean level was much higher than at present. As sea levels lowered, many of these deposits were reworked by meandering rivers and streams that originated in the western part of the state and flowed to the east. In general, the closer to the coast, the nearer the water table is to the soil surface. Soils in the coastal plain are acidic, infertile, highly weathered, and vary from sandy textures to very clayey textures. Soil types are mostly silts and sands deposited in low energy environments which make them potentially susceptible to liquefaction in an earthquake; however, site specific liquefaction potential is highly variable.

Most landscapes are nearly level to gently sloping and because of this feature the soils are not as susceptible to erosion.

6.5.2 Infrastructure Analsysis

The seismic evaluation of the potential failure of roadway infrastructure identified 40 bridges and roadway segments that could fail under the postulated conditions. The major areas where problems occur are in and around the urban area of Williamsburg, Virginia. Figure 158 shows an example of a bridge that could potentially fail under the earthquake conditions. Figure 159 shows the transportation network and the locations of the affected roadway segments and bridges.

Table 25 provides a description of each of the roadway segments and bridges that could fail.

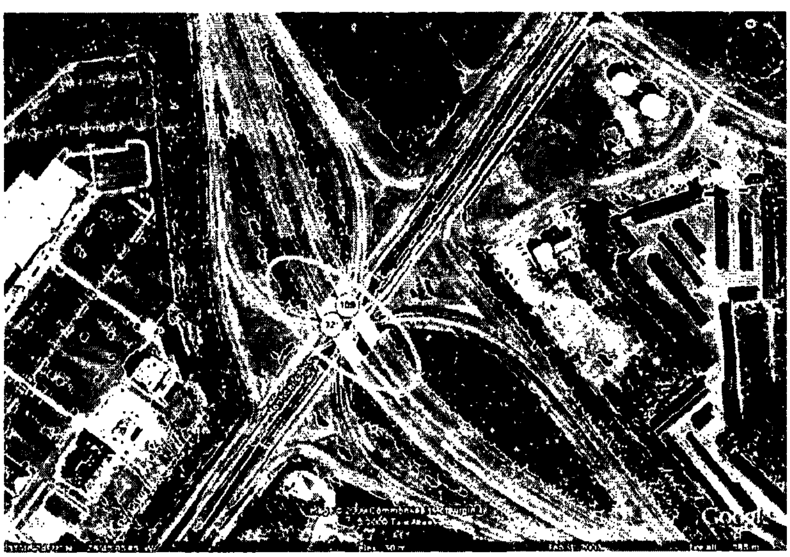

Figure 158 Highway 199 Over Highway 321 (Bridge 13 on Figure 159)

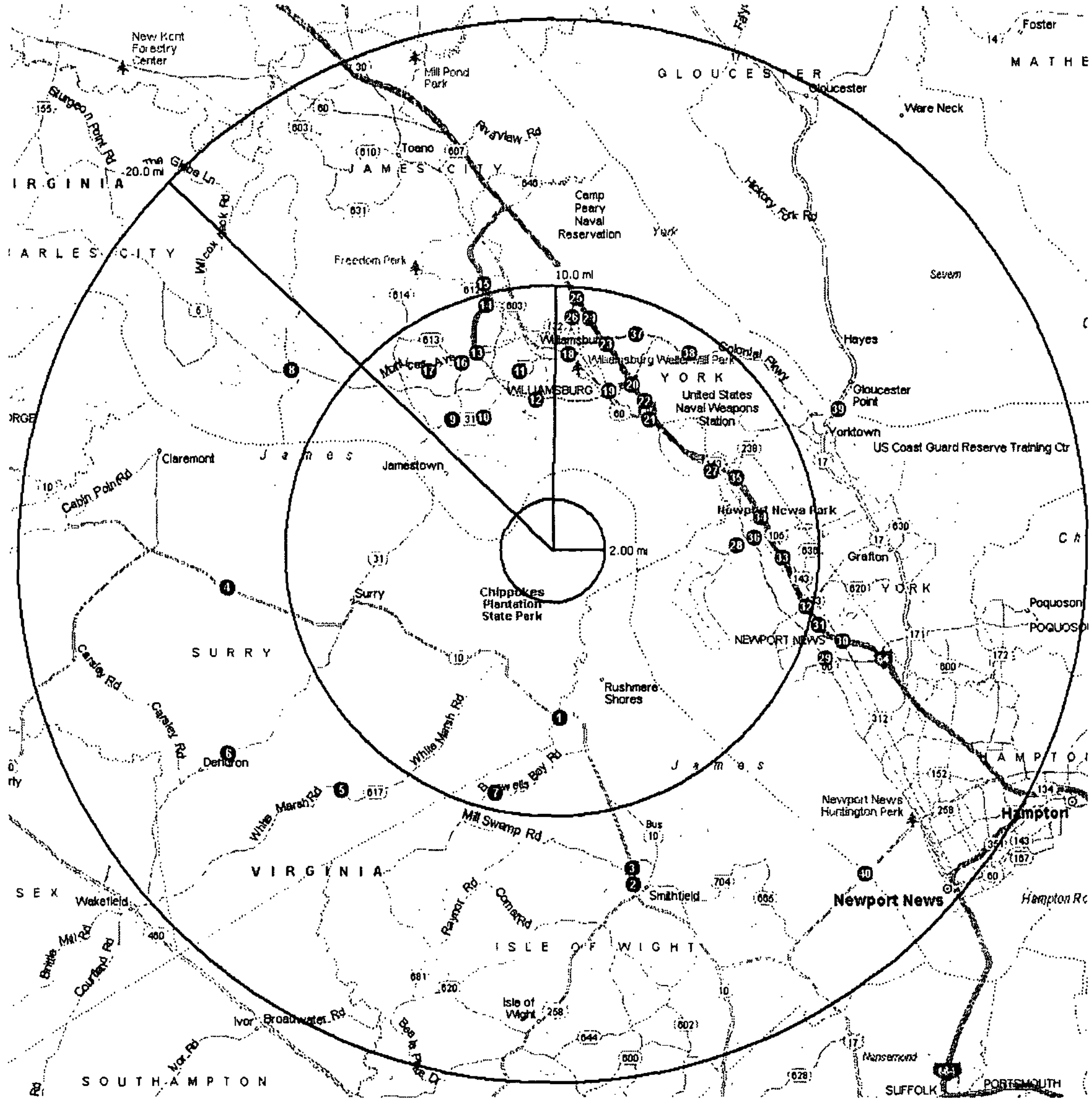

Figure 159 Roadway Network Identifying Potentially Affected Roadways and Bridges

The earthquake may cause structural damage in other areas of the EPZ. The structures within the EPZ are primarily light commercial and residential housing, both of which would largely be expected to stay intact. However, there are areas of larger commercial facilities and theme parks which could sustain damage. The urban setting is also likely to experience localized fires caused by ruptured gas lines.

Table 25 Description of the Potential Evacuation Failure Locations

Location	Description
1	Small bridge on Highway 10 across pond
2	Overpass on Highway 10 near Smithfield
3	Bridge on Highway 10 near Smithfield
4	Stream culvert and boggy area under Highway 10
5	Road 617 (Whitemarsh Rd) culvert and bog beneath roadway
6	Highway 31 south of Highway 10- Bridge over pond
7	Road 621 (Burwell Bay Rd) pond and bog
8	Bridge over Route 5 (John Tyler Highway)
9	Bridge on Route 31 (Jamestown Rd)
10	Bridge over small lake on Route 31
11	Route 31 across small dam
12	Bridge over river on Highway 199-2 bridges in this area
13	Highway 199-overpass over Route 321 (Monticello Ave)
14	Highway 199-overpass over Longhill road
15	Highway 199-overpass Old Towne Rd
16	Route 321 (Monticello Ave) adjacent to Lake Powhaton, Potential slope failure/slumping
17	Route 321 (Monticello Ave) bridge over bog
18	Highway 60 in Williamsburg-Overpass over rail tracks
19	Overpass US 199 over US 60 in Williamsburg
20	US 199/I-64 Interchange in Williamsburg
21	I-64/US 60 Interchange in Williamsburg
22	I-64 overpass near Williamsburg
23	I-64 Bridges over Colonial Parkway, edge of Williamsburg
24	I-64 Bridge over river and swampy area
25	I-64/Route 143 overpass
26	Route 143 ridge over river and swampy area
27	US 60-Dike over lake, southwest of Williamsburg
28	US 60 and SR 105 interchange
29	Small dam above US 60
30	I-64 overpass with SR 143 (Jefferson Ave)
31	I-64 overpass - Bland Ave.
32	I-64 Overpass, SR 173 (Denbigh Ave)
33	I-64 bridges over Industrial Park Drive
34	I-64 bridge over lake
35	I-64 and SR 238 (Old Williamsburg Rd) overpass
36	SR 105 (Ft. Eustis Blvd) bridge over lake
37	Colonial Historic Parkway- road on dam
38	Colonial Historic Parkway- bridge over creek
39	Highway 17 bridge across York River
40	Highway 17/258 across James River

6.5.3 Electrical and Communications

There are many high voltage power lines traversing the Surry EPZ. It is assumed transformers and switchgear fail; however, power lines and related structures are assumed to not fail to a degree that they affect the emergency response (i.e., it is assumed that power lines do not fall across roadways potentially affecting evacuation routes). The siren system at Surry includes backup batteries which would be sufficient to sound the sirens upon declaration of a GE.

Loss of power limits the potential for residents to receive instructions via the Emergency Alert System (EAS) messaging. Televisions, household radios, and some telephones will not operate; although battery operated radios and car radios will operate. It is expected that the public will utilize these means of communications as well as societal forms, such as neighbor to neighbor, propagating the Emergency Alert System (EAS) message throughout the EPZ. The alert and notification would be supplemented by route alerting conducted by OROs. This is a planned back up form of communication for the EPZ, and research shows this is effective and can be conducted in a timely manner. As experienced in other large scale disasters, residents will inundate the emergency telecommunications systems with questions and requests for help, many of which will require emergency responder support [45]. This may cause cell phone service to be overloaded and may delay communications. However, for a localized event such as an earthquake, it is assumed that cell phone service is restored quickly.

The loss of power will affect traffic signalization within the EPZ. Typically, traffic signals default to red/red in a power outage requiring all directions to stop prior to entering an intersection. This effectively turns signalized intersections into four-way stop signs. Four-way stop, as an intersection control, is less effective signalization for moving large numbers of vehicles, particularly when traffic is present on multiple approaches [47].

6.5.4 Emergency Response

The assumption on the event timing is a mid-week winter day in which the public is at work and children are at school. The primary shift of emergency responders would be on duty and immediately available at the time of the incident. There is an initial need to assess damage and respond to life threatening needs. These initial priorities for emergency response personnel may delay implementation of traffic control to support an evacuation. It is expected that route alerting would not be appreciably delayed, because Surry has backup batteries for the siren system. Route alerting may be needed in localized areas.

During large scale emergencies, OROs routinely supplement staff with on-call and off-duty personnel. Although communications are assumed to be initially limited, radios are available to contact needed staff, and off-duty responders are expected to report for duty during such emergencies. By the time an evacuation is ordered, it is expected that OROs would have been augmented with additional staff; however, the effect on the infrastructure within the EPZ will require that OROs initially support activities that are protective of health and safety. It is assumed that response personnel are not immediately available to support traffic control for an evacuation.

6.5.4.1 Evacuation Time Estimate

Evacuation times are affected by bridges and roadways that fail, traffic signalization, and EAS messaging is not disseminated timely to inform evacuees of protective actions and evacuation routes. There are 40 locations identified as potential failures of infrastructure. Although evacuations are planned and conducted to move the public radially away from the NPP, evacuation following this postulated earthquake will be constrained to the few unobstructed access routes out of the EPZ. West of the James River, the population is sparse and infrastructure failures are relatively isolated. The result is a negligible affect on the evacuation time for this area. However, the infrastructure failures north and east of the river will have a pronounced affect on the movement of vehicles and requires that an ETE for this area be developed to reflect the conditions.

In review of the Surry 2001 ETE report [20], no significant traffic congestion was noted in the areas west of the James River; however, significant traffic congestion exist in all zones that are north and east of the James River. Congestion is particularly concentrated in areas to the northeast of Surry. The James River varies in width from about 1.5 to 5.5 miles with only three crossings along a 50-mile stretch between Hopewell and Portsmouth. For this reason, the evacuation network north and east of the James River is essentially disconnected from the network west of the river.

The major road system on the east side of the river, as discussed in the Surry 2001 ETE report [20], is oriented northwesterly and southeasterly, parallel with the peninsula. Primary evacuation routes east of the river include U.S. Highway 17, Interstate 64, U.S. Highway 60, and Route 143 and the Surry ETE report identifies all of these roadways as congested under evacuation conditions. All of these roadways are further affected by the seismic event.

As indicated in Figure 159, 32 of the 40 affected roadways and bridges are located east of the James River and are clustered in and around Williamsburg, Virginia. Of particular importance is the major affect on Interstate 64 through this section of the EPZ. Most of the bridges and overpasses on this interstate are assumed to fail which causes some very difficult issues with an assessment of the evacuation time. To truly understand the effect of such damage on the ETE, the roadway system should be modeled. However, a basic vehicle/capacity approach is provided to develop a reasonable ETE and associated evacuation speeds. This assessment considers the timing and activities of residents, but may not fully account for factors such as driver confusion over which routes are accessible.

This scenario is a mid-day mid-week event where the interstate can be assumed to be moderately traveled. Following the timeline of the event, a priority for emergency response personnel will be assisting those who are in life threatening conditions such as occupants of vehicles that are stranded on or under the sections of Interstate 64 between the failed bridge segments. The assistance in removing the vehicles has a two fold effect of tying up emergency response personnel and creating additional congestion around the roadways leading to the interstate.

Even before SAE has been declared, the failure of the interstate bridges will affect traffic and cause a gridlock within the area. The interstate is unusable, and the underpasses to the interstate also become unusable which are major impediments to an evacuation. The significant failure of infrastructure causes the limiting factor of the ETE to be the queuing and loading of the evacuating vehicles at the points at which evacuation routes are available near the edge of the EPZ. At approximately the point at which Interstate 64 crosses the 10 mile EPZ boundary, there is no further damage to the Interstate. This is true at both the north and south ends of the EPZ, and at this point Interstate 64 is available to support the evacuation with 3 lanes northbound and 5 lanes southbound.

Theoretical lane capacity for the Interstate would approximate 2,400 passenger cars per lane per hour; however, studies of lane capacity during planned evacuations, such as the evacuation in response to Hurricane Katrina, concluded that observed peak interstate flows were between 1,350 and 1,500 passenger cars per lane per hour [44]. There are an estimated 90,000 evacuating vehicles, as derived from the Surry ETE report [20]. Applying a rate of 1,500 passenger cars per lane per hour, the 8 available lanes would support all of the vehicles in approximately 7.5 hours; however, vehicles cannot simply access the interstate, so this simple Interstate capacity analysis only confirms that once on the Interstate, traffic will be in a free-flow state.

The controlling point in the evacuation is the access capacity to Interstate 64. As vehicles travel to the available Interstate onramps, the lines of traffic, or queue, saturate the roadway network. To evaluate onramp capacity, the segment-flow density function is applied for oversaturated conditions following the procedure in the Chapter 22 of the Highway Capacity Manual [47]. Using a saturated traffic density estimated at 160 passenger cars per mile per lane, which is appropriate for gridlock conditions, a corresponding onramp capacity of 500 passenger cars per hour per lane is obtained. Using aerial mapping, there are an estimated 10 onramps to Interstate 64 within a short distance of the EPZ boundary. Ten onramps with 500 passenger cars per hour per lane can load 90,000 in 18 hours. Because outbound capacity once on the Interstate is not a limiting factor and the loading points are beyond the limits of the EPZ, it is assumed that the time to load the vehicles is effectively the ETE.

Clearly this is a simplified approach to evacuation under the seismic conditions identified. However, for purposes of understanding the effects of protective actions, the value of 18 hours to complete the evacuation appears reasonable.

6.5.5 Development of WinMACCS parameters

Traffic movement was approximated in each grid element by assigning a direction and speed for the vehicles within the grid. To account for the potential loss of bridges and roadway sections, the routing patterns in the WinMACCS model were adjusted to divert traffic around the locations identified.

6.5.5.1 Relocation Outside the Evacuation Area

In the event of a significant release, the population in the region outside the evacuation area would be moved if their potential dose exceeds protective criteria based on field measurements. The MACCS2 code uses hotspot and normal relocation, which is a dose based rather than distance based protective action. The values used in the earthquake analysis are the same as those used in the baseline analysis.

For hotspot relocation, individuals beyond twenty miles are relocated 24 hours after plume arrival if the total lifetime dose commitment for the weeklong emergency phase exceeds 0.05 Sv (5 rem). For the normal relocation, individuals are relocated 36 hours after plume arrival if the total lifetime dose commitment exceeds 0.01 Sv (1 rem). Review of the accident sequence timelines suggest that OROs would not be available earlier to assist with relocation due to higher priority tasks in the evacuation area.

6.5.5.2 Shielding Factors

Shielding factors are the same as those used in the baseline analyses. It may be expected that the damage to structures caused by an earthquake of this magnitude would include broken windows and some structural damage. Additionally, earthquakes frequently cause residents to go outside until they are more certain of the extent of structural damage that may have occurred. These factors would reduce the shielding capacity; however, because of the limited time residents within the seismic area are assumed to shelter, no adjustments in the modeling were made.

6.5.6 Seismic Analysis STSBO with TI-SGTR

The emergency response timeline for the unmitigated STSBO with TI-SGTR seismic scenario is shown in Figure 160. For this scenario, EAL SS1.1 specifies that if all offsite power and all onsite AC power is lost for greater than 15 minutes an SAE is declared. If restoration of power is not likely within 4 hours, EAL SG1.1 establishes that a GE be declared. It is assumed the SAE is declared in about 15 minutes, and plant operators would recognize rather soon that restoration of power within 4 hours is unlikely. A 2 hour period from loss of power was selected as a reasonable time for declaration of a GE. It is assumed that notification to OROs is timely and the logistics of preparing and sounding sirens occurs approximately 45 minutes after declaration of GE. From the MELCOR analysis, core damage is evidenced by the first fission product gap release about three hours into the event with the SGTR occurring 30 minutes later. The duration of specific protective actions for each cohort is summarized in Figure 161.

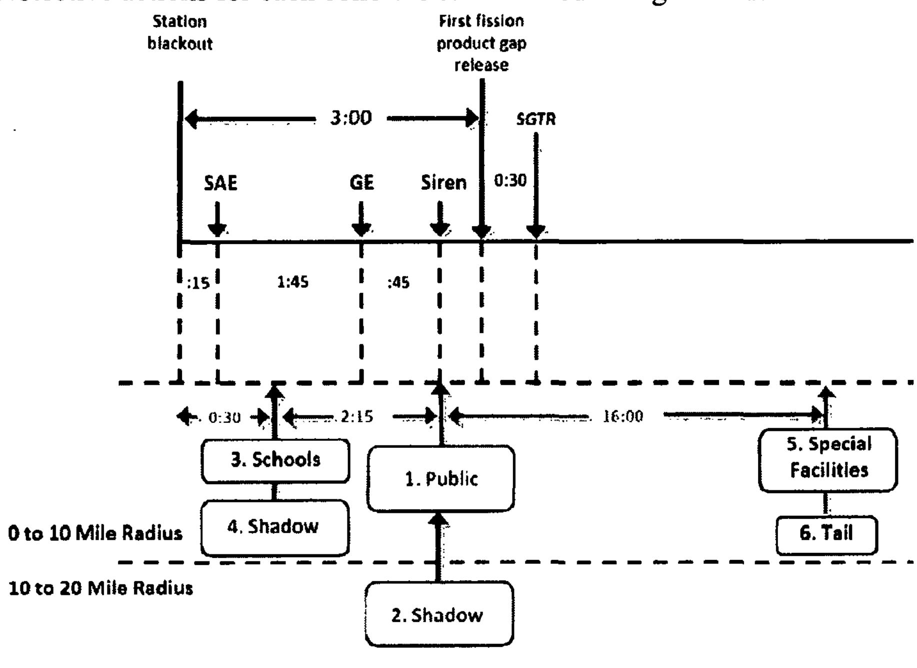

Figure 160 STSBO with TI-SGTR Emergency Response Timeline (Seismic Scenario)

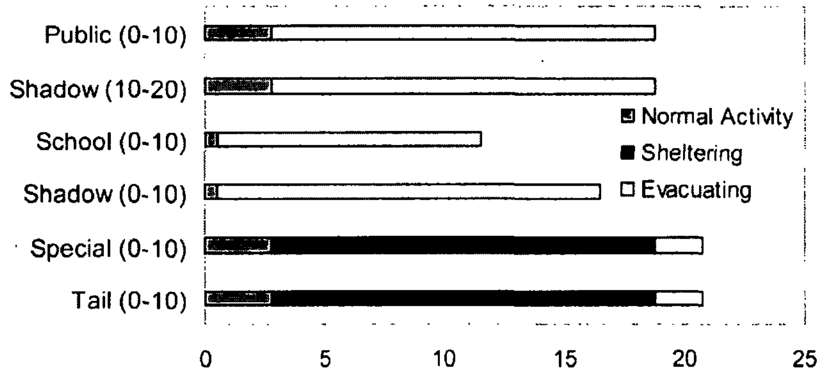

Figure 161 Protective Action Durations for STSBO with TI-SGTR (Seismic Scenario)

The timeline identifies points at which cohorts would receive instruction from OROs to implement protective actions. While protective actions within the EPZ can be modeled in accordance with procedures, assumptions were made that reasonably approximate those actions that could be taken due to the effects of the earthquake; however, the actual decisions made by OROs could differ.

The emergency response procedures for Surry provide for the sounding of sirens for a declaration of General Emergency. Back up batteries are available to support the sounding of sirens. It is assumed the large earthquake will be felt by everyone within the EPZ, and individuals will begin to prepare for an evacuation prior to receiving official notice.

Cohort 1: **0 to 10 Public.** The 0 to 10 Public is assumed to begin evacuating when the sirens sound. OROs would prepare and broadcast an EAS message, but the loss of power and infrastructure may limit the range of the broadcast. It is assumed that the effects of the earthquake are severe such that members of the public, knowing they live within an EPZ, begin preparations for evacuation shortly after the earthquake and are ready to leave when sirens sound.

Cohort 2: **10 to 20 Shadow.** This cohort is assumed to begin movement at the same time as the 0 to 10 Public after sirens have sounded within the EPZ and when widespread media broadcasts are underway. It is assumed that the shadow population increases to 30 percent of the public in the area beyond the EPZ.

Cohort 3: **0 to 10 Schools.** It is assumed schools take the initiative to prepare to evacuate prior to notification from the Virginia Department of Emergency Management. Buses would be mobilized, and it is assumed schools begin evacuating 30 minutes after the start of the incident; however, traffic congestion resulting from infrastructure failure causes a very slow evacuation speed. The analysis also considers that some drivers may not report due to an inability to get to the bus depot or need to address other immediate concerns. It is also assumed that given the magnitude of the earthquake, parents in the vicinity of the schools will pick up their children, reducing the need for a full compliment of buses. In addition, it is expected schools will respond as needed and make do with the resources that arrive to evacuate the children in a single wave. This may include placing more than the normal 50 to 70 children on a bus and / or using school administrator's and teacher's vehicles to augment transportation needs.

Cohort 4: 0 to 10 Shadow. This cohort is assumed to begin movement first. They experience the earthquake and quickly begin to evacuate avoiding the congestion.

Cohort 5: 0 to 10 Special Facilities. The Special Facilities cohort is assumed to depart at the same time as the evacuation tail. Special Facilities need to have transportation resources mobilized, some of which are very specialized. Inbound lanes on roadways will be useable for emergency support vehicles, but localized congestion will delay the arrival of specialized vehicles. Special Facilities are assumed to leave at the same time as the evacuation tail; however, as discussed earlier, this is a simplification of the analysis because Special Facilities would realistically evacuate individually as resources are available.

Cohort 6: 0 to 10 Tail. The Tail takes longer to evacuate for many valid reasons such as the need to return home from work to evacuate with the family; the need to shut down farming or manufacturing operations prior to evacuating; and for the earthquake, the need to move rubble or other items prior to evacuating. However, with the extent of damage within the Surry EPZ, the tail simply becomes a continuous extension of the evacuating public.

Cohort 7: Non-evacuating public. This cohort group represents the portion of the 0 to 10 public who may refuse to evacuate and is assumed to be 0.5 percent of the population.

Table 26 provides a summary of the evacuation timing for each cohort. The values in the table represent the minimum evacuation speeds corresponding to the area north and east of the site. In general, the cohorts in the seismic study have faster mobilization times but significantly slower evacuation speeds. The delay to shelter represents a delay before people get to the shelter, and delay to evacuation represents the length of the sheltering period prior to evacuation. These delays correspond to the different shielding factors that were applied to each cohort during these timeframes. The speeds and durations in this table represent constant movement speeds for the cohorts. These values are adjusted within each grid element of the WinMACCS model.

Table 26 Cohort Timing STSBO with TI-SGTR Including Speeds East (E) and West (W) of River

Cohort	Delay to Shelter DLTSHL (hr)	Delay to Evacuation DLTEVA (hr)	DURBEG (hr)	DURMID (hr)	ESPEED early (mph)		ESPEED middle (mph)	
					E	W	E	W
0-10 Public	2.75	0	0.25	15.75	5	15	0.7	2.1
10 to 20 Shadow	2.75	0	0.25	15.75	5	15	0.7	2.1
0-10 Schools	0.50	0	1.00	10.00	5	15	0.7	2.1
0 to 10 Shadow	0.50	0	0.25	10.00	10.0	30	0.7	2.1
0-10 Special Facilities	2.75	16	1.00	1.00	0.7	2.1	10.0	30.0
0-10 Tail	2.75	16	1.00	1.00	0.7	2.1	10.0	30.0

Non-evac	NA	NA	NA	NA	NA	NA	NA	NA

As indicated in Table 26, the evacuation speeds east of the James River are significantly slower than the speeds west of the river. The timeline identifies the point at which it is assumed cohorts begin to take action. The actions taken by each cohort last for a given period as indicated in the timing table. To model the difference in speeds between the east and west sides of the river, the speeds were developed for the east side and the WinMACCS multipliers were applied to the west side. Because a multiplier is used rather than calculated speeds for the west side of the river, the values are approximate.

6.6 Accident Response and Mitigation of Source Terms

SOARCA MELCOR analyses, which reflected best-estimate thermal hydraulics and accident progression parameters, were conducted assuming the onsite emergency response organization (ERO) takes no mitigative action other than to notify offsite authorities. The staff does not believe the licensee would not implement mitigative measures, and this is inconsistent with the realistic assumptions that have been used elsewhere in SOARCA. However, staff did not perform a human reliability assessment or a detailed seismic damage assessment for implementation of mitigative measures. The staff believed it appropriate to perform the sensitivity analysis to further understanding of core melt sequences, source term evolution and offsite response dynamics. Perhaps the most important objective of the sensitivity analyses is to quantify the benefit of mitigation enhancements. To further support the expectation of mitigative response, a detailed discussion of the expansive resources available to support a national incident is provided below.

This analysis describes the likely national response to a severe nuclear plant accident and provides a basis for truncating the release no later than 48 hours after the accident begins. The discussion presents a timeline for bringing resources onto the Surry site in order to flood the reactor building to a level above a hypothetically melted core. Specific options are discussed but there could be a number of additional efforts led by multiple organizations should it be necessary. While the staff believes it is most likely that plant personnel would mitigate the accident before core melt, if efforts were unsuccessful the national level response would mitigate the source term.

The NRC has onsite inspectors that are available to provide first hand knowledge of accident conditions. Concurrently, the NRC region office, would send a full Site Team to the licensee's EOF to support the response. A Site Team would include reactor safety experts and protective measures experts to review actions taken to mitigate the accident and to review protective action decisions that will be recommended to the public to assure the most appropriate actions are taken. Although a Site Team would arrive after protective actions within the EPZ have been initiated, the Site Team would be available to support decisions on mitigation measures.

Surry is part of the Dominion fleet which includes a remote EOF that would be activated and has access to fleet wide emergency response personnel and equipment, including equipment from sister plants following 10 CFR 50.54(hh) reactor security requirements to mitigate the effects of large fires and explosions. Significant resources would be made available to the site to mitigate the accident. In addition to those directly involved in the incident and those agencies that fully test and exercise response plans, the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site. These efforts would provide knowledgeable personnel and an extensive array of equipment would be available and as such are considered in the decision to truncate the release at 48 hours.

The National Response Framework (NRF) establishes a coordinated response of national assets. Under established agreement, the DHS would be the coordinating agency and NRC would be the primary cooperating agency for an event in which a General Emergency is declared. Some of the other agencies cooperating in an incident include EPA, FEMA, HHS, and any other Federal agency that may be needed. The assets of the Department of Energy (DOE) would be activated and brought to bear on the accident. Every licensee participates with many of these organizations in a full onsite and offsite exercise biennially. The NRC has an extensive well trained and exercised emergency response capability that would support, and under unusual circumstances, direct licensee efforts. Communications systems require battery backup in accordance with 10 CFR 50.54 Appendix E, and multiple communication bridge lines would be established to facilitate structured communication among the various response teams. Satellite phones, cell phones, radios, and other means are available for those instances where communications have been affected.

Offsite mitigation strategies are different from onsite mitigation strategies as indicated in

Table 27. An onsite mitigation strategy is specific to the accident scenario, has only onsite resources, and has a limited amount of time to prevent core melt and radiological release. For the unmitigated cases, the SOARCA project assumes onsite mitigative efforts are not effective and that a radiological release occurs. Offsite mitigation strategies would bring national resources to the site and take more time than onsite measures. The mitigation strategy considered in this evaluation is to fill the containment building with water and cover core debris in order to scrub the radiological release.

Table 27 Onsite and Offsite Mitigation Strategies

	Onsite Mitigation	Offsite Mitigation
Objective	Prevent release	Stop release
Strategy	Specific to accident	Cover molten core
Time	Less	More
Equipment Available	Onsite	Onsite & portable offsite

6.6.1 External Resources

The primary focus of the site and utility ERO would be mitigating core damage, and State and local resources would focus on the public evacuation. However, it is typical, as demonstrated in drills and exercises, for EROs to develop contingency plans in case initial onsite mitigative actions are not successful. The NRC ERO would focus on protection of the public and methods to reduce consequences reviewing the licensee and ORO information, actions, and decisions while performing independent analyses. If the site ERO is not successful with the onsite mitigative actions, as the sensitivity study assumes, various EROs would be considering in parallel the availability of portable power and pumping capacity from offsite locations. Virginia has a statewide mutual aid agreement for assistance from every fire department in the commonwealth.

The Surry volunteer fire department and Rushmere volunteer fire department are both 15 miles from the site. The fire departments are in opposite directions, but they share a six-mile stretch of road to the power plant. Figure 162 shows Surry and Rushmere Volunteer Fire Departments in relation to Surry Nuclear Power Plant.

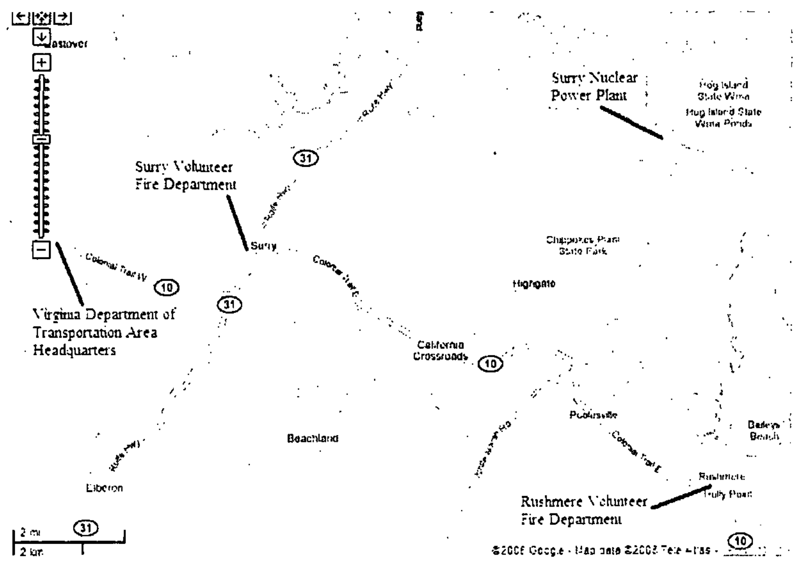

Figure 162 Surry and Rushmere Fire Departments

The Surry volunteer fire department practices annual response drills with Surry Nuclear Power Plant. This fire department has five pumper trucks, each with a capability to pump 500 to 1000 gpm and has the ability to draft water directly from the river. A fire truck typically provides a pressure of up to 125 psi when limited by the firefighter controlling the nozzle. Fire hoses have a test pressure of at least 300 psi and a burst pressure of between 600 and 1000 psi. To obtain an Underwriter's Laboratory (U.L.) Certification, a fire pump inside the truck must meet the following rated capacity:

Table 28 Rated Flowrate of Fire Pumps

Pressure (psi)	Rated Flowrate
150	100%
165	100%
200	70%
250	50%

Additional resources are available through the State of Virginia Office of Emergency Management (OEM) which has an emergency services contract for the delivery of significant amounts of equipment within eight hours. OEM has access to high-pressure, high capacity pumps with 8 and 12 inch flanges that could readily move the water necessary for containment flooding. Such equipment is available for immediate State emergency use from many locations including Norfolk, Elizabeth City, and Raleigh.

The initiating event for the reactor accident is an earthquake in close proximity to the plant. This event causes significant ground motion and damage to certain types of structures. Long span bridges could be lost, but smaller bridges, culverts and most housing stock would likely survive the event. The loss of bridges as far as 7 miles from the site is assumed likely due to the close proximity necessary for this earthquake to affect the site in the manner assumed. However, the loss of these bridges would not affect the ability to truck equipment into the site. Roads themselves would likely not be compromised due to the earthquake.

The EPZ environment is generally rural, and the roadway system is substantial. There are small bridges and culvert crossings on the plant access route that could be damaged, but these would not prevent passage by heavy trucks. A Virginia department of transportation area headquarters is 3 miles from the Surry fire department and has six dump trucks, two front-end loaders, a road grader, and other heavy equipment. Staff at the area headquarters have the capability to open roads and could immediately respond. Even if the culverts were to collapse, personnel at the transportation facility believe they could fix the road so that equipment could pass in less than 10 hours. In addition, Norfolk Naval Station has airlift resources that could be used to bring equipment into the plant. Fire trucks, which can weigh 35,000 lbs, are too heavy for air lift with MH-53 Helicopters which have a lift capacity of about 13,000 pounds. Airlift is not expected to be necessary, but if it was, the EROs would work together to identify appropriate pump and electrical equipment resources. The State has fire pumps and generators of all types available, and these could be on site within about 10 hours.

In addition, there are six fireboats in the Newport News/Norfolk area each with the pumping capacity of 1500 to 3000 gpm, and a marine fire fighting capability exists for three more boats equipped for firefighting, if needed. These fireboats have tremendous pumping capacity but may need to use long lengths of hose to support the site. It is likely that fire trucks and other portable equipment can be trucked onto the site more rapidly than the fire boats can be deployed, but they are a viable option should they be needed. EROs may pursue several options to ensure success.

The state of Virginia has access to diesel generators that range from 15 kW to 15 MW and electrical generators are available from commercial sources. Large portable generators could be brought onto the site within 10 hours.

The timing of the hypothetical radiological release is scenario dependant, but when it occurs the site would be contaminated and working conditions more difficult. However, plant staff are trained in radiological work and the full staff of health physics technicians would be available. Within about 12 hours, staff from Dominion fleet plants could be at the site as well as staff from the DOE. Additional radiological technicians could likely be obtained from neighboring plants or perhaps from the Norfolk Naval Shipyard approximately 40 miles away should they be needed.

6.6.2 Mitigation Strategies

Strategies focus on filling the containment with water above the molten core preferably by injection of water into the reactor vessel or into containment via the containment spray system. This action would reduce the release potential through the cooling and scrubbing action of water. The site, utility and NRC EROs would identify various water injection methods corresponding to the damage circumstances. An effective option may be to inject water via the containment spray system. The two containment spray systems at Surry each have the capacity to inject more than 3000 gpm into containment. This would have an immediate effect of lowering pressure and scrubbing radionuclides from the air and would suppress the release. The containment spray system has 8 inch flanges which can accept a milled flange and fire hose arrangement. The containment spray system has two sections: the containment spray (CS) subsystem and the recirculation spray (RS) subsystem. The recirculation spray subsystem also has two sections, inside and outside. Both the inside recirculation spray (ISRS) and the outside recirculation spray (OSRS) are designed for 100 percent capacity.

There are several water sources but earthquake based scenarios assume that most water supply tanks fail. If nothing else is available, the intake or discharge canals would provide a water source. The site ERO would perform a damage survey to determine tank status and either find usable tanks or make repairs. The utility would connect a water tank to a containment spray connection and make up to the water tank from the river if necessary. It may also be possible to draw water directly from the river or the fire suppression system through some installed or field fabricated means. The availability of portable electrical power sources was discussed above, and portable power would likely be available at about the same time tank repairs and pumping capacity would be arranged.

Mitigation of the ISLOCA scenario is unique in that there is no widespread onsite damage to prevent the use of normal systems and water sources. The containment spray system would be essentially available as installed without the need for any ad hoc actions, i.e., normal procedures could be used to initiate the system. Actually, most emergency core cooling systems would be available to prevent core damage or truncate the release. The utility could simply align one of many water sources to one of many injection systems for truncating a release.

In addition to the containment spray pumps, Surry has a portable low-pressure pump with a capacity of 2000 gpm, and two portable high-pressure pumps with a capacity of 150 to 400 gpm (depending on the pressure). There is also a diesel-driven fire pump available at the Surry site. The utility has procedures to align installed pumps, portable pumps or the fire pump to the containment spray system and would use this strategy for the LTSBO and STSBO accident sequences. ISLOCA and STSBO TI-SGTR on the other hand are bypass events, so this strategy would not stop the release of radioactive material. However, connecting the fire pump to the containment spray system could be effective for any sequence after the molten core breaches the vessel.

A high-pressure pump would be more effective because of the elevation of the containment spray nozzles and containment pressure. A high-pressure, high capacity fire pump (available from the State) connected to either the CS or the OSRS subsystems would immediately start to lower the pressure, scrub the volatiles, and eventually cover the core. Figure 163 shows the analyzed pressure of the unmitigated STSBO accident.

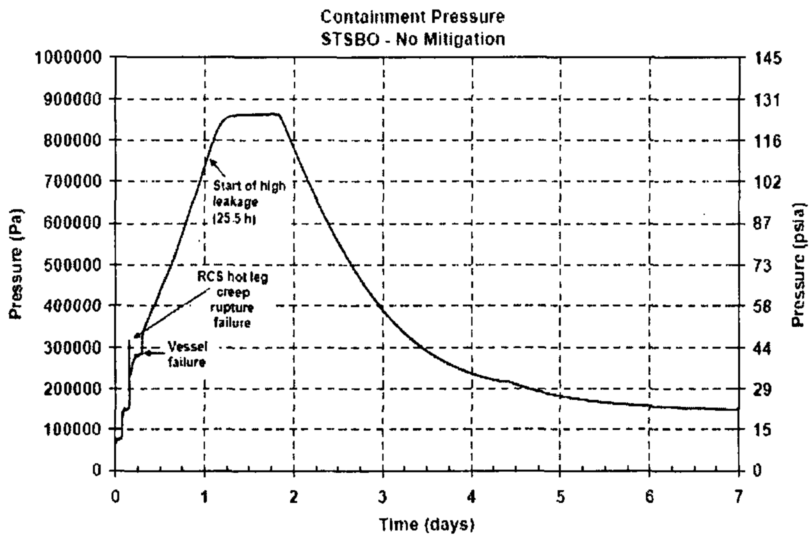

Figure 163 Containment Pressure for Unmitigated STSBO

The ERO could also use a fire truck for pumping capacity. The utility has guidelines for using a fire truck to feed plant systems such as auxiliary feed water. However, to obtain a desirable flow rate in containment spray it would be necessary to gang fire trucks, and high capacity high-pressure portable pumps may be a better solution.

Another method for release truncation would be to inject water directly into the primary system. This may be preferred because there is little elevation head to overcome and water would flow directly to the breach in the reactor vessel bottom head and cover the molten core scrubbing volatiles from the release. However, containment pressure will continue to increase due to decay heat and the site ERO will need a means to remove heat (e.g., air coolers or heat exchangers via containment spray system). As necessary, all EROs would work together to identify other measures for mitigation interacting with plant personnel who know the plant well and may identify innovative solutions to inject water into containment.

The staff noted a difference between the two Surry units with respect to the volume of water needed to cover the molten core. Unit-1 has an opening in the reactor vessel pedestal so that water can flow between the reactor cavity and the rest of containment but Unit-2 does not. In Unit-2 the water cannot overflow into the reactor cavity until it reaches a level 25 feet from the bottom of the sump. In Unit-2 about 1.75 million gallons of water, as extrapolated from Figure 164, would be necessary before it would reach the cavity.

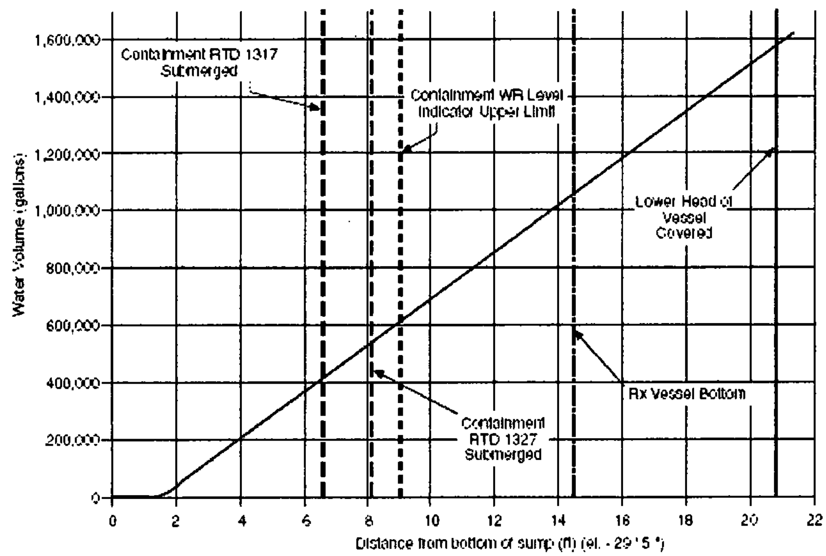

Figure 164 Containment Water Level vs. Volume

Table 29 Cumulative Water Voume vs. Elevation for Unit 1

ELEVATION (sea level)	Distance from bottom of sump (ft)	Water volume (gal)	Water volume (ft^3)
-29' 4 7/8"	0	0	0
-29' 4"	0.07	52	7
-29' 3"	0.16	108	14.4
-29' 3"	0.24	164	21.9
-29' 3"	0.41	277	37
-28' 7"	0.82	558	74.6
-28' 2"	1.24	965	129
-27' 10"	1.57	1,831	244.8
-27' 7"	1.82	9,113	1,218.2
-26' 7"	2.82	87,990	11,762.6
-25' 7"	3.82	166,867	22,306.9
-24' 7"	4.82	245,744	32,851.3
-23' 7"	5.82	324,620	43,395.5
-22' 7"	6.82	404,997	54,140.4
-21' 10 1/2"	7.53	461,933	61,751.6
-20' 1"	9.32	600,688	80,300.5
-10' **	19.41	1,381,588	184,692.0

** last data point extrapolated

For Unit-2, 1.75 million gallons would be necessary to fill the building to cover the molten core. The location of radiological release from containment may be expected in the vicinity of the equipment hatch, but the location of the increased leakage is not expected to depressurize the containment immediately. Therefore, in the unlikely event that the increased leakage is at or below the level of the water, the containment would still be capable of retaining water. In any case, if water is injected into containment through containment spray or primary injection it would still dramatically suppress the release.

Figure 165 shows the Surry containment. Figure 166 shows the time it would take to pump 2 million gallons of water at various pumping capacities. At 3,000 GPM, it would take about 10 hours.

PIPING AND COMPONENTS ELEVATIONS SPRAY SYSTEMS

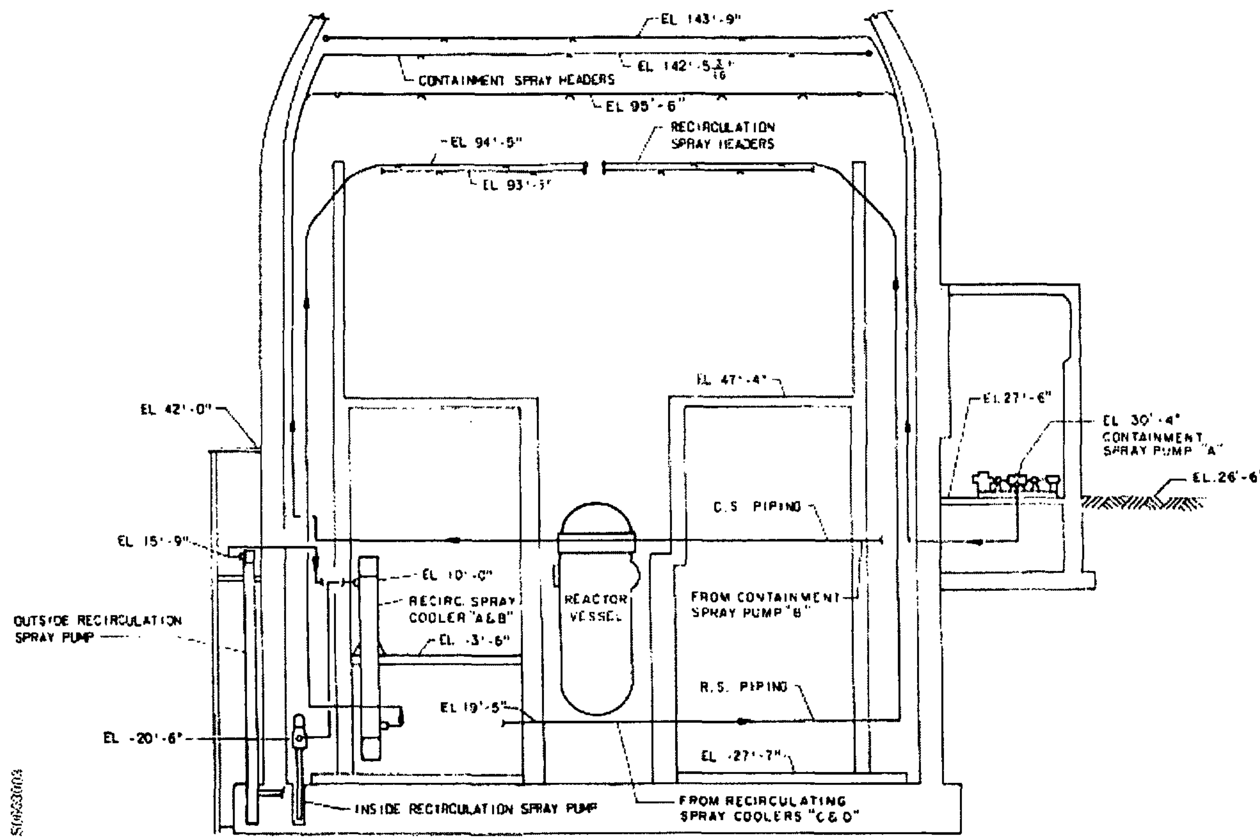

Figure 165 Surry Containment System

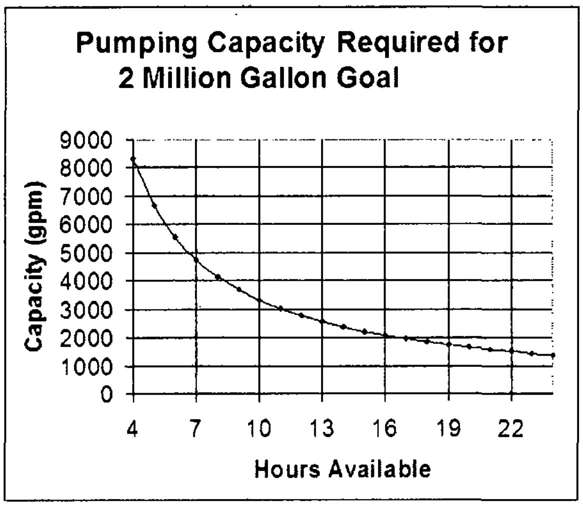

Figure 166 Pumping Capacity

6.6.3 Truncation Summary

The NRC believes the onsite mitigative actions would limit core damage and reduce the release magnitude; however, this analysis is not definitive. Some of the actions identified are described within emergency response plans and some are ad hoc. The availability of the equipment is likely but not certain. If core damage is not prevented from the onsite mitigative measures, this truncation analysis demonstrates the offsite mitigative measures would likely take no more than 48 hours from accident initiation to truncate the radiological release. Based on the approach provided, which considers in detail the timing of many activities underway during an event, it is reasonable to bound the truncation of the accidents at 48 hours.

6.7 Emergency Preparedness Summary and Conclusions

Advancements in consequence modeling provide an opportunity to integrate realism in the application of protective action decisions applied for discrete population segments to represent implementation of protective actions. To best utilize these advancements, detailed information was developed and / or obtained from response of OROs to assure the quality of input data. Consequence modeling now provides for analysis of individual population segments and a user interface has been added to the consequence model to facilitate input detailed information that incorporates differences in the response to protective actions by various population segments. These advancements are significant because they now allow the modeling of response activities, timing of decisions, and implementation of protective actions across a wide range of population segments.

Licensees develop ETEs to support emergency planning and help assure the most appropriate protective actions are implemented in an emergency. These ETEs provide detailed information regarding the evacuation of the general public, schools, special facilities and the evacuation tail. The improvements to consequence modeling and improved understanding of implementation of protective actions now allows use of this detailed information when modeling potential consequences of reactor accidents. For the first time, consequence modeling can represent the actions of OROs and the timing of public response to an emergency with a defensible basis provided for the timing of these actions.

In this analysis, 6 cohorts were modeled for each of the accident scenarios and a seventh cohort was added for the seismic analysis. Protective action factors were applied to each specific cohort.

- For the general public, shielding factors appropriate for the region were applied during normal, sheltering, and evacuation times and speeds were derived from the Surry 2001 ETE.
- Schools are notified directly in accordance with offsite emergency response plans and buses are mobilized to support expedited evacuation of schoolchildren. Mobilizing school resources early allows the evacuation of schools to occur first, prior to roadways becoming congested from evacuation of the general public. Therefore, the speed of the school cohort was established based on relatively little traffic on the roadways at the time.
- Special facilities are also notified early, but respond quite differently than schools. Transportation resources for special facilities are quite specialized, can be limited, and typically take extra time to mobilize. Evacuation of these facilities starts later than schools and continues longer than the evacuation of the general public. This is because transportation resources take longer to mobilize and make return trips to evacuate each facility independently. A benefit of special facilities is the robust nature of the structures of these nursing homes, hospitals, etc. The shielding protection values are increased for these facilities. For this analysis, this cohort is sheltered until the point at which evacuation begins which for calculation purposes was set at the same time as the evacuation tail begins.
- The evacuation tail was treated as a separate cohort and includes those members of the public who take longer to evacuate and are the last to leave the area. Indoor shielding values were applied to this and they were evacuated late in the emergency moving at faster speeds because of the lower volume of traffic on the roadways at this time. The timing of the evacuation tail was derived from the ETE as the time at which the last 10 percent of the public begin to evacuate.
- Recent data published by the NRC provides a quantitative value of the shadow evacuation. The shadow evacuation was modeled representative of their occurrence in previous large scale evacuation.
- Consistent with Sample Problem A, evacuees received a dose until they traveled to a point 10 miles beyond the analysis area, which for SOARCA was 30 miles.
- For the seismic analysis, it may be expected that a shadow evacuation of residents from within the EPZ may occur prior to the issuance of an evacuation order. This additional shadow evacuation was included in the analysis.
- A non-evacuation cohort was also included in the analysis assuming that a very few members of the public may refuse to evacuate. Normal shielding values were applied to this cohort.

The Surry EALs were obtained for each of the accident scenarios modeled to best reflect the timing of the declaration of SAE and GE.

The following accident scenarios were modeled:

1. STSBO
2. Unmitigated STSBO with TI-SGTR
3. Mitigated STSBO with TI-SGTR
4. LTSBO
5. ISLOCA
6. Sensitivity 1, ISLOCA with evacuation to 16 miles
7. Sensitivity 2, ISLOCA with evacuation to 20 miles
8. Sensitivity 3, ISLOCA with a delay in implementation of protective actions
9. Seismic Unmitigated STSBO with TI-SGTR

For each of these accident scenarios, the specific EAL information and cohort movement was applied and the WinMACCS files were complied for the consequence analysis.

The sensitivity analyses were performed to identify differences when varying selected parameters. This included expanding the limits of the evacuation and adjusting the timing of implementation of protective actions. In the first sensitivity analysis, the limits of the evacuation were extended to 16 miles. In order to evaluate such a protective action, an ETE was developed using OREMS. Each cohort was adjusted to reflect the appropriate distance from the plant. A second sensitivity analysis was conducted assuming an evacuation of 20 miles from the plant. The ETE was developed using OREMS, and for this case, there was no shadow evacuation assumed due to the extreme distance from the point of the accident.

All of the initiating events for these accidents was loss of power and the resulting EALs were similar. An additional sensitivity analysis was conducted to assess the sensitivity of the timing of ORO decisions to consequences. This required increasing the delay times for cohorts to take action.

7. OFF-SITE CONSEQUENCES

7.1 Introduction

The MACCS2 consequence model was used to calculate the effects of offsite doses to members of the public. MACCS2 was developed at Sandia National Laboratories for the NRC for use in probabilistic risk assessments for commercial nuclear reactors to simulate the impact of accidental atmospheric releases of radiological materials on humans and on the surrounding environment. The principal phenomena considered in MACCS2 are atmospheric transport using a straight-line Gaussian plume model, short-term and long-term dose accumulation through several pathways including cloudshine, groundshine, inhalation, deposition onto the skin, and food and water ingestion. The ingestion pathway was not treated in the analyses reported here because uncontaminated food and water are in abundant supply within the US and it is unlikely that the public would consume contaminated food or water. The doses that are included in the reported risk metrics are as follows:

- Cloudshine during plume passage
- Groundshine during the emergency and long-term phases from deposited aerosols
- Inhalation during plume passage and following plume passage from resuspension of deposited aerosols. Resuspension is treated during both the emergency and long-term phases.

Additional enhancements were made to MACCS2 [24] as an element of the SOARCA project. In general, these enhancements reflect recommendations obtained during the SOARCA external review and also reflect needs identified by the broader consequence analysis community. The code enhancements that were done for SOARCA are primarily to enhance fidelity, improve code performance, and enhance existing functionality. They do not represent a major phenomenological model development effort. Nevertheless, these enhancements are anticipated to have a significant effect on the fidelity of the analyses performed under the SOARCA project.

MACCS2 previously allowed up to three emergency-phase cohorts in the EARLY module plus a long-term cohort in the CHRONC module. Each emergency-phase cohort represents a uniform group of the population who behave in the same manner. For example, a cohort might represent a fraction of the population who rapidly evacuate after officials instruct them to do so. To create a higher-fidelity model for SOARCA, the number of emergency-phase cohorts was increased as described in the previous Emergency Response Section. This allowed significantly more variations in emergency response, e.g., variations in preparation time prior to evacuation and more accurately reflects the movement of the public during an emergency. In a similar way, modeling evacuation routes using the network-evacuation model adds a greater degree of realism than in previous analyses that used the simpler, radial-evacuation model.

7.2 Surry Source Terms

Brief descriptions of the source terms for the Surry accident scenarios are provided in Table 30. For comparison, the largest source term from the Sandia Siting Study (SST1) [43] is also shown. Of the Surry source terms shown in the table, the unmitigated interfacing-systems loss-of-coolant accident (ISLOCA) is the largest in terms of release magnitude, but the release begins more than 9 hours after accident initiation. Release begins earliest for the two thermally induced steam generator tube rupture scenarios (TISGTR), only 3.6 hr after accident initiation, but the magnitudes are very small. The unmitigated short-term and long-term station blackout (STSBO and LTSBO) sequences begin very late in time and have very small release magnitudes.

By comparison, the SST1 source term is significantly larger in magnitude than the largest of the Surry source terms. Moreover, it begins only 1.5 hours after accident initiation, about 2 hours earlier than the fastest release of the set of Surry source terms. Thus, it is clear that the current understanding of accident progression has lead to a very different characterization of release signatures than was current at the time of the Sandia Siting Study [43].

Table 30 Brief Source-Term Description for Surry Accident Scenarios and the SST1 Source Term from the Sandia Siting Study

	Integral Release Fractions by Chemical Group									Release Timing	
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Unmitigated STSBO	0.518	0.001	0.000	0.006	0.006	0.000	0.000	0.000	0.000	25.5	48.0
Unmitigated STSBO with TISGTR	0.592	0.004	0.000	0.009	0.007	0.000	0.001	0.000	0.000	3.6	48.0
Mitigated STSBO with TISGTR	0.085	0.004	0.000	0.005	0.004	0.000	0.001	0.000	0.000	3.6	48.0
Unmitigated LTSBO	0.537	0.000	0.000	0.003	0.006	0.000	0.000	0.000	0.000	45.3	72.0
Unmitigated ISLOCA	0.942	0.092	0.002	0.095	0.111	0.002	0.024	0.000	0.000	9.2	48.0
SST1	1.000	0.670	0.070	0.450	0.640	0.050	0.050	0.009	0.009	1.5	3.5

For comparison purposes, a consequence analysis using the old SST1 source term is presented in this chapter and in the summary report. This allows a direct comparison, using the same modeling options and result metrics, between the outdated SST1 source term and the current, best-estimate source terms.

7.3 Consequence Analyses

Five baseline accident scenarios and three sensitivity analyses are reported in this chapter. Two sensitivity analysis for the unmitigated STSBO scenario show the influence of the size of the evacuation zone and of a delay in emergency response on the risks for the unmitigated STSBO. The third sensitivity analysis is based on the unmitigated TISGTR scenario and provides an estimate of the effect of seismic activity on emergency response and effect on risk. Also, a separate analysis of the SST1 source term [43] (shown in Table 30) allows older source term assumptions to be compared with the current state-of-the-art methods for source term evaluation using otherwise equivalent assumptions and models. This analysis does not try to reproduce the Sandia Siting Study results; it merely overlays the older source term onto what are otherwise SOARCA assumptions for dose-response modeling, emergency response, etc.

The results of the consequence analyses are presented in terms of risk to the public for each of the five accident scenarios that were identified for Surry. Both unconditional and conditional risks are tabulated. The conditional risks assume that the accident occurs and show the risks to an average individual as a result of the accident. The unconditional risks are the product of the core damage frequency and the conditional risks. Unconditional risks are estimates of the likelihood of an individual within a specified radius of the plant receiving an excess fatal cancer or early fatality per year of plant operation from a potential plant accident. The word excess carries the meaning of additional risk over and above normal cancer risk. Conditional risks are expressed as probabilities that are dimensionless; unconditional risks are expressed as frequencies (i.e., probabilitiws per uear of plant operation).

The risk metrics are latent-cancer-fatality and early-fatality risks to residents in circular regions surrounding the plant. Population and economic data used in these analyses are projected for the year 2005. These risk values are averaged over weather data for the year 2004. They are also averaged over the entire population within the circular region. The risk values represent the predicted number of fatalities divided by the population for four choices of dose-truncation level. Thus, these risk metrics account for the distribution of the population within the circular region and for the interplay between the population distribution and the wind rose probabilities. The risk metrics do not account for typical commuting patterns; rather, they are based on the locations where people reside.

7.3.1 Unmitigated Long-Term Station Blackout

The unmitigated long-term station blackout (LTSBO) scenario is similar to the STSBO scenario described above except that cooling of the primary is maintained until the batteries wear down, so degradation of the fuel and subsequent failure of the pressure boundary are delayed. The source term is later and smaller for this scenario than for the STSBO. In fact, the source term for this scenario begins more than 45 hr after accident initiation. This source term is also unique in that it was truncated at 72 hr rather than 48 hr, like all of the other source terms.

Table 31 displays the conditional, mean, latent-cancer-fatality risks to residents within a set of concentric circular areas centered at the Surry site for the long-term station blackout (LTSBO) scenario. Four values of dose-truncation level are shown in the table: linear, no threshold (LNT), i.e., a dose-truncation level of zero; 10 mrem/yr; the average, annual, US-background radiation (including average medical radiation) of 620 mrem/yr; and the Health Physics Society (HPS) recommended dose truncation of 5 rem/yr, with a lifetime limit of 10 rem.

Table 32 is analogous to Table 31, but displays unconditional rather than the conditional risks. In the case of the Surry LTSBO scenario, the mean core damage frequency is $2 \cdot 10^{-5}$/yr. This frequency is used to scale the results in Table 32, as described above.

The values in Table 31 are plotted in Figure 167. The plot shows that for all dose-truncation levels, the risk is greatest for those closest to the plant and diminishes monotonically as distance increases. The trends shown in this figure are the same as those shown in Figure 169 for the unmitigated STSBO.

Table 31 Conditional, Mean, Latent-Cancer-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Probabilities are for the Long-Term Station Blackout Scenario. Mean Core Damage Frequency is $2 \cdot 10^{-5}$/yr.

Radius of Circular Area (mi)	*LNT*	*10 mrem/yr*	*620 mrem/yr*	*5 rem/yr; 10 rem lifetime*
10	$4.7 \cdot 10^{-5}$	$2.7 \cdot 10^{-5}$	$4.0 \cdot 10^{-7}$	$1.5 \cdot 10^{-9}$
20	$2.6 \cdot 10^{-5}$	$1.5 \cdot 10^{-5}$	$1.4 \cdot 10^{-7}$	$4.1 \cdot 10^{-10}$
30	$1.7 \cdot 10^{-5}$	$9.6 \cdot 10^{-6}$	$7.8 \cdot 10^{-8}$	$2.3 \cdot 10^{-10}$
40	$1.1 \cdot 10^{-5}$	$5.7 \cdot 10^{-6}$	$4.0 \cdot 10^{-8}$	$1.2 \cdot 10^{-10}$
50	$8.1 \cdot 10^{-6}$	$4.2 \cdot 10^{-6}$	$2.7 \cdot 10^{-8}$	$7.9 \cdot 10^{-11}$

Table 32 Unconditional, Mean, Latent-Cancer-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks are for the Long-Term Station Blackout Scenario. Mean Core Damage Frequency is $2 \cdot 10^{-5}$/yr.

Radius of Circular Area (mi)	**LNT**	**10 mrem/yr**	**620 mrem/yr**	**5 rem/yr; 10 rem lifetime**
10	$7.1 \cdot 10^{-10}$	$4.1 \cdot 10^{-10}$	$6.0 \cdot 10^{-12}$	$2.2 \cdot 10^{-14}$
20	$3.8 \cdot 10^{-10}$	$2.2 \cdot 10^{-10}$	$2.1 \cdot 10^{-12}$	$6.2 \cdot 10^{-15}$
30	$2.6 \cdot 10^{-10}$	$1.4 \cdot 10^{-10}$	$1.2 \cdot 10^{-12}$	$3.4 \cdot 10^{-15}$
40	$1.6 \cdot 10^{-10}$	$8.6 \cdot 10^{-11}$	$6.0 \cdot 10^{-13}$	$1.8 \cdot 10^{-15}$
50	$1.2 \cdot 10^{-10}$	$6.3 \cdot 10^{-11}$	$4.1 \cdot 10^{-13}$	$1.2 \cdot 10^{-15}$

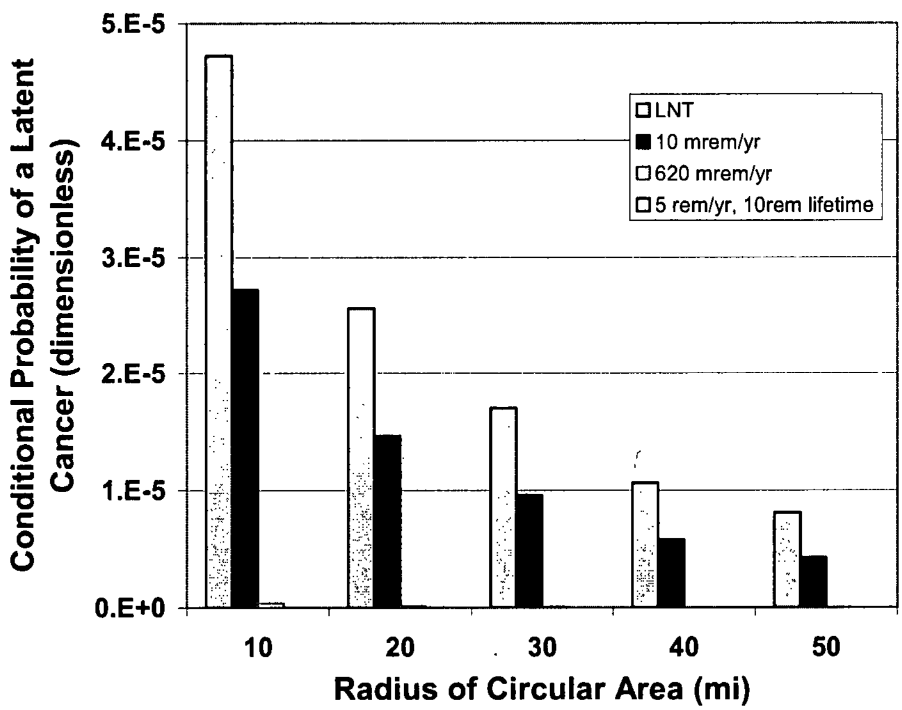

Figure 167 Conditional, mean, latent-cancer-fatality probabilities from the Surry LTSBO scenario for residents within a circular area of specified radius from the plant. The plot shows four values of dose-truncation level.

Figure 168 shows the latent-cancer-fatality risks as a function of the radius from the plant for the emergency phase (EARLY), the long-term phase (CHRONC), and the combined phases (sum of the two). The figure shows that the emergency response is very effective within the EPZ and that the long-term phase dominates the overall risks. The habitability (i.e., return) criterion, which is implemented as a limit of 4 rem in the first 5 years after returning to live in a residential area, controls the overall risk to the public for this accident scenario. The trends shown in this figure are the same as those shown in Figure 170 for the unmitigated STSBO. In both cases, evacuees have ample time to evacuate before release begins.

All of the emergency-phase risk within the 10-mile EPZ is for the nonevacuating cohort. This is because all of the other cohorts avoid exposure to the plume. Thus, for this accident scenario, the residents within the EPZ who comply with the request to evacuate have no increased risk prior to the long-term phase.

The prompt-fatality risks are identically zero for this accident scenario. This is because the release fractions (shown in Table 30) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5% of the population that does not evacuate.

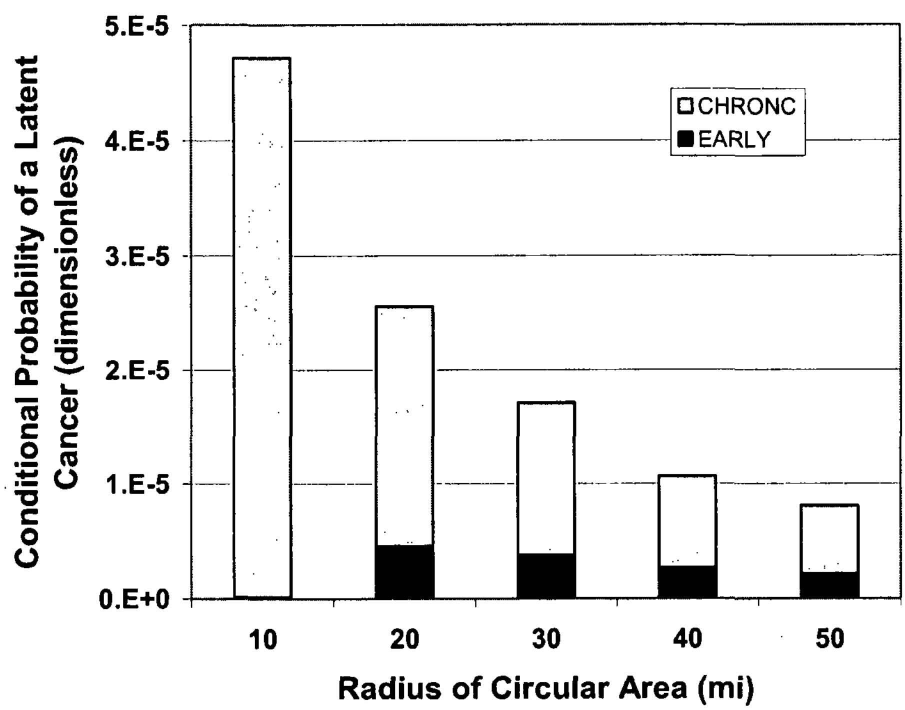

Figure 168 Conditional, mean, LNT, latent-cancer-fatality probabilities from the Surry LTSBO scenario for residents within a circular area of specified radius from the plant. The

plot shows the risks from the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined.

7.3.2 Unmitigated Short-Term Station Blackout

Table 33 displays the conditional, mean, latent-cancer-fatality risks to residents within a set of concentric circular areas centered at the Surry site for the unmitigated short-term station blackout (STSBO) scenario. Four values of dose-truncation level are shown in the table: linear, no threshold (LNT), i.e., a dose-truncation level of zero; 10 mrem/yr; the average, annual, US-background radiation (including average medical radiation) of 620 mrem/yr; and the Health Physics Society (HPS) recommended dose truncation of 5 rem/yr, with a lifetime limit of 10 rem.

The HPS dose-truncation level is more complex that the others because it involves both annual and lifetime limits. According to the recommendation, annual doses below the 5-rem truncation level do not need to be counted toward health effects; however, if the lifetime dose exceeds 10 rem, all annual doses, no matter how small, count toward health effects.

Table 34 is analogous to Table 33, but displays unconditional rather than the conditional risks. In the case of the Surry unmitigated STSBO, the mean core damage frequency is $2 \cdot 10^{-6}$/yr. This frequency is used to scale the results in Table 34, as described above.

The values in Table 33 are plotted in Figure 169. The plot shows that for all dose-truncation levels, the risk is greatest for those closest to the plant and diminishes monotonically with distance.

Table 33 Conditional, Mean, Latent-Cancer-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Probabilities Are for the Unmitigated STSBO Scenario, which Has a Mean Core Damage Frequency of $2 \cdot 10^{-6}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$9.4 \cdot 10^{-5}$	$7.0 \cdot 10^{-5}$	$3.4 \cdot 10^{-6}$	$1.4 \cdot 10^{-8}$
20	$4.8 \cdot 10^{-5}$	$3.3 \cdot 10^{-5}$	$1.5 \cdot 10^{-6}$	$4.9 \cdot 10^{-9}$
30	$3.2 \cdot 10^{-5}$	$2.1 \cdot 10^{-5}$	$8.4 \cdot 10^{-7}$	$2.7 \cdot 10^{-9}$
40	$2.0 \cdot 10^{-5}$	$1.2 \cdot 10^{-5}$	$4.3 \cdot 10^{-7}$	$1.4 \cdot 10^{-9}$
50	$1.5 \cdot 10^{-5}$	$8.9 \cdot 10^{-6}$	$2.9 \cdot 10^{-7}$	$9.4 \cdot 10^{-10}$

Table 34 Unconditional, Mean, Latent-Cancer-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks Are for the Unmitigated STSBO Scenario, which Has a Mean Core Damage Frequency of $1.5 \cdot 10^{-6}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$1.4 \cdot 10^{-10}$	$1.1 \cdot 10^{-10}$	$5.1 \cdot 10^{-12}$	$2.1 \cdot 10^{-14}$
20	$7.2 \cdot 10^{-11}$	$4.9 \cdot 10^{-11}$	$2.3 \cdot 10^{-12}$	$7.3 \cdot 10^{-15}$
30	$4.8 \cdot 10^{-11}$	$3.1 \cdot 10^{-11}$	$1.3 \cdot 10^{-12}$	$4.0 \cdot 10^{-15}$
40	$2.9 \cdot 10^{-11}$	$1.8 \cdot 10^{-11}$	$6.5 \cdot 10^{-13}$	$2.1 \cdot 10^{-15}$
50	$2.2 \cdot 10^{-11}$	$1.3 \cdot 10^{-11}$	$4.4 \cdot 10^{-13}$	$1.4 \cdot 10^{-15}$

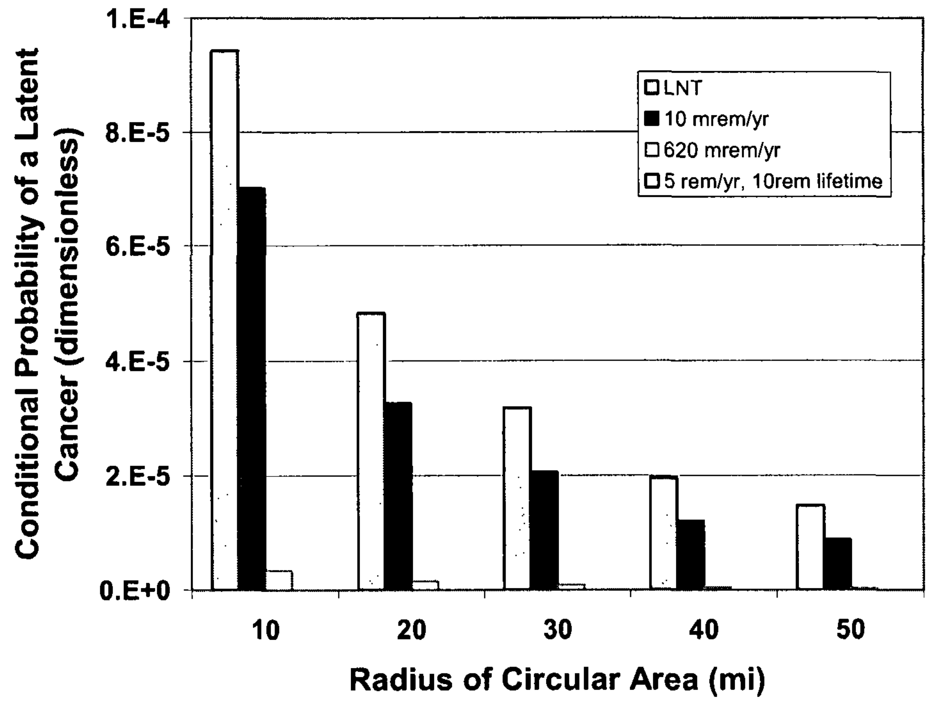

Figure 169 Conditional, mean, latent-cancer-fatality probabilities from the Surry unmitigated STSBO scenario for residents within a circular area of specified radius from the plant. The plot shows four values of dose-truncation level.

Figure 170 shows the latent-cancer-fatality risks as a function of the radius from the plant for the emergency phase (EARLY), the long-term phase (CHRONC), and the combined phases (sum of the two). The figure shows that the emergency response is very effective within the EPZ and that the long-term phase dominates the overall risks. The habitability (i.e., return) criterion, which is implemented as a limit of 4 rem in the first 5 years after returning to live in a residential area, controls the overall risk to the public. This explains why the risks using 620 mrem/yr and HPS dose-truncation criteria are so much lower than the risks for the two other criteria shown in Figure 156. Only the doses received during the first year (or possibly two years) of the long-term phase are counted toward health effects using these dose-truncation criteria.

All of the emergency-phase risk within the 10-mile EPZ is for the nonevacuating cohort. This is because all of the other cohorts avoid exposure to the plume. Thus, for this accident scenario, the residents within the EPZ who comply with the request to evacuate have no increased risk prior to the long-term phase.

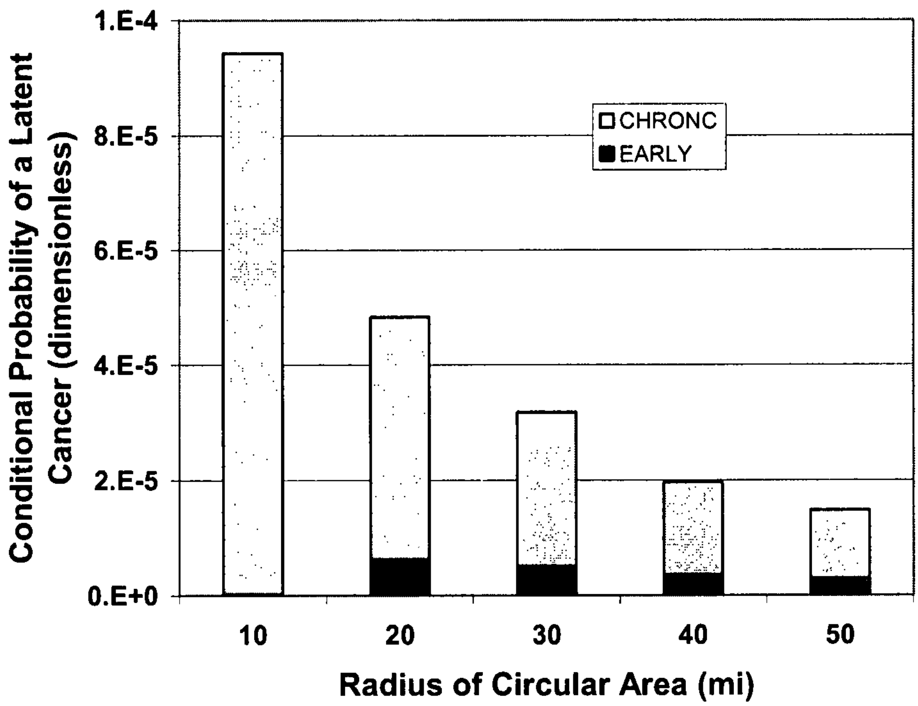

Figure 170 Conditional, mean, LNT, latent-cancer-fatality probabilities from the Surry unmitigated STSBO for residents within a circular area of specified radius from the plant. The plot shows the probabilities from the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined.

The prompt-fatality risks are identically zero for this accident scenario. This is because the release fractions (shown in Table 30) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5% of the population that does not evacuate.

7.3.3 Unmitigated Short-Term Station Blackout with Thermally Induced, Steam-Generator-Tube Rupture

Table 35 displays the conditional, mean, latent-cancer-fatality risks to residents within a set of concentric circular areas centered at the Surry site for the short-term station blackout (STSBO) initiated thermally induced, steam-generator-tube rupture (TISGTR) scenario. Four values of dose-truncation level are shown in the table: linear, no threshold (LNT), i.e., a dose-truncation level of zero; 10 mrem/yr; the average, annual, US-background radiation (including average medical radiation) of 620 mrem/yr; and the Health Physics Society (HPS) recommended dose truncation of 5 rem/yr, with a lifetime limit of 10 rem.

Table 36 is analogous to Table 35, but displays unconditional rather than the conditional risks. In the case of the Surry unmitigated STSBO with TISGTR scenario, the mean core damage frequency is $4 \cdot 10^{-7}$/yr[36]. This frequency is used to scale the results in Table 36, as described above.

Table 35 Conditional, Mean, Latent-Cancer-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Probabilities are for the unmitigated STSBO with TISGTR. Mean Core Damage Frequency is $4 \cdot 10^{-7}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$3.2 \cdot 10^{-4}$	$3.0 \cdot 10^{-4}$	$7.4 \cdot 10^{-5}$	$1.3 \cdot 10^{-5}$
20	$1.9 \cdot 10^{-4}$	$1.7 \cdot 10^{-4}$	$4.0 \cdot 10^{-5}$	$4.5 \cdot 10^{-6}$
30	$1.3 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$	$2.5 \cdot 10^{-5}$	$2.5 \cdot 10^{-6}$
40	$8.4 \cdot 10^{-5}$	$7.2 \cdot 10^{-5}$	$1.4 \cdot 10^{-5}$	$1.3 \cdot 10^{-6}$
50	$6.5 \cdot 10^{-5}$	$5.4 \cdot 10^{-5}$	$9.9 \cdot 10^{-6}$	$8.6 \cdot 10^{-7}$

Table 36 Unconditional, Mean, Latent-Cancer-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks are for the unmitigated STSBO with TISGTR. Mean Core Damage Frequency is $3.75 \cdot 10^{-7}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$1.2 \cdot 10^{-10}$	$1.1 \cdot 10^{-10}$	$2.8 \cdot 10^{-11}$	$5.0 \cdot 10^{-12}$
20	$7.2 \cdot 10^{-11}$	$6.5 \cdot 10^{-11}$	$1.5 \cdot 10^{-11}$	$1.7 \cdot 10^{-12}$
30	$4.9 \cdot 10^{-11}$	$4.4 \cdot 10^{-11}$	$9.5 \cdot 10^{-12}$	$9.2 \cdot 10^{-13}$
40	$3.2 \cdot 10^{-11}$	$2.7 \cdot 10^{-11}$	$5.3 \cdot 10^{-12}$	$4.7 \cdot 10^{-13}$
50	$2.4 \cdot 10^{-11}$	$2.0 \cdot 10^{-11}$	$3.7 \cdot 10^{-12}$	$3.2 \cdot 10^{-13}$

The values in Table 35 are plotted in Figure 173. The plot shows that for all dose-truncation levels, the risk is greatest for those closest to the plant and diminishes monotonically with distance. The feneral trends in this figure are very similar to those shown in Figure 171 in the previous subsection.

[36] The frequency of the Surry short-term station blackout is 1 to $2 \cdot 10^{-6}$/yr. The conditional probability of a thermally induced steam generator tube rupture is 0.1 to 0.4. The mean core damage frequency of $3.75 \cdot 10^{-7}$/yr represents the product of the mid points of these two ranges, i.e., $0.25 * 1.5 \cdot 10^{-6}$/yr.

Figure 172 shows the latent-cancer-fatality risks as a function of the radius from the plant for the emergency phase (EARLY), the long-term phase (CHRONC), and the combined phases (sum of the two). The figure shows that the emergency response does not entirely eliminate doses within the EPZ, but nonetheless, the risks are very small compared with the long-term phase risks. The doses received during the early phase stem from the relatively early release for this accident scenario, which begins 3.6 hr after accident initiation (cf., Table 30). The habitability (i.e., return) criterion, which is implemented as a limit of 4 rem in the first 5 years after returning to a residential area, controls the overall risk to the public for this accident scenario. The general trends in this figure are very similar to those shown in Figure 170 in the previous subsection, with one exception. The risk from exposure during the emergency phase (EARLY) within a 10-mile radius is very small compared with the other distances shown in Figure 170; it is larger within a 10-mile radius than it is for the larger radii in Figure 172. The difference is directly related to the source-term characteristics, particularly the initiation of release, as discussed in the following paragraph.

Most of the emergency-phase risk within the 10-mile EPZ is for the evacuees. This is because release begins at 3.6 hr after accident initiation; the public begins to evacuate at 3.75 hr. Thus, some of the evacuees travel through the plume. By comparison, release begins 25.5 hours after accident initiation in the unmitigated STSBO discussed in the previous subsection while evacuation begins at the same time for both sequences.

The nonevacuating cohort represents 1.4% of the overall emergency-phase risk using the LNT hypothesis. This is a larger percentage of the overall risk than the population fraction represented by this cohort, which is 0.5%. This is expected, i.e., that the nonevacuating cohort should represent a greater risk than the cohorts who evacuate.

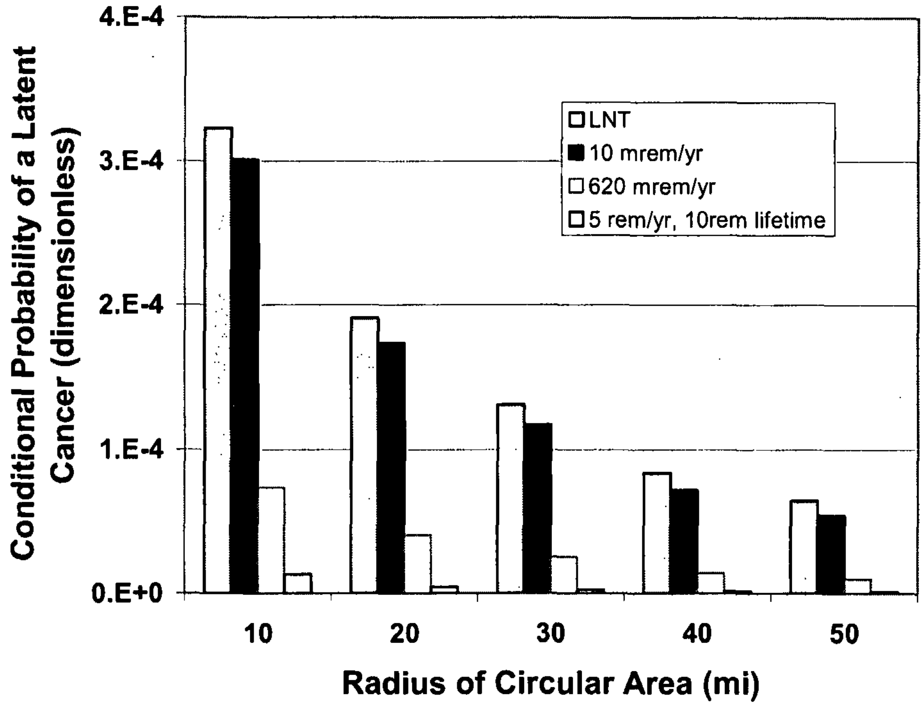

Figure 171 Conditional, mean, latent-cancer-fatality probabilities from the Surry unmitigated STSBO with TISGTR scenario for residents within a circular area of specified radius from the plant. The plot shows four values of dose-truncation level.

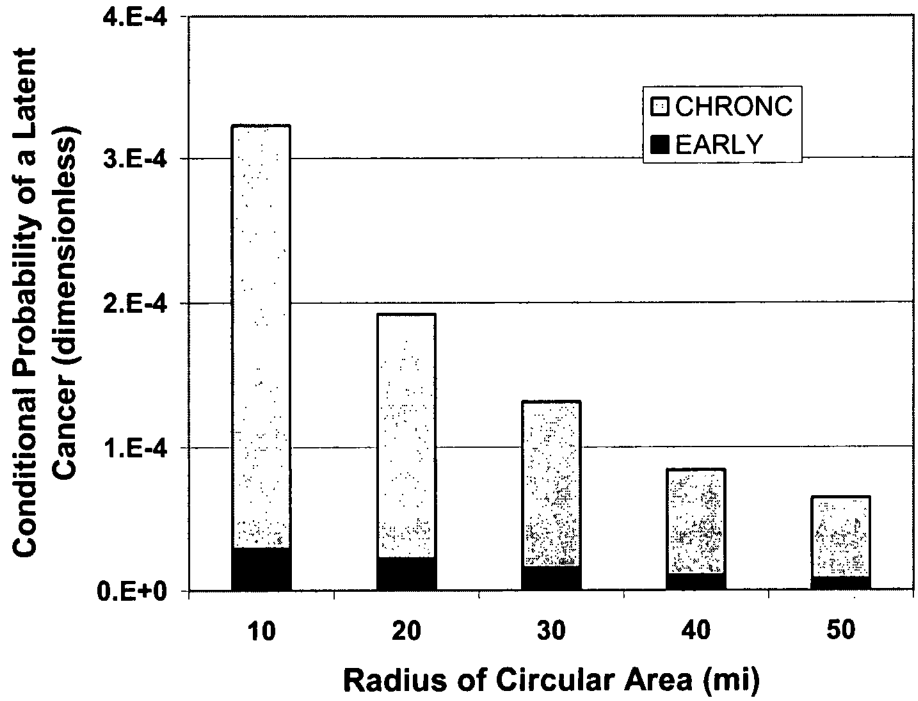

Figure 172 Conditional, mean, LNT, latent-cancer-fatality probabilities from the Surry unmitigated STSBO with TISGTR scenario for residents within a circular area of specified radius from the plant. The plot shows the probabilities for the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined.

The prompt-fatality risks are identically zero for this accident scenario. This is because the release fractions (shown in Table 30) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5% of the population that does not evacuate.

7.3.4 Mitigated Short-Term Station Blackout with Thermally Induced, Steam-Generator-Tube Rupture

Table 37 displays the conditional, mean, latent-cancer-fatality risks to residents within a set of concentric circular areas centered at the Surry site for the mitigated STSBO with TISGTR scenario. This scenario is similar to the one in the previous section except that it is mitigated by operator actions to restore containment sprays. Because of the restored containment sprays, the risks are slightly lower than those shown in the previous subsection. The values in this table are plotted in Figure 173. The trends are identical to those shown in the previous subsection for the unmitigated STSBO with TISGTR.

Four values of dose-truncation level are shown in the table: linear, no threshold (LNT), i.e., a dose-truncation level of zero; 10 mrem/yr; the average, annual, US-background radiation (including average medical radiation) of 620 mrem/yr; and the Health Physics Society (HPS) recommended dose truncation of 5 rem/yr, with a lifetime limit of 10 rem.

Table 38 is analogous to Table 37, but displays unconditional rather than the conditional risks. In the case of the Surry mitigated STSBO with TISGTR scenario, the mean core damage frequency is $4 \cdot 10^{-7}$/yr. This frequency is used to scale the results in Table 38, as described above.

Table 37 Conditional, Mean, Latent-Cancer-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Risks are for the Mitigated STSBO with TISGTR Scenario. Mean Core Damage Frequency is $4 \cdot 10^{-7}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$2.8 \cdot 10^{-4}$	$2.7 \cdot 10^{-4}$	$7.1 \cdot 10^{-5}$	$1.4 \cdot 10^{-5}$
20	$1.7 \cdot 10^{-4}$	$1.6 \cdot 10^{-4}$	$3.8 \cdot 10^{-5}$	$4.5 \cdot 10^{-6}$
30	$1.1 \cdot 10^{-4}$	$1.1 \cdot 10^{-4}$	$2.4 \cdot 10^{-5}$	$2.5 \cdot 10^{-6}$
40	$7.3 \cdot 10^{-5}$	$6.5 \cdot 10^{-5}$	$1.3 \cdot 10^{-5}$	$1.3 \cdot 10^{-6}$
50	$5.6 \cdot 10^{-5}$	$4.9 \cdot 10^{-5}$	$9.3 \cdot 10^{-6}$	$8.7 \cdot 10^{-7}$

Table 38 Unconditional, Mean, Latent-Cancer-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks are for the Mitigated STSBO with TISGTR Scenario. Mean Core Damage Frequency is $4 \cdot 10^{-7}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$1.0 \cdot 10^{-10}$	$1.0 \cdot 10^{-10}$	$2.7 \cdot 10^{-11}$	$5.1 \cdot 10^{-12}$
20	$6.2 \cdot 10^{-11}$	$5.9 \cdot 10^{-11}$	$1.4 \cdot 10^{-11}$	$1.7 \cdot 10^{-13}$
30	$4.3 \cdot 10^{-11}$	$4.0 \cdot 10^{-11}$	$8.9 \cdot 10^{-12}$	$9.3 \cdot 10^{-13}$
40	$2.7 \cdot 10^{-11}$	$2.4 \cdot 10^{-11}$	$4.9 \cdot 10^{-12}$	$4.8 \cdot 10^{-13}$
50	$2.1 \cdot 10^{-11}$	$1.8 \cdot 10^{-11}$	$3.5 \cdot 10^{-12}$	$3.3 \cdot 10^{-13}$

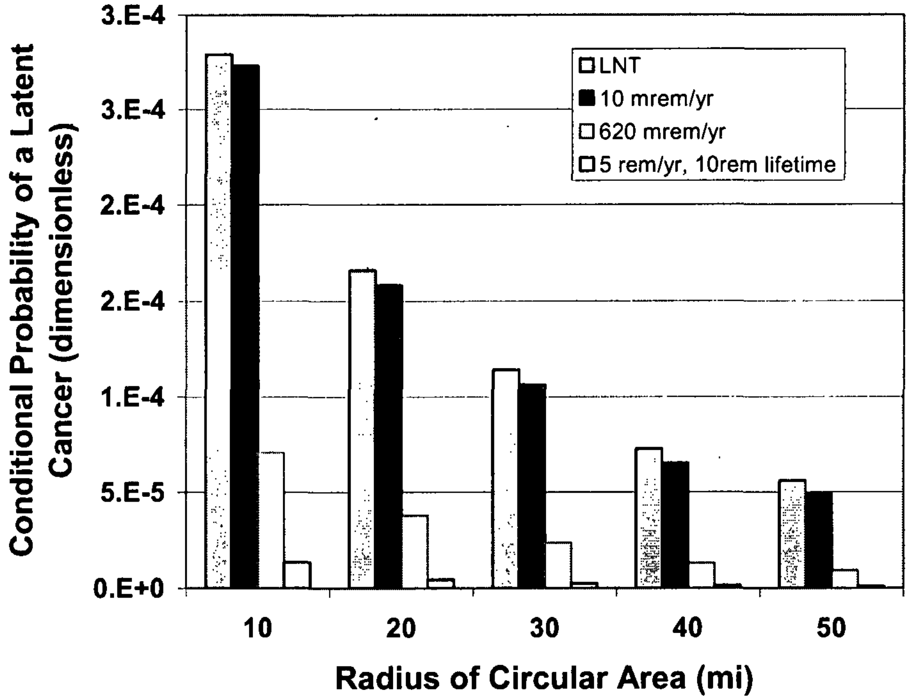

Figure 173 Conditional, mean, latent-cancer-fatality probabilities from the Surry mitigated STSBO with TISGTR scenario for residents within a circular area of specified radius from the plant. The plot shows four values of dose-truncation level.

Figure 174 shows the latent-cancer-fatality risks as a function of the radius from the plant for the emergency phase (EARLY), the long-term phase (CHRONC), and the combined phases (sum of the two). The figure shows that the emergency response does not entirely eliminate doses within the EPZ, but nonetheless, the risks are very small compared with the long-term risks. The doses received during the emergency phase stem from the relatively early release for this accident scenario, which begins 3.6 hr after accident initiation (cf., Table 30). The habitability (i.e., return) criterion, which is implemented as a limit of 4 rem in the first 5 years after returning to live in a residential area, controls the overall risk to the public for this accident scenario. The trend shown in Figure 176 are identical to those shown in Figure 174 for inmitigated scenario.

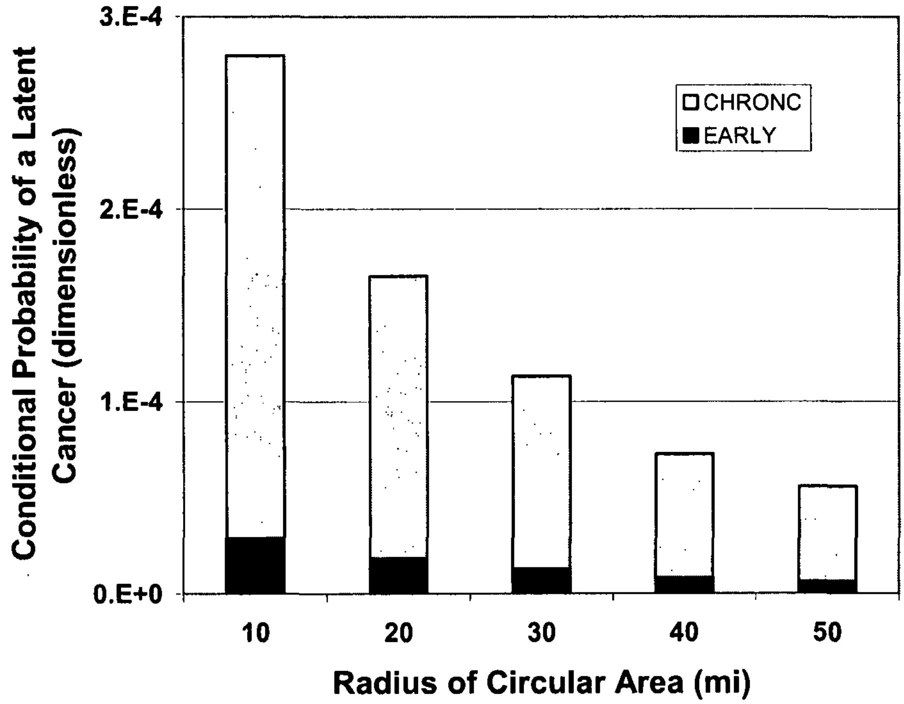

Figure 174 Conditional, mean, LNT, latent-cancer-fatality probabilities from the Surry mitigated STSBO with TISGTR scenario for residents within a circular area of specified radius from the plant. The plot shows the risks from the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined.

The prompt-fatality risks are identically zero for this accident scenario. This is because the release fractions (shown in Table 30) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5% of the population that does not evacuate.

Most of the emergency-phase risk within the 10-mile EPZ is for the evacuees. This is because release begins at 3.6 hr and, as a result, most of the evacuees are unable to avoid exposure to the plume. The nonevacuating cohort represents 1.1% of the overall emergency-phase risk using the LNT hypothesis. This is clearly a larger percentage of the overall risk than the population fraction represented by this cohort, which is 0.5%. It is expected that the nonevacuating cohort should have a greater risk than the cohorts who evacuate.

7.3.5 Unmitigated ISLOCA

The unmitigated interfacing systems loss of coolant accident (ISLOCA) has the largest predicted releases and the release is earlier than for the SBO scenarios without TISGTR. The release for this scenario begins at 9.2 hours after accident initiation. Even so, emergency response is very effective and essentially no early fatalities are predicted. However, predicted latent-cancer-fatality risks are larger than those for the scenarios described in the previous subsections.

Table 39 displays the conditional, mean, latent-cancer-fatality risks to residents within a set of concentric circular areas centered at the Surry site for the unmitigated ISLOCA scenario. Four values of dose-truncation level are shown in the table: linear, no threshold (LNT), i.e., a dose-truncation level of zero; 10 mrem/yr; the average, annual, US-background radiation (including average medical radiation) of 620 mrem/yr; and the Health Physics Society (HPS) recommended dose truncation of 5 rem/yr, with a lifetime limit of 10 rem.

Table 40 is analogous to Table 39, but displays unconditional rather than the conditional risks. In the case of the Surry unmitigated ISLOCA scenario, the mean core damage frequency is $3 \cdot 10^{-8}$/yr. This frequency is used to multiply the results in Table 32, as described above[37].

The values in Table 39 are plotted in Figure 175. The plot shows that for all dose-truncation levels, the risk is greatest for those closest to the plant and diminishes monotonically with distance from the plant. The trends shown in this figure are similar to those shown in the preceding subsections, with one notable exception. The individual risk for the population within a 10-mile radius of the plant is slightly greater using the HPS criteria than using the background truncation level. This trend can occur in some cases because of the lifetime limit of 10 rem that is part of the HPS criteria. When annual doses are less than 5 rem, but the total lifetime dose exceeds 10 rem, all annual doses, no matter how small, are used to estimate risk using the HPS criteria. On the other hand, only the annual doses that exceed 620 mrem are used with background-dose truncation. Since doses diminish in time, this latter truncation criterion generally excludes at least a portion of the LNT dose. Thus, it is not difficult to see how dose truncation using the HPS criteria can sometimes produce greater risks than using the background radiation level.

Figure 176 shows the latent-cancer-fatality risks as a function of the radius from the plant for the emergency phase (EARLY), the long-term phase (CHRONC), and the combined phases (sum of the two). The figure shows that the emergency response does not entirely eliminate risks within the within the EPZ. This is because release begins at 9.2 hours after accident initiation, which is before evacuation is complete. Figure 149 shows that the public evacuates from 3 hr to 13 hr after accident initiation and the Special and Tail Cohorts only begin to evacuate at 13 hr after accident initiation. Clearly, there is a potential for exposure to the plume during evacuation for this accident scenario. Ths accounts for the emergency-phase risk within the 10-mile EPZ shown in Figure 176.

Nonetheless, the long-term phase dominates the overall risks, even within the EPZ. The habitability (i.e., return) criterion, which is implemented as a limit of 4 rem in the first 5 years after returning to live in a residential area, controls the overall risk to the public for this accident scenario.

Most of the emergency-phase risk within the 10-mile EPZ is for the evacuees. The nonevacuating cohort represents 5.1% of the overall emergency-phase risk using the LNT hypothesis. This is an order-of-magnitude larger percentage of the overall risk than the

[37] The licensee quotes a somewhat larger core damage frequency of $7 \cdot 10^{-7}$/yr, which motivated inclusion of this scenario in the Surry analysis.

population fraction represented by this cohort, which is 0.5%. This result is expected, that is, that the individual risk for the cohort that does not evacuate should be greater than for the cohorts who do evacuate.

Table 39 Conditional, Mean, Latent-Cancer-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Probabilties are for the Unmitigated Interfacing System LOCA Scenario. Mean Core Damage Frequency is $3 \cdot 10^{-8}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$7.9 \cdot 10^{-4}$	$7.6 \cdot 10^{-4}$	$2.6 \cdot 10^{-4}$	$2.7 \cdot 10^{-4}$
20	$6.6 \cdot 10^{-4}$	$6.4 \cdot 10^{-4}$	$2.7 \cdot 10^{-4}$	$2.3 \cdot 10^{-4}$
30	$5.3 \cdot 10^{-4}$	$5.1 \cdot 10^{-4}$	$2.0 \cdot 10^{-4}$	$1.4 \cdot 10^{-4}$
40	$4.1 \cdot 10^{-4}$	$3.9 \cdot 10^{-4}$	$1.4 \cdot 10^{-4}$	$7.8 \cdot 10^{-5}$
50	$3.5 \cdot 10^{-4}$	$3.4 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$	$5.5 \cdot 10^{-5}$

Table 40 Unconditional, Mean, Latent-Cancer-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks are for the Unmitigated Interfacing System LOCA Scenario. Mean Core Damage Frequency is $3 \cdot 10^{-8}$/yr.

Radius of Circular Area (mi)	LNT	10 mrem/yr	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$2.4 \cdot 10^{-11}$	$2.3 \cdot 10^{-11}$	$7.7 \cdot 10^{-12}$	$8.0 \cdot 10^{-12}$
20	$2.0 \cdot 10^{-11}$	$1.9 \cdot 10^{-11}$	$8.0 \cdot 10^{-12}$	$6.8 \cdot 10^{-12}$
30	$1.6 \cdot 10^{-11}$	$1.5 \cdot 10^{-11}$	$6.0 \cdot 10^{-12}$	$4.3 \cdot 10^{-12}$
40	$1.2 \cdot 10^{-11}$	$1.2 \cdot 10^{-11}$	$4.3 \cdot 10^{-12}$	$2.3 \cdot 10^{-12}$
50	$1.0 \cdot 10^{-11}$	$1.0 \cdot 10^{-11}$	$3.5 \cdot 10^{-12}$	$1.7 \cdot 10^{-12}$

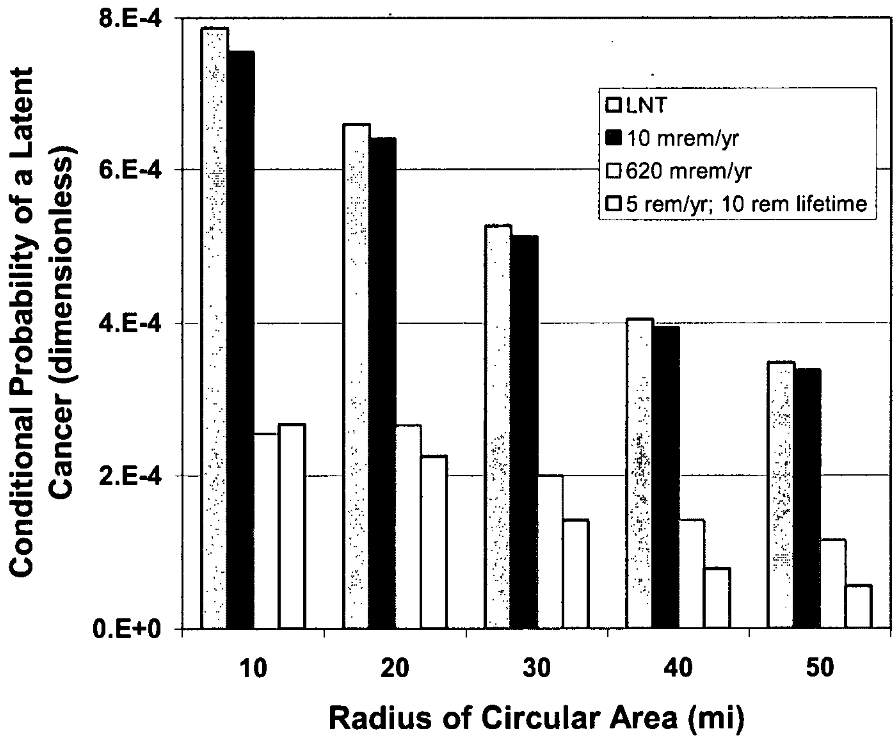

Figure 175 Conditional, mean, latent-cancer-fatality probabilities from the Surry unmitigated ISLOCA scenario for residents within a circular area of specified radius from the plant. The plot shows four values of dose-truncation level.

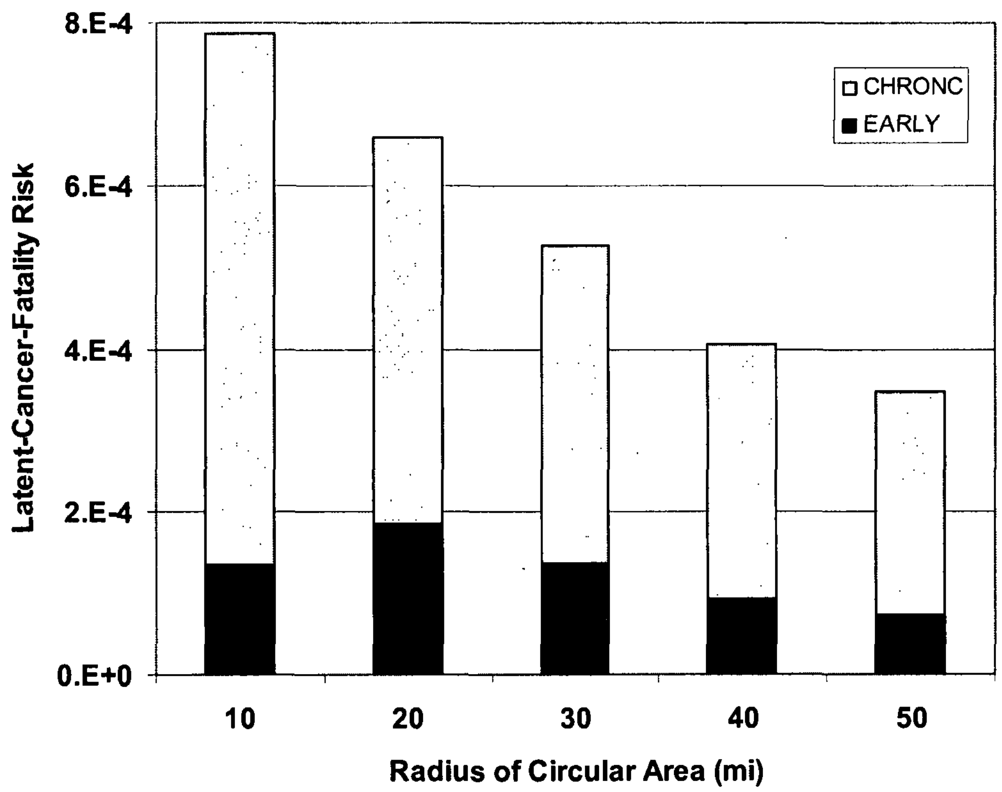

Figure 176 Conditional, mean, LNT, latent-cancer-fatality probabilities from the Surry unmitigated ISLOCA for residents within a circular area of specified radius from the plant. The plot shows the risks from the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined.

The prompt-fatality risks are essentially zero for this accident scenario. The releases are close to the threshold for early fatalities. There is no prompt-fatality risk for the cohorts that evacuate. Conditional prompt-fatality probabilities are shown in Table 41 as a function of distance from the plant. Unconditional prompt fatality risks are shown in Table 7-12.

Table 41 Conditional, Mean, Prompt-Fatality Probabilities (dimensionless) for Residents within the Specified Radii of the Surry Site. Risks Are for the Interfacing Systems LOCA Scenario. Mean Core Damage Frequency is $7 \cdot 10^{-7}$/yr.

Radius of Circular Area (mi)	Prompt Fatality Risk
1.3	$1.7 \cdot 10^{-7}$
2.0	$1.4 \cdot 10^{-7}$
2.5	$7.6 \cdot 10^{-8}$

Table 42 Unconditional, Mean, Prompt-Fatality Risks (1/reactor year) for Residents within the Specified Radii of the Surry Site. Risks Are for the Interfacing Systems LOCA Scenario. Mean Core Damage Frequency is $3 \cdot 10^{-8}$/yr.

Radius of Circular Area (mi)	Prompt Fatality Risk
1.3	$5.1 \cdot 10^{-15}$
2.0	$4.2 \cdot 10^{-15}$
2.5	$2.3 \cdot 10^{-15}$

The NRC quantitative health object (QHO) for prompt fatalities is generally interpreted as the unconditional risk within 1 mile of the exclusion area boundary. For Surry, the exclusion area boundary is 0.35 miles from the reactor building from which release occurs, so the outer boundary of this 1-mile zone is at 1.35 miles. The closest MACCS2 grid boundary to 1.35 miles used in this set of calculations is at 1.3 miles. Using the risk within 1.3 miles should reasonably approximate the risk within 1 mile of the exclusion area boundary. The unconditional risk of a prompt fatality for this source term is approximately $5 \cdot 10^{15}$, which is well below the QHO. There is a very small risk of a prompt fatality within 2.5 miles from the plant, which is about 2.1 miles beyond the site boundary; the prompt-fatality risk from this accident scenario is zero beyond 2.5 miles from the plant.

7.3.6 Sensitivity Analyses on the Size of the Evacuation Zone

The baseline analyses included evacuation of the 10-mile EPZ, a partial shadow evacuation between 10 and 20 miles, and relocations of the remaining members of the public. For the unmitigated ISLOCA scenario, three additional calculations were performed to assess variations in the protective actions.

Sensitivity #1: Evacuation of a 16-Mile Area
For sensitivity case #1, evacuation of a 16-mile area around the NPP is evaluated. In addition, a shadow evacuation occurs from within the 16- to 20-mile area and the remaining members of the public in that area are assumed to shelter for a period of 24 hours after plume arrival, at which point this remaining population group also evacuates.

Sensitivity #2: Evacuation of the 0- to 20-Mile Area
In this calculation, the evacuation zone is expanded to 20 miles. No shadow evacuation is considered.

Sensitivity #3: Delayed Evacuation of a 10-Mile Area

This calculation is identical to the baseline case described above, with the exception that implementation of protective action is delayed by 30 minutes. The results of the sensitivity analyses are presented in Table 43.

Table 434 shows that very little benefit results from increasing the size of the evacuation zone beyond the standard 10 miles.

Table 43 Effect of Size of Evacuation Zone on Conditional, Mean, LNT, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Surry Site. Risks Are for an Unmitigated Interfacing Systems LOCA Scenario.

Radius of Circular Area (mi)	10-Mile Evacuation	16-Mile Evacuation	20-Mile Evacuation	Delayed 10-Mile Evacuation
10	$7.9 \cdot 10^{-4}$	$8.1 \cdot 10^{-4}$	$8.2 \cdot 10^{-4}$	$8.2 \cdot 10^{-4}$
20	$6.6 \cdot 10^{-4}$	$5.7 \cdot 10^{-4}$	$5.5 \cdot 10^{-4}$	$6.7 \cdot 10^{-4}$
30	$5.3 \cdot 10^{-4}$	$4.8 \cdot 10^{-4}$	$4.6 \cdot 10^{-4}$	$5.3 \cdot 10^{-4}$
40	$4.1 \cdot 10^{-4}$	$3.8 \cdot 10^{-4}$	$3.7 \cdot 10^{-4}$	$4.1 \cdot 10^{-4}$
50	$3.5 \cdot 10^{-4}$	$3.3 \cdot 10^{-4}$	$3.3 \cdot 10^{-4}$	$3.5 \cdot 10^{-4}$

7.3.7 *Evaluation of the Effect of the Seismic Activity on Emergency Response*

The effects of seismic activity on emergency response are evaluated in this subsection for the unmitigated TISGTR scenario. Several impacts of the seismic activity are considered. One of these is the effect of collapsed bridges and impassible roadways on the evacuation itself, which is expected to increase risk. Another effect is on the size of the shadow evacuation, which is expected to decrease risk. The overall impact of the seismic activity on emergency response at the Surry site is insignificant, as shown in Table 44. Prompt-fatality risk remains zero for this scenario.

Table 44 Conditional, Mean, LNT, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Surry Site. Risks Are for the unmitigated TISGTR Scenario and Compare the Unmodified Emergency Response (ER) and ER Adjusted to Account for the Effect of Seismic Activity on Evacuation Routes and Human Response.

Radius of Circular Area (mi)	Unmodified ER	ER Adjusted for Seismic Effects
10	$3.2 \cdot 10^{-4}$	$3.3 \cdot 10^{-4}$
20	$1.9 \cdot 10^{-4}$	$1.9 \cdot 10^{-4}$
30	$1.3 \cdot 10^{-4}$	$1.3 \cdot 10^{-4}$
40	$8.4 \cdot 10^{-5}$	$8.4 \cdot 10^{-5}$
50	$6.5 \cdot 10^{-5}$	$6.5 \cdot 10^{-5}$

7.3.8 Evaluation of SST1 Source Term

An additional set of calculations was performed to enable the current, state-of-the-art results to be compared with the older Sandia Siting Study results [43]. In particular, the largest source term from the Sandia Siting Study, the SST1 source term, was selected for comparison. This set of calculations is based on the Surry STSBO scenario, but with the source term replaced by the SST1 source term. No other modeling or parameter changes were made.

The SST1 source term is described in the Sandia Siting Study report as follows:

- Severe core damage
- Essentially involves loss of all installed safety features
- Severe direct breach of containment

However, an exact scenario and containment failure mechanism, e.g., hydrogen detonation, direct containment heating, or alpha-mode failure, are not specified.

Notification time, i.e., sounding a siren to notify the public that a general emergency has been declared, for the Surry unmitigated STSBO occurs at 2.75 hr. Declaration of a general emergency occurs at 2 hr and it takes an additional 45 min to notify the public. Notification of the public is thus after the beginning of release for the SST1 source term (cf., Table 30), which occurs 1.5 hr after accident initiation. Evacuation of the general public begins one hour after notification, or 3.75 hr after accident initiation. The start of evacuation here for this scenario is slightly earlier, but comparable, to that for the largest segment of the population in the Sandia Siting Study, which occurred 4 hr after accident initiation.

While the Sandia Siting Study treated emergency response very simplistically, a major focus of the SOARCA project is to treat all aspects of the consequence analysis as realistically as possible. It was not the intention here to modify the emergency response parameters to be like those used during the Sandia Siting Study. Furthermore, without knowing the specific accident scenario and containment failure mode that corresponds to the SST1 source term, it is not possible to know what notification time would now be considered realistic for Surry. Thus, in the end, the emergency response parameters were chosen to be the same as in the unmitigated STSBO scenario.

Table 45 shows the latent-cancer-fatality risks for a release corresponding to the SST1 source term occurring at Surry. Table 46 compares the LNT risks with those for the unmitigated ISLOCA and the unmitigated STSBO with TISGTR scenarios discussed in preceeding subsections. The LNT risk within 10 miles is about a factor of 15 higher than for the largest Surry source term considered in this study, which is for the ISLOCA; the 10-mile risk using a 620 mrem/yr dose-truncation criterion is a factor of 50 higher (cf., Table 39). At 50 miles the risks are less disparate: a factor of 4.5 for the LNT risks.

Table 45 Conditional, Mean, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Surry Site. Risks Are for the SST1 Source Term from the Sandia Siting Study. This Source Term is Assigned a Frequency of 10^{-5}/yr in the Sandia Siting Study.

Radius of Circular Area (mi)	LNT	620 mrem/yr	5 rem/yr; 10 rem lifetime
10	$1.0{\cdot}10^{-2}$	$9.8{\cdot}10^{-3}$	$1.0{\cdot}10^{-2}$
20	$5.1{\cdot}10^{-3}$	$4.9{\cdot}10^{-3}$	$5.1{\cdot}10^{-3}$
50	$1.5{\cdot}10^{-3}$	$1.3{\cdot}10^{-3}$	$1.4{\cdot}10^{-3}$

Table 46-17 Conditional, Mean, LNT, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Surry Site. Risks Are for the SST1 Source Term from the Sandia Siting Study, the unmitigated ISLOCA, and the unmitigated STSBO with TISGTR scenarios.

Radius of Circular Area (mi)	SST1	Unmitigated ISLOCA	Unmitigated STSBO with TISGTR
10	$1.0{\cdot}10^{-2}$	$7.9{\cdot}10^{-4}$	$3.2{\cdot}10^{-4}$
20	$5.1{\cdot}10^{-3}$	$6.6{\cdot}10^{-4}$	$1.9{\cdot}10^{-4}$
50	$1.5{\cdot}10^{-3}$	$3.5{\cdot}10^{-4}$	$6.5{\cdot}10^{-5}$

The maximum risk for the SST1 source term is within 10 miles, which is partially due to the fact that emergency response is not rapid enough to prevent exposures within the EPZ during the emergency phase. This is expected since release begins before notification of the public and, therefore, before evacuation begins.

A notable feature of the risks presented in Table 45 is that the choice of dose truncation criterion has a minor influence on risk. This is very different than the SOARCA accident scenarios discussed in preceding subsections. Figure 177 provides some insights into this behavior. For the SST1 source term, nearly all of the risk, especially at short distances from the plant, is from exposures that occur during the emergency phase (EARLY). Because a significant fraction of these doses are received over a short period of time and the doses are large due to the large source term, the level used for the dose truncation criterion has little influence on predicted risks. Again, this is a very different trend than is observed for the current, state-of-the-art source terms.

Table 47 shows the risk of prompt fatalities for several circular areas of specified radii centered at the plant. Unlike the source terms presented above, the predicted prompt-fatality risks are significantly greater than zero. The SST1 release fractions are more than large enough to induce prompt fatalities for members of the public who live close to the plant and who do not evacuate quickly.

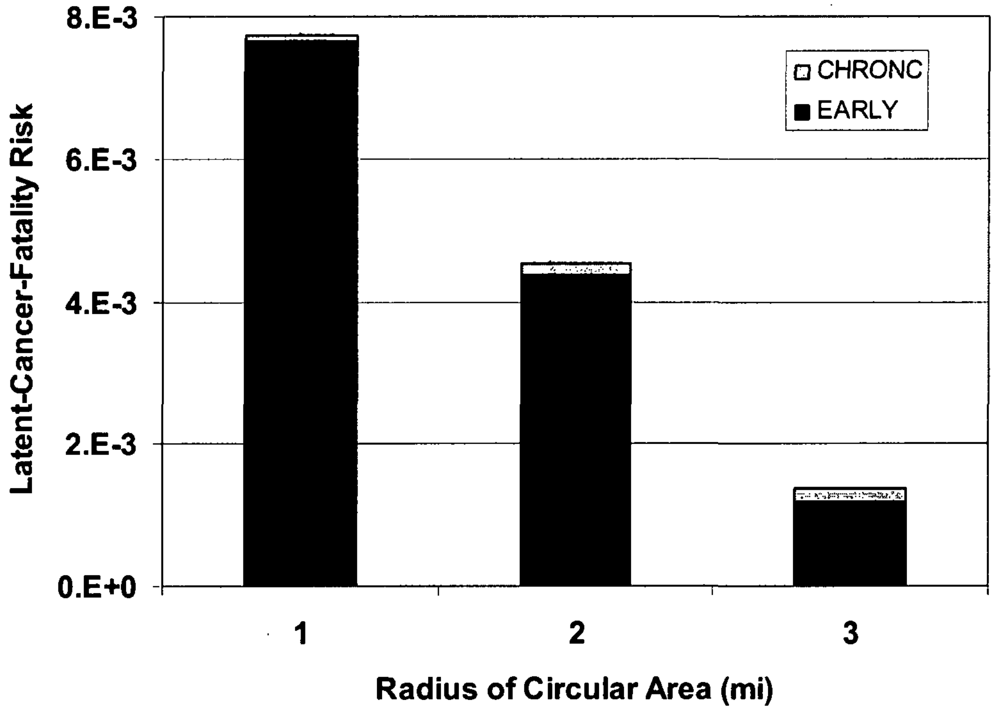

Figure 177 Conditional, mean, LNT, latent-cancer-fatality risks from the SST1 source term for residents within a circular area of specified radius from the Surry plant. The bars show the risks from the emergency phase (EARLY), long-term phase (CHRONC), and the two phases combined (Total).

Table 47 Conditional, Mean, Prompt-Fatality Risks for Residents within the Specified Radii of the Surry Site. Risks Are for the SST1 Source Term from the Sandia Siting Study.

Radius of Circular Area (mi)	Prompt Fatality Risk
1.3	$1.3 \cdot 10^{-2}$
2.0	$1.5 \cdot 10^{-2}$
2.5	$1.1 \cdot 10^{-2}$
3.0	$8.4 \cdot 10^{-3}$
3.5	$5.4 \cdot 10^{-3}$
5.0	$3.7 \cdot 10^{-4}$
7.0	$5.0 \cdot 10^{-5}$
10.0	$1.5 \cdot 10^{-5}$

The NRC quantitative health object (QHO) for prompt fatalities is generally interpreted as the unconditional risk within 1 mile of the exclusion area boundary. For Surry, the exclusion area boundary is 0.35 miles from the reactor building from which release occurs, so the outer boundary of this 1-mile zone is at 1.35 miles. The closest MACCS2 grid boundary to 1.35 miles used in this set of calculations is at 1.3 miles. Using the risk within 1.3 miles should reasonably approximate the risk within 1 mile of the exclusion area boundary. The frequency stated for the SST1 source term in the Sandia Siting Study [43] is 10^{-5}/yr, so the unconditional risk of a prompt fatality for this source term is approximately $1.3 \cdot 10^{-7}$/yr, which is well below the QHO. The actual risk of a prompt fatality (cf.,Table 41), using current best-estimate practices for calculating source terms, is about five orders of magnitude lower than using the SST1 source term would imply (cf., Table 412 and Table 47).

The acute-fatality risks presented in Table 47 are lower than risks that would have been calculated in the Sandia Siting Study. There are two primary reasons for this difference. One is that 30% of the population within the EPZ is assumed to evacuate a full 6 hr after accident initiation in the Sandia Siting Study; here, 97.4% of the population within the EPZ begin to evacuate at least by 3.75 hr after accident initiation. A second reason is that the coefficients in the equations for acute health effects have been updated since the Sandia Siting Study based on more recent expert data [34]. The updated coefficients result in lower predicted acute fatalities across most of the exposure range for which these health effects can occur.

For the types of releases that would be encountered in a severe nuclear accident of sufficient severity, the hematopoietic syndrome would be responsible for most of the acute fatalities. Figure 178 shows the evolution of dose-response curves starting with the Sandia Siting Study, the NUREG-1150 PRA study [1], and now being used for SOARCA. The dose-response curve used in NUREG-1150 predicts the largest number of fatalities over most of the range; the one used at the time of the Sandia Siting Study is intermediate over most of the range; the one used here is the lowest over most of the dose range. However, the current model predicts a higher risk of prompt fatalities at the low end of the dose range than the one used in the Sandia Siting Study. Mean (over weather) peak (around the compass) acute doses to the red marrow, assuming no evacuation, for this case using the SST1 source term range from 48 Gy to the members of the population closest to the site to 1.36 Gy at 10 miles. These doses span the entire range shown in the figure. The maximum radius at which acute health effects occur is slightly greater than 15 miles using the SST1 source term.

Clearly, the SST1 source term and its assumed frequency of occurrence are large compared with the source terms obtained using current, best-estimate practices. This reflects improvements in understanding and modeling capabilities that have been developed since the Sandia Siting Study was conducted.

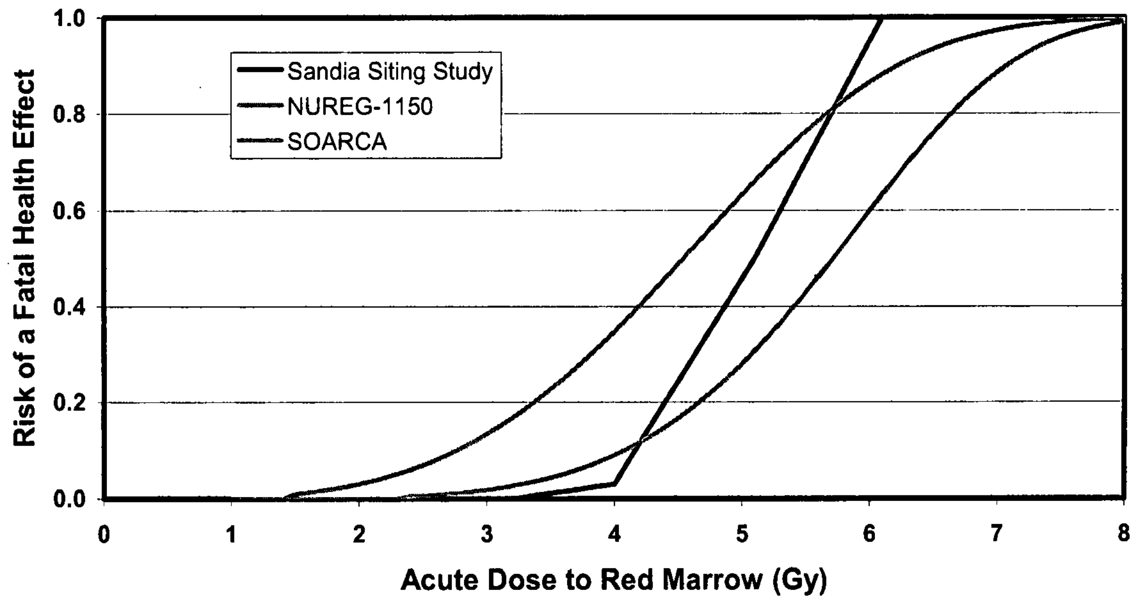

Figure 178 Risk of a fatal occurrence of the hematopoietic syndrome due to an acute dose to the red marrow. The three curves are for the models used at the time of the Sandia Siting Study, NUREG-1150, and in SOARCA.

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APPENDIX B.1 SURRY CONTAINMENT PERFORMANCE

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ACRONYMS

IPE	Individual Plant Examination
lb	pound
psig	pound-force per square inch gauge
psi	pounds per square inch
PWR	Pressurized Water Reactor
SNL	Sandia National Laboratories

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1.0 INTRODUCTION

Containment performance at beyond design basis accident internal pressure and temperature is required as an input for determining the offsite consequences and accident progression of a nuclear power plant during a severe accident. This appendix documents the analysis and assessment of Surry Nuclear Power Plant (SNPP) containment at beyond design basis internal pressures and temperatures developed during a severe accident. The design-specific SNPP containment failure pressure, leakage area, and leakage location as documented is used as an input for the State of-the-Art Reactor Consequences Analysis (SOARCA) of the SNPP.

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2.0 APPROACH

Extensive research and scale model testing of reinforced and prestressed concrete containments to determine behavior at beyond design basis accident pressure has been performed in the last 25 years at SNL [2] and CEGB [3]. Concrete containments start to leak before a complete rupture or failure. It is extremely difficult to accurately predict the location and leakage rate of the concrete containment due to beyond design basis internal pressure and temperature. Hessheimer and Dameron [2], and Dameron, Rashid, and Tang [4] provide guidance for predicting leak area and leak rate in containments. Hessheimer and Dameron [2] recommend a non-linear finite element analysis of the concrete containment to predict containment performance and leakage. In the past, reactor severe accident progression analysis has often assumed that the concrete containment starts to leak through a small hole as soon as containment is pressurized. The area of the small hole is calculated based on nominal design leakage rate of 0.10 to 0.20 percent of containment free volume mass per day at the design internal pressure. The area of the hole is assumed to remain constant until containment failure in the accident progression analysis. Results of concrete containment model tests [5, 6] indicate that leakage area increases appreciably with internal pressure. In addition, if the rate of pressurization is gradual and does not exceed the leakage rate, catastrophic failure of the concrete containment is not possible.

2.1 Concrete Containment Performance under Internal Pressure

A 1:6 scale model of a representative PWR concrete containment was tested at the Sandia National Laboratories (SNL) in July 1987 [5]. Prior to performing the test, 10 international organizations performed an independent and separate (round robin) pretest analyses of the containment [7] to predict containment behavior. A summary of the round robin analyses and test results is presented in Table B.1-1.

Hessheimer and Dameron [2] have concluded that global, free field strain of 1.5 to 2.0% for reinforced and 0.5 to 1.0% for prestressed concrete can be achieved before failure or rupture. In addition, leakage in concrete containment increases appreciably after the rebars and liner plate yield. Furthermore, under gradual increase in internal pressure, containment leakage continues to grow without failure and rupture. Using these criteria, failure and yield pressure for the 1:6 scale model concrete containment were calculated using the following equations.
The results of these simple calculations, as shown in Table B.1-1, are quite consistent with detailed finite element analyses using state of the art computer codes and test data.

$$P_{fail} = (A_{hoop} * Y_{rebar@2\%} + A_{liner} * Y_{liner@2\%}) / R$$

$$P_{yield} = (A_{hoop} * Y_{rebar} + A_{liner} * Y_{liner}) / R$$

where:

P_{fail} = Containment failure pressure

P_{yield} = Containment pressure at which hoop rebars and liner plate yield

A_{hoop} = Area of the hoop rebars

A_{liner} = Area of the liner plate

Y_{rebar} = Yield stress of the rebar
Y_{liner} = Yield stress of the liner plate
$Y_{rebar@2\%}$ = Stress in the rebar at 2% strain
$Y_{liner@2\%}$ = Stress in the rebar at 2% strain
R = Radius of the containment

Table B.1-1 Internal Pressure in 1:6-Scale Reinforced Concrete Containment.

Source	Hoop Rebar and Liner Plate Yield MPa (psig)	Containment Failure MPa (psig)
Round Robin Analyses (Maximum)	0.951 (138)	1.276 (185)
Round Robin Analyses (Minimum)	0.827 (120)	0.883 (128)
Round Robin Analyses (Average)	0.869 (126)	1.076 (156)
Test Data	0.820 (119)	1.00 (145)
Proposed Simplified Analysis	0.876 (127)	0.986 (143)

The simplified analysis approach was then used to determine the behavior of three existing reinforced concrete PWR containments. The comparison of results using the simplified approach with information provided by the three plant licensees in their Individual Plant Examination (IPE) reports is presented in Table B.1-2. A review of this table indicates that failure pressure predicted in the IPE reports for all three containments is 10 to 25 percent higher than the one obtained by simplified approach. Similarly, the pressure at which rebars in the three containments yield, as reported in the IPE reports, varies from the simplified analysis by -4 to 40 percent. These differences in the predictions are similar to the ones reported by the round robin analysts for the 1:6 scale containment, and are essentially due to the use of different criteria for postulated failure. For instance, the licensees have used strains greater than 2 percent to determine failure pressure reported in the IPEs.

Table B.1-2 Internal Pressure at Yield and Failure in Reinforced Concrete PWR Containments.

Item	Containment #1	Containment #2	Containment #3
Internal Pressure at Rebar Yield from IPE Report MPa (psig)	0.758 (110)	1.000 (145)	1.248 (181)
Internal Pressure at Failure from IPE Report MPa (psig)	1.062 (154)	1.048 (152)	1.489 (216)*
Internal Pressure at Rebar Yield from Simplified Analysis MPa (psig)	0.779 (113)	0.848 (123)	1.062 (154)
Internal Pressure at from the Proposed Simplified Analysis MPa (psig)	0.855 (124)	0.958 (139)	1.200 (174)

* IPE confirmatory analysis determined the failure pressure as 158 psi at 1% strain.

To further confirm the validity of the simplified analysis approach, it was applied to the ¼ scale model of a PWR prestressed concrete containment that was tested at Sandia National Laboratories in 2000 [6]. Prior to performing the test, a round robin pretest analysis of the containment [8] was performed by 17 international organizations to predict containment behavior. A summary of the round robin analysis, test results, and results of simplified analysis based on free field strain of 1.0% for failure is presented in Table B.1-3. The simplified approach for prestressed containment is similar to the one described above for reinforced concrete except that effect of prestressing steel is included in the calculations.

Table B.1-3 Internal Pressure at Yield and Failure in the 1:4-Scale Prestressed Concrete Containment.

Source	Hoop Reinforcement Yield – MPa(psig)	Containment Failure MPa (psig) (Leakage >100%)
Round Robin Analyses (Maximum)	1.248 (181)	1.979 (287)
Round Robin Analyses (Minimum)	0.855 (124)	0.814 (118)
Round Robin Analyses (Average)	1.034 (150)	1.413 (205)
Test Data	1.055 (153)	1.296 (188)
Proposed Simplified Analysis	1.062 (154)	1.331(193)

A review of the Table B.1-3 indicates that there is a wide variation in predicted pressures by 17 organizations. The maximum predicted failure pressure is 2.4 times more than the minimum predicted failure pressure. However, the average round robin and the proposed simplified analysis predicted pressures are quite close to the pressures recorded during the test.

The simplified approach described above was also applied to the 1:10 Scale Sizewell B model. The results of proposed simplified analysis are compared with the 3-D finite element analysis and pressure test data in Table B.1-4. The proposed simplified approach results closely match with detailed 3-dimensional non-linear finite element analysis and test data.

Table B.1-4 Internal Pressure at Yield and Failure in the 1:10-Scale Prestressed Concrete Containment.

Item	Test Result [2]	Proposed Simplified Approach	3-D Analysis [2]
Internal Pressure at Rebar Yield MPa (psig)	0.586 (85)	0.683 (99)	0.662 (96)
Internal Pressure at Failure MPa (psig) (Leakage > 100%)	0.772 (112)	0.738 (107)	0.738 (107)

Based on the above discussion, simplified analysis approach provides good agreement with the more detailed finite analysis and test data for the concrete containment performance under internal pressure, and was to determine Surry containment behavior.

2.2 Containment Leakage

The containment performance criteria used for severe accident analysis require prediction of leakage rate as a function of internal pressure and temperature. There is lack of experimental data for containment leakage beyond design pressure. Rizkalla et al. [9], Dameron et al. [4], and others have attempted to quantify leakage through concrete sections. This guidance cannot be used directly to determine leakage thru concrete containments in which steel liner plate is designed to act as a leakage barrier. Detailed 3-dimensional nonlinear analysis of the containments with equipment hatch and other penetrations can determine the local strains in the liner plate and concrete. The results of the 3-dimensional nonlinear analysis can be used to determine airflow through the liner plate and containment concrete. All these complicated analyses will lead to leak rate predictions with large uncertainties due to variation in the properties of the materials, quality and porosity of welds, and concrete placement.

The relationship between containment leakage and internal pressures for reinforced concrete and prestressed concrete containment model tests from References 4 and 5 is shown in Figure A-1 and Figure A-2, respectively. A review of these figures indicates that the concrete containments start to leak appreciably once the liner plate yields. The rate of leakage when the liner plate yields is about 10 times more than normal leakage of 0.10 percent of containment air mass per day at the containment design pressure. The leakage rate increases appreciably with further increase in test pressure. Once the rebars yield, the leakage rate is about 10-15 percent. Thereafter, the leakage rate continues to increase and reaches to about 60-65 percent when the strain in the rebars is about 1-2 percent. Containment pressure does not increase significantly after leakage rate exceeds 60-65 percent of the containment air mass per day. The liner welds and concrete crack after rebars and liner plate yields to create a path for leakage. The leakage occurs in areas such as equipment hatch, personnel airlocks and penetrations where local strains are substantially higher than the global strains.

The results in Figure A-1 and Figure A-2 are for scale model tests of two concrete containments. Rebar and concrete crack spacing, and aggregate size can affect the leakage rates in full size containments. However, based on the results of testing and analyses presented above, it is reasonable to conclude that all concrete containments start to leak once the rebars and liner plate yield. In addition, leakage becomes excessive once the strains in the reinforced and prestressed concrete containments reach about 2 and 1 percent respectively. Based on information of the containment model test results and analyses data presented in Figure A-1and Figure A-2, it is reasonable to assume that containment leakage is about one percent of the containment mass per day when the liner plate yields, this increases to 13 percent of containment mass per day when rebars yield. Similarly, leakage rate of 62 percent can be used in severe accident analysis when the containment global strains are 1-2 percent. Uncertainty in the leakage rate can be accounted for by conservatively reducing the yield and failure pressure calculated by simplified analysis to 85 percent of the calculated value.

The location of the leakage can have a significant effect on the results of the severe accident analysis and dose rates. For instance, if the containment leakage occurs thru penetrations that are located inside adjoining plant buildings, the fission product release into atmosphere would be significantly less as compared to direct leakage to the environment. Previously, some of the severe accident analyses were based on the assumption that the leakage takes place at the top of the containment dome. A more realistic approach is to consider leakage to occur at equipment hatch and other penetrations as demonstrated by tests data, and non-linear analyses.

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3.0 ANALYSIS

3.1 Containment Internal Pressure at Liner Plate Yield

The liner plate material yield strength is less than yield strength of rebars. However, modulus of elasticity of carbon steel liner plate and rebars is about the same. Therefore, the liner plate is likely to yield first under internal pressure. When the liner plate yields, the stress in the rebar and liner plate would be the same and equal to yield strength of the liner plate. Using this approach, the internal pressure at which liner plate will yield $P_{lineryield}$ was calculated as follows

$$P_{lineryield} = (A_{hoop} + A_{liner}) * Y_{liner} / R$$

where:

$P_{lineryield}$ = Containment pressure at which liner plate yield
A_{hoop} = Area of the hoop rebars = 18.777 in^2/ft (Reference 10)
A_{liner} = Area of the 3/8" thick liner plate = 4.5 in^2/ft (Reference 11)
Y_{rebar} = Yield stress of the rebar = 50,000 psi (Reference 12)
Y_{liner} = Yield stress of the liner plate = 32,000 psi (Reference 11)
R = Radius of the containment = 63 feet (Reference 11)

Using the above listed values:

$$P_{lineryield} = 82.10 \text{ psi}$$

To account for uncertainties in material properties and other simplifying assumptions, this pressure at liner plate yield was reduced to 85 percent of the calculated value for the MELCOR analysis.

Therefore:

$$P_{@lineryield} = 69.79 \text{ psi}$$

3.2 Containment Internal Pressure at Rebar Yield

$$P_{yield} = (A_{hoop} * Y_{rebar} + A_{liner} * Y_{liner}) / R$$

Using this equation

$$P_{yield} = 119.36 \text{ psig}$$

To account for uncertainties in material properties and other simplifying assumptions, this pressure at yield will be reduced to 85 percent of the calculated value for the MELCOR analysis.

Therefore:

$P_{@yield} = 101.46$ psi

3.3 Containment Internal Pressure at 2% Strain

$P_{fail} = (A_{hoop} * Y_{rebar@2\%} + A_{liner} \cdot Y_{liner@2\%}) / R$

where:

P_{fail} = Containment failure pressure at 2% strain

$Y_{rebar@2\%}$ = Stress in the rebar at 2% strain = 53,000 psi

$Y_{liner@2\%}$ = Stress in the liner at 2% strain = 34,300 psi

Using the above listed values,

$P_{fail} = 126.71$ psi

To account for uncertainties in material properties and other simplifying assumptions, this pressure at failure will be reduced to 85 percent of the calculated value for the MELCOR analysis.

Therefore:

$P_{@fail} = 107.70$ psi

3.4 Containment Leakage

Surry minimum containment free volume per Table 5.4-24 of Reference 13 = 1, 730,000 ft^3

Density of air at containment pressure of 119.36 psi and 200°F (rebar yield):

$\rho = 0.55\ lb/ft^3$, (Page A-10 of Reference 14)

Mass of air inside containment at Pyield

$Mass_{Pyield} = \rho\ V_{containment}$

$Mass_{Pyield} = 9.515 \times 10^5$ lb

Mass leak rate of the containment at Pyield:

$Massleakrate_{Pyield} = 13\%/day$

$Massleak_{perday} = Mass_{Pyield}\ Massleakrate_{Pyield}$

$Massleak_{perday} = 1.237 \times 10^5$ lb/day

Density of air at 70° F and atmospheric pressure, ρa

$\rho_a = 0.075$ lb/ft^3, (Reference 14)

Leakage flow "Q" calculation

$Q = Massleakperday / \rho_a$

$Q = 1.649 \times 10^6$ lb/day

Therefore, leakage volume

$Volleak_{\%} = Q/V_c$

$Volleak_{\%} = 95.33\%$/day

Table B.1-5 provides a summary of these results for containment air temperature of 200° F.

Table B.1-5 Recommended Leakage Rates and Areas for the Surry Analyses.

Containment Pressure (psig)	Containment Temperature (°F)	Containment Leakage (% Mass/day)
45.00	70	0.1
69.79	200	1.0
101.46	200	13
107.70	200	62
123.20	200	352

4.0 CONCLUSION

Simplified analysis of concrete containment provides good agreement with the more detailed finite analysis and test data for the concrete containment performance under internal pressure, and has been used in this report to determine Surry containment.

Extensive research and scale model testing of reinforced and prestressed concrete containments indicate that concrete containments start to leak before a complete rupture or failure. Unless the rate of pressurization is extremely rapid, concrete containments are not likely to have a catastrophic failure. There is some uncertainty about containment leakage rate; however, concrete containments start to leak significantly after the liner plate and rebar yield. Leakage rate becomes excessive after the global strains in the liner plate and rebar reach 2%. Surry containment leakage rates under different internal pressures have been determined using the results of previous tests performed on different scale models of the concrete containments. The leakage rates are presented in Table B.1-5. Most of the leakage is likely to occur at equipment hatch and other penetrations as demonstrated by tests data, and non-linear analyses.

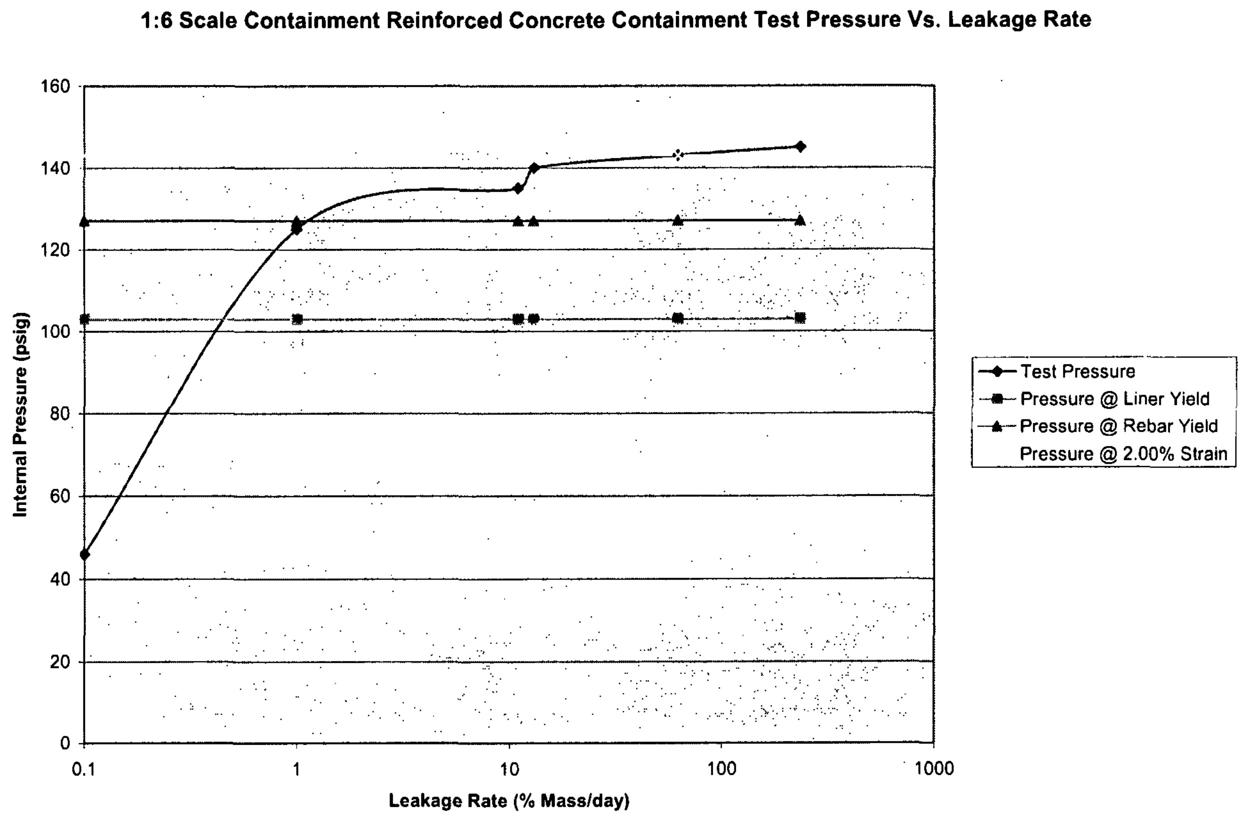

Figure A-1 1:6 Scale Prestressed Concrete Containment Test Pressure Versus Leakage Rate.

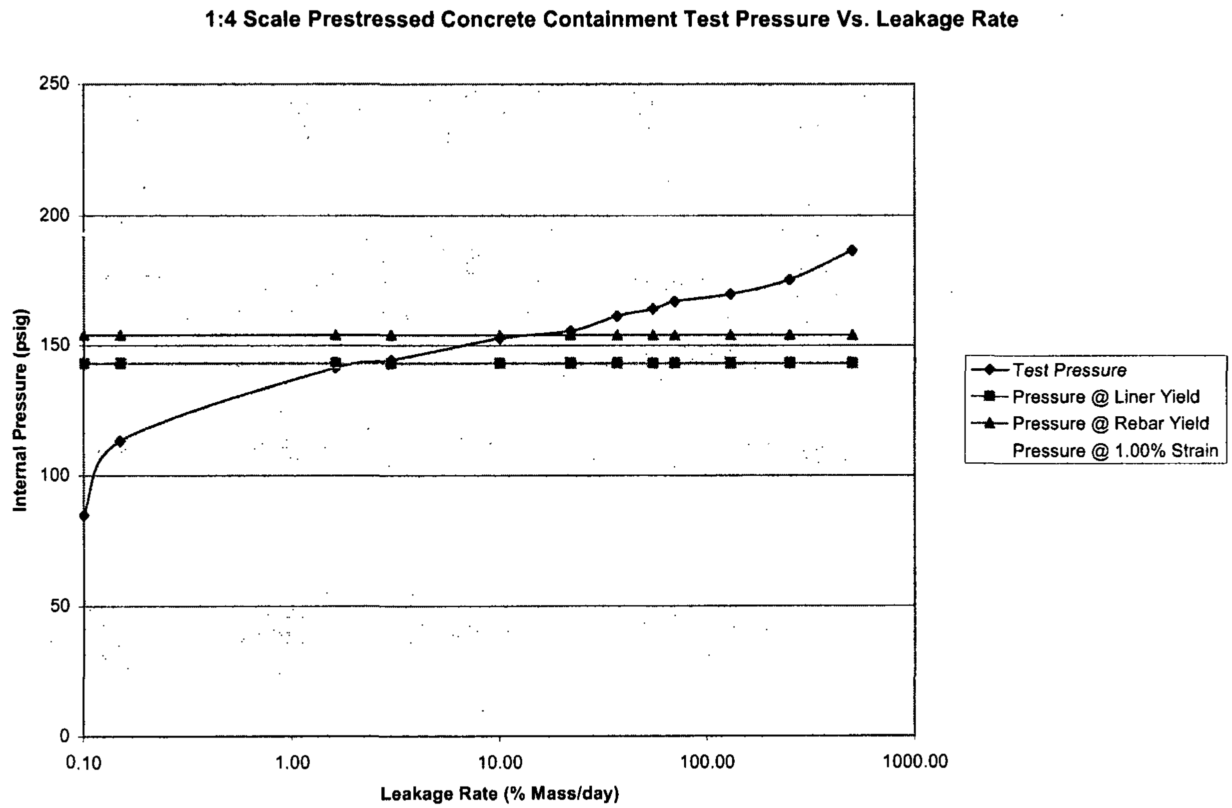

Figure A-2 1:4 Scale Prestressed Concrete Containment Test

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APPENDIX B.2 SURRY RADIONUCLIDE INVENTORY

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The following tables summarize the radionuclide core inventory for the Surry plant at the time of shutdown for each of the accident progression scenarios considered in this report.

Table B.2-1 Surry radionuclide core inventory and class definition.

Radionuclide Class Name	Representative Element	Member Elements	Total Mass (kg)
Noble Gas	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N	448.2
Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu	251.7
Alkaline Earths	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm	187.6
Halogens	I	F, Cl, Br, I, At	17.0
Chalcogens	Te	O, S, Se, Te, Po	40.9
Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni	309.5
Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W	323.5
Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C	1226.0
Trivalents	La	Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf	621.2
Uranium	U	U	66770.0
More Volatile Main Group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi	7.26
Less Volatile Main Group	Sn	Ga, Ge, In, Sn, Ag	9.19

Table B.2-2 Surry noble gas radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Kr-85	2.94E+16
Kr-85m	8.07E+17
Kr-87	1.60E+18
Kr-88	2.14E+18
Xe-133	6.07E+18
Xe-135	1.80E+18
Xe-135m	1.29E+18

Table B.2-3 Surry alkali metals radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Cs-134	4.32E+17
Cs-136	1.57E+17
Cs-137	3.05E+17
Rb-86	5.36E+15
Rb-88	2.16E+18

Table B.2-4 Surry alkali earths radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Ba-139	5.54E+18
Ba-140	5.37E+18
Sr-89	2.98E+18
Sr-90	2.27E+17
Sr-91	3.75E+18
Sr-92	4.00E+18
Ba-137m	2.92E+17

Table B.2-5 Surry halogen radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
I-131	2.78E+18
I-132	4.08E+18
I-133	5.76E+18
I-134	6.48E+18
I-135	5.49E+18

Table B.2-6 Surry chalcogen radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Te-127	2.60E+17
Te-127m	4.22E+16
Te-129	7.79E+17
Te-129m	1.49E+17
Te-131m	5.71E+17
Te-132	4.29E+18
Te-131	2.55E+18

Table B.2-7 Surry platinoid radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Rh-105	2.90E+18
Ru-103	4.61E+18
Ru-105	3.14E+18
Ru-106	1.40E+18
Rh-103m	4.61E+18
Rh-106	1.56E+18

Table B.2-8 Surry early transition element radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Nb-95	5.18E+18
Co-58	4.79E+13
Co-60	2.65E+14
Mo-99	5.68E+18
Tc-99m	5.03E+18
Nb-97	5.24E+18
Nb-97m	4.95E+18

Table B.2-9 Surry tetravalent radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Ce-141	4.87E+18
Ce-143	4.55E+18
Ce-144	3.42E+18
Np-239	5.67E+19
Pu-238	8.31E+15
Pu-239	9.56E+14
Pu-240	1.17E+15
Pu-241	3.39E+17
Zr-95	4.96E+18
Zr-97	5.00E+18

Table B.2-10 Surry trivalent radionuclide class specific isotopic activity at the time of reactor shutdown

Isotope	Activity (Bq)
Am-241	3.43E+14
Cm-242	1.14E+17
Cm-244	1.13E+16
La-140	5.67E+18
La-141	5.10E+18
La-142	4.92E+18
Nd-147	2.04E+18
Pr-143	4.65E+18
Y-90	2.39E+17
Y-91	3.93E+18
Y-92	4.11E+18
Y-93	4.62E+18
Y-91m	2.20E+18
Pr-144	3.63E+18
Pr-144m	5.06E+16