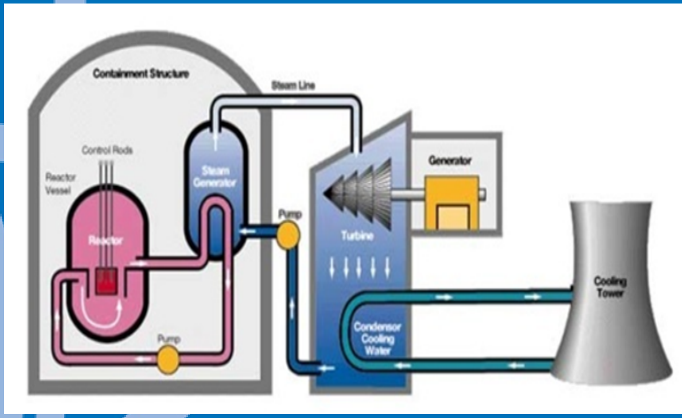
U.S. NRC Level 3 Probabilistic Risk Assessment (PRA) Project

Volume 3c: Reactor, At-Power, Level 2 PRA for Internal Events and Floods



April 2022

United States Nuclear Regulatory Commission

Protecting People and the Environment

ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk analysis (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant, responding to Commission direction in the staff requirements memorandum (SRM) (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities" (ADAMS Accession No. ML11090A039).

As described in SECY-11-0089, the objectives of the L3PRA project are to:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data,[1] that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (NUREG-1150[2]), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low power and shutdown [LPSD] risk, multi-unit risk, other radiological sources).

- Extract new insights to enhance regulatory decision making and to help focus limited NRC resources on issues most directly related to the agency's mission to protect public health and safety.

- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.

- Demonstrate technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs.

The scope of the L3PRA project encompasses all major radiological sources on the site (i.e., reactors, spent fuel pools, and dry cask storage), all internal and external hazards, and all modes of plant operation. Fresh nuclear fuel, radiological waste, and minor radiological sources (e.g., calibration devices) are not included as part of the scope. In addition, deliberate malevolent acts (e.g., terrorism and sabotage) are excluded from the scope of this study.

This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-power, Level 2 PRA model for internal events and internal floods for a single unit. The analyses documented herein are based information for the reference plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained.[3]

[1] "State-of-practice" methods, tools, and data refer to those that are routinely used by the NRC and industry or have acceptance in the PRA technical community. While the L3PRA project is intended to be a state-of-practice study, note that there are several technical areas within the project scope that necessitated advancements in the state-of-practice (e.g., modeling of multi-unit site risk, modeling of spent fuel in pools or casks, and of human reliability analysis for other than internal events and internal fires).

[2] NUREG-1150, "Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants," December 1990.

[3] An overview report, which covers all three PRA levels, has been created for each major element of the L3PRA project scope (e.g., for the combined internal event and internal flood PRAs for a single reactor unit operating at full

A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources. Furthermore, some future advanced light water reactor (ALWR) and advanced non-light water reactor (NLWR) applicants may rely heavily on results of analyses similar to those used in the L3PRA project to establish their licensing basis and design basis by using the Licensing Modernization Project (LMP) (NEI 18-04, Rev. 1) which was recently endorsed via RG 1.233. Licensees who use the LMP framework are required to perform Level 3 PRA analyses. Therefore, another potential use of the methodology and insights generated from this study is to inform regulatory, policy, and technical issues pertaining to ALWRs and NLWRs.

CAUTION: While the L3PRA project is intended to be a state-of-practice study, due to limitations in time, resources, and plant information, some technical aspects of the study were subjected to simplifications or were not fully addressed. As such, inclusion of approaches in the L3PRA project documentation should not be viewed as an endorsement of these approaches for regulatory purposes.

power). These overview reports include a reevaluation of plant risk based on a set of updated plant equipment and PRA model assumptions (e.g., incorporation of the current reactor coolant pump shutdown seal design at the reference plant and the potential impact of the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Mitigation Capability [FLEX], both of which reduce the risk to the public).

FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk analysis (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant, responding to Commission direction in the staff requirements memorandum (SRM) (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment (PRA) Activities” (ADAMS Accession No. ML11090A039).

Licensee information used in performing the Level 3 PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models used in its regulatory decisionmaking. **As such, use of L3PRA project reports to assess the risk from the reference plant is not appropriate and these reports will not be the basis for any regulatory decision associated with the reference plant.**

Each set of L3PRA project reports covering the Level 1, 2, and 3 PRAs for a specific site radiological source, plant operating state, and hazard group is accompanied by an overview report. The overview reports summarize the results and insights from all three PRA levels.

In order to provide results and insights better aligned with the current design and operation of the reference plant, the overview reports also provide a reevaluation of the plant risk based on a set of new plant equipment and PRA model assumptions and compare the results of the reevaluation to the original study results. This reevaluation reflects the current reactor coolant pump (RCP) shutdown seal design at the reference plant, as well as the potential impact of FLEX strategies,[4] both of which reduce the risk to the public.

A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources (i.e., spent fuel in pools and dry storage casks). These insights may be used to further enhance the regulatory framework and decisionmaking and to help focus limited agency resources on issues most directly related to the agency’s mission to protect public health and safety. More specifically, potential future uses of the Level 3 PRA project can be categorized as follows (a more detailed list is provided in SECY-12-0123, “Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC’s Regulatory Framework,” dated September 13, 2012):

- enhancing the technical basis for the use of risk information (e.g., obtaining updated and enhanced understanding of plant risk as compared to the Commission’s safety goals)
- improving the PRA state-of-practice (e.g., demonstrating new methods for site risk assessments, which may be particularly advantageous in addressing the risk from advanced reactor designs, or in supporting the evaluation of the potential impact that a

[4] FLEX refers to the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Mitigation Capability. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from offsite.

multi-unit accident, or an accident involving spent fuel, may have on the efficacy of the emergency planning zone in protecting public health and safety)

- identifying safety and regulatory improvements (e.g., identifying potential safety improvements that may lead to either regulatory improvements or voluntary implementation by licensees)
- supporting knowledge management (e.g., developing or enhancing in-house PRA technical capabilities)

In addition, the overall Level 3 PRA project model can be exercised to provide insights with regard to other issues not explicitly included in the current project scope (e.g., security-related events or the use of accident tolerant fuel). Furthermore, some future advanced light water reactor (ALWR) and advanced non-light water reactor (NLWR) applicants may rely heavily on the results of analyses similar to those used in the L3PRA project to establish their licensing basis and design basis by using the Licensing Modernization Project (LMP) (NEI 18-04, Rev. 1) which was recently endorsed via RG 1.233. Licensees who use the LMP framework are required to perform Level 3 PRA analyses. Therefore, another potential use of the methodology and insights generated from this study is to inform regulatory, policy, and technical issues pertaining to ALWRs and NLWRs.

The results and perspectives from this report, as well as all other reports prepared in support of the Level 3 PRA project, will be incorporated into a summary report to be published after all technical work for the Level 3 PRA project has been completed.

ABBREVIATIONS AND ACRONYMS

AC	alternating current
ACC	accumulator
ACCW	auxiliary component cooling water
AFW	auxiliary feedwater
ALWR	advanced light-water reactor
AOP	abnormal operating procedure
AOV	air-operated valve
ARV	atmospheric relief valve
ATWS	anticipated transient without scram
BAF	bottom of active fuel
BD	blowdown
BMT	basemat melt-through
BOC	beginning-of-cycle
BWR	boiling-water reactor
CA	computational aid
CC	capability category
CCF	common-cause failure
CCFP	conditional containment failure probability
CCU	containment cooling unit
CCW	component cooling water
CD	core damage
CDE	committed dose equivalent
CDF	core damage frequency
CET	containment event tree
CETC	core exit thermocouple column
CF	containment failure
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CHR	containment heat removal
CIF	containment isolation failure
CIS	containment isolation system
CLOOP	consequential loss of offsite power
CR	control room
CRI	control room isolation
Cs	cesium
CS	containment spray
CSFST	critical safety function status tree
C-SGTR	consequential steam generator tube rupture
CST	condensate storage tank
CV	control valve

CVH	control volume hydrodynamic
DBA	design basis accident
DC	direct current
DCH	direct containment heating
DET	decomposition event tree
DF	decontamination factor
DFC	diagnostic flow chart
DG	diesel generator
EAL	emergency action level
ECCS	emergency core cooling system
ECF	early containment failure
ED	emergency director
EDG	emergency diesel generator
EDMG	extensive damage mitigation guidance
EF	error factor
EOC	end-of-cycle
EOF	emergency operations facility
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESFAS	engineered safety features actuation system
EXLOCA	excessive loss of coolant accident
FTC	fail to close
FTO	fail to open
FW	feedwater
FWST	feedwater storage tank
GE	general emergency
HEP	human error probability
HFE	human failure event
HPI	high pressure injection
HPME	high-pressure melt ejection
HPR	high pressure recirculation
HRA	human reliability analysis
HVAC	heating, ventilation and air conditioning
ICF	intermediate containment failure
IE	initiating event
INL	Idaho National Laboratory
IPE	Individual Plant Examination
ISGTR	induced steam generator tube rupture
ISLOCA	interfacing systems loss of coolant accident
IVSE	in-vessel steam explosion
IVR	in-vessel retention

LBLOCA	large-break loss-of-coolant accident
LCF	late containment failure
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LRF	large release frequency
MACCS	MELCOR Accident Consequence Code System
MCCI	molten core concrete interaction
MCR	main control room
MDAFW	motor-driven auxiliary feedwater
MDP	motor-driven pump
MFW	main feedwater
MLOCA	medium loss-of-coolant accident
Mo	molybdenum
MOC	middle-of-cycle
MOV	motor operated valve
MSIV	main steam isolation valve
MSSV	main steam safety valve
MU	model uncertainty
NEI	Nuclear Energy Institute
NOCF	no containment failure
NPP	nuclear power plant
NR	narrow range
NRC	Nuclear Regulatory Commission
NSCW	nuclear service cooling water
ORNL	Oak Ridge National Laboratory
PCT	peak clad temperature
PDS	plant damage state
PI-SGTR	pressure-induced steam generator tube rupture
PORV	power operated relief valve
PPAFES	piping penetration area filtration and exhaust system
PRA	probabilistic risk assessment
PRT	pressurizer relief tank
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
PZR	pressurizer
QHO	quantitative health objective
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RG	regulatory guide
RHR	residual heat removal

RN	radionuclide
RPS	reactor protection system
RPV	reactor pressure vessel
RVLIS	reactor vessel level indicating system
RWST	reactor water storage tank
SAE	site area emergency
SAEG	severe accident exit guideline
SAG	severe accident guideline
SAMG	severe accident mitigation guidelines
SAPHIRE	Systems Analysis Program for Hands-On Integrated Reliability Evaluations
SBO	station blackout
SC	severe challenge
SCG	severe challenge guideline
SCST	severe challenge status tree
SFP	spent fuel pool
SG	steam generator
SGT	steam generator tube
SGTR	steam generator tube rupture
SGTR-UNI	unisolable steam generator tube rupture
SI	safety injection
SLOCA	small loss of coolant accident
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk
SRM	staff requirements memorandum
SRV	safety relief valve
SV	solenoid valve
TAF	top of active fuel
TAG	technical advisory group
TDAFW	turbine-driven auxiliary feedwater
TEDE	total effective dose equivalent
TISGTR	thermally induced steam generator tube rupture
TMI	Three Mile Island
TSC	technical support center
TUPA	trial use and pilot application
UA	uncertainty analysis
UHS	ultimate heat sink
VB	vessel breach
WHO	World Health Organization
XLOCA	excessive loss of coolant accident

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1. Introduction

This report documents the single-unit, reactor, at-power, Level 2 PRA for internal events and internal floods that supports the U.S. Nuclear Regulatory Commission (NRC) full-scope site Level 3 PRA project (L3PRA project) for a two-unit pressurized-water reactor (PWR) reference plant. The results provided in this report are for a single unit—a subsequent report in this series addresses multi-unit risk.

Licensee information used in performing the L3PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. (For example, the L3PRA does not reflect the current versions of the severe accident management guidelines, which could influence the Level 2 PRA modeling.) In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models used in its regulatory decisionmaking. **As such, use of this report to assess the risk from the reference plant is not appropriate and this report will not be the basis for any regulatory decision associated with the reference plant.**

Since the L3PRA project involves multiple PRA models, each of these models should be considered a "living PRA" until the entire project is complete. It is anticipated that the models and results of the L3PRA project are likely to evolve over time, as other parts of the project are developed, or as other technical issues are identified. As such, the final models and results of the project (which will be documented in a summary report to be published after all technical work for the L3PRA project has been completed) may differ in some ways from the models and results provided in the current report.

The series of reports for the L3PRA project are organized as follows:

Volume 1: Summary (to be published last)

Volume 2: Background, site and plant description, and technical approach

Volume 3: Reactor, at-power, internal event and flood PRA
- Volume 3x: Overview
- Volume 3a: Level 1 PRA for internal events (Part 1 – Main Report; Part 2 – Appendices)
- Volume 3b: Level 1 PRA for internal floods
- Volume 3c: Level 2 PRA for internal events and floods
- Volume 3d: Level 3 PRA for internal events and floods

Volume 4: Reactor, at-power, internal fire and external event PRA
- Volume 4x: Overview
- Volume 4a: Level 1 PRA for internal fires
- Volume 4b: Level 1 PRA for seismic events
- Volume 4c: Level 1 PRA for high wind events and other hazards evaluation
- Volume 4d: Level 2 PRA for internal fires and seismic and wind-related events
- Volume 4e: Level 3 PRA for internal fires and seismic and wind-related events

Volume 5: Reactor, low power and shutdown, internal event PRA

Volume 5x: Overview
Volume 5a: Level 1 PRA for internal events
Volume 5b: Level 2 PRA for internal events
Volume 5c: Level 3 PRA for internal events

Volume 6: Spent fuel pool all hazards PRA
Volume 6x: Overview
Volume 6a: Level 1 and Level 2 PRA
Volume 6b: Level 3 PRA

Volume 7: Dry cask storage, all hazards, Level 1, Level 2, and Level 3 PRA

Volume 8: Integrated site risk, all hazards, Level 1, Level 2, and Level 3 PRA

The details of the reactor, at-power, Level 2 PRA for internal events and internal floods, including modeling assumptions, scenario descriptions, and sources of uncertainty are documented in this report. **Section 1.1** provides an overview of the Level 2 PRA model development. **Section 1.2** describes the limitations of the Level 2 PRA model. **Section 1.3** identifies various computer codes and software that were used in the development and application of the Level 2 PRA model. **Section 2** provides the description of the technical elements of the Level 2 PRA and **Section 3** provides a list of references.

CAUTION: While the L3PRA project is intended to be a state-of-practice study, due to limitations in time, resources, and plant information, some technical aspects of the study were subjected to simplifications or were not fully addressed. As such, inclusion of approaches in the L3PRA project documentation should not be viewed as an endorsement of these approaches for regulatory purposes.

1.1. Overview of Level 2 PRA Model Development

The Level 2 PRA for internal events and floods builds from the corresponding Level 1 PRA for internal events (NRC, 2022a) and the Level 1 PRA for internal floods (NRC, 2022b). The probabilistic logic model was extended (in an integrated, single-model fashion) to carry sequences beyond core damage, to their ultimate Level 2 PRA end-state in terms of radiological release to the environment. The Level 2 PRA's primary purpose was to provide a characterization of accident progression and radiological release suitable for use in completing the Level 3 PRA (consequence analysis) portion of the L3PRA project. This situation is depicted in **Figure 1-1**. The Level 2 PRA has many similarities with the severe accident analysis portion of a deterministic consequence analysis. The main difference is that in addition to answering the questions of "what can go wrong?" and "what are the consequences?," the Level 2 PRA also answers the question of "how likely is it to happen?" To do this, the Level 2 PRA generally considers a broad range of phenomena, system failures, and operator actions.

Pieces of the L3PRA project Level 2 PRA have origins in previous work, but the model itself is generally stand-alone. Consistent with the overall L3PRA project, the Level 2 PRA is a state-of-practice study, except in areas where there was no mature state-of-practice (e.g., post-core-damage human reliability analysis).

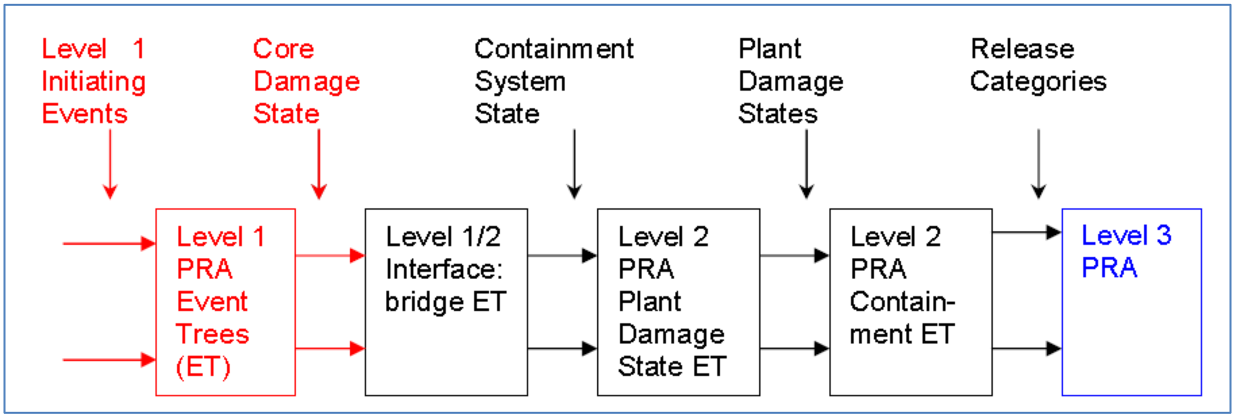

Figure 1-1: Flow of information from Level 1 to Level 3 PRA

1.2. Level 2 PRA Model Limitations

Table 1-1: Key Limitations That Could Impact Potential Applications

Item	Description
Scope	
Airborne pathway	The focus of this model is airborne radiological releases only, and modeling decisions (e.g., release category selection) were made as such. When relevant, surface aqueous releases are noted, but only airborne releases are passed to the offsite consequence analysis.
Level-of-detail	
Surrounding structures	Modeling of the surrounding structures was aimed at providing coarse perspectives on their effect on accident progression and fission product scrubbing. The estimated conditions in these structures (e.g., temperatures and combustible gas concentrations) should be used with caution, owing to the resolution (e.g., nodalization, modeling of internal structures) of their description within the MELCOR model.
Data	Data (unreliability) modeling was not performed for some systems, as discussed further in **Section 2.4.1**. This limits the use of the PRA in assessing the impact of maintenance activities.
Survivability	Survivability of equipment was handled qualitatively, as discussed in **Section 2.3.6**. As such, the model may be limited in its usefulness in assessing the merit of activities related to hardening equipment.
Modeling assumptions	
Numerous modeling assumptions	These were addressed in the form of model uncertainties, which are discussed in **Sections 2.4.7** and **2.5.3** (and in Appendix C).

1.3. Main Computer Codes Used

The two computer codes used most prominently in the Level 2 PRA were MELCOR and Systems Analysis Program for Hands-On Integrated Reliability Evaluations (SAPHIRE). MELCOR's use is discussed predominantly in **Sections 2.3.2** through **2.3.5**. Since the time that the Source Term Code Package was used for the NUREG-1150 analysis, significant development has been undertaken to consolidate and advance the severe accident modeling modules into a single code capable of performing integrated severe accident analysis (i.e., MELCOR). The MELCOR code itself contains the necessary empirical and analytical formulations to capture major reactor thermal-hydraulic and severe accident physics. The MELCOR input includes the geometry of major systems, structures, and components (SSCs), the nominal operating conditions, and the response of major automatic control and protection systems. Initial and boundary conditions were specified for each calculation to define the plant configuration (including equipment and operator successes and failures) that were relevant to the PRA sequence or cutset being analyzed.

SAPHIRE is a fault tree and event tree PRA code designed to support event and condition assessment. It was used here to construct and solve the bridge event tree (**Section 2.1.1**), the plant damage state (PDS) tree (**Section 2.1.2**), the containment event tree (**Section 2.4.2**), and decomposition event trees (**Section 2.4.3**). This was all done in the same SAPHIRE project as the Level 1 PRA, resulting in a single linked Level 1 and Level 2 PRA model. As discussed later, the PDS tree was used strictly as a sorting tool to group similar sequences, whereas the bridge and containment trees were akin to Level 1[5] event trees (i.e., they apportion frequency based on successes and failures). Decomposition event trees also apportion frequency based on successes and failures, but in a different way; each decomposition event tree sequence was assigned to one of a few end-states, each of which maps to one of the branch points for the top event that called the decomposition event tree. Ultimately, a decomposition event tree is a more compact and traceable way of capturing Boolean logic that would otherwise require numerous fault trees. Linkage rules were used to define how branch assignments were made (e.g., forced up-branch, forced down-branch, assignment of a split fraction, or assignment of a fault tree) in all these event tree types. Lastly, process flags were used to define what assumptions SAPHIRE made when quantifying fault trees.

2. Technical Elements

2.1. Level 1/2 PRA Interface – Accident Sequence Grouping

The Level 1/2 PRA Interface consists of five interrelated steps:

1. Development of the bridge event tree
2. Development of plant damage state binning
3. Review of the resulting plant damage states
4. Iteration on the Level 1 PRA modeling, as necessary
5. Establishment of criteria for, and selection of, representative sequences

The objective of the first step is to add additional containment systems to “the end” of the Level 1 PRA sequences. The objective of the second step is to develop the plant damage states used to identify a manageable number of sequences for deterministic investigation. The objective of the third step is to review the resulting plant damage states to ensure adequate transfer of information across the Level 1 / Level 2 interface, such that information important to the Level 2 analysis (e.g., initiator and support system dependencies, operator action dependencies) is transferred and that credit is not being given for equipment or operator actions that were not appropriate for that plant damage state. The objective of the fourth step is to re-visit and refine any Level 1 modeling assumptions that adversely affect the plant damage state binning. The objective of the fifth step is to establish the criteria used for the selection of representative

[5] Occasionally, this report uses the phrases “Level 1” and “Level 2” as short-hand for “Level 1 PRA” and “Level 2 PRA,” respectively.

sequences and selecting those sequences for each plant damage state. Each of these steps are discussed in further detail in this section.

Table 2-1 shows the rank-ordered core damage frequency (CDF) results of the internal events and internal floods Level 1 PRA, by initiator, at a truncation of 1×10^{-11} per reactor-critical-year (/rcy). The Level 1 probabilistic logic model that generated these results was the starting point for the Level 2 PRA.

Table 2-1: Level 1 Core Damage Frequency (CDF) Contribution Results

Initiating event (# of sequences, including transfers)	CDF (/rcy)	Individual contribution	Cumulative contribution
1-FPI-LOOPGR (960 Seqs.)	2.27×10^{-5}	29.5%	29.5%
1-FPI-LOOPSC (960 Seqs.)	1.30×10^{-5}	16.9%	46.5%
1-FPI-LOOPWR (960 Seqs.)	1.10×10^{-5}	14.4%	60.9%
1-FPI-LONSCW (642 Seqs.)	8.82×10^{-6}	11.5%	72.4%
1-FPI-OTRANS (744 Seqs.)	2.96×10^{-6}	3.9%	76.2%
1-FPI-MLOCA (42 Seqs.)	2.64×10^{-6}	3.4%	79.7%
1-FPI-LOOPPC (960 Seqs.)	2.39×10^{-6}	3.1%	82.8%
1-FPI-LO4160VA (486 Seqs.)	1.85×10^{-6}	2.4%	85.2%
1-FPI-SSBO (264 Seqs.)	1.57×10^{-6}	2.1%	87.2%
1-FPI-LO4160VB (486 Seqs.)	1.26×10^{-6}	1.6%	88.9%
1-FPI-LO125BD1 (486 Seqs.)	1.24×10^{-6}	1.6%	90.5%
1-FPI-TTRIP (744 Seqs.)	1.24×10^{-6}	1.6%	92.1%
1-FPI-LOSINJ (630 Seqs.)	1.24×10^{-6}	1.6%	93.7%
1-FPI-RTRIP (642 Seqs.)	1.14×10^{-6}	1.5%	95.2%
1-FPI-LOMFW (744 Seqs.)	5.88×10^{-7}	0.8%	96.0%
1-FPI-LOCHS (486 Seqs.)	5.36×10^{-7}	0.7%	96.7%
1-FPI-LO125AD1 (486 Seqs.)	4.29×10^{-7}	0.6%	97.2%
1-FPI-SLOCA (150 Seqs.)	2.39×10^{-7}	0.3%	97.6%
1-FPI-ISL-RHR-HLS (6 Seqs.)	2.25×10^{-7}	0.3%	97.8%
1-FLI-AB_C113_LF1 (744 Seqs.)	1.77×10^{-7}	0.2%	98.1%
1-FPI-SGTR (186 Seqs.)	1.50×10^{-7}	0.2%	98.3%
1-FLI-AB_C120_LF (744 Seqs.)	1.48×10^{-7}	0.2%	98.5%
1-FPI-SSBI (264 Seqs.)	1.19×10^{-7}	0.2%	98.6%
1-FLI-AB_C115_LF (744 Seqs.)	1.05×10^{-7}	0.1%	98.8%
1-FLI-CB_123_SP (264 Seqs.)	1.03×10^{-7}	0.1%	98.9%
1-FLI-CB_122_SP (264 Seqs.)	1.03×10^{-7}	0.1%	99.0%
1-FPI-XLOCA (6 Seqs.)	1.00×10^{-7}	0.1%	99.2%
1-FPI-ISINJ (774 Seqs.)	9.63×10^{-8}	0.1%	99.3%
1-FPI-LO120VAB (744 Seqs.)	9.61×10^{-8}	0.1%	99.4%
1-FPI-LOACCW (744 Seqs.)	6.16×10^{-8}	0.1%	99.5%
1-FLI-AB_108_SP1 (264 Seqs.)	6.07×10^{-8}	0.1%	99.6%
1-FLI-AB_108_SP2 (264 Seqs.)	6.07×10^{-8}	0.1%	99.6%
1-FPI-ISL-RHR-CLI-A (6 Seqs.)	4.17×10^{-8}	0.1%	99.7%
1-FPI-ISL-RHR-CLI-B (6 Seqs.)	4.17×10^{-8}	0.1%	99.8%
1-FPI-LLOCA (30 Seqs.)	3.76×10^{-8}	0.0%	99.8%
1-FPI-ISL-RCP-S1LO (6 Seqs.)	3.45×10^{-8}	0.0%	99.8%
1-FPI-LOIAS (486 Seqs.)	2.28×10^{-8}	0.0%	99.9%
1-FLI-CB_A60 (264 Seqs.)	1.82×10^{-8}	0.0%	99.9%
1-FLI-CB_A48 (486 Seqs.)	1.66×10^{-8}	0.0%	99.9%
1-FLI-TB_500_LF (486 Seqs.)	1.64×10^{-8}	0.0%	99.9%

Table 2-1: Level 1 Core Damage Frequency (CDF) Contribution Results

Initiating event (# of sequences, including transfers)	CDF (/rcy)	Individual contribution	Cumulative contribution
1-FLI-DGB_101_LF (744 Seqs.)	7.58×10^{-9}	0.0%	100.0%
1-FLI-AB_D74_FP (744 Seqs.)	6.25×10^{-9}	0.0%	100.0%
1-FLI-AB_C118_LF (744 Seqs.)	5.81×10^{-9}	0.0%	100.0%
1-FLI-AB_B08_LF (642 Seqs.)	5.70×10^{-9}	0.0%	100.0%
1-FLI-DGB_103_LF (744 Seqs.)	5.54×10^{-9}	0.0%	100.0%
1-FLI-TB_500_LF-CDS (744 Seqs.)	4.12×10^{-9}	0.0%	100.0%
1-FLI-AB_B50_JI (744 Seqs.)	2.52×10^{-9}	0.0%	100.0%
1-FLI-AB_B24_LF2 (744 Seqs.)	2.52×10^{-9}	0.0%	100.0%
1-FLI-AB_A20 (744 Seqs.)	1.67×10^{-9}	0.0%	100.0%
1-FLI-TB_500_HI2 (744 Seqs.)	5.88×10^{-10}	0.0%	100.0%
1-FLI-TB_500_HI1 (486 Seqs.)	4.86×10^{-10}	0.0%	100.0%
1-FLI-AB_D78_FP (744 Seqs.)	1.89×10^{-10}	0.0%	100.0%
1-FLI-AB_A20_FP (744 Seqs.)	8.83×10^{-11}	0.0%	100.0%
1-FPI-LO120VAC (744 Seqs.)	2.14×10^{-11}	0.0%	100.0%
1-FPI-LO120VAD (744 Seqs.)	2.14×10^{-11}	0.0%	100.0%
1-FPI-LO120VBC (744 Seqs.)	2.14×10^{-11}	0.0%	100.0%
1-FPI-LO120VBD (744 Seqs.)	2.14×10^{-11}	0.0%	100.0%
1-FPI-LO120VCD (744 Seqs.)	2.14×10^{-11}	0.0%	100.0%
1-FPI-ISL-RCP-TBHX (12 Seqs.)	$<1.0\times10^{-11}$	0.0%	100.0%

2.1.1. Step 1 – Development of the bridge event tree

This step functionally added additional top events associated with containment systems onto the end of the Level 1 event trees, forming the first part of the bridge between the Level 1 and Level 2 PRAs. This tree appears in virtually all Level 2 models and is often referred to by the following generally interchangeable terms: bridge event tree, containment systems event tree, containment systems transfer event tree, or extended Level 1 event tree. While this tree was a distinct event tree, use of SAPHIRE's transfer option functionally makes it a continuation of the Level 1 event trees. The containment systems top events define the initial availability/ unavailability or success/failure of these systems at the time of core damage based on the support system information resident in the Level 1 PRA model.[6] By doing this prior to plant damage state binning, it allows consideration of containment system functionality prior to parsing cutsets into bins. The following top events are treated in the bridge event tree (shown in Figure 2-1):

(1) *Containment Isolation System*

(2) *Containment Spray System*

(3) Containment Cooling Unit (CCU) System (i.e., Containment Fan Coolers)

[6] The terminology used to describe the up-branch and down-branch of these top events is complicated , because: (i) success connotes that the system was demanded and did perform, while failure connotes that the system was demanded and failed; (ii) available connotes that the system has all the necessary support to perform but may or may not actually perform if demanded, while unavailable connotes the opposite; and (iii) which of these situations actually applies at the time of core damage for a given top event is sequence-dependent (sometimes the system was demanded and succeeded or failed; sometimes it wasn't yet demanded but it will be and will succeed or fail; sometimes it never will be demanded).

Note that it was decided not to distinguish among the wide range of possible isolation failure sizes in the bridge event tree (i.e., containment was treated as either being isolated or not). However, for determining the probability of containment isolation failure, a fault tree model was constructed that included active isolation failures of 2-inch diameter pipes or larger, and pre-existing tears or maintenance errors of equivalent leakage size diameter of 1.2-inch or greater (more on the assumptions related to isolation failure modeling can be found in Appendix D of this report in Section 2, "Treatment of Bypass and Unisolated Containment Events with Low Frequency," and Section 6, "Containment Leakage, Effective Size Under Normal and Accident Conditions"). For source term characterization, a leakage size equivalent to a 2-inch diameter pipe was used for all containment isolation failures. In particular, the predicted dominance of pre-existing tears or maintenance errors (see Appendix D, Section 2) prompted this simpler treatment. Such leakage pathways can have a spectrum of sizes, but the larger the tear, the more likely it is to be discovered during normal operation.

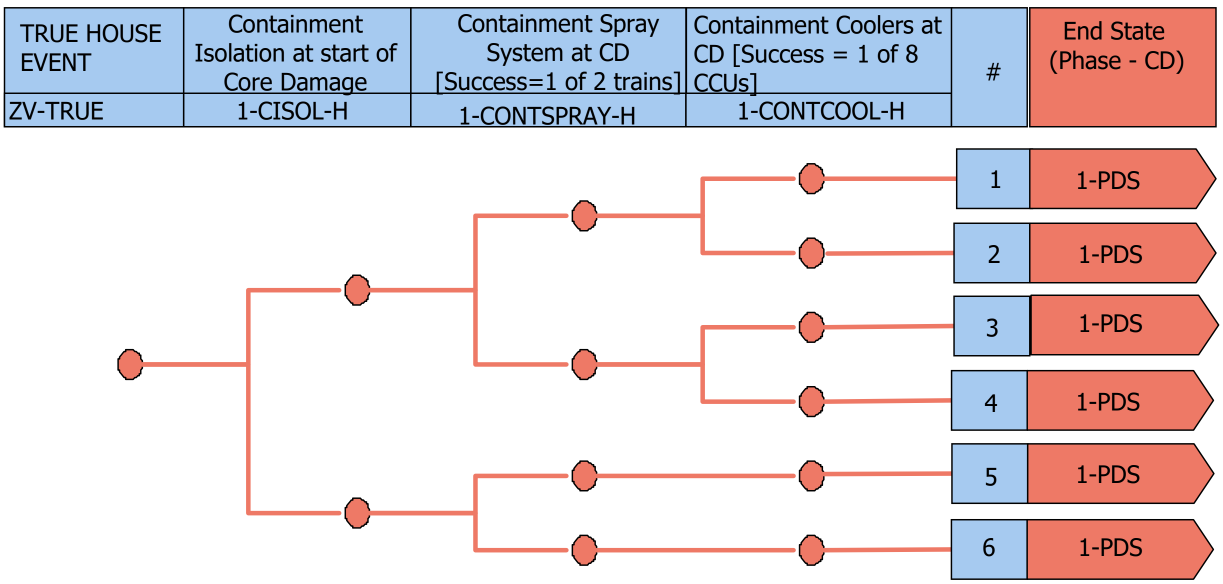

Figure 2-1: Bridge Event Tree

The bridge event tree does not branch on containment cooling availability if the containment is unisolated. The rationale is that containment cooling would be less relevant in the situation where the containment cannot significantly pressurize (due to the increased leakage associated with the isolation failure), whereas containment spray as a form of scrubbing is still of interest.

Note that the bridge event tree excludes some containment systems, namely electric hydrogen recombiners, and containment ventilation and purge systems. The hydrogen recombiners were excluded because they were retired-in-place. The containment ventilation and purge systems were excluded because they were only expected to be used post-core damage in very specific situations, namely to deliberately vent the containment atmosphere, as part of accident management actions. In the Level 2 PRA and human reliability analysis (HRA), these actions did not wind up being prompted and, therefore, were not modeled. This was partially a reflection of simplifications in the accident management modeling, as will be described later in **Sections 2.4.4** and **2.4.5**.

2.1.2. Step 2 – Development of plant damage state binning

This step includes the development of a plant damage state (PDS) binning event tree to determine the availability of certain Level 1 PRA systems that describe the reactor and containment status at the time of core damage. By defining the PDS bins, an interface between the plant systems analysis done as part of the Level 1 PRA and the containment response analysis in the Level 2 PRA was established. In the logic model, Level 1 PRA cutsets were carried forward to the Level 2 PRA, and the PDS binning did not affect this. The PDS quantification (and resulting bins) were used only to establish a “pinch-point” (i.e., a large reduction in the number of sequences that needed to be analyzed deterministically) to achieve tractability in the portions of the work that supported the logic modeling. This concept is depicted in **Figure 2-2**.

Specifically, the PDSs were used to prioritize and select representative sequences, which provided the underlying basis for the following:

- Narrative understanding of post-core-damage response
- Level 2 PRA sequence timing
- Phenomenological evaluations
- Survivability and habitability determinations
- Human reliability analysis
- Source terms characteristics
- Starting points for some model uncertainty sensitivity analyses

All internal events and internal floods initiators modeled in the Level 1 PRA were propagated to the PDSs, and ultimately to the containment event tree (CET).

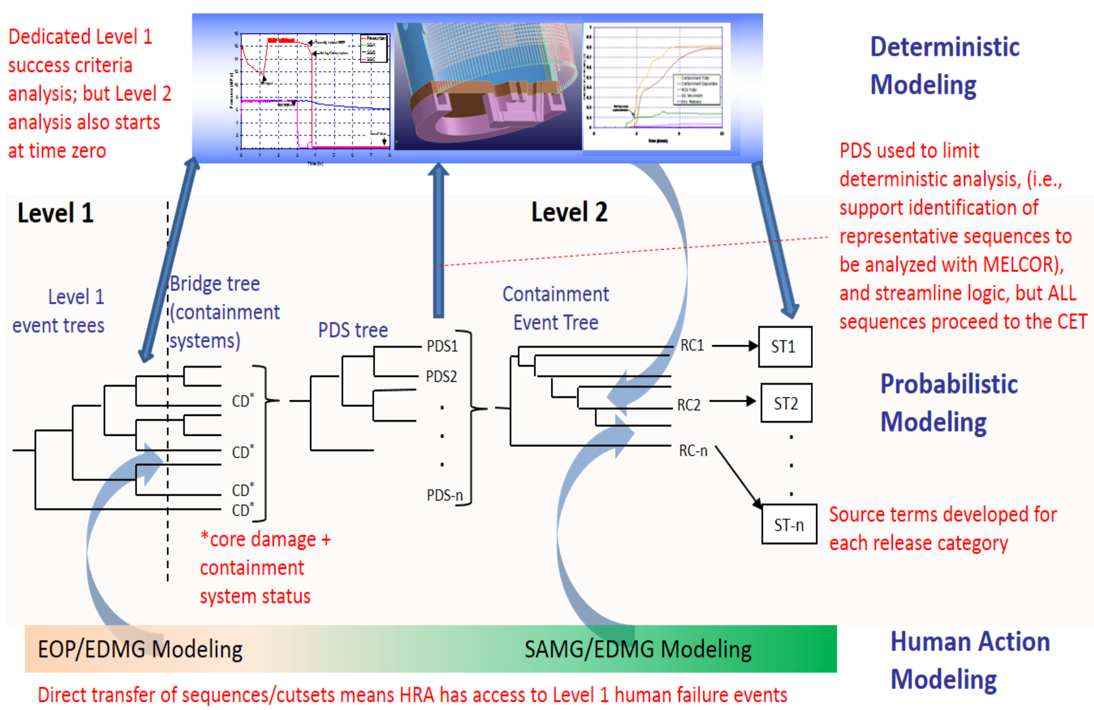

Figure 2-2: Schematic of the Role of PDS binning

The top events in the PDS event tree are those plant response attributes that have an impact on the containment response and/or fission product release to the environment. These include both the PDS binning top events:

- accident type
- steam generator cooling availability/success
- reactor water storage tank (RWST) availability/success
- emergency core cooling system (ECCS) availability/success

along with the bridge event tree top events:

- containment isolation status
- containment sprays status
- containment cooling units status

When quantifying the model for PDS frequencies, the 1-PDS-Q event tree was used that explicitly merged the 1-PDS event tree and the bridge event tree and has end-states associated with the PDS bins. When quantifying the model for release frequency, core damage sequences were routed to the bridge event tree, from there to the 1-PDS event tree, and from there to the containment event tree. The 1-PDS event tree is shown in **Figure 2-3**. A transfer event tree (1-CD-XFER) was used to readily switch between core damage quantification, PDS quantification, PDS event tree linkage rule debugging, and release frequency quantification. As a structural test of the model, and to comply with the Standard (ASME, 2014), **Table 2-2** provides a cross-walk of the modeling aspects covered by Supporting Requirements L1-A1 through L1-A3a of the Standard.

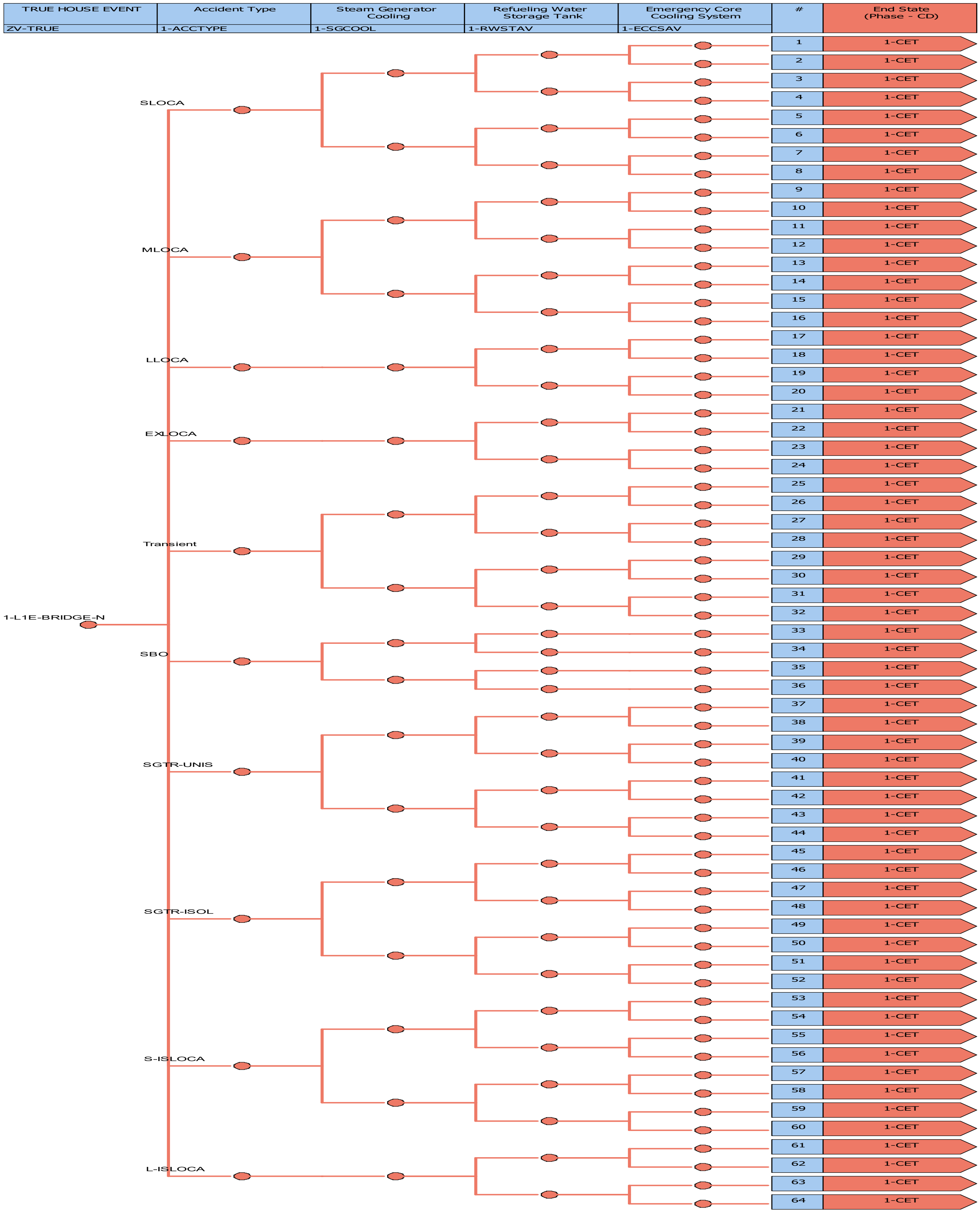

Figure 2-3: The Plant Damage State Event Tree (1-PDS)

Table 2-2: Cross-walk of Physical and Accident Sequence Characteristics from High-Level Requirement L1-A (ASME, 2014)

Physical Characteristics Identified at the Time of Core Damage (L1-A1)	Accident Sequence Characteristics That Determine the Physical Characteristics (L1-A2)	Where the Physical Characteristics are Treated in the Model (L1-A3a)
Reactor coolant system (RCS) pressure	Type of initiating event and subsequent accident sequence characteristics (i.e., accident type, operator depressurization, and availability of steam generator cooling).	The type of initiating event and subsequent accident sequence characteristics that affect the RCS pressure are queried by the PDS tree linkage rules for 1-ACCTYPE and 1-SGCOOL top events. This information is also used in 1-L2-DET-PRESVE (which is a decomposition event tree [DET] supporting the CET).
RCS configuration	Not applicable. At-power operations only.	Not applicable.
ECCS status	Type of initiating event and subsequent accident sequence characteristics, including ECCS and support system availabilities.	The type of initiating event and other accident sequence characteristics that affect the availability of the ECCS are queried by the PDS tree linkage rules for the 1-ECCSAV top event.
Containment isolation status	• System dependencies on supporting systems • System availability	The containment isolation system dependencies and status (i.e., system component failures) are modeled in the supporting fault tree logic for the 1-CISOL-H top event in the bridge tree.
Containment heat removal status	• System dependencies on supporting systems • System availability	The containment cooling unit system dependencies and status (i.e., system component failures) are modeled in the supporting fault tree logic for the 1-CONTCOOL-H top event in the bridge tree.
Containment integrity	Type of initiating event (i.e., whether the containment is bypassed). Open containment situations are not within scope of the at-power model. Induced containment failure is within the scope of the CET. Containment isolation is covered above.	The type of initiating event that affects containment integrity is queried in the PDS tree and CET linkage rules for various top events.
Steam generator pressure	Type of initiating event and subsequent accident sequence characteristics	The type of initiating event and other accident sequence characteristics that affect the steam generator pressure are queried by the PDS tree linkage rules for the 1-SGCOOL top event.
Steam generator water level	Type of initiating event and subsequent accident sequence characteristics	The type of initiating event and other accident sequence characteristics that affect steam generator water level are queried by the PDS tree linkage rules for the 1-SGCOOL top event.

Table 2-2: Cross-walk of Physical and Accident Sequence Characteristics from High-Level Requirement L1-A (ASME, 2014)

Physical Characteristics Identified at the Time of Core Damage (L1-A1)	Accident Sequence Characteristics That Determine the Physical Characteristics (L1-A2)	Where the Physical Characteristics are Treated in the Model (L1-A3a)
Steam generator tube integrity	Type of initiating event, as well as other sequence logic for pressure-induced SGTRs	The type of initiating event, and relevant sequence logic (e.g., 1-RPS, 1-MFW-ATWS and 1-PPR for pressure-induced steam generator tube rupture [PI-SGTR]) during anticipated transient without scram [ATWS]), are queried by the PDS tree linkage rules for the 1-ACCTYPE top event.
Containment pressure	Initiating event information that would affect containment pressure (namely bypass) is passed in the 1-ACCTYPE PDS top. Other aspects are determined within the bridge tree and CET.	Containment isolation status, availability of sprays, and availability of heat removal all affect containment pressure. These are all handled in the bridge tree. Other influencing characteristics, such as energetic events and core-concrete interaction are handled within the CET/DETs.
Availability/accessibility of mitigating equipment	Type of initiating event and subsequent accident sequence characteristics No information on extensive damage mitigation guideline (EDMG) equipment (except for success or failure in blind feeding steam generators [SGs] in relevant station blackout [SBO] situations) was passed from the Level 1 PRA since EDMGs were not generally addressed in the Level 1 PRA.	Information regarding the availability of mitigating equipment is queried by the PDS tree linkage rules. For example, the information passed through for steam generator pressure is also used to determine the availability of auxiliary feedwater (AFW) and the information passed through for ECCS status is used to determine the availability of ECCS. Accessibility is handled as part of the HRA, and indirectly uses Level 1 information. For example, certain actions in the auxiliary building might not be considered for ISLCOAs where the auxiliary building is likely to be flooded.

Table 2-2: Cross-walk of Physical and Accident Sequence Characteristics from High-Level Requirement L1-A (ASME, 2014)

Physical Characteristics Identified at the Time of Core Damage (L1-A1)	Accident Sequence Characteristics That Determine the Physical Characteristics (L1-A2)	Where the Physical Characteristics are Treated in the Model (L1-A3a)
Status of support systems	Type of initiating event and subsequent accident sequence characteristics	The PDS tree linkage rules query the Level 1 accident sequence logic to infer the status of support systems when possible (e.g., status of offsite power from the initiating event). The CET indirectly receives this information by virtue of the fact that the CET linkage rules query the PDS tree sequence logic. Detailed information about the support system availability is passed on to the Level 2 via the Level 1 cut sets; however, this information is not available for the Level 2 probabilistic sequence modeling. Meanwhile, support system information is manually addressed in the Level 2 deterministic analysis (e.g., assuming a support system success or failure for a representative sequence by scrutinizing the Level 1 PRA cut sets).
Time of core damage	No information is passed from the Level 1 PRA.	The role of core damage timing in the Level 2 model is handled via the CET's use of the Level 2-specific deterministic accident progression modeling that includes the pre-core damage phase of the accident.
Status of other non-safety systems	No information is passed from the Level 1 PRA.	Information about the status of non-safety systems is manually addressed in the Level 2 HRA.
Environmental or physical conditions caused by the hazards	Not applicable. This PRA does not address external hazards.	Not applicable. Note that environmental hazards caused by the severe accident itself are considered manually in the Level 2 HRA.
Design and configuration of surrounding structures	Dependencies	The only dependency modeled in this regard is the auxiliary building ventilation system's dependency on alternating current (AC) power (meant here to represent both the auxiliary building ventilation system and piping penetration area filtration and exhaust system).

Table 2-2: Cross-walk of Physical and Accident Sequence Characteristics from High-Level Requirement L1-A (ASME, 2014)

Physical Characteristics Identified at the Time of Core Damage (L1-A1)	Accident Sequence Characteristics That Determine the Physical Characteristics (L1-A2)	Where the Physical Characteristics are Treated in the Model (L1-A3a)
Physical effects of flooding	Type of initiating event and subsequent accident sequence characteristics No information is passed from the Level 1 PRA regarding flooding in the auxiliary building during an interfacing systems loss-of-coolant accident (ISLOCA), as this information is handled manually. Containment flooding and resultant instrument failure are not explicitly modeled.	Information about the physical effects of flooding is queried by the PDS tree linkage rules for the 1-ACCTYPE, 1-SGCOOL, 1-RWSTAV, and 1-ECCSAV top events, in that sequence logic (successes and failures) for flooding sequences are queried in the PDS tree in the same manner as is done for internal events. However, the failures cannot be definitively attributed to flooding damage (versus random failure). In other words, if a flooding initiator causes flooding-induced damage to equipment that would render auxiliary feedwater unavailable, then the Level 2 PDS logic would know that AFW failed (by virtue of the flooding initiator and the down-branch on AFW), but would not know whether the failure was caused by flooding versus other failures modes.

2.1.3. Step 3 – Review the resulting plant damage states

The PDS bins developed in the preceding step were reviewed to gain confidence that critical information was being appropriately considered in the binning process. In the logic model, 384 unique PDS bins are possible (6 bridge event tree paths x 64 PDS event tree paths). Quantification of the model yielded approximately 50,000 cutsets above a truncation of 1×10^{-11}/rcy in the model. Also, 18 PDS bins have a frequency of greater than 1.0×10^{-7}/rcy, and the top 7 PDS bins comprise 95.5 percent of the PDS frequency. This situation is represented in **Figure 2-4**.

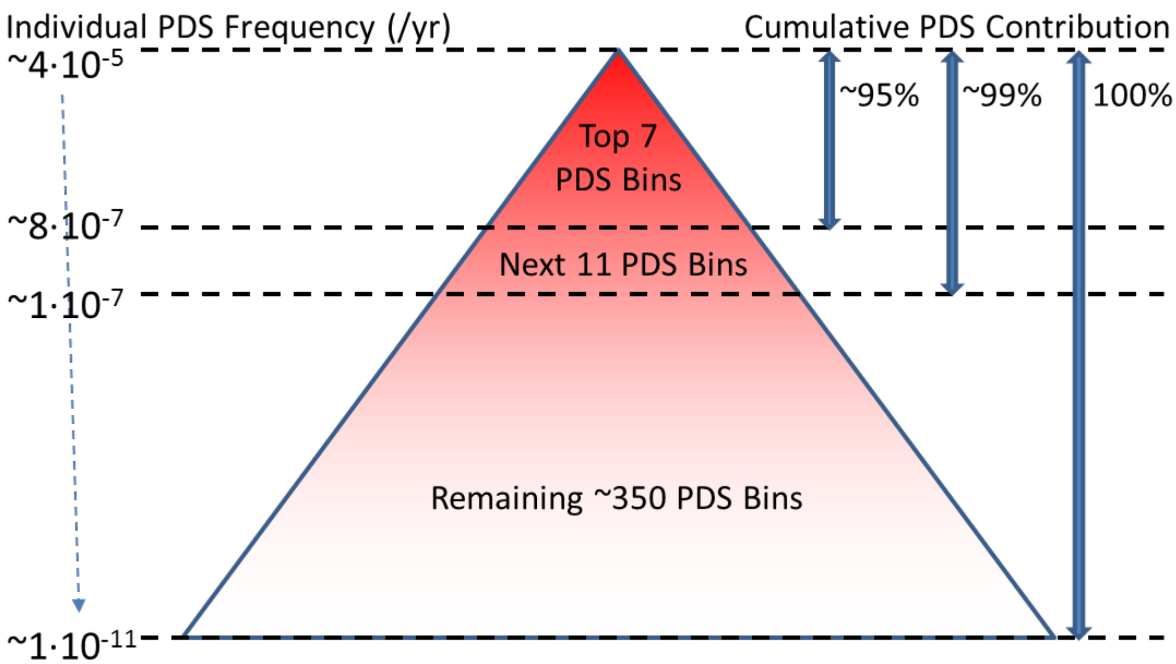

Figure 2-4: Schematic Showing Breakdown of PDS Frequency

The top 19 contributing PDS bins are catalogued in **Table 2-3**, along with some observations. Specifically, scrutiny of the PDS binning further highlights some limitations in the transfer of information across the Level 1/2 interface, as follows:

- Since the Level 1 PRA treats offsite power recovery at the event tree level, rather than the fault tree level, it is not captured by the containment systems evaluations in the bridge event tree. So, in some cases, containment systems are treated as unavailable, whereas they could be potentially available following timely restoration of offsite power, if they are not affected by the operator failure to restore systems following restoration of offsite power, which dominates the relevant cutsets (e.g., see PDS-29-4 and PDS-05-4).[7]

[7] Understand that in the PDS naming scheme, the set of numbers after the first hyphen is the PDS event tree path (1-64) and the number after the second hyphen is the bridge event tree path (1-6). In reality, PDS quantification was done using the 1-PDS-Q event tree, a merger of the PDS and bridge tree, but end-states were numbered/assigned to retain this naming scheme. For example, PDS-29-4 is path 29 through the PDS tree, and path 4 through the bridge tree.

- In some cases, there is insufficient information in the Level 1 PRA sequence logic to judge the availability of feedwater or ECCS, and numerous assumptions are encoded in the PDS binning logic. Exceptions to these assumptions manifest in the PDS binning of cutsets (e.g., see PDS-25-4).
- A limitation in the engineered safety features actuation system (ESFAS) dependency modeling in the containment spray system fault tree leads to situations where sprays are treated as available, but in actuality may not be (e.g., see PDS-12-2).
- Since the Level 1 PRA treats consequential loss-of-offsite power (CLOOP) at the fault tree level, rather than the event tree level, it is not captured by the PDS binning logic. Rather, assumptions are hard-wired into the model for sequences that are dominated by CLOOP-induced SBOs, and situations that violate these assumptions manifest in the PDS binning of cutsets (e.g., see PDS-35-1).

As will be seen later in the release category cutset review and the presentation of results: (i) some instances of these limitations did not manifest into significant contributors and; (ii) other limitations on the model prompted caveats on how precisely to interpret the results. As such, these limitations were determined to be acceptable.

Of the total PDS frequency, 1.2 percent comes from internal flooding initiators (the same ratio applied to the CDF results). The highest-contributing internal flooding cutset contributes less than 0.1 percent to the total PDS frequency. Based on this small contribution, effects unique to flooding events are not considered in the PDS binning (beyond the loss of equipment specifically captured in the Level 1 flooding modeling), and generally are not considered in the CET. Recall that PDS binning was performed as a convenient approach for making modeling assumptions and guiding deterministic analysis, but all Level 1 sequences and cutsets are carried forward to the CET.

Table 2-3: Listing of Top Internal Events and Floods PDS

	PDS #	PDS %	Evaluated using Level 1 sequence logic				Evaluated using fault trees			Notes
			Accident Type	SG Cooling	RWST	ECCS	Containment Isolation	Containment Sprays	Containment Coolers	
1	PDS-35-4	58.8%	SBO	Failed	Available	N/A	Success	Unavailable	Unavailable	SBO with immediate or delayed loss of turbine-driven auxiliary feedwater (TDAFW) and no power recovery
2	PDS-02-4	17.7%	Small LOCA (SLOCA)	Success	Available	Unavailable	Success	Unavailable	Unavailable	Transient (most notably loss of nuclear safety cooling water [NSCW]) leading to loss of reactor coolant pump (RCP) seal integrity with loss of containment heat removal
3	PDS-29-4	10.6%	Transient	Failed	Available	Available	Success	Unavailable	Unavailable	SBO with AC power recovery but operators fail to restore systems (the model does not handle resumed availability of containment sprays/coolers upon AC power recovery – though they may not be available given the operator failure)
4	PDS-29-1	3.2%	Transient	Failed	Available	Available	Success	Available	Available	Transient/loss-of-offsite power (LOOP) with failure of AFW and failure to establish feed-and-bleed cooling

Table 2-3: Listing of Top Internal Events and Floods PDS

	PDS #	PDS %	Evaluated using Level 1 sequence logic				Evaluated using fault trees			Notes
			Accident Type	**SG Cooling**	**RWST**	**ECCS**	**Containment Isolation**	**Containment Sprays**	**Containment Coolers**	
5	PDS-05-4	2.2%	SLOCA	Failed	Available	Available	Success	Unavailable	Unavailable	SBO with loss of RCP seal integrity, with AC power recovery but operators fail to restore systems (the model does not handle resumed availability of containment sprays/coolers upon AC power recovery – though they may not be available given the operator failure)
6	PDS-11-3	1.8%	MLOCA	Success	Unavailable	Available	Success	Unavailable	Available	Medium loss-of-coolant accident (MLOCA) with successful ECCS injection, but failure to establish recirculation (renders containment sprays unavailable due to operator action dependency modeling)
7	PDS-25-4	1.3%	Transient	Success	Available	Available	Success	Unavailable	Unavailable	Transient with dominant cut sets for this PDS involving failures of both trains of safety-related electrical power
8	PDS-12-1	0.7%	MLOCA	Success	Unavailable	Unavailable	Success	Available	Available	MLOCA with failure of ECCS dominated by sequence-related failures, but resulting in continued availability of one train of NSCW; unavailability of RWST is a simplifying and inclusive assumption for all MLOCA sequences

Table 2-3: Listing of Top Internal Events and Floods PDS

	PDS #	PDS %	Evaluated using Level 1 sequence logic				Evaluated using fault trees			Notes
			Accident Type	SG Cooling	RWST	ECCS	Containment Isolation	Containment Sprays	Containment Coolers	
9	PDS-63-1	0.5%	L-ISLOCA	n/a	Unavailable	Available	Success	Available	Available	Residual heat removal (RHR) interfacing system loss-of-coolant accident (ISLOCA), a.k.a. a large ISLOCA (L-ISLOCA), where ECCS injects, but recirculation is not possible
10	PDS-12-2	0.5%	MLOCA	Success	Unavailable	Unavailable	Success	Available	Unavailable	MLOCA with failure of ECCS and containment coolers dominated by ESFAS-related failures; containment sprays may also be unavailable, but are not captured due to simplifications in the ESFAS dependency modeling in the containment spray system model
11	PDS-35-1	0.3%	SBO	Failed	Available	N/A	Success	Available	Available	These are generally not SBO events; however, they are lower-order contributors from sequences that are dominated by CLOOP-induced SBOs and are attributed as being such.
12	PDS-07-1	0.3%	SLOCA	Failed	Unavailable	Available	Success	Available	Available	Transient-induced small loss-of-coolant accidents (SLOCAs) (e.g., successful feed and bleed) with electrical/operator failures leading to no feedwater (FW)

Table 2-3: Listing of Top Internal Events and Floods PDS

	PDS #	PDS %	Evaluated using Level 1 sequence logic				Evaluated using fault trees			Notes
			Accident Type	SG Cooling	RWST	ECCS	Containment Isolation	Containment Sprays	Containment Coolers	
13	PDS-25-1	0.3%	Transient	Success	Available	Available	Success	Available	Available	Myriad of comparably-contributing cutsets, including safe/stable failures and dual-train electrical issues; it shares some of the same weaknesses as PDS 25-4 (row 7 of this table)
14	PDS-03-1	0.3%	SLOCA	Success	Unavailable	Available	Success	Available	Available	Mix of transient-induced SLOCAs with operators failing to maintain ECCS injection and pipe ruptures with ECCS failures (may result from operator errors, rather than hardware failures)
15	PDS-12-4	0.2%	MLOCA	Success	Unavailable	Unavailable	Success	Unavailable	Unavailable	MLOCA with failure of ECCS dominated by sequencer-related failures resulting in NSCW failure (and containment spray/cooler failure)
16	PDS-43-1	0.2%	SGTR-UNIS	Failed	Unavailable	Available	Success	Available	Available	Secondary-side breaks or transients with coincident ATWS, leading to PI-SGTR (i.e., an unisolable SGTR [SGTR-UNIS])
17	PDS-11-1	0.2%	MLOCA	Success	Unavailable	Available	Success	Available	Available	MLOCA with successful ECCS injection but subsequent failures (e.g., ECCS recirculation failure)

Table 2-3: Listing of Top Internal Events and Floods PDS

	PDS #	PDS %	Evaluated using Level 1 sequence logic				Evaluated using fault trees			Notes
			Accident Type	SG Cooling	RWST	ECCS	Containment Isolation	Containment Sprays	Containment Coolers	
18	PDS-29-3	0.2%	Transient	Failed	Available	Available	Success	Unavailable	Available	Transients with failure of FW, failure to establish feed and bleed, and failure to align containment spray recirculation
19	PDS-23-1	0.2%	EXLOCA[1]	n/a	Unavailable	Available	Success	Available	Available	Excessive loss-of-coolant accident (XLOCA) (i.e., reactor pressure vessel [RPV] rupture) occurs and ECCS is available

[1] EXLOCA is used in the Level 2 PRA as a term generally interchangeable with the Level 1 PRA's use of XLOCA, with the only distinction being their place in the model: XLOCA is an initiator, EXLOCA is a PDS attribute.

2.1.4. Step 4 – Iteration on the Level 1 PRA modeling, as necessary

The development of the Level 2 PRA model required iteration between the Level 1 and Level 2 PRA models. This section discusses some of the changes made to the Level 1 PRA model specifically to support the Level 2 PRA model.

Several issues related to the interface between the Level 1 and Level 2 models were considered, investigated, and/or modified during the course of the model development. These include, but are not limited to:

- The modeling of LOCA-induced LOOP events
- Modeling of blind feeding SGs using turbine-driven auxiliary feedwater (TDAFW) during an SBO
- Plant response following degraded NSCW conditions
- The modeling (e.g., initiator frequency, systems considered, pre-core damage mitigation) of ISLOCAs, and the potential for severe accident-induced ISLOCAs
- The availability of RWST and ECCS in sequences where their status was not directly discernible in the Level 1 PRA sequence logic
- The modeling of PI-SGTR prior to core damage
- The isolation of faulted SGs prior to and following core damage
- Potential effects of steamline flooding during SGTR events
- The recovery of offsite power prior to, and after, battery depletion
- Level 1-to-Level 2 human error probability (HEP) dependency
- Instances where a cooldown and depressurization may be operationally pursued but was not modeled in the Level 1 PRA because it did not impact the core damage determination

Closely related to the final bullet is the issue of late depressurization for the many Level 1 sequences involving failure of either feed-and-bleed injection or recirculation. Feed and bleed is modeled in numerous places in the Level 1 PRA as a means of removing decay heat given failure of normal SG cooling using AFW/main feedwater (MFW). After establishing feed-and-bleed heat removal, plant procedures direct operators to make repeated attempts to restore sources of feedwater (AFW, MFW, condensate feed, or other low-pressure sources). (Successful restoration of these sources was also not modeled in the Level 1 PRA.) Once the RWST indicated level reaches 29 percent, and without having re-established feedwater, operators perform switchover to sump recirculation. If feed-and-bleed injection or recirculation is unsuccessful, the PRA model assumes core damage with no additional top events queried. In reality, low vessel water level and high core-exit thermocouple temperatures will trigger a higher-priority functional restoration procedure that addresses inadequate core cooling. The relevant actions from the higher level inadequate core cooling procedure applicable to this situation are simplified as:

- Fully opening all SG atmospheric relief valves (ARVs) and
- Opening all pressurizer power-operated relief valves (PORVs), and barring immediate temperature decrease, opening head vents and aligning letdown and excess letdown lines.

Whether these actions would apply to specific PRA cutsets is a human reliability analysis (HRA) issue. There is a general human reliability challenge associated with transitioning between different event response procedures. Also, if feed and bleed/feed-and-bleed recirculation failed due to a human failure event (HFE), then the evaluation of these actions should consider the dependency on the preceding feed-and-bleed HFE.

Another question exists as to whether the inadequate core cooling actions would actually prevent core damage for various PRA cutsets. The potential impact on core damage is unclear. Feed and bleed reliability is strongly influenced by the reliability of the associated operator action. Therefore, the level of dependency between the feed-and-bleed HFE and the inadequate core cooling procedure HFE will significantly impact the quantification. Given the above considerations, there is reason to conclude that the amount of credit for successful inadequate core cooling procedure actions in these cases may be small, and that the likelihood of averting core damage given these actions also may be small. If an HRA evaluation were to lead to significant credit for the inadequate core cooling procedure actions being taken, this may increase the likelihood of:

- depressurization of the steam generators leading to high primary-side pressure and low secondary side pressure – conditions associated with a higher likelihood of PI-SGTR
- depressurization of the primary-side that could cause these sequences to no longer be high-dry-low situations.

These two considerations have competing effects with respect to the likelihood of a consequential SGTR. Resolving all of the issues described herein would represent a significant undertaking with potentially little change to risk-significant sequences, and for this reason, the model was not altered. Rather, these issues are identified as a general model uncertainty for the Level 1 and Level 2 PRA.

2.1.5. Step 5 – Establishment of criteria for, and selection of, representative sequences

The previous steps take tens of thousands of cutsets and bin them into a smaller group (hundreds) of PDSs. From each of the significant PDSs (as defined below), one (or a few) "representative" sequences were selected for deterministic treatment in the Level 2 PRA. These sequences are surrogates for the numerous cutsets that, while carried forward in the integrated probabilistic model, were not explicitly modeled by deterministic accident progression analysis. Therefore, it is important that they reflect the general characteristics of the numerous cutsets. The role of the representative sequences was previously discussed in **Section 2.1.2** (i.e., they provide a narrative of post-core damage behavior, support human reliability analysis, etc.).

While the representative sequences are intended to encompass the individual traits associated with the Level 1 PRA cutsets, across the full scope of a Level 2 PRA it is often difficult to anticipate the effects of a particular assumption. Therefore, judgment is necessarily a part of representative sequence selection. In some cases, variability was addressed through the identification of new sources of model uncertainty (see **Section 2.4.7**) or identification of sensitivities. PDSs were selected for deterministic evaluation (i.e., were translated to representative sequences or sensitivities) if they:

- Comprised an important portion of PDS frequency, or

- Were of potentially high conditional consequence based on projection of release magnitude or timing, or
- Illustrated or yielded data or phenomenological insights (e.g., combustible gas accumulation, RCS piping creep rupture behavior) into items of interest to the CET modeling.

The representative sequences for the current model were chosen based on the results of a preliminary (2014) version of the model (simply due to project timing/progress considerations). The previous PDS results that led to these representative sequences were comprised of various station blackout sequences, dual-train electrical transients, loss of nuclear service cooling water sequences, and ISLOCAs. With the exception of ISLOCAs (which remain of interest due to being bypass events), these contributors remain important to the current PDS results. Meanwhile, medium LOCA has risen in importance (to ~3.5 percent of the current PDS frequency), but wasn't addressed by the prior representative sequences. As demonstrated below, re-development of representative sequences based on the newer PDS results would be expected to result in a generally similar set of selections.

The detailed deterministic specification (covered later in **Section 2.3.3**) is based on scrutiny of the cutset contributions of the associated PDS. Where a range of representative sequence boundary conditions could apply to a PDS, conditions that would generally be expected to pessimistically affect the radiological source term were chosen, or else the modeling selection was specifically identified as a model uncertainty; however, there was no systematic attempt to bias boundary conditions in a conservative direction. Section 4 of Appendix C provides additional perspective on the extent of conservatism (or lack thereof) in the deterministic modeling, by comparing a large number of different accident simulations for various release categories.

In addition to the 8 base representative sequences, a total of 22 additional permutations and 6 recovery cases were ultimately simulated (not to mention the additional sensitivities covered in Appendix C), to fill all information needs related to development of the probabilistic model and population of release category representative source terms. The full list of cases associated with the representative sequence selection is provided in **Table 2-4** below. Appendix B provides a brief overview of each of the cases outlined in **Table 2-4**.

Table 2-5 maps the previously tabulated top PDSs (i.e., the top nine contributors in **Table 2-3**) to these MELCOR cases in order to show whether the representative sequences and sensitivities adequately cover the dominant PDSs. **Table 2-5** also illustrates the mapping of the containment bypass and isolation failure-oriented representative sequences to relevant PDS. As can be seen, the highest-contributing PDS bins are well-represented, if one credits the 6-series cases as providing relevant information for both PDS-29-1 and PDS-29-4. As intended, some low-contributing PDS bins are also well-represented because they involve bypass events (ISLOCA, SGTR, containment isolation failure). Several PDS bins contributing on the order of 1 percent should be better represented, but are not due to the timing of the representative sequence selection relative to the availability of the final PDS binning results. Nevertheless, it is felt that the existing MELCOR calculations provide more-than-sufficient coverage to support development of the Level 2 PRA.

A key decision in this process was to not simulate ATWS accidents. This decision was made based on a combination of the very low contribution of ATWS sequences and the additional complexities and limitations associated with modeling ATWS events in MELCOR. While this decision is believed to be justified, it does represent a limitation on the scope of the deterministic portion of the analysis. ATWS is modeled probabilistically, but with the assumption that the ultimate post-core damage response is generally similar to non-ATWS accidents (e.g., ATWS-related fuel damage is ultimately over-taken by overheating fuel damage). It should be noted that the ATWS-induced PI-SGTR was specifically considered in the Level 1 PRA and carried forward in the Level 2 PRA.

For each simulation, the associated accident sequence analysis is covered in **Section 2.3.3**, and recovery calculations related to the effectiveness of accident management actions are the subject of **Section 2.3.5**.

Table 2-4: Representative Sequence and Sensitivity/Recovery Case Descriptions

Case	Description
1	SBO with 21 gpm per RCP Seal LOCA, Indefinite AFW, and Rapid Depressurization
1A	Base-case SBO with Eventual Loss of AFW
1A1	Base-case SBO with Eventual Loss of AFW, and Suppressed Deflagrations
1A2	Base-case SBO with Eventual Loss of AFW and Late Combustion-Induced Containment Failure
1B	Base-case SBO with Initial Loss of AFW, and No Depressurization
1B1	Base-case SBO with Initial Loss of AFW, No Depressurization, and 182 gpm per RCP seal LOCA
1B2	Base-case SBO with Initial Loss of AFW, No Depressurization, and Stuck-Open power-operated relief valve (PORV)
2	Transient Induced by Total Loss of NSCW, 182 gpm per RCP Seal LOCA, AFW, and Controlled Depressurization
2R1	Base-case with severe accident mitigation guideline (SAMG)-prompted Additional Secondary-Side Cooldown During Core Damage
2R2	Base-case with SAMG-prompted Firewater-based Containment Spray Following Vessel Breach
2A	Base-case with Containment Failure Forced at the Time of Vessel Breach
3	Transient Initiated by Loss of Main Feedwater, AFW Lost at 3 Hours, and ECCS Unavailable
3A1	High-Pressure Transient, with Instrument Tube Failure
3A2	High-Pressure Transient, with Induced Rupture of Steam Generator Tubes (SGTs)
3A3	High-Pressure Transient, with Induced Ruptures of SGTs and Hot Leg Nozzle
3A4	High-Pressure Transient, with All Induced RCS Failure Paths Disabled
4	Transient Induced by Electrical Distribution and NSCW Failures, 182 gpm per RCP Seal LOCA, AFW, and Controlled Depressurization
5	Interfacing System LOCA (ISLOCA) with Submerged Break
5A	Base-case ISLOCA but with Uncovered Break
5B	Base-case ISLOCA but with Double-Ended Eight-Inch Break
5C	Base-case ISLOCA with Plugging of Piping Penetration Area Filtration and Exhaust System (PPAFES) Filters
5D	Base-case ISLOCA but with Double-Ended Eight-Inch Uncovered Break

Table 2-4: Representative Sequence and Sensitivity/Recovery Case Descriptions

Case	Description
6	Transient Initiated by Loss of Offsite Power, AFW Lost at 6 Hours, ECCS Available, and Containment Cooling Available
6R1	Base-case with SAMG-prompted Low Pressure Injection Initiated During Core Damage
6A	Base-case with Containment Sprays Actuating After Core Damage
6B	Base-case with Deflagrations Suppressed
6C	Base case with Containment Failure Forced at the Time of Vessel Breach
6D	Case 6A with Containment Failure Forced at the Time of Vessel Breach
7	SBO with 21 gpm per RCP Seal LOCA, AFW Lost at 4 hours, and Containment Isolation Failure
7A	Base case with Portable Pump Injection through Containment Spray Lines
8	Un-isolated Steam Generator Tube Rupture with AFW
8R1	Base-case with SAMG-prompted Flooding of Ruptured SG During Core Damage
8R2	Base-case with SAMG-prompted Flooding of Ruptured SG Following Vessel Breach
8A	Base-case SGTR with AFW Supplied to Affected Steam Generator
8B	Base-case SGTR with Stuck-Open Relief Valve in Affected Steam Generator
8BR1	Case 8B with SAMG-prompted Flooding of Ruptured SG During Core Damage

Table 2-5: Mapping of Representative Sequences to PDS Bins

PDS #	PDS %	Accident Type	SG Cooling[1]	RWST	ECCS	Containment Isolation	Containment Coolers	Containment Sprays	MELCOR Cases[2]
PDS-35-4	58.8%	SBO	Failed	Available	N/A	Success	Unavailable	Unavailable	1[3], 1A, 1A1, 1A2, 1B, 1B1, 1B2
PDS-02-4	17.7%	SLOCA	Success	Available	Unavailable	Success	Unavailable	Unavailable	2, 2A, 4
PDS-29-4	10.6%	Transient	Failed	Available	Available	Success	Unavailable	Unavailable	Similar to PDS-29-1 in terms of RCS response
PDS-29-1	3.2%	Transient	Failed	Available	Available	Success	Available	Available	6, 6A, 6B, 6C, 6D
PDS-05-4	2.2%	SLOCA	Failed	Available	Available	Success	Unavailable	Unavailable	Not explicitly covered by the existing representative sequences; rather, modeling was based on the relevant aspects of other simulations
PDS-11-3	1.8%	MLOCA	Success	Unavailable	Available	Success	Unavailable	Available	
PDS-25-4	1.3%	Transient	Success	Available	Available	Success	Unavailable	Unavailable	
PDS-12-1	0.7%	MLOCA	Success	Unavailable	Unavailable	Success	Available	Available	
PDS-63-1	0.5%	L-ISLOCA	n/a	Unavailable	Available	Success	Available	Available	5, 5A, 5B[4], 5C, 5D[4]
…									
PDS-35-6	0.1%	SBO	Failed	Available	Unavailable	Failure	Unavailable	Unavailable	7, 7A
PDS-38-4	<0.1%	SGTR-UNIS	Success	Available	Unavailable	Success	Unavailable	Unavailable	8, 8A. 8B
PDS-30-4	<0.1%	Transient	Failed	Available	Unavailable	Success	Unavailable	Unavailable	3, 3A1, 3A2, 3A3, 3A4

1 Recall that failure includes cases where feedwater was successful initially but lost at the time of battery depletion.

2 Recovery cases are not included here, because they are identical to their parent cases through the onset of core damage.

3 The PDS binning is unaware of this situation (SG blind feeding), as it is only addressed in 1-L2-REC.

4 In this case, both RHR pumps are unavailable due to the location of the break, but this is a Level 2 modeling decision, not something that is discernible from the Level 1 PRA sequence.

2.2. Containment Capacity Analysis

This subtask consists of four interrelated steps:

1. Assessment of preliminary failure modes and failure locations of interest
2. Development of a finite element model of the containment
3. Development of containment fragilities associated with severe accident conditions
4. Assessment of structural responses to severe accident conditions in adjoining buildings

The objective of the first step is to develop an initial state-of-knowledge about the containment's more likely failure modes and locations, to guide development of a finite element model. The objective of the second step is to develop a finite element model of the containment structure that is adequate for assessing the containment's capacity relative to over-pressure (long time-scales and dynamic) and over-temperature conditions. The objective of the third step is to apply the finite element model to arrive at cumulative distribution functions that can be used in specifying failure likelihoods/characteristics for use in the containment event tree modeling and to be used in establishing the containment failure response within the MELCOR model. The objective of the fourth step is to use available information to establish failure characteristics of the auxiliary building, again for use in the MELCOR model and containment event tree. Each of these steps is discussed in further detail in this section.

Note that there was a built-in presumption that new finite element analysis was needed. The reason for this presumption is the significant additional containment testing and containment integrity analysis performed since the original analysis, in conjunction with maintaining and extending staff capabilities in this area. The presumption was not based on any a priori knowledge about how significant this new information would be to the results.

2.2.1. Step 1 – Assess preliminary failure modes and locations of interest

Table 2-6 provides a brief cross-walk of the containment failure mechanisms that were (or were not) considered in this study. These failure mechanisms were identified as having the potential to result in containment failure, which is defined as an opening or breach in the containment boundary resulting in a containment leakage area greater than the equivalent area for the containment design-basis maximum allowable leakage rate. The reference plant's maximum allowable leakage rate corresponds to an opening area of approximately 0.2-inch diameter.

A containment over-pressure characterization performed for the reference plant yielded the fragility curve shown in **Figure 2-5**, which was used for its Individual Plant Examination (IPE). This information, as well as the containment fragility curves used for the IPEs of plants with similar containment designs, and experimental results obtained from NRC-sponsored containment testing at Sandia National Laboratories, were reviewed to develop the overall containment characterization used for this study.

Table 2-6: Brief Catalogue of Containment Failure Mechanisms

Mechanism	Consideration in this study
Direct mechanisms	

Table 2-6: Brief Catalogue of Containment Failure Mechanisms

Mechanism	Consideration in this study
Hydrogen combustion	The quasi-static over-pressure fragility curve was used for situations involving deflagrations, while situations involving detonations were assumed to always fail containment. A new finite element analysis specifically for combustion loads was not performed, and this limitation is identified as a source of model uncertainty in Sections 4.2, 4.4, and 4.7 of Appendix C.
High temperatures	Discussed briefly in this sub-section (not modeled), except for direct containment heating effects.
Hydrodynamic loads	Not considered; these are expected to be small in magnitude or consequence due to the size and relative strength of the containment design and the lack of large submerged discharge points associated with some pressure suppression containment designs.
Dynamic interactions between core debris and water	Discussed in Appendix E and later sections of this report – found to have very low likelihood of generating forces sufficient to fail containment.
Direct core debris impingement	Discussed in Appendix E and has a very low likelihood of challenging containment integrity, primarily due to the geometry of the reactor cavity and lower containment.
Concrete cracking and liner tearing	Considered in the quasi-static over-pressure analysis described in this sub-section. **This was the only mechanism for which detailed, quantitative structural modeling was performed.** This was not evaluated separately for combustion loads.
Radiation damage to containment sealant material	Not considered, for the same geometry arguments made for core debris impingement (i.e., no line-of-sight between where debris is likely to deposit and the containment boundary).
Indirect mechanisms	
Vessel displacement due to molten core concrete interaction (MCCI) attack of the cavity	Discussed in Section 11 of Appendix D .
Displacement caused by vessel failure at high-pressure	Discussed in Section 9 of Appendix E.
Other	Tearing of the steamline penetration due to steamline flooding is considered in Section 19 of Appendix D. Note, the reference plant does not have a concrete pedestal, and seismic hazards are outside the scope of the internal events/floods model.

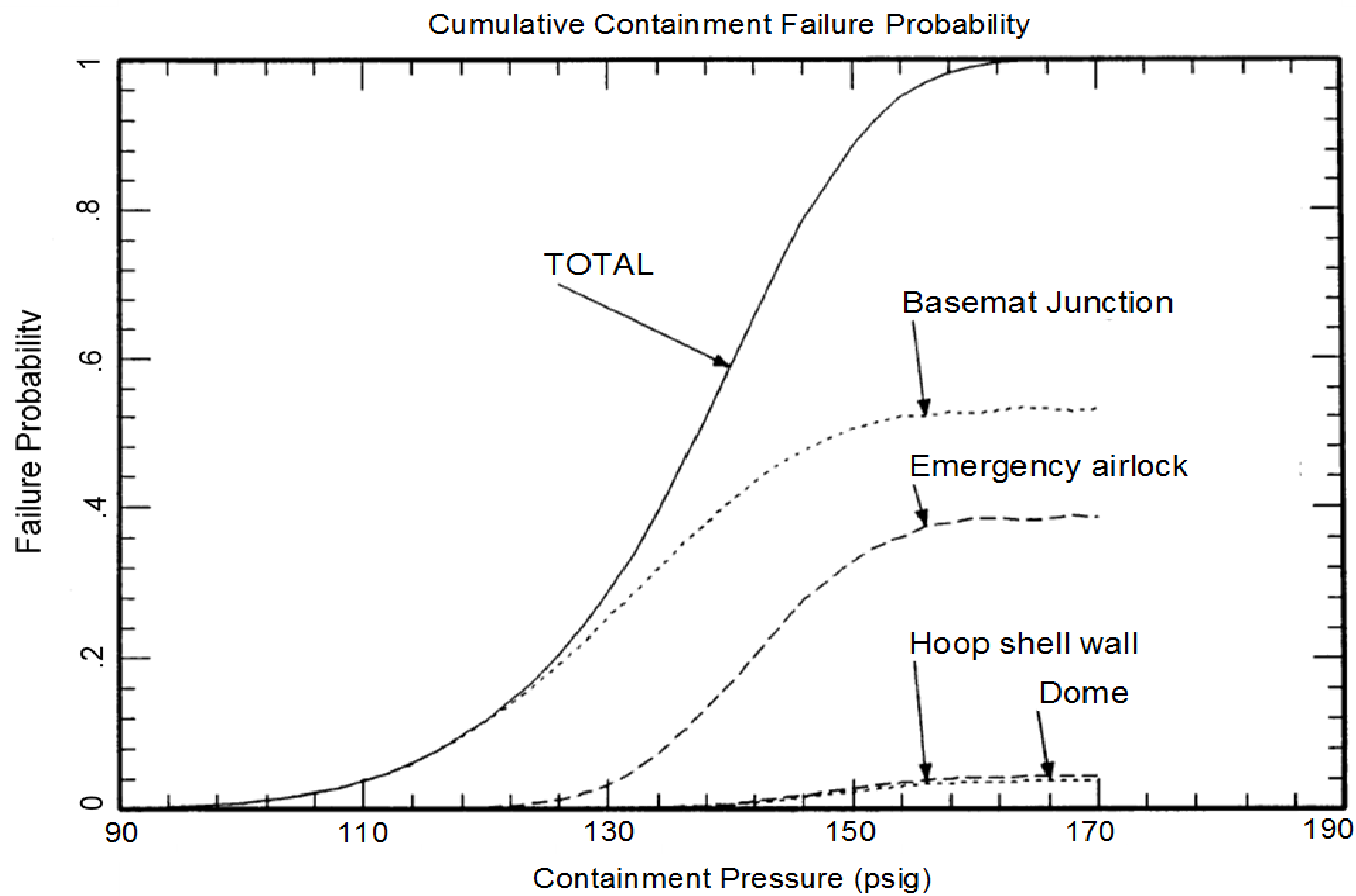

Figure 2-5: Containment Fragility Curves

Regarding integrated leakage rate tests (ILRTs) as a potential source of information, it was concluded that test results showing the actual leak rates below the $1L_a$ allowable value would not significantly alter the project analysis, because the current results demonstrate that the higher design-basis assumptions are still effective at retaining radiological material (i.e., results are ultimately expected to be driven by containment failure and bypass situations). A sensitivity on the design leakage rate was performed, as briefly discussed in Section 7 of Appendix C.

Key insights from the structural work performed to confirm and refine the IPE-vintage information are provided below:

- The IPE assessment focused on identifying the most probable failure location; the present analysis goes beyond that to provide failure characterization.
- The present analysis concludes that the key failure locations from the IPE are valid, but the relationship between those results and the present results is complex.
- The IPE captured more catastrophic failure modes, but then used a 5th-percentile failure pressure for evaluating containment failure.
- The present analysis captures liner tearing failure modes that occur at lower internal pressures relative to the more catastrophic modes; however, the median internal pressure estimates from these modes turn out to be reasonably similar to the 5th percentile values from the IPE (e.g., 112 psig versus 128 psig).
- The present analysis supports treatment of liner tearing initiation as being equally likely between the area near the equipment hatch, the area near the personnel airlock, the area near the emergency airlock, and the wall-basemat junction, due to relatively small differences in the median pressures calculated for these different failure pressure/region combinations.
- Liner tearing initiated at the wall-basemat junction would lead to concrete cracking that would create a leakage path into the tendon gallery.
- For the pressurization rates expected during sustained MCCI, liner tears would quickly grow to a point where further pressurization would not be expected, with containment breach sizes in the range of 0.3 to 1.0 ft^2 estimated.
- If pressurization did occur at a rate where enhanced liner tear leakage did not prevent continued pressurization, more catastrophic failure modes would be reached at roughly 160 psig.

Note that the above work used a constant temperature in assessing the containment pressure response. As a simplification, ambient temperature was used; however, NUREG/CR-6906 (Hessheimer, 2006) found that temperatures in the ranges predicted to occur have only a small effect on the ultimate pressure capacity of both reinforced and pre-stressed containment structures and only a minor effect on typical rebar (or steel containment shell) properties. The design-basis temperature for the reference plant is within this temperature range. Typical containment peak temperatures calculated in this study are also in this range.

The spectrum of containment leakage sizes that arise for various aspects of the Level 2 PRA model are synopsized in Section 6 of Appendix D to this report.

Regarding temperature-induced failure of penetrations, this was analyzed as part of the reference plant IPE and dismissed (see the synopsis in Section 15 of Appendix E).

2.2.2. Step 2 – Development of a finite element model of the containment

A finite element model was developed for analysis using the LS-DYNA software. The model was used to perform the analysis described in [Section 2.2.1](#). The model, shown in **Figure 2-7**, contains the containment structure itself, but not the containment internals. The model includes the cylindrical portion of the containment, the dome, the equipment and personnel hatches, the anchor gallery, the tendon gallery, the reactor foundation, the pre-stressed tendons, and the reinforcement. A total of roughly 750,000 elements were used. This degree of modeling is sufficient for developing the over-pressure/over-temperature failure criteria. The analysis provides the basis for modeling the containment leakage pathway in the MELCOR model for scenarios with containment overpressure failure.

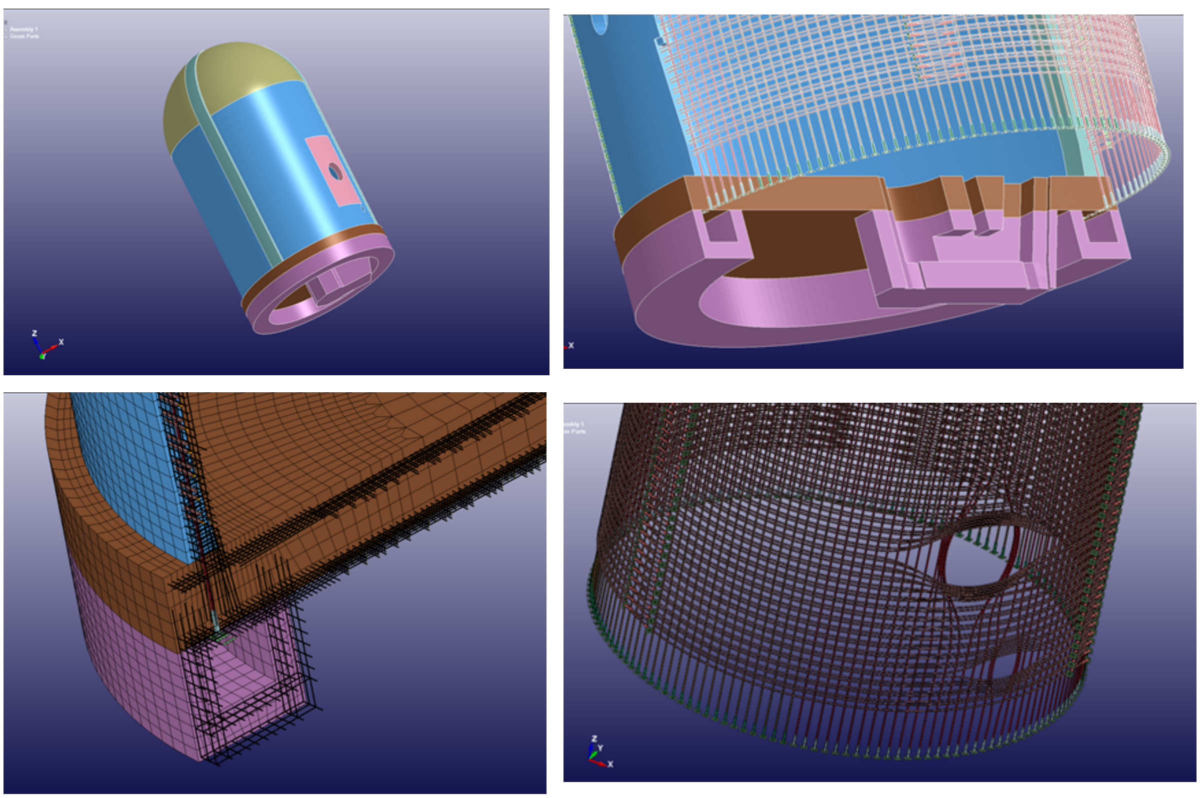

Figure 2-6: Containment Finite Element Model

2.2.3. Step 3 – Development of containment fragilities for severe accident conditions

The development of the containment fragility for severe accident conditions was discussed in **Section 2.2.1**. Additional containment fragility analysis was contemplated, which could have used the finite element model to further investigate high-temperature effects on penetrations to confirm the existing understanding, as well as the effects of dynamic forces in order to use a less constraining response assumption (containment failure is assumed for all detonation events). Ultimately, it was determined that such work is beyond the current state-of-practice, and forgoing the additional containment fragility analysis is not expected to have a fundamental effect on the results of the project.

2.2.4. Step 4 – Structural responses to severe accident conditions in adjoining buildings

The only work performed in this regard was the investigation of steamline flooding effects (within containment and up to the main steam isolation valves (MSIVs)) that led to the conclusion that torqueing of the steamline affecting steamline integrity or containment penetration damage was not likely. This is explained further in Section 19 of Appendix D.

The deterministic (MELCOR) analysis uses a subjective rather than mechanistic value for the over-pressure failure of surrounding buildings, namely that the building will fail locally for pressures greater than 1.1 bar-abs (a very low pressure). Notionally, this failure pressure is assumed to represent pressure-induced damage to the filtration system filter banks and trip of the associated fans, resulting in a large, unfiltered pathway to the environment through the heating, ventilation and air conditioning (HVAC) intake and exhaust. This failure pressure is on the same order of magnitude as has been estimated for other plants (e.g., 1.02 bar-abs in both (NRC, 2013b) and (Barto, 2014)). However, those plants' designs are different than the L3PRA project reference plant design, so there is a high degree of uncertainty in this subjective assignment, discussed further in the Appendix C sections on ISLOCA modeling (Section 4.10) and release pathway modeling (Section 4.13). There are no other internal or external sub-compartment or building failure analyses from different parts of the L3PRA project (namely, internal fires and internal flooding) that would have been useful in the Level 2 PRA model development.

For containment over-pressure failures into the tendon gallery, no credit was taken for leak-tightness of the associated access shaft doors, since they are not designed to withstand pressurization. If a particular accident should result in pressurization of the auxiliary building outside the compartments serviced by the PPAFES, credit was also not taken for equipment hatch covers on preventing pressurization of adjoining floors. Other aspects of structural performance of SSCs under severe accident conditions are discussed in **Section 2.3.6**.

2.3. Severe Accident Progression Analysis

The severe accident progression analysis consists of six interrelated steps:

1. SCALE analysis for decay heat and radionuclide inventory parameters
2. Development of a plant-specific MELCOR model
3. Accident progression modeling for the representative Level 2 sequences
4. Phenomenological evaluations for logic model construction
5. Assessment of post-core damage recovery actions
6. Evaluation of equipment survivability

The objective of the first step is to develop the necessary information regarding fuel decay heat and radionuclide inventories (masses and activities of each major radioisotope) for use in the MELCOR model's COR and RN packages. The objective of the second step is to develop a plant-specific MELCOR model of the RCS, steam generators and steam lines, ECCS, containment and containment systems, reactor protection system (RPS) and ESFAS logic, and auxiliary building. The objective of the third step is to exercise this MELCOR model in performing accident progression analysis for the representative sequences selected in **Section 2.1.5**, to inform the development of the containment event tree, provide timelines for use by the HRA, provide source terms, etc. The objective of the fourth step is to exercise this same MELCOR model, and other specialized separate effects tools, for analyzing specific phenomena, to guide development of the Level 2 logic model and split fractions. The objective of the fifth step is to model the effect of actions identified by the HRA on accident progression and source terms. The objective of the sixth step is to evaluate what equipment would be adversely affected by the conditions (e.g., temperature and humidity) associated with the accident. Each of these steps is discussed in further detail in this section.

2.3.1. Step 1 – SCALE analysis for decay heat and radionuclide inventory parameters

The purpose of this task is to take detailed fuel design information and utilize the ORIGEN-ARP routine in SCALE (ORNL, 2011), with supporting routines, to calculate the quantity and composition of various radionuclides in each assembly. Results are then grouped to provide the relevant inputs to the MELCOR model for each of the five radial assembly groups specified by the MELCOR core nodalization.

A plant-specific SCALE analysis was performed by Oak Ridge National Laboratory (ORNL) using detailed fuel information provided by the reference plant for decay heat (used by MELCOR), radionuclide inventories (used by MELCOR), and radionuclide activities (used by MELCOR Accident Consequence Code System [MACCS]). Data sets were developed for the reactor core (both from an at-power trip condition and a planned shutdown leading to a refueling outage), as well as all spent fuel in the Unit 1 and Unit 2 spent fuel pools. The coast-down to refueling conditions were chosen to mimic the actual 2012 outage information. Analysis was performed on an assembly-by-assembly basis for the more than 2,500 assemblies processed (from the Unit 1 core, the Unit 1 spent fuel pool [SFP], and the Unit 2 SFP). However, the results were grouped in a manner convenient for their use by MELCOR and MACCS.

ORNL also performed several sensitivity analyses to examine key uncertainties. The results of the sensitivity analyses are provided in **Table 2-7**, and show the estimated change in the decay heat that would be expected from making various alternative modeling assumptions.

Table 2-7: Results of SCALE Sensitivity Analyses

Alternative Assumption Relative to Base Case Analysis	Increased Pool Decay Heat	Increased Core Decay Heat
Using cycle-average power for all but one cycle	< 0.2%	N/A
Using VANTAGE+ assembly design for all pool assemblies	< 1.7%	N/A
Not including burnable absorbers in analysis (WABA+IFBA)[a]	< 5.9%[b]	< 0.8%
Not including axial power and moderator distributions	< 0.2%	< 0.1%
Including approximate hardware activation model	< 0.1%	< 0.8%

[a] WABA = wet annular burnable absorber; IFBA = integral fuel burnable absorber

[b] The bounding case assumes burnable absorbers inserted at all times, which is unrealistic. If more realistic reactivity increases are assumed (1 percent Δk from single-cycle and 3 percent Δk from all-cycle cases), then the increase in pool heat load is < 2 percent. This is a more realistic result.

2.3.2. Step 2 – Development of a plant-specific MELCOR model

2.3.2.1. *Use of MELCOR*

Severe accident progression calculations have been performed with the MELCOR computer code (Humphries, 2015a-c). MELCOR is a fully integrated, engineering-level computer code that models the progression of accidents in light water reactors. Phenomena modeled by MELCOR include:

- the thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and surrounding buildings
- core uncovery, fuel heatup, cladding oxidation, fuel degradation, and core melting and relocation
- thermal and mechanical loading and failure of the lower head
- core-concrete attack
- hydrogen production, transport, and combustion
- fission product release, transport, and deposition
- the impact of engineered safety features (e.g., containment sprays, containment fan coolers, and filters) on thermal-hydraulic and radionuclide behavior

Not all phenomena considered in the Level 2 PRA are modeled within MELCOR (e.g., vessel rocketing or probability of hydrogen detonation). These other phenomena were addressed using stand-alone analysis, often informed by MELCOR estimates of the relevant accident conditions.

MELCOR has undergone extensive validation against severe accident experiments, as described in (Humphries, 2015c). MELCOR has been used in numerous severe accident analyses, including the State-of-the-Art Reactor Consequence Analyses study (Chang, 2012) and studies to support new reactor licensing (e.g., the AP1000 design certification review (Esmaili, 2004)). In addition, MELCOR has been used extensively as part of efforts to understand the accidents at the Fukushima Daiichi site in Japan (e.g. (Gauntt, 2012)).

One such effort involved a systematic comparison of in-vessel accident progression results from MELCOR and MAAP for the Fukushima Daiichi Unit 1 accident (EPRI, 2014), and some observations are repeated here to illustrate the types of differences that may be observed between the code calculations done here, versus those done by licensees. This report identified several areas in which MELCOR 2.1 and MAAP5 differ. Most notable are differences in flow areas through a severely degraded core and heat transfer surface areas for particulate debris (i.e., debris from collapsed fuel rods). MAAP5 assumes that blockages form at higher elevations in the core that significantly reduce steam flow through the core and heat transfer from the core debris. In contrast, MELCOR 2.1 assumes that there will always be some flow through the core, so it sets a lower limit to the porosity (0.05 by default) used in the Ergun equation to calculate flow through a debris bed. Further, MELCOR 2.1 assumes a larger particulate debris surface area than MAAP. As a result, there is more steam flow through the core and more heat removal from the debris in MELCOR simulations than in MAAP simulations. The greater steam flow through the core results in much greater (2-3 times that predicted by MAAP) zircaloy oxidation and hydrogen production. The greater heat removal by the flowing gas leads to higher gas temperatures in the upper plenum and in the hot leg, which increases the calculated likelihood of hot leg creep rupture. These differences also have downstream effects at the time of vessel breach on RCS pressure, the temperature of the relocating melt, and the amount of material that re-locates.

It is important to note that both MELCOR and MAAP were benchmarked against small-scale experiments. Differences in in-vessel accident progression models in MELCOR and MAAP are the result of attempts to extrapolate these small-scale results to the reactor scale. In other words, these differences reflect uncertainty in our current understanding of in-vessel melt progression. Parameters expected to be significantly impacted by uncertainties in in-vessel accident progression are identified as part of the overall treatment of uncertainty (see **Section 2.4.7** and Appendix C, Section 4.16).

2.3.2.2. *MELCOR Model*

An existing MELCOR model for a similarly designed plant was taken as the starting point, and then heavily modified, to represent the reference plant for the L3PRA project and to extend the model's capabilities to severe accident analysis. This included modification of system geometries, system flow rates, component masses, flow areas, set-points, etc., using reference-plant-specific information. In some cases, very detailed plant-specific information (e.g., vessel internal geometries) was not available, and the following information from other plants or models was used:

- Details of much of the RPV internals geometry
- Details of some steam generator internals geometry
- Details of the secondary-side piping and steam dump geometry
- Some details of the vessel head vent system
- The full-open pressure for the pressurizer PORVs

A review of the modeling was performed against the documented MELCOR best practices as of (Ross, 2014). All these best practices are matched in the L3PRA project model, except for (i) the eutectic temperatures for ZrO_2 and UO_2 and (ii) the secondary-side decontamination factors. These are two areas where the state-of-practice has changed, and continues to change, since the publication of (Ross, 2014). As previously mentioned, the SCALE results were used to compare various modeling selections in MELCOR for decay heat. Some more specific aspects related to the model, such as its treatment of consequential steam generator tube rupture and the related assumptions about instrument tube failure, are documented in Section 5 of Appendix D.

2.3.2.3. *Mapping of SAMG Parameters*

There was a need to process the MELCOR output in a manner that facilitates its use in the Level 2 HRA for navigating through the SAMGs. The detailed mapping of MELCOR output streams to the plant conditions later used in the HRA is provided in Section 14 of Appendix D. Unless otherwise stated in the notes section in the associated table in Section 14 of Appendix D, all instrument readouts are available to the main control room (MCR), technical support center (TSC), and emergency operations facility (EOF), and the instruments are powered by Class 1E instrument power. Also note that during a loss of normal AC power event, the TSC receives its backup power from the security diesel generator system, which is also the system that provides the same function for the security system.

2.3.3. Step 3 – Accident progression modeling for the representative Level 2 sequences

Eight representative sequences, and numerous additional sensitivity cases, were previously defined, in Section 2.1.5 (**Table 2-4**). The MELCOR model described in the previous section was used to analyze these representative sequences and sensitivity cases. A summary of boundary conditions and accident progression results are presented in **Table 2-8** and **Table 2-9**.

Some of the more noteworthy initial and boundary conditions are captured below, to provide a distilled version of some of the modeling aspects that are complex:

- Battery depletion time for TDAFW is assumed to be 4 hours, to be consistent with the Level 1 PRA.
- When RCP leaks greater than 21 gpm/RCP were identified in the PRA sequence, a leakage size of 182 gpm/RCP was used. The 182 gpm/RCP leakage size is the most probable enhanced leakage rate (by a factor of 20) from the WOG 2000 model, of the failure sizes germane to situations with no O-ring failure (for the L3PRA project model, the RCPs do not have O-rings).[8]
- For equipment that is out of service for maintenance (specifically the 1AA02 4.16KV bus in the 3-series cases), the time it is out of service prior to the occurrence of the initiating event was arbitrarily chosen to be the mid-point of the allowed outage time.
- For ISLOCA cases, turbulent deposition of radionuclides in the connected piping is not modeled. Other retention uncertainties (auxiliary building failure, break coverage, and filter plugging) were investigated via sensitivity studies (see Section 5 of Appendix B for more detail).
- Two ISLOCA break sizes are considered. The first size is a 2-inch equivalent break, which is on the upper end of the break range for breaks not expected to over-pressurize the piping penetration area during initial blowdown.[9] The second size is an 8-inch equivalent break, which is the cross-sectional area of the RHR piping in question outside of containment. No flow restrictions are known to exist upstream of the pipe's exit from containment.
- In all cases except the 3-series cases, a 0.1 in^2 leakage area is assumed when a steam generator is isolated (MSIVs and relief valves closed). In the 3-series cases, a 0.5 in^2 leakage area is assumed to evaluate the effects of enhanced steam generator leakage on these high RCS pressure accident sequences. In all cases, this leakage is assumed to be directly to the environment.

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[8] For reference, the draft Surry UA (SNL, 2016) included four RCP leak rates, probabilistically-weighted the same as is done here, and found this to be one of the less influential sampled uncertainties in that study's scenario (a station blackout with early loss of AFW and failure to mitigate).

[9] The currently estimated threshold is 2.5-inch-equivalent, and subject to significant uncertainty in the thermal-hydraulic and structural modeling.

- Containment isolation failure, when modeled (in the 7-series cases), is assumed to be directly to the environment with a leakage area equivalent to a 2-inch diameter circle.
- For the 6-series cases, if one or more emergency diesel generators (EDGs) fail to run, it is arbitrarily assumed that they fail two hours after event initiation.
- Melting of the in-core instrument tubes was tracked in all calculations. However, only in one case (3A1) is it assumed to manifest in a temporary flow path between the RPV and the seal table room.
- In all cases (except where noted otherwise), creep failure of the hot leg and surge lines are tracked based on the Larson-Miller correlation, and if failure conditions are predicted, then a failure is modeled. Conversely, creep-induced steam generator tube rupture was tracked, but not invoked. (The probabilistic modeling addresses creep-induced steam generator tube failure and provides the boundary conditions for sensitivity calculations 3A2 and 3A3 discussed in Section 3.2 and Section 3.3 of Appendix B.)
- In general, relief valves are allowed to cycle without failing open, closed, or partially-closed. In cases where a notable number of cycles occurs, and where failure of the valve could be expected to have a significant impact on the radiological release, a sensitivity case was run assuming valve failure. (Failure of relief valves is addressed more broadly in the probabilistic model.)
- Except where otherwise noted, combustion is enabled, including random ignition. This means that if combustible gas and inertant gas compositions support combustion, it is predicted to occur. (The probabilistic modeling discussed later addresses the situation where an ignition source is not present.)
- The containment over-pressure failure pathway is assumed to be into the tendon gallery, which in turn connects to both the environment and the auxiliary building, with the leakage area apportioned through the three access shafts (two to the environment and one to the auxiliary building).[10, 11]
- Since fission product retention in secondary structures (including the secondary-side of the reactor, the tendon gallery, and the auxiliary building) is more speculative than retention within the RCS and containment, owing to the modeling resolution and state-of-knowledge, it is important to point out cases with the highest attributed retention (cases with >10 percent of the initial core inventory of cesium/iodine retained) in these secondary structures. These are:

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10 The apportionment of leakage to the environment and auxiliary building is similar for normal containment leakage (70 percent and 30 percent, respectively), but for different reasons. For normal containment leakage, the apportionment is based on the approximate surface area surrounding the containment.

11 In viewing the auxiliary building retentions direct comparison to environmental releases can be misleading because of the inclusion (in some cases) of aqueous leakage into the auxiliary building prior to containment failure. Also keep in mind that the leakage area does not linearly correlate to the leakage flow, particularly when the tendon gallery is effectively vented (i.e., large opening to the atmosphere). In this case, flow may preferentially go to the environment (which was confirmed by the MELCOR calculations) due to small pressure differences between the environment (atmospheric pressure) and the auxiliary building (whose normal leakage area would not be large enough to prevent some pressurization).

- Case 2R2 that had very large retention in the auxiliary building due to transit of highly-contaminated water
- Cases 3A2, 8R1, and 8BR1 that had very large retention in the secondary-side
- 5-series cases that had very large retention in the auxiliary building
- Cases 6C and 6D that had very large retention in the tendon gallery

- The default MELCOR MCCI and cavity modeling was employed. This manifests in two important ways in these calculations:
  - If steel and zircaloy become fully oxidized during MCCI, other metals (most notably those in the Molybdenum chemical class) show large volatility, and thus large environmental releases in some cases.
  - An infinite amount of concrete is assumed, meaning that basemat melt-through was not mechanistically predicted. Nevertheless, the report cites the time when the radial or axial ablation exceeds the known thickness of the cavity wall and basemat.
- The simulation duration in almost all cases is 7 days after the initiating event occurs. This selection is discussed further in Section 21 of Appendix D.

The analysis results can be summarized in the following observations:

- In all cases, the aerosol retention inside the reactor coolant system and the containment building (when not bypassed) is significant and contributes to lowering the releases over the short term; however, revaporization of the more volatile radionuclides (e.g., iodine) control, the releases associated with late containment failure.
- The 3-series cases (high pressure transients investigating the effect of primary-side component failures) show that instrument tube failure early may lower the likelihood of a creep-induced SGTR (by lowering primary-side pressure during core damage). These cases also show the benefit of hot leg creep rupture in lowering releases, if a creep-induced SGTR occurs. Section 4.11 of Appendix C briefly describes a case investigating the effect of SGTR timing relative to hot leg creep rupture.
- For ISLOCAs, there is significant retention in the auxiliary building (when relevant), especially in the PPAFES system, which retains most of the particulates. It is noted that the analysis results for radionuclide releases resulting from ISLOCA accident sequences may be over-estimated, since the MELCOR model does not include inertial impaction of aerosols in the RHR piping leading into the auxiliary building. On the other hand, absence of a detailed representation of the auxiliary building in the MELCOR model limits the fidelity of predicting combustible gas distribution in the building and ventilation ducts, making the analyses potentially non-conservative regarding combustion-induced auxiliary building failure.
- The series of cases with containment heat removal available (the 6-series cases) clearly demonstrates the benefit of these systems in controlling containment pressure and minimizing radiological releases, if containment does not fail due to an energetic event (as it is assumed to do in Cases 6C and 6D).
- Regarding the containment isolation failure cases (the 7-series cases), the case with fission product scrubbing (Case 7A) has a notably, but not dramatically, smaller

radiological release compared to the unscrubbed case (Case 7). This is due to the nature of the sprays credited (use of a portable pump with less capability than the installed containment spray system), the time spray is started (5 hours after hot leg creep rupture), and the duration of spray (30 hours, rather than indefinitely). Given these conditions, this is likely an accurate representation of the effectiveness of that accident management capability when deployment is delayed.

- The cases with an SGTR occurring prior to core damage (the 8-series cases) illustrate the potential benefit of early SAMG action (in 8R1 and 8BR1 vessel breach is prevented), and these cases also demonstrate the detrimental effect of a stuck-open secondary-side relief valve.

One means of establishing the relationship between this MELCOR analysis and the probabilistic modeling (beyond the comparison of MELCOR calculations to the PDS results as was done in **Section 2.1.5**), is to identify which MELCOR analyses relate to the significant accident progression sequences as determined in the dissection of the final PRA results. The significant accident progression sequences are provided in **Table 2-10**. This table shows that many of the significant accident progression sequences have clear MELCOR case analogues, while in some cases there are deviations in conditions that are acceptable. Another key location for material related to the treatment, uncertainty, and impact of various modeling assumptions is in the PRA's treatment of uncertainty (see **Section 2.4.7** and Appendix C).

A final key point is the validity of these simulations, beyond the general MELCOR validation issue addressed previously. For additional information on this issue, see Section 4 of Appendix D, which makes comparisons of various aspects of the MELCOR simulation results to a handful of other information sources, in order to provide confidence that the MELCOR results are reasonable for the purposes for which they are being used.

Table 2-8: Summary of MELCOR Analysis Boundary Conditions and Results for Cases 1-4

<table>
<tr><th></th><th>1</th><th>1A</th><th>1A1</th><th>1A2</th><th>1B</th><th>1B1</th><th>1B2</th><th>2</th><th>2R1</th><th>2R2</th><th>2A</th><th>3</th><th>3A1</th><th>3A2</th><th>3A3</th><th>3A4</th><th>4</th></tr>
<tr><td>Initiator</td><td colspan="7">Station blackout (SBO)</td><td colspan="10">Transient</td></tr>
<tr><td>RCS pipe rupture size (in.)</td><td colspan="17">n/a</td></tr>
<tr><td>Seal leakage flow (gpm/RCP)[1]</td><td colspan="5">21</td><td>182</td><td>21</td><td colspan="4">182</td><td colspan="5">n/a</td><td>182</td></tr>
<tr><td>AFW run time (hr)</td><td>∞</td><td colspan="3">4</td><td colspan="3">0</td><td colspan="4">∞</td><td colspan="5">3</td><td>∞</td></tr>
<tr><td>Early cooldown/depress. (y/n)</td><td colspan="4">Yes</td><td colspan="3">No</td><td colspan="4">Yes</td><td colspan="5">No</td><td>Yes</td></tr>
<tr><td>SAMG recovery action (y/n)</td><td colspan="8">No</td><td colspan="2">Yes</td><td colspan="7">No</td></tr>
<tr><td>Containment heat removal (y/n)</td><td colspan="17">No</td></tr>
<tr><td>Scrubbing from engineered sys.</td><td colspan="9">-</td><td>Spray</td><td colspan="7">-</td></tr>
<tr><td>Auxiliary building intact (y/n)</td><td colspan="17">Yes</td></tr>
<tr><td>Time of reactor trip (hr)</td><td>0</td><td>0</td><td>0</td><td>0</td><td>0</td><td>0</td><td>0</td><td>0.1</td><td>0.1</td><td>0.1</td><td>0.1</td><td>0</td><td>0</td><td>0</td><td>0</td><td>0</td><td>2</td></tr>
<tr><td>Time of core uncovery (hr)</td><td>76</td><td>13</td><td>13</td><td>13</td><td>2.4</td><td>1.9</td><td>2.4</td><td>6.5</td><td>6.5</td><td>6.5</td><td>6.5</td><td>8.5</td><td>8.5</td><td>8.5</td><td>8.5</td><td>8.5</td><td>7.8</td></tr>
<tr><td>Core exit thermocouple column (CETC) = 1200F (hr)</td><td>136</td><td>16</td><td>16</td><td>16</td><td>3.5</td><td>2.8</td><td>3.2</td><td>14</td><td>14</td><td>14</td><td>14</td><td>10</td><td>10</td><td>10</td><td>10</td><td>10</td><td>15</td></tr>
<tr><td>Peak clad temperature (PCT) = 2200F (hr)</td><td>139</td><td>16</td><td>16</td><td>16</td><td>3.9</td><td>3.0</td><td>3.3</td><td>15</td><td>15</td><td>15</td><td>15</td><td>11</td><td>11</td><td>11</td><td>11</td><td>11</td><td>15</td></tr>
<tr><td>Hot leg creep rupture (hr)</td><td>142</td><td>17</td><td>17</td><td>17</td><td>4.5</td><td>3.5</td><td>-</td><td>-</td><td>-</td><td>-</td><td>-</td><td>11</td><td>13</td><td>-[6]</td><td>11</td><td>-[6]</td><td>-</td></tr>
<tr><td>Vessel breach (hr)</td><td>152</td><td>21</td><td>21</td><td>21</td><td>7.7</td><td>6.9</td><td>6.1</td><td>22</td><td>30</td><td>22</td><td>22</td><td>15</td><td>15</td><td>13</td><td>15</td><td>13</td><td>23</td></tr>
<tr><td>Combustion events (y/n)</td><td>Yes</td><td>Yes</td><td>No[4]</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes[5]</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td><td>Yes</td></tr>
<tr><td>Cause of containment failure[3]</td><td>n/a</td><td>OP</td><td>OP</td><td>F</td><td>OP</td><td>OP</td><td>OP</td><td>OP</td><td>n/a</td><td>OP</td><td>F</td><td>OP</td><td>OP</td><td>BY</td><td>BY</td><td>OP</td><td>OP</td></tr>
<tr><td>Time of containment failure (hr)</td><td>>7d</td><td>68</td><td>68</td><td>28</td><td>48</td><td>49</td><td>56</td><td>90</td><td>-</td><td>120</td><td>22</td><td>56</td><td>52</td><td>10</td><td>10</td><td>65</td><td>91</td></tr>
<tr><td>Cumulative noble gas release (%)</td><td><1</td><td>83</td><td>81</td><td>99</td><td>91</td><td>91</td><td>89</td><td>68</td><td>1</td><td>86</td><td>99</td><td>86</td><td>91</td><td>95</td><td>87</td><td>86</td><td>70</td></tr>
<tr><td>Cumulative Cs release (%)</td><td><1</td><td><1</td><td><1</td><td>3.2</td><td>1.0</td><td>1.0</td><td>4.3</td><td><1</td><td><1</td><td><1</td><td>16</td><td>1.5</td><td><1</td><td>9.2</td><td>3.8</td><td><1</td><td><1</td></tr>
<tr><td>Cumulative I release (%)</td><td><1</td><td><1</td><td><1</td><td>4.3</td><td>1.2</td><td>1.6</td><td><1</td><td><1</td><td><1</td><td><1</td><td>15</td><td>2.4</td><td><1</td><td>23</td><td>7.6</td><td>1.5</td><td><1</td></tr>
<tr><td>End of calculation (hr)</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>140</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td><td>168</td></tr>
</table>

[1] This refers to the equivalent leakage at full system pressure, not the actual leakage once the system has depressurized

[2] PPAFES is operating until the filters are assumed to be over-loaded

[3] OP = long-term overpressure; BY = bypass; F = forced by sequence definition

[4] Combustion is deliberately suppressed in this calculation

[5] No burns occur inside of containment, but one or more does occur in the tendon gallery or auxiliary building

[6] Hot leg nozzle creep rupture is deliberately suppressed in this calculation

[7] The calculation terminated during in-vessel recovery due to numerical problems; no significant changes in the results after this time are expected

Table 2-9: Summary of MELCOR Analysis Boundary Conditions and Results for Cases 5-8

	5	5A	5B	5C	5D	6	6R1	6A	6B	6C	6D	7	7A	8	8R1	8R2	8A	8B	8BR1
Initiator	Interfacing Systems LOCA					Transient						SBO		SGTR (as an initiating event)					
RCS pipe rupture size (in.)	2		8	2	8	n/a													
Seal leakage flow (gpm/RCP)[1]	n/a																		
AFW run time (hr)	∞					6						4		∞			84	∞	
Early cooldown/depress. (y/n)	No											Yes		No					
SAMG recovery action (y/n)	No						Yes		No		Yes	No			Yes		No		Yes
Contain. heat removal (y/n)	Yes											No							
Scrub. from engineered sys.	PPAFES		-	-[2]	-			Spray	-		Spray	-	Spray	-	Feedwater (FW)			-	FW
Auxiliary building intact (y/n)	Yes		No	Yes	No	Yes													
Time of reactor trip (hr)	~0	~0	~0	~0	~0	0	0	0	0	0	0	0	0	0.2	0.2	0.2	0.2	0.2	0.2
Time of core uncovery (hr)	7.6	7.5	1.2	7.6	1.2	12	12	12	12	12	12	13	13	38	38	38	71	38	38
CETC = 1200F (hr)	9.5	9.5	2.9	9.5	2.9	15	15	15	15	15	15	16	16	49	49	49	95	49	49
PCT = 2200F (hr)	9.5	9.5	2.8	9.5	2.8	15	15	15	15	15	15	16	16	50	50	50	96	50	50
Hot leg creep rupture (hr)	-	-	-	-	-	16	16	16	16	16	16	17	17	52	-	52	98	-	-
Vessel breach (hr)	13	13	6.2	13	6.0	20	-	20	20	20	20	21	21	58	-	58	106	59	-
Combustion events (y/n)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No[4]	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	No
Cause of containment failure[3]	BY	BY	BY	BY	BY	BM	n/a	BM	BM	F	F	CIF	CIF	BY	BY	BY	BY	BY	BY
Time of contain. failure (hr)	0	0	0	0	0	129	-	135	139	20	20	0	0	0	0	0	0	0	0
Cumul. noble gas release (%)	99	99	86	87	86	<1	<1	<1	1.1	40	17	98	97	20	2.2	15	27	92	28
Cumulative Cs release (%)	<1	<1	9.2	<1	13	<1	<1	<1	<1	<1	<1	3.4	2.5	1.1	<1	1.1	<1	25	1.0
Cumulative I release (%)	<1	<1	12	<1	14	<1	<1	<1	<1	<1	<1	4.2	3.3	1.2	<1	1.1	<1	34	<1
End of calculation (hr)	72[8]	72[8]	72[8]	72[8]	72[8]	168	23[7]	168	168	168	168	168	168	168	59[7]	168	168	168	61[7]

1 This refers to the equivalent leakage at full system pressure, not the actual leakage once the system has depressurized

2 PPAFES is operating until the filters are assumed to be over-loaded

3 BY = bypass; F = forced by sequence/scenario definition; BM = basemat melt-through; CIF = containment isolation failure

4 Combustion is deliberately suppressed in this calculation

5 No burns occur inside of containment, but one or more does occur in the tendon gallery or auxiliary building

6 Hot leg creep rupture is deliberately suppressed in this calculation

7 The calculation terminated during in-vessel recovery due to numerical problems; no significant changes in the results after this time are expected

8 For ISLOCAs, run-time was longer and environmental releases assymptoted much more quickly, so 3 days is used as a modeling convenience

Table 2-10: Comparison of Significant Accident Progression Sequences to MELCOR Cases (see Appendix B)

Sequence	Description	MELCOR Cases
1-CET-068	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, high RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and eventual containment over-pressure failure due to the sustained MCCI. As such, it is binned in the 1-REL-LCF release category.	1A, 1A1, 1B, 1B1, 3, 3A2, 3A3, 3A4
1-CET-067	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, high RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and containment failure caused by a large combustion well after vessel breach. Thus, it is binned in the 1-REL-ICF-BURN release category.	1A2
1-CET-044	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, medium RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and eventual containment over-pressure failure due to the sustained MCCI. As such, it is binned in the 1-REL-LCF release category.	1B2, 3A1
1-CET-021	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, low RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and eventual containment over-pressure failure due to the sustained MCCI. As such, it is binned in the 1-REL-LCF release category.	1B2 and 3A1 (though RCS pressure was higher in these cases)
1-CET-017	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, low RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, scrubbing during the ex-vessel phase, and eventual containment over-pressure failure due to the sustained MCCI, though potentially delayed via the benefit of the scrubbing source on containment pressure control. As such, it is binned in the 1-REL-LCF-SC release category.	Aspects were covered by 1B2 (in-vessel) and 2R2 (ex-vessel) – no case covers the full spectrum

Table 2-10: Comparison of Significant Accident Progression Sequences to MELCOR Cases (see Appendix B)

Sequence	Description	MELCOR Cases
1-CET-020	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, low RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and containment failure caused by a large combustion well after vessel breach. Thus, it is binned in the 1-REL-ICF-BURN release category.	1A2 (though RCS pressure was higher in this case)
1-CET-043	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, intermediate RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, no scrubbing during the ex-vessel phase, and containment failure caused by a large combustion well after vessel breach. Thus, it is binned in the 1-REL-ICF-BURN release category.	1A2 (though RCS pressure was higher in this case)
1-CET-064	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, high RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment remaining intact during in-vessel melt progression, no scrubbing during in-vessel melt progression, no containment failure at the time of vessel breach, sustained MCCI, scrubbing during the ex-vessel phase, and eventual containment over-pressure failure due to the sustained MCCI, though potentially delayed via the benefit of the scrubbing source on containment pressure control. As such, it is binned in the 1-REL-LCF-SC release category.	Aspects were covered by 1A, 1A1, 1B, 1B1, 3, 3A2, 3A3, 3A4 (in-vessel) and 2R2 (ex-vessel) – no case covers the full spectrum
1-CET-072	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment initially intact, high RCS pressure during core damage, failure of in-vessel recovery (when applicable), and the occurrence of a creep-induced SG tube rupture. As such, it is binned in the 1-REL-ISGTR release category.	3A2, 3A3
1-CET-128	This sequence includes cutsets in which an ISLOCA leads to core damage, RCS pressure was low during core damage, there was scrubbing (from an overlying pool of water) during the in-vessel and ex-vessel phases, sustained MCCI occurs following vessel breach, and the auxiliary building fails. Therefore, it is binned in the 1-REL-V-F-SC release category.	5B
1-CET-136	This sequence includes cutsets in which an ISLOCA leads to core damage, RCS pressure was low during core damage, there was no scrubbing during the in-vessel or ex-vessel phase, sustained MCCI occurs following vessel breach, and the auxiliary building fails. Therefore, it is binned in the 1-REL-V-F release category.	5D

Table 2-10: Comparison of Significant Accident Progression Sequences to MELCOR Cases (see Appendix B)

Sequence	Description	MELCOR Cases
1-CET-098	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment isolation failure prior to core damage, high RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment not sustaining any additional failures during in-vessel melt progression, no scrubbing during in-vessel melt progression, sustained MCCI, and no scrubbing during the ex-vessel phase. As such, it is binned in the 1-REL-CIF release category.	7, 7A
1-CET-088	This CET sequence includes cutsets with no extension of TDAFW (when applicable), containment isolation failure prior to core damage, intermediate RCS pressure during core damage, failure of in-vessel recovery (when applicable), containment not sustaining any additional failures during in-vessel melt progression, no scrubbing during in-vessel melt progression, sustained MCCI , and no scrubbing during the ex-vessel phase. As such, it is binned in the 1-REL-CIF release category.	7, 7A (though RCS pressure was higher in these cases)
1-CET-122	This CET sequence includes cutsets with a SGTR occurring prior to core damage, intermediate RCS pressure and no scrubbing during the in-vessel melt progression, isolation failure for the ruptured SG, sustained MCCI, scrubbing during the ex-vessel phase, and a stuck-open secondary relief valve. Since the bulk of the environmental release occurs during the in-vessel melt progression, the 1-REL-SGTR-O release category is assigned despite the late scrubbing.	8B (though no late scrubbing occurs in the MELCOR calculation)
1-CET-124	This CET sequence includes cutsets with a SGTR occurring prior to core damage, intermediate RCS pressure and no scrubbing during the in-vessel melt progression, isolation failure for the ruptured SG, sustained MCCI, no scrubbing during the ex-vessel phase, and a stuck-open secondary relief valve. Thus, the 1-REL-SGTR-O release category is assigned.	8B
1-CET-140	This sequence includes cutsets in which an ISLOCA leads to core damage, RCS pressure was in the intermediate range during core damage, there was scrubbing (from an overlying pool of water) during the in-vessel and ex-vessel phases, sustained MCCI occurs following vessel breach, and the auxiliary building fails. Therefore, it is binned in the 1-REL-V-F-SC release category.	5B (though RCS pressure was low in this case) and 5C
1-CET-148	This sequence includes cutsets in which an ISLOCA leads to core damage, RCS pressure was in the intermediate range during core damage, there was no scrubbing during the in-vessel or ex-vessel phase, sustained MCCI occurs following vessel breach, and the auxiliary building fails. Therefore, it is binned in the 1-REL-V-F release category.	5D (though RCS pressure was low in this case)

2.3.4. Step 4 – Phenomenological evaluations for logic model construction

The phenomenological evaluations for split fraction assignment and logic model construction rely on: (i) past analytical and experimental investigations, (ii) performance of sensitivity calculations based on the representative sequences, and (iii) stand-alone analytical investigations. Appendix E provides a brief description of how each phenomenon was approached, and where additional documentation iss provided.

Split fractions in the probabilistic model represent, by definition, the likelihood that a certain phenomenon will occur. They represent the sequence-to-sequence variability of plant conditions, in combination with any simplifying assumptions that are made. Phenomena that are perceived to have moderate to large uncertainty are still identified as uncertainties in **Section 2.4.7**, irrespective of the selection of their split fraction. In other words, the split fraction is an attempt to develop the expected response; the model uncertainty or parameter distribution is an attempt to express the uncertainty about that estimate.

In doing the above phenomenological evaluations, there are cases where simplifying assumptions were made. Sometimes these assumptions were made because the additional resources to go to the next degree of mechanistic modeling was not warranted (e.g., assuming that a detonation fails containment rather than modeling the likelihood of this). The context of these modeling decisions is important if they are extended beyond their application to this particular PRA (e.g., if comparing to statements in rulemaking technical bases or petitions for rulemaking evaluations [e.g., NRC, 2013a]). Similarly, it is also important to recognize that these assumptions are in the context of a PRA, which deliberately goes beyond the design-basis of the plant in answering the questions of what can go wrong, how likely it is to happen, and what the consequences are.

2.3.5. Step 5 – Assessment of Post-Core Damage Recovery Actions

The analyses described in **Section 2.3.3** simulate the accident progression for various accident sequences largely in the absence of operator intervention. They are used as the basis for the human reliability analysis discussed further in **Section 2.4.4** and **Section 2.4.5**. The human reliability analysis is used to develop the set of most likely operator actions. Simulations are then re-run modeling the specific actions expected for each accident sequence. Due to resource and timing considerations, actions are only identified for the main representative sequences, and only one action each for pre- and post- vessel breach. Resources are further saved by not performing calculations that are fundamentally similar to other recovery calculations already performed. The resulting set of recovery calculations are included with the representative sequences and sensitivities (see Sections 2.1, 2.2, 6.1, 8.1, and 8.2 of Appendix B). Several different actions are considered, and they are estimated to have varying degrees of effectiveness.

Among those sequence/action combinations that had the largest effect are:

- Depressurizing and feeding two steam generators using a condensate pump before vessel breach in Representative Sequence 2 (Case 2R1)
- Feeding the faulted steam generator and dumping steam to the condenser before vessel breach in Representative Sequence 8 (Case 8R1)

The actions in Case 2R1 decrease fission product releases to the environment by two orders of magnitude by preventing containment overpressure failure. The lower containment pressure is due to heat transfer from containment to the steam generators. The actions in Case 8R1 end fission product releases to the environment by scrubbing releases through the broken steam generator tube and diverting the remaining releases to the condenser. The actions also prevent hot leg creep rupture and RPV lower head failure.

Among those sequence/action combinations that had the least effect are:

- Using firewater to provide containment spray in Representative Sequence 2 (Case 2R2)
- Feeding the faulted steam generator and dumping steam to the condenser after vessel breach in Representative Sequence 8 (Case 8R2)

The actions in Case 2R2 delayed containment failure, but do not have a large effect on the results otherwise, due to the limited capacity assumed (in terms of both flow rate and available inventory). The actions in Case 8R2 have little impact because environmental releases in the unmitigated case effectively stop before water is added to the faulted steam generator.

In addition, calculations were run early in the project to investigate the effect of flooding the reactor cavity late in the accident to terminate the accident prior to basemat melt-through or long-term containment over-pressurization. In general, they show that these late actions are likely to significantly reduce further radiological releases and prevent long-term containment over-pressurization, but are not likely to prevent basemat melt-through.

2.3.6. Step 6 – Evaluation of equipment survivability

This section addresses the issue of equipment survivability under severe accident conditions. Separate sections are provided for background information, the relatively simplistic approach employed in the L3PRA project, and potential enhancements to improve the level of rigor of the analysis.

Equipment Survivability Background Information

In developing the approach used for equipment survivability, several sources were reviewed with respect to generic methodologies, accident lessons learned, etc., such as:

- NUREG/CR-5513, “Accident Management Information Needs” (1990)
- NUREG/CR-5691, “Instrumentation Availability for a Pressurized Water Reactor with a Large Dry Containment During Severe Accidents” (1991)
- NUMARC-87-00, Revision 1, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors” (1991)
- EPRI TR-102371, "Instrument Performance Under Severe Accident Conditions: Ways to Acquire Information from Instrumentation Affected by an Accident" (1993)
- EPRI TR-103412, "Assessment of Existing Plant Instrumentation for Severe Accident Management" (1993)
- Westinghouse AP1000 Design Control Document, Revision 17, Tier 2, Chapter 19, Appendix 19D, “Equipment Survivability Assessment” (2008).[12]
- IAEA TECDOC-1661, “Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants” (2011)

[12] This source in turn references additional relevant information in EPRI NP-4354 and NUREG/CR-5334.

- SECY-12-0025, Enclosure 3, "Enhanced Reactor and Containment Instrumentation Withstanding Beyond-Design-Basis Conditions" (2012)
- INL/EXT-13-28043, "TMI-2 – A Case Study for PWR Instrumentation Performance During a Severe Accident" (2013)
- ORNL/TM-2013/154, "Fukushima Daiichi – A Case Study for [Boiling Water Reactor] Instrumentation and Control Systems Performance during a Severe Accident" (2013)
- EPRI Report No. 3002005385, "Severe Nuclear Accidents: Lessons Learned for Instrumentation, Control, and Human Factors," (2015)

These studies generally support the notion of developing quantitative survivability estimates by doing environmental load and capacity comparisons, but do not provide a clear path forward for assessing instrument reliability when the load exceeds the environmental qualification (EQ) capacity (i.e., assessing the margin above the EQ capacity). Of note, the Three Mile Island (TMI) accident and the Fukushima Dai-ichi nuclear power plant (NPP) accident case studies demonstrate that significant instrumentation availability issues can arise during a severe accident and that environmental effects on instrument cabling[13] can be important to instrument performance. Specific concerns in this regard stemming from the Fukushima Dai-ichi NPP response include:

- Reference leg boiloff leading to false Unit 1 reactor pressure vessel water level indication (EPRI, 2015)
- A ruptured vacuum breaker valve on Unit 2 (TEPCO, 2015)
- The apparent drop in Unit 2 suppression chamber pressure on March 15, 2011, believed to be due to a sensor failure (but at the time, correlated to suppression chamber failure based on misleading coincidental timing with the Unit 4 combustion event) (NAS, 2014)
- An unexpected enabling signal for automatic depressurization on high suppression chamber pressure in Unit 3 (TEPCO, 2013)
- Erratic behavior of certain reactor pressure vessel temperature thermocouple circuits (EPRI, 2015)

Two reports, GEND-INF-023, Volume 1, "Investigation of Hydrogen-Burn Damage in the Three Mile Island Unit 2 Reactor Building," June 1982, and GEND-INF-023, Volume 2, "Estimated Temperatures of Organic Materials in the TMI-2 Reactor Building During Hydrogen Burn," December 1982 (both EG&G reports), discuss the limited hydrogen burn that took place during the TMI-2 accident, with a focus on damage to organic materials. As part of this work, material temperatures were estimated to reach the 300 degrees Fahrenheit to 500 degrees Fahrenheit range in specific locations in the TMI-2 containment, and seemingly random damage to some material and equipment was noted, but without any obvious loss-of-function to mitigation equipment.

L3PRA Project Equipment Survivability Approach

To orient the analyst in the types of spatial equipment survivability issues that could arise in modeling operator actions post core-damage, the physical plant layout and sample

[13] In addition to pressure, temperature, or humidity/flooding effects on cabling, another potential issue is direct contact with molten debris in accident sequences with high-pressure melt ejection. The results of the L3PRA project support that RPV failure at very high pressure would be exceedingly unlikely due to the prevalence of hot leg nozzle creep rupture in these accident sequences.

environmental loads were investigated, along with comparing the resolution of the reference plant's design-basis EQ envelope relative to the environmental conditions predictable using the MELCOR model. This work is described in Section 8 of Appendix D. The interplay between equipment survivability, accident analysis modeling, structural performance, and accident management actions is also described in (Helton, 2014).

The information captured above was assimilated, along with knowledge of how the various equipment responds to environmental challenges (to the extent practicable at the level of reference-plant-specific information available), to consider whether:

- The hardware is likely to experience conditions that would challenge it
- The hardware is likely to fail for specific accident sequences
- The response is indeterminate based on the current information

The last category is only addressed via model uncertainty identification.

Table 8-1 of Appendix D captures specific observations related to survivability for the instrumentation used to guide SAMG navigation. Two areas of concern are the survivability of the hydrogen sampling lines following vessel breach (in reference to Severe Challenge Guideline (SCG)-3, Severe Accident Guideline (SAG)-7, CA-3, CA-7) and core-exit thermocouples following core damage (in reference to SAG-3). In both cases, the survivability determination is intertwined with how the TSC interprets the information and what other relevant information it has available to make determinations. As such, the detailed description of the treatment of these two issues is described in the human reliability model development section (i.e., **Section 2.4.4**) of this report, ultimately leading to the bottom-line treatment that the hydrogen sampling lines are plugged and unavailable following vessel breach, while sufficient temperature information is available to guide entrance into SAG-3.

An overview of specific survivability issues considered during the screening HRA are provided in **Table 2-11**.

Table 2-11: Specific survivability issues from screening HRA

Rep. Seq. #	Item	Basis
1 (Extended SBO)	In the base accident sequence no credit is taken for any HRA actions initiated following core damage. In the MELCOR simulation, manual operation of TDAFW and condensate storage tank refill initiated prior to core damage are assumed to continue following core damage.	
2 (Loss of all NSCW pumps)	Use of PORVs in Severe Accident Guideline (SAG)-2 implementation prior to vessel breach	It is assumed that the potential effects of high-temperature gasses would be one factor leading to the use of a different means of depressurizing.
	Use of reactor head vents in SAG-2 implementation prior to vessel breach	It is assumed that the potential effects of high-temperature gasses would be one factor leading to the use of a different means of depressurizing.
	"Bumping" RCPs in SAG-3 implementation-prior to vessel breach	It is assumed that the loss of seal cooling and indications of seal leakage would preclude use of the RCPs.
	"Bumping" RCPs in SAG-3 implementation following vessel breach	It is assumed that the loss of seal cooling and indications of seal leakage would preclude use of the RCPs.

Table 2-11: Specific survivability issues from screening HRA

Rep. Seq. #	Item	Basis
3 (Dual-train electrical w/ AFW fail-to-run and seal leakage)	Opening SG ARVs in SAG-1 to allow low-pressure SG injection	High temperatures are expected on the secondary side due to dry SGs during core damage; however, for the base case, where induced SG tube rupture is not modeled, no specific survivability issues are known (e.g., weakening of MSIVs).
4 (Loss of AC bus with NSCW fans out for maintenance)	"Bumping" RCPs in SAG-3 implementation prior to vessel breach	It is assumed that the loss of seal cooling and indications of seal leakage would preclude use of the RCPs.
	"Bumping" RCPs in SAG-3 implementation following vessel breach	It is assumed that the loss of seal cooling and indications of seal leakage would preclude use of the RCPs.
5 (Interfacing system LOCA)	Using the unaffected (by the initial Interfacing System LOCA break) train of ECCS in SAG-3 implementation prior to vessel breach	It is assumed that a mixture of survivability and habitability concerns would prevent the use of the other train of ECCS due to its proximity to the damaged train (flooding, radiation shine).
	"Bumping" RCPs in SAG-3 implementation prior to vessel breach	It is assumed that the RCPs will remain available for the very brief action of bumping them (to push water in the crossover leg into the vessel downcomer), but no credit is eventually given because this action does not meet the screening criteria.
	Using the unaffected (by the initial ISLOCA break) train of ECCS in SAG-3 implementation following vessel breach	It is assumed that a mixture of survivability and habitability concerns would prevent the use of the other train of ECCS due to its proximity to the damaged train (flooding, radiation shine).
	"Bumping" RCPs in SAG-3 implementation following vessel breach	It is assumed that the RCPs will be unavailable due to continued harsh conditions in the RCS.
6 (Transient with ECCS and no feed)	Use of centrifugal charging pump, safety injection (SI), or RHR pumps in recirculation mode prior to vessel breach	It is assumed that the RHR pump is preferentially used, in part due to it being less susceptible to affects from sump debris (below the size that would be captured by the sump strainers); the possibility of sump clogging is acknowledged, but not assumed to occur.
8 (Steam generator tube rupture)	Use of steam dump valves in Severe Challenge Guideline (SCG)-1 prior to vessel breach	Steam dump valves are assumed to survive, based on their distance downstream in the secondary piping, but their use could be impacted by the open main steam isolation valve (MSIV) on the affected SG.
	Isolating the affected SG in SCG-1 prior to vessel breach	Not credited, in part because secondary piping instrumentation may be affected by radiation contamination.
	Gagging the affected SG relief valves in SCG-1 prior to vessel breach	Not credited, in part because relief valves may be affected by radiation contamination.

Table 2-11: Specific survivability issues from screening HRA

Rep. Seq. #	Item	Basis
	Use of steam dump valves in SCG-1 following vessel breach	Steam dump valves are assumed to survive, based on their distance downstream in the secondary piping, but their use could be impacted by the open MSIV on the affected SG.
	Isolating the affected SG in SCG-1 following vessel breach	Not credited, in part because secondary piping instrumentation may be affected by radiation contamination.

Next, installed systems (or components) that may be directly affected by harsh environments and were assumed to automatically operate (or actuate) following core damage are catalogued in **Table 2-12**.

Table 2-12: Survivability assumptions for automatically-demanded installed systems

System/component	Accident sequences	Comments
Control room and TSC ventilation and filtering systems	All	The HRA assumes that these systems operate in non-SBO situations to a sufficient degree to support MCR actions and TSC decision-making – see Section 9 of Appendix D for more information.
Piping penetration area filtration and exhaust system	ISLOCA	Survivability of this system is considered in the context of failure of the auxiliary building, and whether an ISLOCA release is scrubbed.
Containment hydrogen sampling	These are not credited following vessel breach	
Containment hydrogen recombiners	These are not credited anywhere in the model	They are retired in place at the plant.
Fire detection and suppression systems	These are not considered by the model	
Containment cooling units	S5 (ISLOCA): 8/8 CCUs available S6: 4/8 CCUs available by the time of core damage In all other representative sequences, the support system failures that made other equipment unavailable (leading to core damage) also made both trains of CCUs unavailable.	A concern here would be with airborne debris being ingested by the cooling units and leading to fouling of the cooling units. Due to the high intake elevation (above the operating deck) and the large heat exchanger surface area, this is not expected to be a significant concern. In both representative sequences in question, combustions were predicted within containment after vessel breach, and the ability of the CCUs to withstand dynamic pressure loads has not been assessed; they are designed to withstand quasi-static pressure loads for final safety analysis report Chapter 6 accidents.

Table 2-12: Survivability assumptions for automatically-demanded installed systems

System/component	Accident sequences	Comments
Other, including containment sprays		Not credited unless specifically called out in the HRA (in which case survivability is considered therein), except for Case 6A where they are credited.

Regarding systems that might be used late in the accident to attempt to flood the reactor cavity (see Section 21 of Appendix D), survivability of this equipment is not explicitly modeled. This is, in part, because no single set of equipment is relied on, and because the ultimate use of differing truncation times only conceptually relies on these late actions (i.e., the actions themselves are not explicitly modeled).

Potential Enhancements to the Treatment of Equipment Survivability

Regarding equipment survivability treatment more generally, the logical next step in rigor would be to:

- Identify the subset of assumptions that have the largest impact on the quantified model, either using importance measures (in the case of parameter uncertainty) or sensitivity analyses (in the case of modeling uncertainty)

- Use MELCOR results to quantitatively estimate the predicted environmental loads (pressure and temperature) in terms of ambient gas temperature time-history, and separately, to estimate the activity of radiological material that will contribute to the equipment's dose

- Use analytical or numerical one-dimensional heat transfer equations to estimate the component peak temperature, and separately, estimate the equipment dose.[14]

- Compare the calculated pressure and temperature load to the equipment location's EQ limits

However, gaps in environmental load characterization, plant-specific equipment information, and project resources prevent further progress in this area. In any event, this would still leave cable routing and the potential satisfactory performance of equipment beyond its EQ limit as key gaps.

A related issue is the treatment of RCS components (e.g., surge line, hot leg nozzle, in-core instrument tubes) due to high temperatures, but this issue is treated directly in the MELCOR modeling and related sensitivity analyses.

[14] An example application of the MAAP5-DOSE tool is discussed in P. Maka, et al., "Usage of MAAP5-DOSE to Support Equipment Survivability Assessments," presented at PSA 2015, Sun Valley, ID, April 26-30, 2015. Given the uncertainties and limitations (e.g., core debris is not considered) in this type of analysis, it is best suited for providing quantitative results that can be used to develop qualitative guidance for making equipment dose evaluations. A significant amount of design information is required to set up this type of model.

2.4. Probabilistic Treatment of Accident Progression

This subtask consists of seven interrelated steps:

1. Reliability of SSCs not considered in the Level 1 PRA
2. Construction of the containment event tree (CET)
3. Development of support trees
4. Human reliability model development
5. Human reliability analysis (HRA)
6. Level 2 model quantification
7. Uncertainty characterization

The objective of the first step is to consider the reliability of SSCs not considered in the Level 1 PRA. The objective of the second step is to develop the set of accident progression event tree (a.k.a., CET) top events, and the tree's logic structure. The objective of the third step is to develop the severe accident phenomena and system response logic modeling (i.e., decomposition event trees [DETs]) needed to support the CET top events. The objective of the fourth step is to develop the HRA model to be used in considering post-core damage actions. The objective of the fifth step is to exercise the HRA for the representative sequences. The objective of the sixth step is to quantify the Level 2 PRA, to arrive at release category frequencies (since the release categories are the end-states of the CET). The objective of the seventh step is to identify sources of parameter and model uncertainty, characterize these sources, and use uncertainty propagation and sensitivity analyses to assess the effects of key sources of uncertainty. Each of these steps is discussed in further detail in this section.

2.4.1. Step 1 – Reliability of SSCs not considered in the Level 1 PRA

The reliability of containment systems to perform their engineered, design-basis accident functions is captured in the model by including containment sprays, containment fan coolers, and containment isolation in the bridge event tree. The underlying fault tree models in the bridge event tree capture these systems' reliability following the lead of the Level 1 PRA's treatment of data issues, common-cause failure, etc. (NRC, 2022a). Their failure is explicitly captured in the sequences (and cutsets) leading into the containment event tree.

This leaves systems that are modeled (or otherwise considered) within the containment event tree (and more precisely within the supporting decomposition event trees). These systems are those that either actuate automatically based on their design or are manually operated as part of the post-core damage accident management (SAMGs, EDMGs, or accident termination efforts). Reliability models were not developed for these systems for the following combination of reasons:

- Human reliability and equipment/instrument survivability considerations for the use of this equipment are discussed in **Section 2.3.6**. These considerations lead to failure probabilities that are generally much higher (0.1 to 1.0 range) than those that would be estimated in random equipment failure models (typically lower than 0.01 for engineered design-basis systems).
- The above-mentioned human reliability evaluations include consideration of system and component availability, such as system unavailabilities due to sequence-specific support system failures (e.g., AC power, NSCW cooling).

- In most cases, data for these systems and components is not obtained as part of the NRC's data collection processes (either because it is not reportable in the underlying data streams or because it is not relevant for Standardized Plant Analysis Risk (SPAR) model maintenance). As such, reliability evaluations would be resource-intensive and/or data-starved. (The use of surrogate data for similar systems or components was considered, but not pursued, in light of the considerations above.)

In other words, detailed reliability evaluations for these systems would likely not have a first-order effect on their overall probability of being successfully employed and would thereby have limited value. The systems for which this decision applies are provided in **Table 2-13**.

A separate data-oriented question is the treatment of offsite power recovery during loss-of-offsite power events. Information on this issue is routinely used in Level 1 PRAs (including the SPAR models). For instance, NUREG/CR-6890, Tables A-4 through A-7 provide this information from the 1986 – 2004 reporting period (Eide, 2005). There are decisions made along the way that may influence the data's applicability to longer time-frames (e.g., curve fit selection), since it was designed for use in Level 1 PRA models that are typically focused on a 24-hour mission time. In addition, there is much less data in the underlying dataset for LOOPs lasting longer than 24 hours. For these reasons, significant additional effort would be needed to address this.

Based on the above factors, the Level 2 PRA does not model AC power recovery (beyond that already modeled in the Level 1 PRA), as a simplifying assumption. This is identified as a model uncertainty in Section 4.3.4 of Appendix C. The offsite power recovery paradigm in the Level 1 PRA is that once direct current (DC) power has depleted in station blackout accidents, AC power cannot be restored (e.g., it has not been demonstrated that the relevant breakers can all be closed mechanically).

Finally, on another related matter, the Level 2 PRA did not treat equipment repair, or recovery of out-of-service equipment. As discussed later in **Section 2.4.4**, the Level 2 HRA assumes a preference for strategies that rely on using available equipment over repair or recovery of damaged or out-of-service equipment. This simplifying assumption is consistent with the Level 1 PRA and represents a limitation in the state-of-practice. For instance, the combined Level 1/large early release frequency (LERF) standard (ASME, 2013) has supporting requirements related to crediting repair and recovery that require a justification based on analysis or data examination (SY-A24) and the assimilation of applicable operating experience (DA-C15). Very limited operating experience for severe accident conditions exists, and the application of operating experience from other industries in a means that would meet these types of requirements would require significant effort and is not part of the Level 2 PRA state-of-practice.

Table 2-13: Systems for which reliability estimates were not generated

System/component	Comments
Post-accident monitoring instrumentation	No credit is given for actions when no AC/DC power is available post-core damage.

Table 2-13: Systems for which reliability estimates were not generated

System/component	Comments
Security diesel	The security diesel provides power to both the security system and the TSC in the case of a LOOP. No credit is given for actions when no AC/DC power is available post-core damage (i.e., SBO), so the effect of this is limited to LOOPs where the EDGs function but the security diesel does not. Also, HEPs for modeled post-core damage actions are high, and alternative locations (in the case of the TSC) and plant access (in the case of the security system) exist. For these reasons, the impact of not modeling the security diesel reliability is expected to be small.
Main control room and TSC ventilation and filtering systems	When TSC ventilation is out-of-service, a backup location is specified. The quantitative impact of using the backup location during an actual emergency on human error probabilities would be very speculative.
Maintenance and EOF availability	As above, the impacts of this on human error probabilities would be very speculative.
Effluent radiation monitors (e.g., main plant stack monitor)	These are not relied on in the modeling because virtually every release is from an unmonitored location.
Piping penetration area filter and exhaust system (PPAFES) and auxiliary building ventilation system	These are only relevant for substantial leaks into the auxiliary building (namely, containment isolation failures and ISLOCA) for which the HVAC system is not treated as damaged by static or dynamic over-pressure.
Containment hydrogen sampling	These are not credited following vessel breach.
Containment hydrogen recombiners	These systems are retired-in-place and are not credited anywhere in the model.
Portable equipment (e.g., diesel-driven trailer-mounted pump)	See preceding discussion regarding the effect of high HEPs (i.e., the bulleted list at the beginning of the section).
Valves used in accident management	See preceding discussion regarding the effect of high HEPs (i.e., the bulleted list at the beginning of the section).
Fire detection and suppression systems	These are not considered by the model.
Some installed equipment not within the scope of the Level 1 PRA that may be used for accident management	Examples include: condensate pumps, hand-crank operators on SG ARVs

2.4.2. Step 2 – Construction of the Containment Event Tree

One CET is used for managing all Level 2 PRA sequences. Top events in the CET are organized approximately in a chronological or causal order.

- *In-Vessel Time Frame:* This period starts from the beginning of core damage and lasts up until (but not including) the time of vessel breach. Potentially important phenomena include hydrogen combustion; in-vessel steam explosions; and temperature-induced creep rupture of the hot leg nozzle, pressurizer surge line, or SG tubes.
- *Vessel Failure Time Frame:* This period includes the time of vessel breach, as well as the time associated with the containment transient just after vessel breach (typically a duration of less than 30 minutes). Potentially important phenomena accompanying

vessel breach include direct containment heating, hydrogen combustion, vessel rocketing, and ex-vessel steam explosions.

- *Initial Ex-Vessel Time Frame:* This period begins at the end of vessel blowdown (i.e., the end of the vessel failure time frame), and lasts for approximately 8 to 12 hours thereafter. The duration of this time frame was chosen such that it includes the majority of the ex-vessel core debris oxidation and fission product release. Potentially important phenomena in this time frame include ex-vessel steam explosions and combustion of hydrogen and/or carbon monoxide generated during molten core-concrete interaction (MCCI).
- *Very Late Time Frame:* This period extends from the end of the initial ex-vessel time frame until the end of the Level 2 PRA sequence. The potentially important phenomena in this time frame include quasi-static pressurization of the containment due to MCCI and decay heat, revaporization of radionuclides from surfaces (only relevant to source term analysis), and potential Basemat Melt-Through (BMT) during MCCI.

An issue that quickly arises is when to terminate the Level 2 PRA sequence relative to the time of simulation (e.g., stop all sequences 36 hours after SAMG entry or carry all sequences to their ultimate end-point). This issue is discussed at length in Section 21 of Appendix D. As described there, sequences are carried either to the point that the core is recovered in-vessel or ex-vessel (i.e., further core degradation or containment cavity basemat attack has been terminated), or to a pre-established stop time (3 days for ISLOCAs, and 7 days for other initiators[15]). Results are generally presented based on these run-times, but the effect of truncating these times on the release frequency profile and relevant risk surrogates is also provided.

One of the inputs in the development of the Level 2 logic model is the reference plant Level 2 PRA, circa 2011. Some aspects of the reference plant's Level 2 PRA model were reviewed in detail, including its treatment of:

- RCS pressure at the time of core damage, and SAMG-based depressurization
- Steam generator cooling
- Power recovery during SBO
- Energetic events that have the potential to lead to containment failure
- Fission product scrubbing
- Long-term coolability of ex-vessel debris

A single CET is used to handle all Level 2 PRA sequences in the L3PRA project. The CET contains the following top events:

1. Special treatment for extremely slowly-developing accident sequences (1-L2-REC) - This top event is used as a means of giving special treatment[16] to extremely slowly-developing accident sequences. It only applies to indefinite blind feeding of SGs past

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[15] The use of 3 days for ISLOCAs is simply a modeling convenience employed because these simulations generally ran more slowly and environmental releases assymptoted much more quickly, relative to other simulations.

[16] This term is simply used to communicate that these sequences are treated differently in the CET than the remainder of the Level 2 PRA sequences.

battery depletion that dramatically delays the time of core damage for SBO accident sequences without elevated RCP seal leakage. Successful blind feeding at this top event is routed straight to a no containment failure end state, whereas the full accident progression treatment is performed otherwise.

2. Containment status at the time of core damage (1-L2-SUM-CONTINT) - This event summarizes the status of the containment or containment bypass as of the time of core damage, using information from the bridge and plant damage state event trees (i.e., containment isolation status from the former and accident type from the latter). This information is necessary for the downstream logic structure to properly account for differences in the behavior of issues such as bypass versus non-bypass sequences.

3. RCS pressure before vessel breach (1-L2-DET-PRESVE) - This event questions the RCS primary-side pressure during the time frame between the start of core damage and before vessel breach. It has three potential branches (low, medium, and high pressure) and was evaluated via a decomposition event tree (DET).

4. In-vessel recovery after the start of core damage (1-L2-IVREC) - This event evaluates the potential to arrest core damage by means of post-core damage action in time to prevent core relocation and vessel breach. Its success is evaluated based on the RCS pressure, hardware availabilities, and the amount of time available for accident management. For ISLOCA sequences, no in-vessel recovery is considered since RCS inventory cannot be maintained indefinitely. Recovery for SGTRs is not explicitly treated in the 1-L2-IVREC top event, but the same effect is essentially covered by the scrubbing top events associated with these sequences (in the sense that the accident simulation that involves successful SGTR scrubbing prior to vessel breach effectively results in termination of radiological releases at that point).

5. Containment status during in-vessel phase (1-L2-DET-CONTVE) - This event evaluates whether containment failure occurs during the time frame after the start of core damage and before vessel breach. It has three branches corresponding to: containment intact, containment failed due to overpressure or energetic event (e.g., hydrogen combustion), and containment bypassed due to a pressure/temperature-induced steam generator tube rupture. It is evaluated using a DET.

6. Scrubbing of radionuclide release during in-vessel and vessel failure phase (1-L2-DET-SCRUBE) - This event questions whether the phase of the radiological release occurring during the time frame at or before vessel breach is mitigated by either a pool of water overlying the break location (in the case of containment bypass) or by containment systems (in the absence of containment bypass). It is evaluated using a DET.

7. Containment status at vessel breach (1-L2-DET-CONTE) - This event evaluates the potential for containment failure at or around the time of vessel breach due to phenomena such as in-vessel or ex-vessel steam explosion, vessel rocketing, direct containment heating, or hydrogen combustion. It is evaluated using a DET.

8. Molten core-concrete interaction (1-L2-MCCI) - This event questions whether sustained basemat attack occurs in the reactor cavity by core-concrete interactions following vessel breach. It is evaluated using logic rules based upon the presence of water in the cavity. In addition, energetic events at vessel breach, such as steam

explosions, could disperse the debris, rendering it coolable, and prevent concentrated basemat attack.

9. Scrubbing of radioactive release after vessel failure (1-L2-DET-SCRUBL) - This event evaluates whether mitigation of the release by sprays or water pools occurs in the time frame following vessel breach. It is otherwise similar to event SCRUBE (above), with the additional factor that water in the reactor cavity may be present to scrub the ex-vessel release. It is evaluated using a DET.

10. Containment status well after vessel failure (1-L2-DET-CONTL) - This event evaluates whether containment failure occurs in the late time frame following vessel breach, via overpressure or energetic event (e.g., hydrogen combustion), or by basemat melt-through. It is evaluated using a DET.

11. Atmosphere relief valve status (1-L2-ARV) - This event determines whether SG relief/safety valves are predominantly closed or cycling during release, or whether they remain open due to deliberate action or failure. This is important only for SGTR sequences, in order to assign them to a proper release category.

12. Auxiliary building status (1-L2-DET-AB) - This event evaluates whether the auxiliary building fails due to overpressure during the post-core damage accident progression (e.g., due to hydrogen combustion). It is important in assigning the proper release category for ISLOCA sequences, and it is evaluated using a DET.

These top events combine the functionality of accident sequence characterization with that of a release categorization tree. As such, the end-states of the CET are the release categories (as opposed to a transfer to a release categorization tree). There are 148 sequences mapped to 16 release categories. The CET is shown in **Figure 2-8** through **Figure 2-11**.

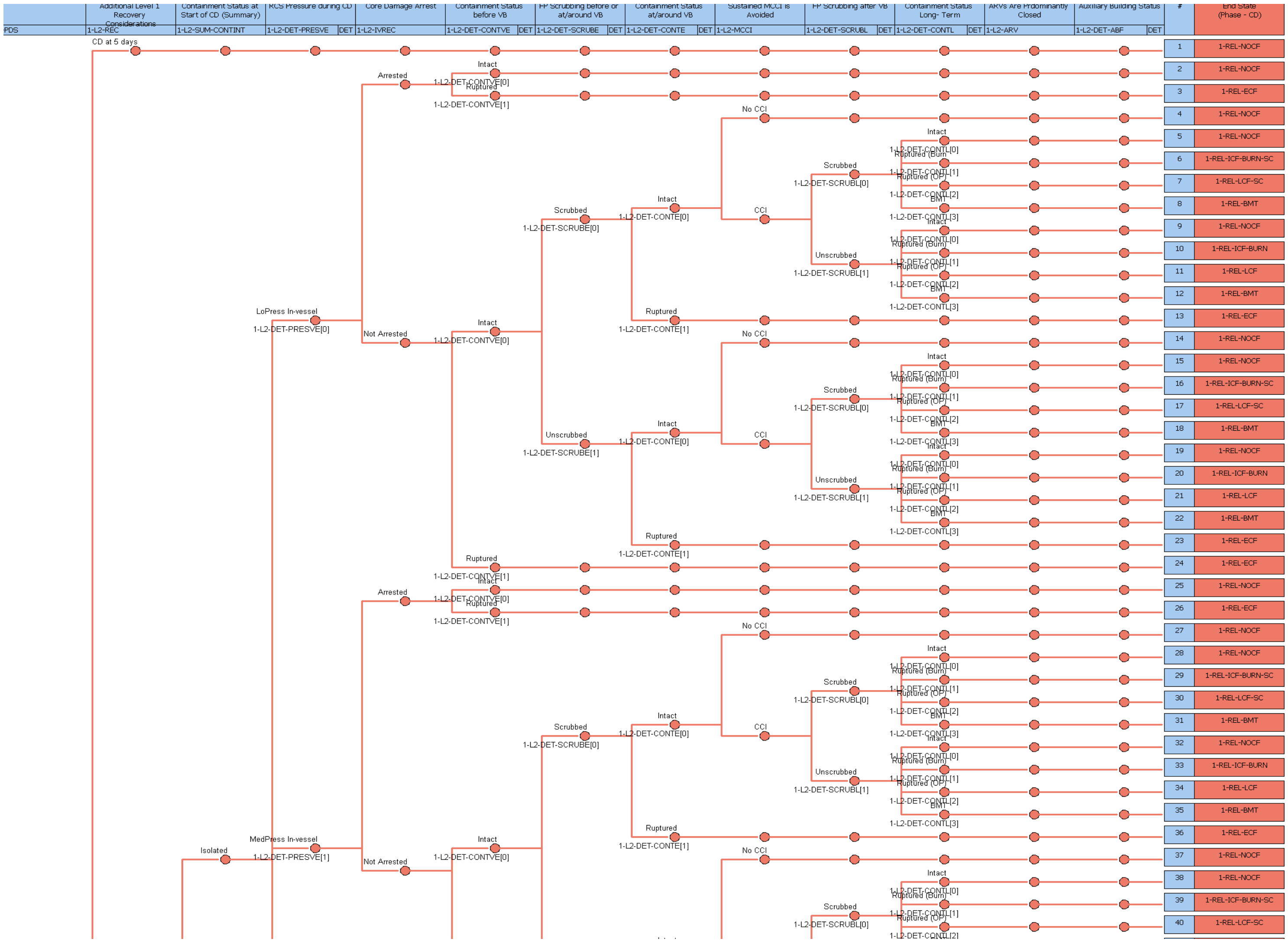

Figure 2-7: Containment Event Tree (1 of 4)

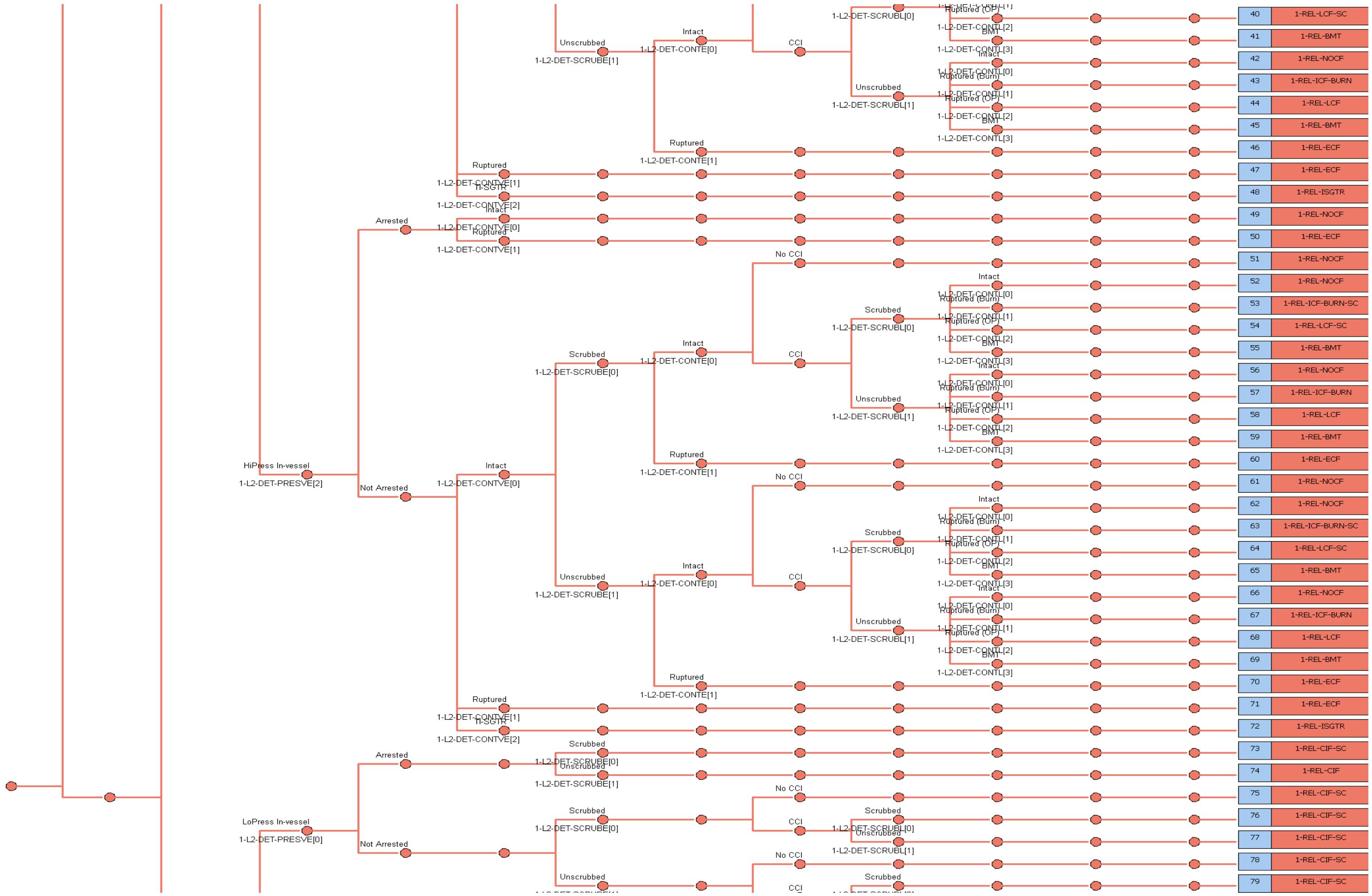

Figure 2-8: Containment Event Tree (2 of 4)

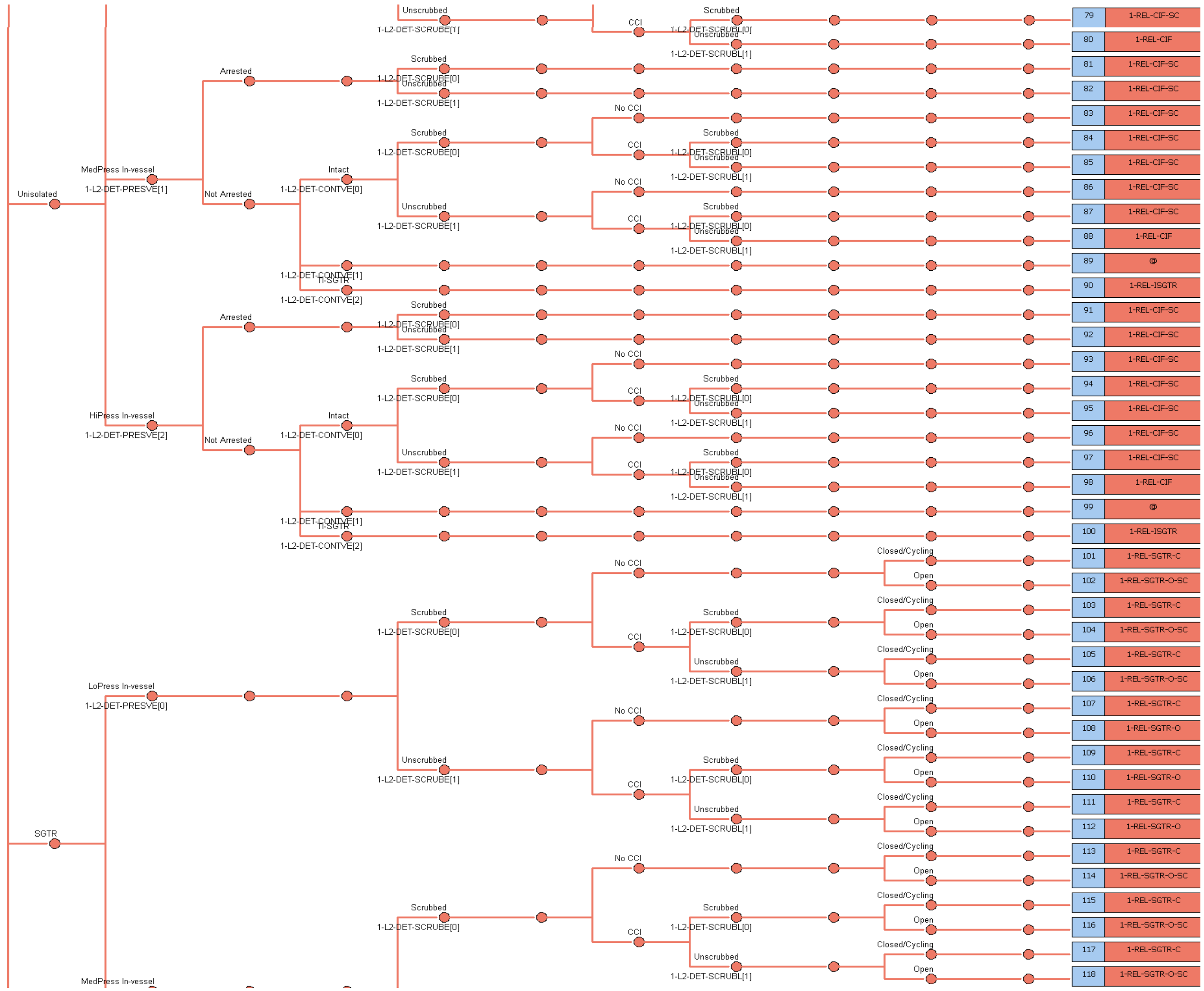

Figure 2-9: Containment Event Tree (3 of 4)

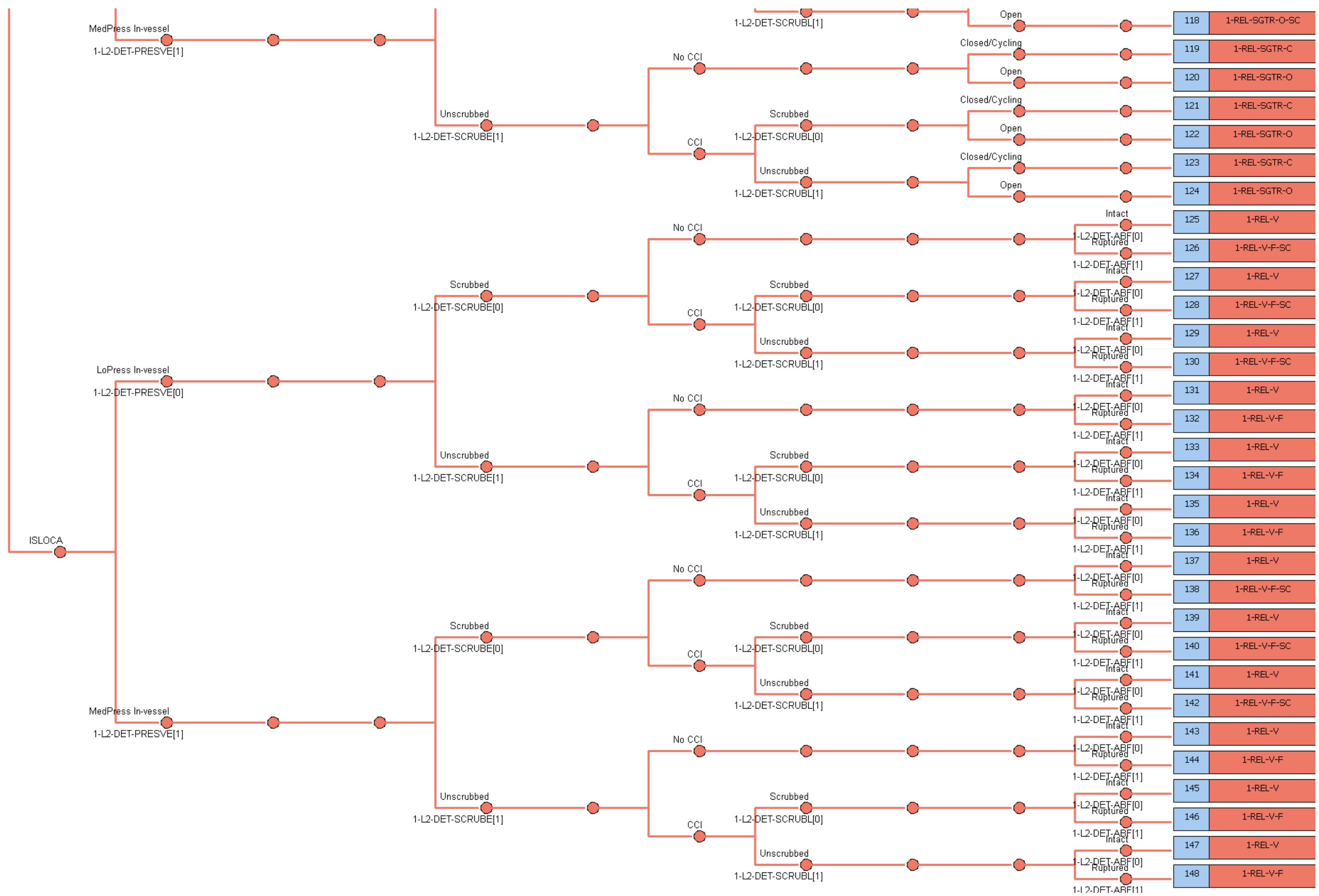

Figure 2-10: Containment Event Tree (4 of 4)

2.4.3. Step 3 – Development of support trees

The support trees used take the following forms:

- Fault trees to support the top events in the bridge event tree – these are discussed in **Section 2.1.1**
- Linkage rule logic to support the plant damage state tree – these are discussed conceptually in **Section 2.1.2** and **Section 2.1.3**
- Fault trees and/or linkage rules to support some CET top events – these are briefly mentioned in **Section 2.4.2**
- DETs to support the remaining CET top events – these are discussed briefly in this section

The seven CET top events represented by DETs are repeated here:

1. RCS pressure before vessel breach (1-L2-DET-PRESVE)
2. Containment status during in-vessel phase (1-L2-DET-CONTVE)
3. Scrubbing of radionuclide release during in-vessel and vessel failure phase (1-L2-DET-SCRUBE)
4. Containment status at vessel breach (1-L2-DET-CONTE)
5. Scrubbing of radioactive release after vessel failure (1-L2-DET-SCRUBL)
6. Containment status well after vessel breach (1-L2-DET-CONTL)
7. Auxiliary building status (1-L2-DET-ABF)

A DET functionally expands a single top event in a main event tree (i.e., the CET, in the L3PRA project) into a secondary event tree with its own top events, logic structure, and logic rules. DETs are employed as part of the Level 2 PRA model for the L3PRA project because they enable the main CET to be simplified into only 12 top events for determining release categories, with details of individual phenomena handled by the DETs. This representation also simplifies the presentation of the model by reducing the number of branches in the CET. Logically, SAPHIRE treats each DET as if it were inserted into the main event tree in place of the main tree's top event that represents the DET.

2.4.4. Step 4 – Human reliability model development

Within the area of nuclear reactor PRA, the focus of HRA has been primarily on at-power, internal events, post-initiator, control room operator actions that are taken while following emergency operating procedures. There are very few HRA applications that have supported PRA for post-core-damage accident sequences. Also, there has been very limited HRA method development aimed at supporting such PRA studies (e.g., [Baumont, 2000]). Consequently, for the most part, Level 2 PRA HRA applications to-date were performed using the existing HRA methods that are intended for use in supporting at-power, Level 1 internal events PRAs.

The fact that no state-of-practice method exists in Level 2 HRA[17] is the subject of several articles, such as (Raganelli, 2014) and (Boring, 2015). These articles discuss:

[17] As used in this report, the term "Level 2 HRA" is short-hand for an HRA performed for a Level 2 PRA, and the term "Level 1 HRA" is short-hand for an HRA performed for a Level 1 PRA.

- The potential for capturing instrument survivability and human performance feedback using existing methods like NARA (Nuclear Action Reliability Assessment) and SPAR-H (Standardized Plant Analysis Risk – HRA)
- The practice of capturing post-core damage actions at a high level, not akin to the task breakdown level-of-detail of Level 1 HRA (thus reducing the resolution of the human performance context)
- The role of higher degrees of uncertainty (e.g., phenomenological uncertainty) in governing the level-of-detail for Level 2 HRA
- The need to address 'stress' or 'burden' as a fundamental indicator of human performance, and how this may limit the utility of expert judgment estimation
- Use of existing methods in some Level 2 PRAs (e.g., THERP [Technique for Human Error Rate Prediction], ASEP [Accident Sequence Evaluation Program], and MERMOS, SPAR-H)
- The lack of any existing method for handling Level 1 to Level 2 dependency
- Dominant determiners of human performance, posited to be information and operator team preparedness (training/teamwork)

For the L3PRA project, it is important to generate human reliability estimates so that operator actions can be integrated into the overall PRA model. As such, a post-core-damage HRA approach was developed for this study that uses L3PRA project team experience (including HRA experts from the NRC staff, Sandia National Laboratories, and Idaho National Laboratory) and aspects of existing methods. However, it is acknowledged that for the reasons cited above, the human reliability estimates generated through application of this approach are very uncertain. A summary of the HRA approach and some additional background information are provided below.

2.4.4.1. *Preparatory work*

As part of the effort to understand the accident management framework for the reference plant, fundamental steps were taken, including reviewing the guiding procedures and guidelines, walking down portions of the plant germane to accident management, and discussing accident management training, exercising, philosophies, and emergency preparedness drill insights with site personnel. Also, a licensee internal document describing a previous emergency preparedness drill for the reference plant, which included limited use of the EDMGs and SAMGs, was provided to the L3PRA project team. In addition, the project team performed MELCOR simulations to provide further context to the plant response expected for the accident sequence used in the drill. The review of the emergency preparedness drill provided insights on human performance aspects that might otherwise have been missed or de-emphasized, such as:

- The vacating of areas where system re-alignments are taking place, when those re-alignments are expected to change the local radiation fields (and their corresponding effect on repair or local manual actions taking place in those same areas)
- Delays that can be introduced in the time required to make repairs or take local manual actions when a Health Physics or security escort is needed
- Unanticipated complications that can arise in the field (in this case, as part of the Field Monitoring Team's activities)

- Potential delays introduced by maintenance personnel not being familiar with the equipment used in the EDMGs

The review was also helpful in identifying many things that went smoothly during the drill, as well as providing a greater familiarity with the anecdotal aspects of the accident management approach helpful for guiding reference plant staff interviews.

Finally, (Echeverria, 1994) provides a reference handbook for use by NRC inspectors to help determine the impacts of specific environmental conditions on licensee personnel performance (Volume 1) and a companion literature review (Volume 2). The environmental conditions investigated includes vibration, noise, heat, cold, and lighting. While generally oriented toward more routine occupational demands (i.e., less challenging than the insults possible during some specific accident management actions), this provides useful reference material on the effects of these challenges on human performance. The Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Human and Organizational Factors (WGHOF) was conducting research in this area at the time of the L3PRA project. A workshop was held in March 2014 focused on human performance under extreme conditions.[18] Topics discussed there that are of relevance here include:

- Decisions regarding repair versus alternative equipment during the Fukushima Daiichi nuclear power plant (NPP) accident response (e.g., pursuing the use of fire engines rather than repairing damaged installed equipment)
- Human performance complications during the Fukushima Daiichi NPP accident response, including:
    - Lack of, and conflicting, information available on the existing systems
    - Challenges with radiation fields
    - Availability of procedures and staff with appropriate technical capability
- Gaps in human performance modeling capabilities

This activity was documented in (NEA, 2015).

2.4.4.2. *SAMG structure and navigation*

The reference plant SAMGs (circa 2012), relied on in the L3PRA project, follow the Westinghouse SAMGs in place since 1998 (i.e., prior to the planned upgrade based on lessons learned from the Fukushima Daiichi NPP accident and the associated completed update of the industry SAMG basis codified in [EPRI, 2012]). The basic structure of the SAMGs is depicted in **Figure 2-12**.

[18] For the purposes of that workshop, extreme conditions were defined as events characterized by one or more of the following attributes: (i) unexpected accidents, not covered by training or procedures, (ii) beyond design-basis, loss of safeguards and safety barriers, (iii) dynamic, rapidly changing, escalating, accumulating, insufficient and unreliable information, (iv) complex, long-term duration, (v) challenging the organization (on-site and off-site), and (vi) potential loss of health or life.

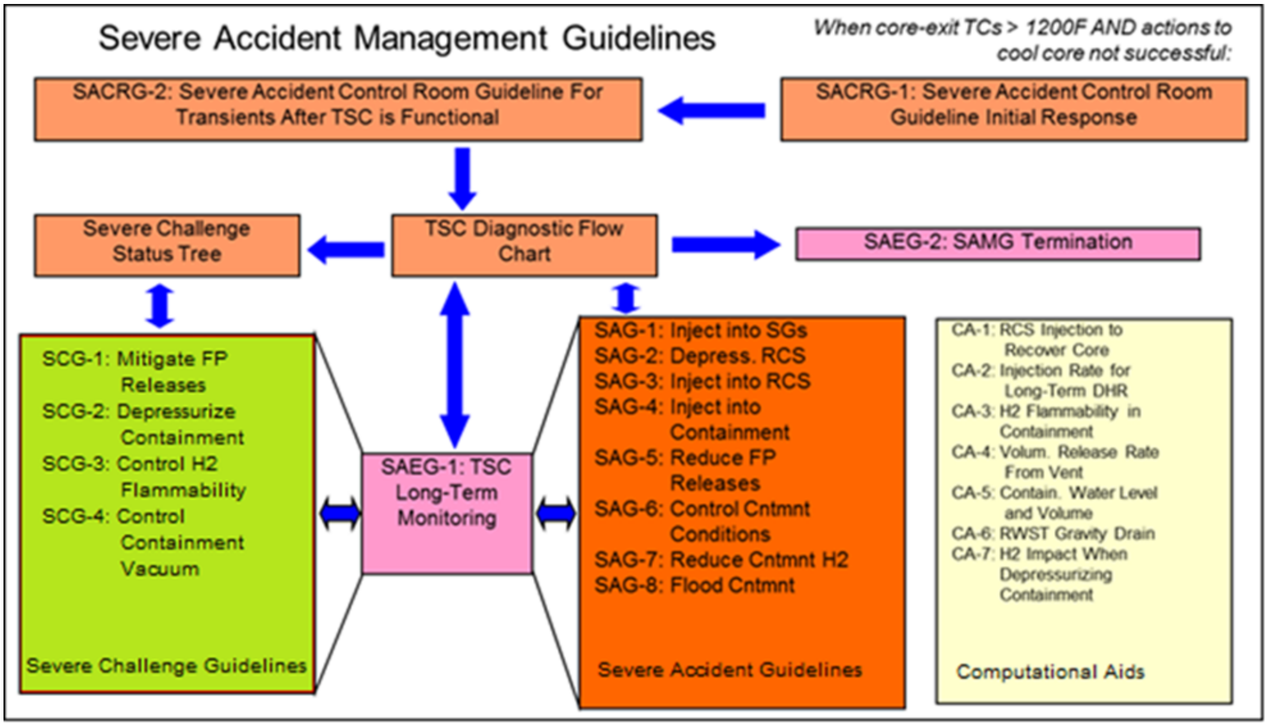

Figure 2-11: SAMG Structure

Of particular importance to the HRA, is that the SAMGs are hierarchical; that is, the 12 high-level actions encapsulated in the Severe Challenge Guidelines (SCGs) and Severe Accident Guidelines (SAGs) are prioritized. Their entrance is governed (through the Diagnostic Flow Chart [DFC] and Severe Challenge Status Tree [SCST]) by specific plant parameters, and they are ordered in a manner to reflect their importance, with SCG-1 being the highest priority (when entrance criteria are met) and SAG-8 being the lowest priority. Two control room guidance documents (with or without the TSC active), two exit guidelines (long-term monitoring of actions and concerns, and SAMG termination), and seven computational aids round out the guidance set.

Also of importance is the fact that the EDMG strategies applicable to the reactor can be viewed as specific strategies that support a high-level action identified by the SAMGs. That is, the EDMGs can be viewed as a means to an end identified (albeit without explicit reference) by the SAMGs, as opposed to a completely independent set of guidelines that must be considered in parallel to the SAMGs. This perspective can be seen by looking at the specific EDMG strategies, as identified in **Table 2-14**.[19]

Table 2-14: Mapping of EDMG strategies to SAMG guidance

EDMG Strategy	Relevance to SAMGs for the L3PRA Project
App A – Extensive Damage Mitigation Guideline	n/a (command and control)

[19] The mapping in this table is not intended to be generically applicable to other plants, or even at the reference plant in other contexts.

Table 2-14: Mapping of EDMG strategies to SAMG guidance

EDMG Strategy	Relevance to SAMGs for the L3PRA Project
App B – RWST Makeup Using External Sources	Numerous SAMG guidelines reference RWST makeup
App C – Containment Flooding Using Godwin* Pump	Supports SCG-1 (mitigate fission product releases**), SCG-2 (depressurize containment**), SAG-4 (inject into containment), SAG-5 (reduce fission product releases**), SAG-6 (control containment conditions**), and SAG-8 (flood containment)
App D – Makeup to Condensate Storage Tank	Supports SAG-1 (inject into SGs) and SAG-2 (depressurize RCS***)
App E – Fire System Strategies	General supporting function for other EDMGs (or SAMG actions that utilize fire water)
App F – Manual Operation of TDAFW Pump	Supports SAG-1 (inject into SGs) and SAG-2 (depressurize RCS***)
App G – Manually Depressurize SGs	Supports SAG-1 (inject into SGs) and SAG-2 (depressurize RCS***)
App H – AFW Supply Using Godwin Pump	Supports SAG-1 (inject into SGs) and SAG-2 (depressurize RCS***)
App I – Spraying Down Release Points	Supports SCG-1 (mitigate fission product releases), and SAG-5 (reduce fission product releases)
App J – Godwin* Pump Setup and Operation	General supporting function for other EDMGs
App K – Godwin* Pump and Accessories	General supporting function for other EDMGs
App L – SFP Makeup Using Internal Sources	n/a
App M – SFP Makeup or Spray Using External Sources	n/a
App N – SFP Leakage Control Strategies	n/a
App O – SG Indication Measurement	Supports SAG-1 (inject into SGs) and SAG-2 (depressurize RCS***)
App P – Rx Vessel Head Vent Strategy	Supports SCG-3 (control H2 flammability), SCG-4 (control containment vacuum), SAG-2 (depressurize RCS), SAG-3 (inject into RCS), and SAG-6 (control containment conditions)

* Godwin is the manufacturer's name for the portable trailer-mounted diesel-driven pump used by many plants to meet the regulatory requirements of 10 CFR 50.54(hh)(2); it is used here (and elsewhere) synonymously with "EDMG pump" or "B5b pump".

** This strategy uses the Godwin pump to inject water through the containment spray headers, so it is a potential means for scrubbing fission products, condensing steam, and increasing sump water level.

*** This refers to the role of feedwater in actions involving primary-side depressurization via dumping steam.

2.4.4.3. *Special considerations in modeling SAMG navigation*

Most of the thresholds specified in the SAMG DFC and SCST are plant conditions that can be directly extracted from MELCOR simulations for the purposes of informing human reliability analysis, and the translation done for the L3PRA project is described in **Section 2.3.2**. There are a few exceptions to this, and a few aspects that warrant further discussion. These are covered in Section 14 of Appendix D and are briefly synopsized here.

Entry into SCG-1 (Mitigate Fission Product Releases) or SAG-5 (Reduce Fission Product Releases) is based on site doses. Significant work was undertaken to model their entry as realistically as possible, using the type of information that would be available to plant staff during accidents involving unmonitored releases. This work resulted in the development of a decision flow-chart (**Figure 2-13**) that was used in the HRA to govern entrance to these guidelines.

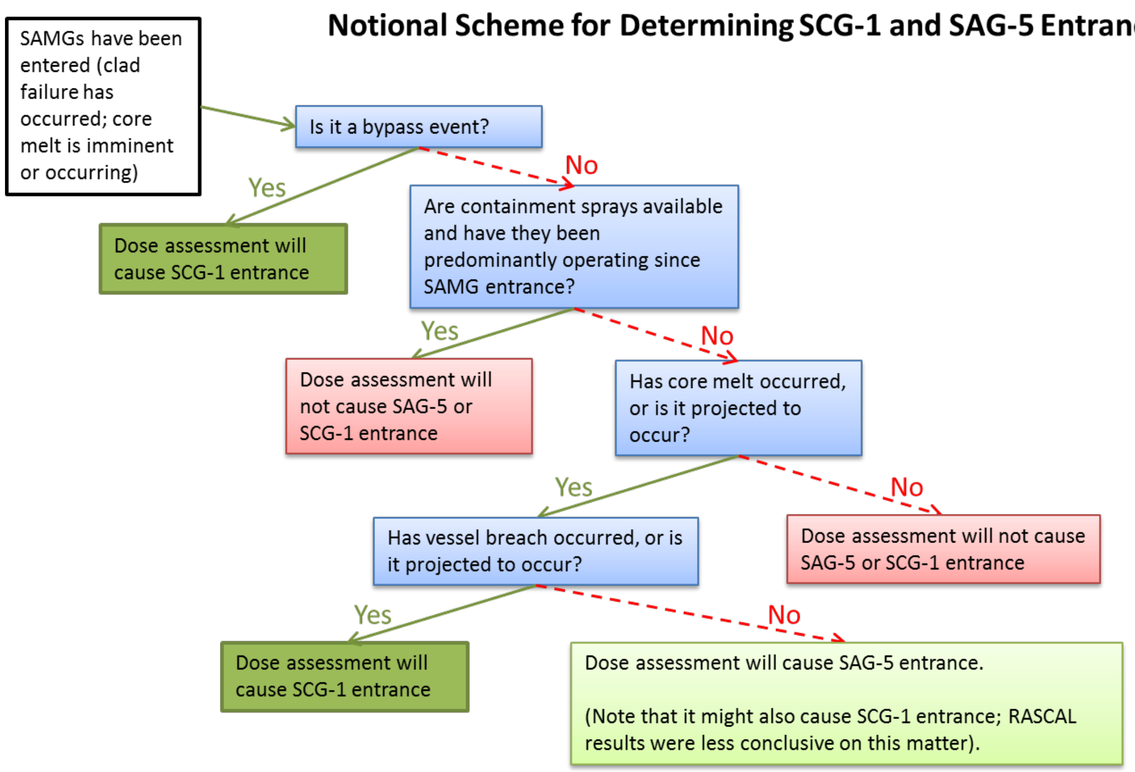

Figure 2-12: Modeling of Decision for SCG-1 and SAG-5 Entry

There are also complexities related to the entry into SCG-3 (Control Hydrogen Flammability) and SAG-3 (Inject into the RCS) that led to the following judgments:

- Regarding the evaluation of SCG-3 entry:
  - From the time of SAMG entrance to vessel breach, hydrogen monitoring and sampling are assumed to be available, so long as DC power is available
  - From the time of vessel breach to the end of the accident, hydrogen monitoring and sampling may be compromised by plugging, and are assumed unavailable (default computation aid curves are used instead)
  - In terms of SCG-3 entry, no distinction is made based on how far from the flammability line the conditions are (i.e., just above the line is treated the same as well above the line in terms of denoting that a flammability hazard exists)
- Regarding the evaluation of SAG-3 entry:
  - Sufficient information would be available to guide this decision (unless DC power is unavailable)
  - The guideline would be entered when conditions prompt, regardless of the status of vessel failure

2.4.4.4. *Additional background regarding E-Plan and SAMG interface*

An assumption needed to be made regarding whether the timing of TSC activation may impact the time to diagnose and start the execution of the first post-core damage actions. This issue is closely coupled to the assumptions related to emergency action level (EAL) classification, since declaration of an “Alert” is the formal trigger for activating the TSC. This topic is discussed in greater detail in Section 2.5. That discussion provides the basis for why it is assumed here that the TSC is already activated at the time of SAMG entrance, unless a set of conditions leads to SAMG entrance occurring within 1 hour of the first off-normal condition, which would be rare. The modeling assumption related to this is embedded in the identified source term modeling assumption related to the timing of EAL declarations.

In light of all of the preceding discussion, below is further discussion on the relationship between the Emergency Plan and the EOPs/SAMGs, in terms of the onsite response infrastructure.

- The entry into the SAMGs is not the triggering event for transferring command and control responsibilities to the Emergency Director (ED); it is the declaration of an Alert (or higher).
- Typically, Emergency Plans have the ED in charge. Early in the event, the Shift Manager (SM) serves this role, but will transfer command and control to another qualified ED once one becomes available with adequate support. This transition occurs regardless of whether the plant is in the SAMGs. Except for very fast accident sequences or cases where the control room missed an EAL declaration, the TSC should normally be manned and someone other than the SM would have the ED role (along with overall command and control) by the time the SAMGs are entered. [The L3PRA project assumptions and basis related to TSC activation are described in **Section 2.5.2** of this report.]
- Upon entrance to the SAMGs, there would be a shift in how accident management actions are identified. Once this transition is made, the response follows the less prescriptive SAMGs that require an evaluation function from the TSC. Prior to this transition, the operators follow the procedural framework provided by the EOPs (which do not generally require the TSC evaluation function, although the TSC and ED will serve in an advisory and consent role even under the EOPs).
- The SAMG framework includes provisions for the Main Control Room crew to deal with post-core damage sequences prior to the point at which the TSC has developed a strategy. This is covered by the SACRG-1 procedure that repeats many steps contained in the EOPs, and is focused on:
  - having the crew place the plant in a situation where equipment will not automatically actuate (e.g., placing non-operating equipment in pull-to-lock)
  - establishing support systems that may be needed to carry out SAMG actions (e.g., instrument air to containment)
  - taking actions to place the plant in a safer state (e.g., RCS depressurization and injection) – although many of these may not be feasible given that they were not already taken or were ineffective in response to similar emergency operating procedure direction
  - establishing periodic monitoring of information of relevance to the SAMGs (e.g., sump sampling)
- Once the TSC has developed its initial strategy, the Main Control Room Crew transitions to SACRG-2.

2.4.4.5. *Overall HRA process overview*

Figure 2-14 below shows how the screening and detailed HRA fits in the overall PRA.

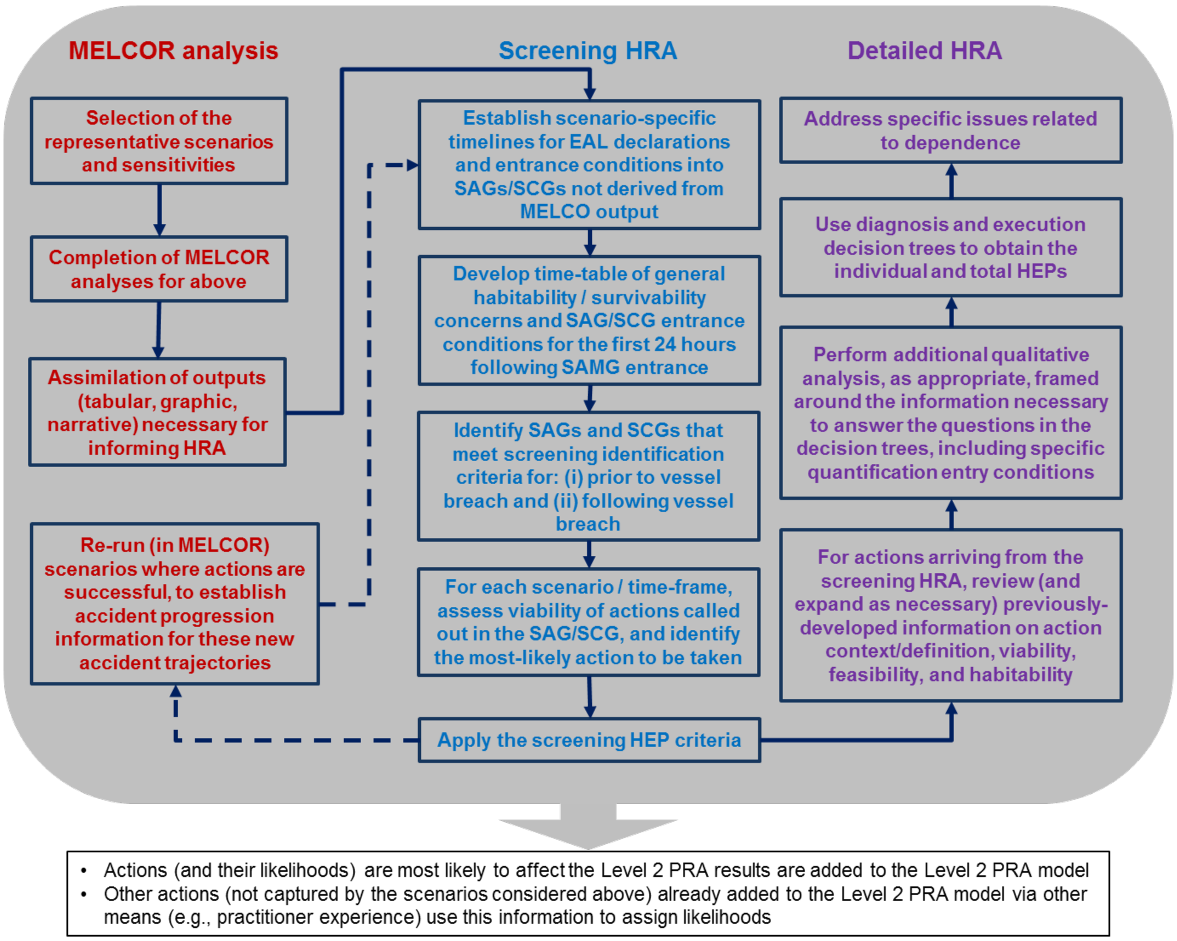

Figure 2-13: Overall Human Reliability Analysis Application Framework

In the above process, simulations are run using MELCOR, and the results of relevance to the HRA are identified. These results are used to establish accident timings, including when conditions are present for entrance into the 12 guidelines contained in the SAMGs. This process is depicted in the red text and is described in more detail in **Section 2.3.3** of this report. (The figure also denotes the feedback loop by which identified actions lead to new MELCOR analyses to assess their effectiveness, as described further in **Section 2.3.5**.)

A set of screening assumptions is then used to establish which potential actions will receive further scrutiny. For each distinct combination of accident sequence, timeframe, and guideline considered, a viability assessment is performed to determine the most likely action to be taken. This viability assessment considers issues such as habitability, survivability, and availability. A screening HEP is then assigned. This process is depicted in the blue text above.

Next, the identified actions are processed through a more detailed methodology to ensure their feasibility. Finally, a more accurate estimate of their probability of successful diagnosis and implementation is developed using decision trees, taking into account key performance shaping factors. This process is depicted in the purple text.

Additionally, simplified timings are assumed to prescribe the deterministic modeling of the recovery actions, as follows:

- Guideline entry:
  - 30 minutes after conditions exist for SCG-1 and SAG-5 due to dose projection / field monitoring team interplay
  - Otherwise, 15 minutes after conditions exist
- Diagnosis:
  - 30 minutes to work through pros/cons, and get ED buy-in
- Start of implementation:
  - 15 minutes for control room actions (to allow relaying of direction and any clarification)
  - 30 minutes for straight-forward local actions
  - One hour for local actions involving harsh environments, set up of equipment, etc.

These assumed timings were discussed with staff from the reference plant. In general, the plant personnel felt that the assumed timings were reasonable. They noted that some preparatory activities would already be in motion as a part of the EOP-based response. They also noted that any instances where actions were not using pieces of already existing procedures would take substantially longer in the diagnosis phase. For out-in-the-field implementation, plant personnel noted that this would be very dependent on staffing. Due to these considerations, and the time required for transit and briefings, they ultimately felt that generic best estimates should probably add 15-45 minutes to the overall times that one would estimate when summing the times listed above. This, in turn, placed additional emphasis on the step in the detailed HRA where the team compared these time estimates to the time available (and found there to be adequate margin).

A key assumption in the screening HRA and recovery modeling is that the feedback loop envisioned within a given SCG or SAG is not modeled. For instance, if two means of accomplishing a strategy are available (opening SG ARVs or opening a PORV in order to depressurize the RCS), whichever means is determined to be the more likely decision by the TSC is enacted, with no potential that it would be observed after some time to be ineffective and supplanted by the other means.

Except for survivability considerations, the inoperability or unavailability of needed equipment is not accounted for in the probabilistic estimation of the failure likelihood for the recovery actions. This is discussed more in **Section 2.4.1**, and is viewed as reasonable given the high probabilities associated with failure to diagnoses/execute (0.1 and higher in the screening HRA). Note, this refers to availability from the perspective of random hardware failures or maintenance of the equipment in question; unavailability due to support system failures would be captured in the availability assessment performed in the screening HRA.

On a related note, there is an implicit assumption regarding availability of adequate communications equipment, which includes plant telephones, status loop communications, voice over internet protocol (VOIP) phones, satellite phones, and/or sound-powered phones. The importance of this assumption is greatly reduced by the lack of credit given for SAMG actions during SBO events.

Also, in selecting the most probable action, emphasis is placed on using available alternative equipment over repair of damaged equipment or returning out-of-service equipment to service. In subsequent discussions with reference plant staff, they agreed that this was a reasonable assumption, particularly with respect to damaged equipment. However, they stressed that so long as adequate staff was available, they would pursue repair in parallel to use of alternative equipment. They also stressed the wide-spread understanding of the status of out-of-service equipment during normal operations (through maintenance management software accessible by plant staff, the TSC, and the EOF), which would foster efficiency in returning necessary equipment to service. Nevertheless, this level of detail is not captured in the current HRA approach.

Finally, the following related topics are documented in Appendix D of this report:

- Section 9, “Habitability in Accident Management” (i.e., role of hazardous environments in limiting accessibility for local actions) – the perspective described therein guided qualitative determinations on that front relative to plant access for taking local actions
- Section 21, “Termination of Radiological Releases” (i.e., role of human actions in ultimately terminating the accident) – the perspective described therein guided the treatment of human actions late in the event

2.4.5. Step 5 – Human reliability analysis

The preceding section described the process used for the Level 2 HRA. The application of the HRA approach to arrive at screening and detailed HRA results included the feasibility assessment, qualitative analysis, and quantitative analysis for each of the modeled actions, along with the underlying context development leveraging the previously-described MELCOR analyses. A summary of the resulting HFEs and HEPs is provided here.

As previously identified in Section 2.1.3, the internal flooding contributions for the current PRA are not significant enough to warrant special attention to flooding-unique effects. The application of the HRA method here has considered impediments to accessibility to areas for local action from several factors (including ISLOCA-induced flooding), but not from the perspective of internal flooding associated with the initiating event. As such, the HRA results (and the current Level 2 PRA in general) are limited in terms of their application to internal flooding events.

Table 2-15 summarizes the results of applying the screening and detailed HRA to the representative sequence results.

Table 2-15: HEPs by Representative Sequence

Rep. Seq. #	Modeled Action	Screening HEP	Detailed HEP (rounded)
S1	SBO event – no action modeled due to complete lack of instrumentation (battery depletion) by the time core damage occurs		
S2	SAG-2: Open all SG ARVs prior to vessel breach (VB)	0.1	0.03
	SCG-1: Firewater thru containment spray nozzles following VB*	0.5	0.6
S3	SAG-1: Open ARVs on 2 SGs and feed SGs with condensate prior to VB	0.5	0.4
	SCG-1: Firewater thru containment spray nozzles following VB*	0.5	0.6
	SAG-1: Open ARVs on 2 SGs and feed SGs with condensate following VB*	0.5**	0.4
S4	SAG-2: Open all SG ARVs prior to VB	0.1	0.1

Table 2-15: HEPs by Representative Sequence

Rep. Seq. #	Modeled Action	Screening HEP	Detailed HEP (rounded)
	SCG-1: Firewater thru containment spray nozzles following VB*	0.5	0.6
S5	No actions found that both met the screening criteria and were viable/relevant	-	-
S6	SAG-3: Start RHR and align alternate inventory	0.9***	-
	SCG-1: Start containment sprays and align alternate inventory	0.5***	-
S7	SBO event – no action modeled due to complete lack of instrumentation (battery depletion) by the time core damage occurs		
S8	SCG-1: Feed and bleed SG prior to VB	0.5	0.1
	SCG-1: Feed and bleed SG following VB*	0.5	0.5

* These cases do not include success of the modeled action prior to VB

** This HFE was not originally assigned an HEP during the screening HRA, but one was retroactively estimated, since the HFE received a detailed HRA. An HEP of 0.5 was selected because non-safety DC power was available (thus not 1.0), the action was the highest priority prior to VB (thus not 0.9), and a habitability concern exists (thus not 0.1).

*** The screening HRA for this sequence was originally deferred due to time constraints, its low (relative) contribution to CDF, and the completion of other intact containment sequences. While eventually completed, this did not occur in time to allow a detailed HRA analysis

Table 2-16 further distills these HRA results by PRA sequence applicability, to guide their inclusion in the Level 2 PRA logic model. It provides the basic event names assigned to these operator actions, and these basic events are ultimately used by the top events of the relevant DETs. In this way, **Table 2-16** provides the bridge between the HEP development and its application in the model.

Table 2-16: Breakdown of HEPs by Accident Sequence Characteristics

Sequence	Action modeled	HEP	Basic event name/origin; other notes
Station blackout	None	n/a	Informed by S1, S7 Determined not feasible based on lack of instrumentation
SGTR (as an initiating event)	Feeding SGs prior to VB	0.1	1-L2-OP-SCG1-2; Informed by S8
	Feeding SGs following VB	0.5	1-L2-OP-SCG1-3; Informed by S8
ISLOCA	None	n/a	Informed by S5.
Non-SBO / Non-bypass / Isolated containment, w/ FW at CD	Depressurize RCS prior to VB	0.07	1-L2-OP-SAG2-1; Informed by S2, S4 (simple average of 0.03 and 0.1)
	Firewater through spray headers following VB	0.6	1-L2-OP-SCG1-1; Informed by S2, S4
Non-SBO / Non-bypass / Isolated containment, w/out FW at CD (alternate FW not possible, but ECCS available)	Inject into RCS prior to VB	0.9*	Not presently used in the model; Informed by S6
	Containment sprays following VB	0.1**	1-L2-OP-SCG1-4

Table 2-16: Breakdown of HEPs by Accident Sequence Characteristics

Sequence	Action modeled	HEP	Basic event name/origin; other notes
Non-SBO / Non-bypass / Isolated containment, w/out FW at CD (alternate FW not possible and ECCS unavailable)	Containment sprays following VB	0.1**	1-L2-OP-SCG1-4; Not actually covered by any representative sequences
Non-SBO / Non-bypass / Isolated containment, w/out FW at CD (alternate FW possible)	Feed SGs and dump steam prior to VB	0.4	1-L2-OP-SAG1; Informed by S3
	Firewater through spray headers following VB	0.6	1-L2-OP-SCG1-1; Informed by S3
Other	Hydrogen control late	1.0	1-L2-OP-H2CTL-L; did not manifest itself in the HRA
Other	RCS primary-side depressurization during core damage	1.0	1-L2-OP-PRIDEPRES-VE; timing of depressurization in HRA is generally after the onset of core damage

* HEP based on simplified HRA (no detailed HRA performed) – the high HEP may be realistic in that this action may have a high degree of dependency with failed Level 1 actions (but since it is not used, dependency has not been assessed)

** No detailed HRA was performed, but the results of a detailed HRA were extrapolated using the detailed HRA for S2 and S4; specifically (relative to S2 and S4), the diagnosis branch on procedural support changes from Fair to Good, and the diagnosis HEP = 0.08; the execution branch on the location of the action changes from Ex-CR to MCR, the complexity changes from High to Low, and the execution HEP = 0.01; therefore, the total HEP is ~0.1

2.4.6. Step 6 – Level 2 model quantification

2.4.6.1. *Quantification background information*

The default treatment in SAPHIRE is to use system logic for the failure branch and the delete term for the success branch. Such treatment, in effect, assigns a conditional probability of 1 to the success branch, and thus does not identically conserve total input frequency. The effect of this approximation is small for low failure probabilities (since the corresponding success probabilities would be very nearly 1.0); however, Level 2 PRA often involves events with very large failure probabilities (sometimes approaching unity) for which use of the delete term would result in unacceptable effects. Therefore, the approach taken in this analysis is to use SAPHIRE's "Y" process flag (developed event for failure, complement of developed event for success) for all basic events with a probability of approximately 0.01 or higher. The general topic of process flag settings in light of high failure probabilities is discussed further in (Ma, 2014).

The model also uses rule-based switches to allow the user to solve for a specific level of result without the need to change numerous event tree end-states. By adjusting the linking rules, the user can readily switch between core damage quantification, PDS quantification, PDS event tree linkage rule debugging, and release frequency quantification. The "1-CD-XFER" event tree is used to accomplish this.

No event-tree-specific post-processing rules are applied to the Level 1 event trees. However, there are several global post-processing rules that are applied to these trees. These global post-processing rules operate on cutsets, rather than sequences, in the following ways:

- Disallowed maintenance combinations are deleted
- R-calc generated CCF cross-terms are deleted
- Macros are defined (which do not directly affect quantification)
- Some emergency diesel generator (EDG) recovery cutsets are manipulated to address incorrect behavior that is difficult to fix in the FTs
- HFE substitutions are performed to address HEP dependency
- Convolution correction rules are applied

While these rules do not affect sequence logic and, therefore, do not affect the Level 2 PRA's use of sequence information, they do have the potential to impact quantification. Level 2 cutsets have additional Level 2 basic events appended to the Level 1 basic events (thus reducing the cutset frequency), prior to executing these rules. This creates the potential for cutsets to be truncated prior to having their frequency increased back above the truncation level by these rules. However, the Level 2 truncation (i.e., convergence) study generally addresses this concern.

2.4.6.2. *Quantification results*

The combined Level 1 and Level 2 PRA model discussed previously was linked for all internal initiating events and internal floods and solved using a cutoff frequency of 1×10^{-11}/rcy. The quantification resulted in a total release frequency of approximately 7×10^{-5}/rcy for a single unit, approximately the same as CDF, by virtue of including intact containment end-states.

Table 2-17 provides the release category bin contributions (the release category scheme itself is discussed in **Section 2.5.1**) for the combined internal events and floods initiators. These are radiological release results; they do not include offsite consequence results, which will be part of the Level 3 portion of the L3PRA project. **Table 2-17** shows these contributions for all three truncation times considered in the L3PRA project. The SAPHIRE model only generated the 7-day-after-initiator results. The other two sets of results were generated manually by observing that late containment failure (1-REL-LCF) occurs between 36 and 60 hours after SAMG entry, while scrubbed late containment failure (1-REL-LCF-SC) and basemat melt-through (1-REL-BMT) occur between 60 hours after SAMG entry and 7 days after the initiator. **Figure 2-15** shows the same information in graphical form.

As can be seen, most of the release frequency is attributable to intact containment and late containment failure states, with contribution moving from the former to the latter as the accident truncation time is extended. Meanwhile, combustion-induced containment failure in the timeframe well after vessel breach but prior to late containment failure accounts for ~16 percent of the release frequency. The remaining modes of containment failure or bypass are small contributors to release frequency.

Table 2-17: Release Category (RC) Contributions - Tabular

RC Bin	Shorthand description	Individual contribution to overall release frequency		
		36 hrs after SAMG entry	60 hrs after SAMG entry	7 days after initiator
V-F	ISLOCA, aux. bldg. failed	0.1%	0.1%	0.1%
V-F-SC	ISLOCA, aux. bldg. failed, break submerged	0.3%	0.3%	0.3%
V	ISLOCA, aux. bldg. intact	<0.1%	<0.1%	<0.1%
Total ISLOCA	(All ISLOCAs)	0.5%	0.5%	0.5%
SGTR-O	SGTR, direct relief path	<0.1%	<0.1%	<0.1%
SGTR-O-SC	SGTR, direct relief path, break submerged	0.3%	0.3%	0.3%
SGTR-C	SGTR, no direct relief path	0.1%	0.1%	0.1%
ISGTR	Severe accident-induced SGTR	0.8%	0.8%	0.8%
Total SGTR	(All SGTRs)	1.2%	1.2%	1.2%
CIF	Containment isolation failure	0.1%	0.1%	0.1%
CIF-SC[1]	Containment isolation failure, scrubbed	<0.1%	<0.1%	<0.1%
Total CIF	(All containment isolation failures)	0.1%	0.1%	0.1%
ECF	Early containment failure	<0.1%	<0.1%	<0.1%
Total ECF	(All early containment failure)	<0.1%	<0.1%	<0.1%
ICF-BURN	Intermediate containment failure – combustion	12.4%	12.4%	12.4%
ICF-BURN-SC	Inter. contain. failure – combustion - scrubbed	3.5%	3.5%	3.5%
Total ICF	(All intermediate containment failure)	15.9%	15.9%	15.9%
LCF	Late containment failure	0.0%	41.8%	41.8%
LCF-SC	Late containment failure, scrubbed	0.0%	0.0%	4.7%
Total LCF	(All late containment failure)	0.0%	41.8%	46.5%
BMT	Basemat melt-through	0.0%	0.0%	1.2%
NOCF	Containment intact	82.3%	40.5%	34.6%
Total BMT/NOCF	(All BMT / intact containment)	82.3%	40.5%	35.8%
Total		100.0%	100.0%	100.0%

[1] CET sequence number 74 should have been assigned to the 1-REL-CIF-SC release category rather than the 1-REL-CIF release category, since in-vessel recovery occurs. The change would raise the 1-REL-CIF-SC frequency to 5×10^{-9}/rcy, while reducing the 1-REL-CIF frequency to ~5.8×10^{-8}/rcy (the 1-CET-074 sequence frequency is ~5×10^{-9}/rcy), discounting any minimization that would occur. This would not affect the release categories' percent contribution (<0.1 percent and 0.1 percent, respectively).

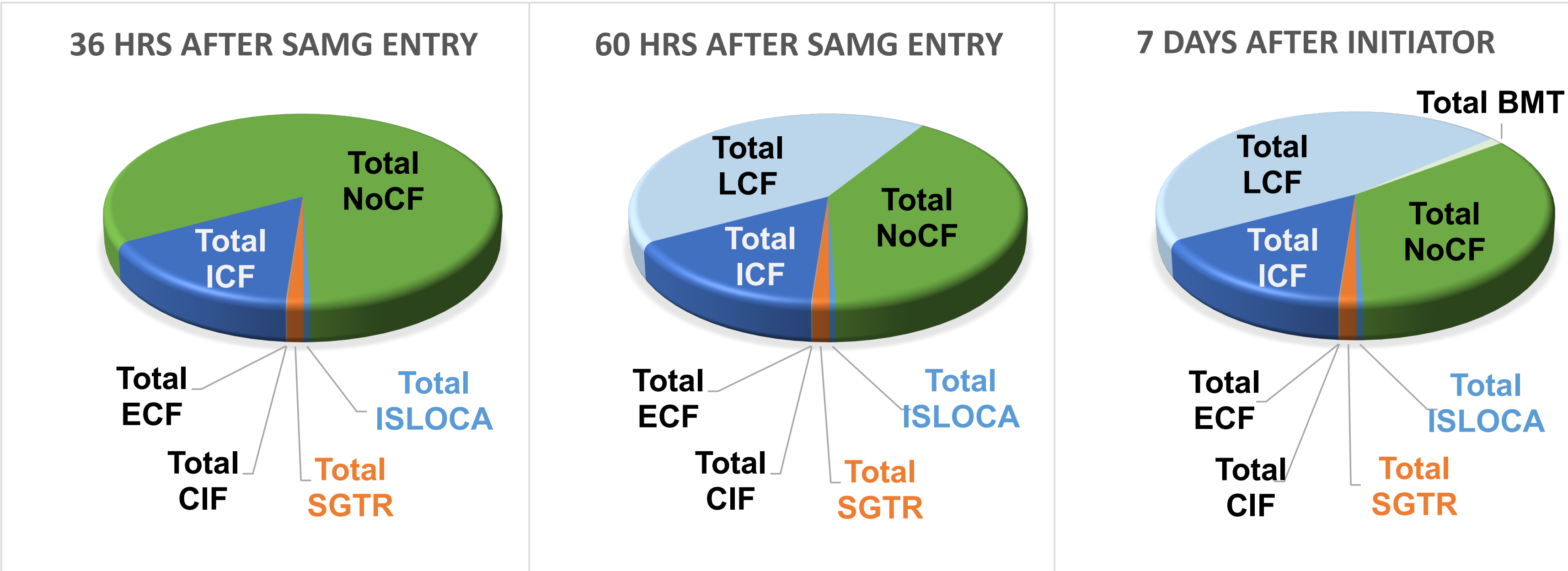

Figure 2-14: Release Category Contributions - Graphical

2.4.7. Step 7 – Uncertainty characterization

Level 1 PRA revolves around a Boolean-based logic model, relying heavily on data-based component failure probabilities and the use of success criteria to represent offline deterministic simulations. The Level 3 portion of a Level 3 PRA involves deterministic offsite consequence simulations, with probabilistic sampling to generate the needed conditional probability outputs. The Level 2 PRA is the transition point between these two methodologies and is a mix of logic modeling and deterministic simulation. This situation leads to differences in terms of capturing uncertainty between a fundamentally Boolean model and a fundamentally simulation-based model. A Boolean model makes it much more straight-forward (and less costly) to capture uncertainty, but is arguably poorly suited for quantifying accident progression modeling uncertainty (and this is the reason that contemporary Level 1 PRA models rely on consensus models or sensitivity analyses to consider uncertainty in the success criteria or other deterministic inputs [e.g., seal LOCA modeling]).

It is also important to note that there are fundamentally two related but distinct types of uncertainty in Level 2 PRA. The first is the uncertainty associated with the probabilistic model that manifests itself in the uncertainty in release frequencies (e.g., where the frequency of some release category is 1×10^{-5}/rcy with a 5^{th} and 95^{th} percentile of 2×10^{-7}/rcy and 8×10^{-5}/rcy, respectively). The second is the uncertainty in the deterministic simulations that manifests itself in the source term characteristics (e.g., where the start of release for some release category is 8 hours, with a 5^{th} and 95^{th} percentile of 11 hours and 5 hours, respectively). To complicate matters, uncertainty associated with the deterministic simulation can be represented as the uncertainty in the probabilistic model. For example, 10 closely-related simulations were run, and a hydrogen deflagration occurs in 1; therefore, the split fraction for hydrogen deflagration will be 0.1. Alternatively, an example of probabilistic model uncertainty being represented in the deterministic model is the assumption that the relief valve sticks open after 70 cycles because that is the median of the probability density function for the valve's failure probability.

Even so, the L3PRA project attempts to address uncertainty in a format analogous to its treatment in state-of-practice Level 1 PRAs. **Figure 2-16** depicts the basics of this approach schematically. Distributions are assigned (based on practitioner judgment) to basic events in the PRA model, resulting in release category frequency uncertainty distributions. Sensitivity analyses are performed in MELCOR, arriving at the nominal source term plus alternative possibilities. This blending of epistemic and aleatory uncertainties is not ideal, but provides a practical path forward to demonstrating the general precision (or lack thereof) in the PRA results.

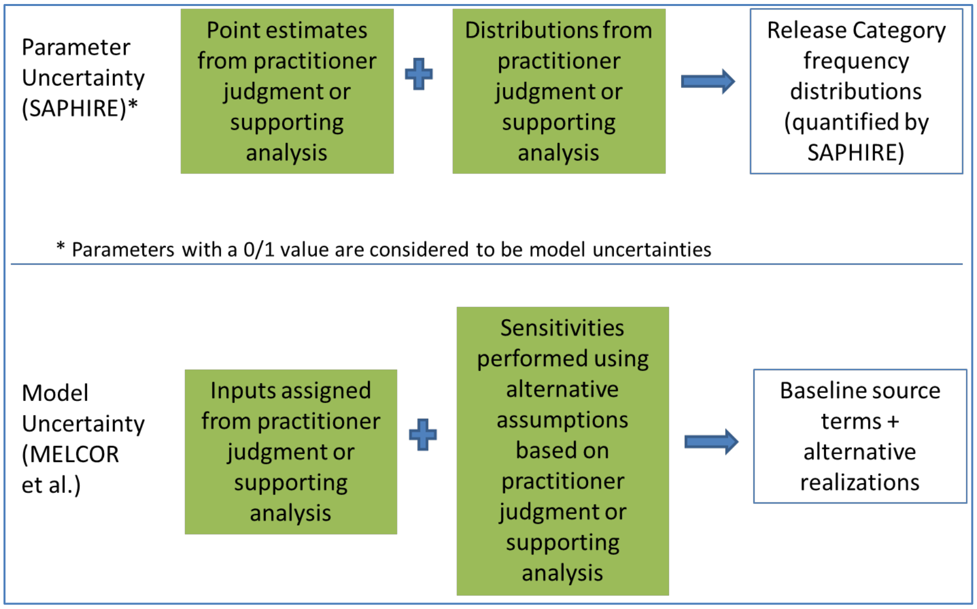

Figure 2-15: Level 2 PRA Scheme for Treatment of Uncertainty

The application of this approach for the Level 2 PRA (i.e., identification of parameter and model uncertainties, characterization of those uncertainties, propagation of parameter uncertainty, and sensitivity analysis of selected model uncertainties) can be found in Appendix C. As discussed in that document, propagation of parameter uncertainties has some limitations, as not all modeling assumptions and dependencies are represented by parameter uncertainties. Nevertheless, and as expected, the parameter uncertainty propagation performed is useful for demonstrating relative similarity between the mean, median, and point estimates, while showing large spreads between the 5th and 95th percentiles.

Appendix C also provides a number of sensitivity analyses using either the SAPHIRE model, MELCOR analyses, or hand calculations to investigate the effects that key modeling assumptions have on the results. Examples of sensitivity analyses include:

- PDS binning assumptions
- Not considering subsequent containment failure modeling in CET-001 (i.e., assigning that sequence directly to the NOCF release category without considering the lower-probability phenomena analyzed for most other sequences)
- Systematically changing Level 2 HEPs upward or downward
- How long-term recovery might affect LERF, large release frequency (LRF), and conditional containment failure probability (CCFP), and closely related to this, the impact of simulation end-time on the same
- Exploring various modeling assumptions associated with modeling containment combustion events
- The impact of the dominant containment isolation failure event

There were also several MELCOR sensitivity analyses that were performed in developing the source terms. These MELCOR sensitivity analyses are synopsized later in this report (Section 2.5.3), because they relate to the uncertainty in the source terms discussed in Section 2.5.

2.5. Radiological Source Term Analysis

The Radiological Source Term Analysis consists of three interrelated steps:

1. Definition of the release category binning logic
2. Development of source terms for the various release categories
3. Consideration of uncertainties in the source term development

The objective of the first step is to develop the logic that will be used for assigning the CET end-states to release categories for use in the Level 3 PRA. The objective of the second step is to take the MELCOR source terms that are a natural outcome of the PDS representative sequence analyses and specify which source terms will be designated as the representative source term for each release category. The objective of the third step is to re-visit the issue of uncertainty in the context of source term uncertainties.

2.5.1. Step 1 – Definition of the release category binning logic

As with the PDS binning, the release category binning steps represent another phase in the analysis where the explosion of sequences needs to be pared back into a manageable set of end states. Rather than handling this as a separate logic model, this consolidation of sequences is done directly in the structure and end-state definitions of the CET.

The release category binning scheme uses functional sequence characterizations related to:

- the status of the core
- the status of containment and (if applicable) surrounding structures, which in turn governs the release path for airborne releases to the environment[20]
- fission product scrubbing, either from containment systems or an overlying pool
- timing of release

The breakdown of the release categories is shown in **Table 2-18**. Meanwhile, **Table 2-19** provides a description of each release category.

Table 2-18: Mapping of Release Categories to Accident Characteristics

	Unscrubbed	Scrubbed
Containment intact	NOCF[1]	
Containment bypass – ISLOCA	V[1]	
	V-F	V-F-SC
Containment bypass – SGTR	SGTR-C[1], ISGTR[1]	
	SGTR-O	SGTR-O-SC
Containment not isolated	CIF	CIF-SC
Containment fails at or before vessel breach	ECF[1]	
Containment fails hours after vessel breach due to a global deflagration or detonation	ICF-BURN	ICF-BURN-SC
Containment fails late due to over-pressure	LCF	LCF-SC

[20] The focus of this model is airborne radiological releases only. When relevant, surface aqueous releases are noted, but only airborne releases are passed to the offsite consequence analysis.

Table 2-18: Mapping of Release Categories to Accident Characteristics

	Unscrubbed	Scrubbed
Containment fails late due to BMT	BMT[1]	

[1] These release categories do not differentiate with respect to scrubbing because either (a) the associated source terms are already very small without scrubbing or (b) the underlying phenomena occur so quickly that there is insufficient time for natural scrubbing phenomena or operator actions to have a large impact.

Table 2-19: Description of Release Categories

Name	Description
1-REL-NOCF	Containment is not bypassed or failed, and radiological release to the environment occurs via design-basis containment leakage only. This release may or may not benefit from any aerosol scrubbing.
1-REL-ECF	The containment fails before or around the time of vessel breach due to an energetic event. This release may or may not benefit from any aerosol scrubbing.
1-REL-ICF-BURN	The containment fails hours after vessel breach due to a global deflagration or detonation. Releases to the environment are not mitigated significantly by sprays or water pools.
1-REL-ICF-BURN-SC	The containment fails hours after vessel breach due to a global deflagration or detonation. Releases to the environment benefit from scrubbing.
1-REL-LCF	The containment fails tens of hours after the time of vessel breach due to long-term quasi-static overpressure. Releases to the environment are not mitigated significantly by sprays or water pools.
1-REL-LCF-SC	The containment fails tens of hours after the time of vessel breach due to long-term quasi-static overpressure. Releases to the environment are mitigated by sprays and/or water pools.
1-REL-BMT	The containment eventually fails due to basemat ablation due to sustained core-concrete interaction. Only the airborne component of release to the environment (which stems from normal containment leakage while the containment is pressurized) is modeled.
1-REL-CIF	Release from the containment to the environment occurs via a containment penetration that fails to be isolated by the containment isolation system, or a pre-existing leakage path. The release is unmitigated.
1-REL-CIF-SC	Release from the containment to the environment occurs via a containment penetration that fails to be isolated by the containment isolation system, or a pre-existing leakage path. The release is mitigated.
1-REL-SGTR-C	Release from the RCS to the environment occurs via ruptured steam generator tube(s), where the rupture occurs prior to core damage. Atmospheric relief valves (ARVs) and main steam relief valves remain predominantly closed.
1-REL-SGTR-O	Release from the RCS to the environment occurs via one or more ruptured SG tubes, where the rupture occurs prior to core damage. The release is not mitigated by water above the break point on the secondary side of the affected SG. One or more secondary-side relief valves are kept open during release as a deliberate action or fail in the open position.
1-REL-SGTR-O-SC	Release from the RCS to the environment occurs via one or more ruptured SG tubes, where the rupture occurs prior to core damage. The release is mitigated by water above the break point on the secondary side of the affected SG. One or more secondary-side relief valves are kept open during release as a deliberate action or fail in the open position.

Table 2-19: Description of Release Categories

Name	Description
1-REL-ISGTR[21]	Release to the environment occurs via a thermally-induced rupture of one or more steam generator tubes after the time of core damage.
1-REL-V	Release occurs from the RCS to the auxiliary building via interfacing systems LOCA. The break point may or may be not submerged. The auxiliary building remains intact.
1-REL-V-F	Release occurs from the RCS to the auxiliary building via interfacing systems LOCA. The break point was not submerged. The auxiliary building fails.
1-REL-V-F-SC	Release occurs from the RCS to the auxiliary building via interfacing systems LOCA. The break point was submerged. The auxiliary building fails.

2.5.2. Step 2 – Development of source terms for the various release categories

This step provides the assignment of source terms to release categories. It starts by providing the criteria for the selection process. It then provides the preliminary emergency action level declarations for the various sequences. Following this, the source term mapping is provided, followed by additional subsections that provide the rationale for this mapping in cases where multiple candidate source terms exist. Next, information about the aerosol size distribution and the release points to the environment is provided.

2.5.2.1. *Release category source term criteria*

Criteria for assigning a calculation's source term to a release category include:

- The calculation boundary conditions and results must be consistent with the release category description (e.g., candidates for 1-REL-LCF-SC must result in predicted late containment failure and with scrubbing of the release).
- Among possible candidates, the chosen calculation should ideally be free from known, significant modeling inaccuracies (e.g., see pressurizer relief tank [PRT] dryout discussion below for the late containment failure source term selection).
- Among possible candidates, the chosen calculation's predicted accident progression should ideally be consistent with the assumptions of the Level 2 logic model.
- Among possible candidates, the chosen calculation should ideally look more reflective in its particulars of the highest contributing cutsets of the release category.
- All the above factors being equal, the candidate chosen should consider the balance of timing and magnitude to select a source term that conservatively bounds the range of outcomes for that release category.

2.5.2.2. *Emergency action level declarations*

Timely monitoring and declaration of EALs is assumed, and the basis for this is provided in Section 7 of Appendix D of this report. During the development of the Level 2 PRA, preliminary EAL timings were developed for the eight representative sequences during the screening HRA. These preliminary estimates (provided in **Table 2-20**) were confirmed as part of the Level 3 PRA (consequence analysis) activities. Note, recovery cases have the same EAL timings

[21] In this report and elsewhere, the acronyms ISGTR and TI-SGTR are used interchangeably. Meanwhile, the term C-SGTR is used to more broadly capture both TI-SGTRs and the PI-SGTRs considered in the Level 1 PRA.

because the recovery actions all follow SAMG entrance and all representative sequences were found to have General Emergency declarations prior to SAMG entrance.

Table 2-20: EAL Classifications

Representative Sequence No.	Alert Declaration* (hh:mm)	Site Area Emergency Declaration* (hh:mm)	General Emergency Declaration* (hh:mm)
1	~0	0.25	3
1A	~0	0.25	3
1A1	~0	0.25	3
1A2	~0	0.25	3
1B	~0	0.25	3
1B1	0	0.25	2.5
1B2	0	0.25	3
2	1	7	8
2A	1	7	8
3	0.25	5	8
3A1	0.25	5	8
3A2	0.25	5	8
3A3	0.25	5	8
3A4	0.25	5	8
4	2.25	8	17
5	0.25	0.25	7.5
5A	0.25	0.25	7.5
5B	0.25	0.25	1.25
5C	0.25	0.25	7.5
5D	0.25	0.25	1.25
6	2.5	13	13
6A	2.5	13	13
6B	2.5	13	13
6C	2.5	13	13
6D	2.5	13	13
7	~0	0.25	3
7A	~0	0.25	3
8	0.25	38	47
8A	0.25	70	90
8B	0.25	38	47

* Timings are relative to the time of occurrence of the initiating event and are based on plant-specific information that is typically proprietary and not included here.

2.5.2.3. *Mapping source terms to release categories*

Table 2-21 shows (for each accident simulation) the relationship between the declaration of a General Emergency (GE) (preliminary estimate) and illustrative timings in terms of the radiological release, to provide a sense of the timing considerations in selecting a representative source term. As expected, this shows that bypass events (ISLOCAs, SGTRs, and C-SGTRs) have the greatest potential to lead to releases that are closer in time to the GE declaration. These timings also highlight the importance of the Case 7 timings (Case 7 involves an SBO initiator, AFW operation for 4 hours, and containment isolation failure), wherein the EAL timing is driven by SBO conditions rather than accident progression timings (leading to an earlier GE declaration). A simulation similar to Case 7 in which AFW is never available, would lead to a radiological release more proximate to the GE declaration; however, such a sequence would also have a lower frequency than Case 7 (which already has a low relative frequency). For this reason, some fraction of the containment isolation failure frequency could also notionally be considered early.

Table 2-22 maps the source terms (whose case descriptions were previously provided in **Table 2-4**) to the appropriate release categories. When multiple source terms are available for a given release category, the selection of one as the representative source term is justified in the ensuing sub-sections, which also include graphical representations of the source terms.

To inform the source term selection for the L3PRA project Level 2 model, the preliminary Level 3 consequence analysis results that were available were scrutinized to learn lessons about within-release-category variability and representative source term selection, using all release categories where multiple candidate source terms existed.

To identify instances of large within-release category variability, this activity looked across all reported consequence metrics, and isolated those instances where the representative source term result was more than twice or less than half of the release category's average result (when considering all candidate source terms). Considering the span of ratios across the different consequence metrics, it was identified in the preliminary results that the V-F-SC, late containment failure (LCF), and LCF-SC release categories had higher-than-desired variability. For V-F-SC, the variation between the three possible source terms related directly to the underlying uncertainties in the ISLOCA modeling in general. In particular, the initial break size and the building response to blowdown were significant sources of variability. For LCF, the variability in results was likely caused by the protracted timeline for containment failure of these calculations in conjunction with the associated increased uncertainty in the phenomenological modeling. Also, this release category represented a large fraction of the overall release frequency as well as combustion-induced and non-combustion-induced failures, and thus logically contained more variability. This was part of the motivation for separating the post-vessel breach combustion failures from the late quasi-static overpressure failures in the release categorization scheme, and this change directly addressed part of the previously-identified variability. For LCF-SC, the large degree of variability was attributed to the same reasons as described above for LCF, but was of less importance, since the consequence results were generally significantly lower than the LCF counterparts.

To identify instances where the representative source term may under-represent the release category significance, the representative source term consequence results were compared to the peak of all source terms within the same release category. If the representative source term was more than a factor of two below the peak for multiple consequence metrics, this release category was further scrutinized; this was the case for SGTR-C and LCF-SC. For SGTR-C, the problem was caused because the source term that determined the peak and drove up the average was deliberately not selected, since the bulk of the (late) release was due to combustion-induced containment failure rather than SGTR. This should have resulted in a modification to the logic model to capture the probability of late combustions failing containment during SGTR accident sequences, and then assigned that portion of the frequency to the LCF release category. However, the modification was not made because the frequency of SGTR-C was four orders of magnitude lower than LCF. For LCF-SC, Case 2R2 was chosen over other possible representative source terms because its releases were larger prior to containment failure. There were variations to Case 3 that were contemplated to result in cesium releases that were significantly higher following containment failure, but those MELCOR calculations were not completed due to issues with running the code for the specific modeling conditions. The judgement was made that the post-containment-failure results were more speculative, and Case 2R2 was retained as the chosen representative source term. In light of the preliminary Level 3 consequence analysis results, which show that the total magnitude of the release is somewhat more important than the timing of release, the higher magnitude post-containment-failure releases (i.e., Case 3R2) may have warranted additional consideration in chosing the representative source term.

All the above supports that the process used to select representative source terms is reasonable, with the main lesson learned being that total magnitude is somewhat more important than timing for the conditions present in the results. This insight is carried forward in the binning of the present results.

Table 2-21: Comparison of Preliminary GE Time to Release Timings

	Time (hr)				
Represent. Sequence #	**Time of reactor trip**	**Preliminary GE Declaration**	**PCT > 1204 C**	**Xe release > 10%**	**I release > 1%**
1	0	3	139	Never	Never
1A	0	3	16	75	Never
1A1	0	3	16	76	Never
1A2	0	3	16	28	33
1B	0	3	3.9	55	158
1B1	0	2.5	3.0	56	146
1B2	0	3	3.3	64	Never
2	0.1	8	15	99	Never
2R1	0.1	8	15	Never	Never
2R2	0.1	8	15	128	Never
2A	0.1	8	15	22	22
3	0	8	11	62	140
3A1	0	8	11	58	151
3A2	0	8	11	11	11
3A3	0	8	11	11	11
3A4	0	8	11	71	156
4	2	17	15	99	Never
5	~0	7.5	9.5	10	Never

Table 2-21: Comparison of Preliminary GE Time to Release Timings

	Time (hr)				
Represent. Sequence #	Time of reactor trip	Preliminary GE Declaration	PCT > 1204 C	Xe release > 10%	I release > 1%
5A	~0	7.5	9.5	10	Never
5B	~0	1.25	2.8	3.5	3.2
5C	~0	7.5	9.5	10	Never
5D	~0	1.25	2.8	3.5	3.2
6	0	13	15	Never	Never
6R1	0	13	15	Never	Never
6A	0	13	15	Never	Never
6B	0	13	15	Never	Never
6C	0	13	15	40	Never
6D	0	13	15	74	Never
7	0	3	16	21	18
7A	0	3	16	21	18
8	0.25	47	50	52	52
8R1	0.25	47	50	Never	Never
8R2	0.25	47	50	52	52
8A	0.25	90	96	97	Never
8B	0.25	47	50	51	51
8BR1	0.25	47	50	51	Never

Table 2-22: Mapping of Source Terms to Release Categories Without Truncation

Release Category	Candidate Source Terms	Chosen Source Term
1-REL-V-F	5D	5D
1-REL-V-F-SC	5B, 5C	5B
1-REL-V	5, 5A	5
1-REL-SGTR-O	8B	8B
1-REL-SGTR-O-SC	8BR1	8BR1
1-REL-SGTR-C	8, 8A, 8R1, 8R2	8
1-REL-ISGTR (a.k.a., C-SGTR)	3A2, 3A3	3A2
1-REL-CIF	7	7
1-REL-CIF-SC	7A	7A
1-REL-ECF	2A, 6C, 6D	2A
1-REL-LCF	1A, 1A1[ii], 1B, 1B1, 1B2, 2, 3, 3A1, 3A4, 4	1B
1-REL-LCF-SC	2R2	2R2
1-REL-ICF-BURN	1A2[i]	1A2[i]
1-REL-ICF-BURN-SC	None - 1A2 truncated at the time of containment failure can be used as a surrogate	1A2 truncated at the time of containment failure (~28 hrs)

Table 2-22: Mapping of Source Terms to Release Categories Without Truncation

Release Category	Candidate Source Terms	Chosen Source Term
1-REL-BMT	6, 6A, 6B[ii]	6
1-REL-NOCF	1, 2R1, 6R1; *BMT source terms could be used as surrogates*	2R1

[i] This case involves containment failure 7 hours after vessel breach due to a large deflagration.

[ii] The source terms from these calculations should be viewed with caution because the cases non-mechanistically suppress combustion to generate information for the probabilistic treatment of hydrogen.

2.5.2.4. *Selection of ISLOCA (V-F-SC & V) release category representative source terms*

In the MELCOR calculations performed for ISLOCA accident sequences, some did or did not include the user-input assumption that the break would be covered by water. Also, in case of sufficiently large breaks, the auxiliary building was predicted to fail relatively early. Therefore, sufficient combinations exist to assign source terms to the release categories V, V-F, and V-F-SC. (Cases with an intact auxiliary building do not differentiate based on break submergence because the source terms are very low in either case.) The source terms across these different simulations do show the relative trends one would expect: V-F is higher magnitude than V-F-SC, which is higher than V. **Figure 2-17** to **Figure 2-20** show the cesium and iodine releases for the two release categories that have multiple candidate source terms.

MELCOR calculations for sequences 5B and 5C entail auxiliary building failure via overpressure and ventilation system failure, respectively. They generally reflect the two logical extremes of release category V-F-SC, in that 5B involves a large ISLOCA leading to early core damage and immediate auxiliary building failure, while 5C involves a smaller ISLOCA leading to later core damage (and a correspondingly later GE declaration) and delayed auxiliary building failure (occurring after most of the volatile fission products have been retained). As such, 5B has a much larger and earlier source term. The dissection of the probabilistic results demonstrates that early auxiliary building failure is a much larger contributor to 1-REL-V-F and 1-REL-V-F-SC than is late combustion-induced failure. Considering this, Case 5B was selected as the representative source term, but it was acknowledged that it is a somewhat conservative representation of the overall release category.

Regarding release category V, the source terms for sequences 5 and 5A are very similar and, in part on this basis, no differentiation is made between submerged and non-submerged breaks when the auxiliary building remained intact. Sequence 5 was selected as the representative source term because it is more conservative in that it has higher iodine releases.[22]

It should be noted that none of these calculations model the potential effects of fission product retention due to turbulent deposition in the RHR piping.

[22] Higher iodine releases could theoretically lead to surpassing thresholds that trigger additional protective actions (and therefore lower exposures). Therefore, Case 5 is not inherently conservative for the full range of Level 3 PRA outputs.

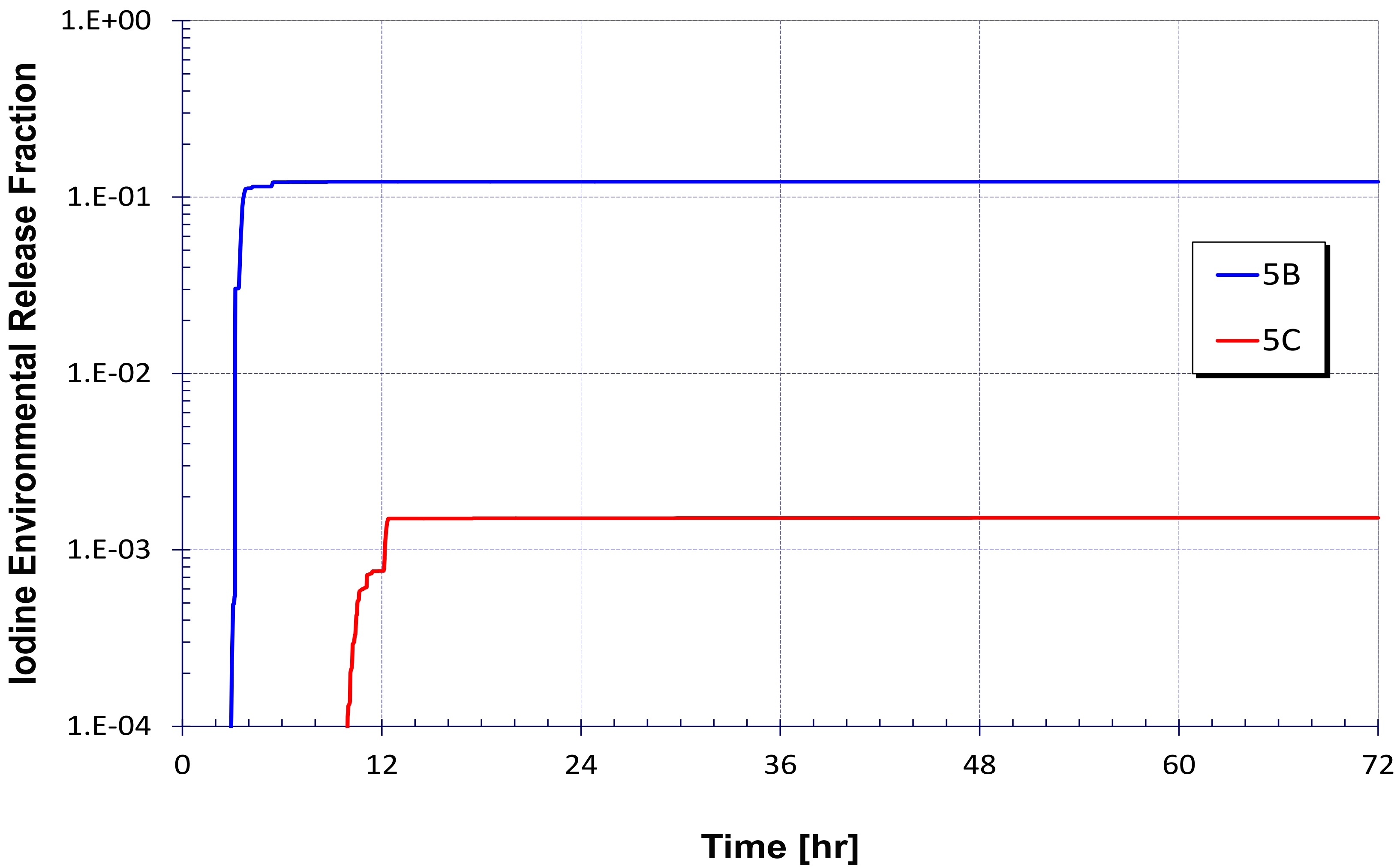

Figure 2-16: Iodine Releases for RC V-F-SC

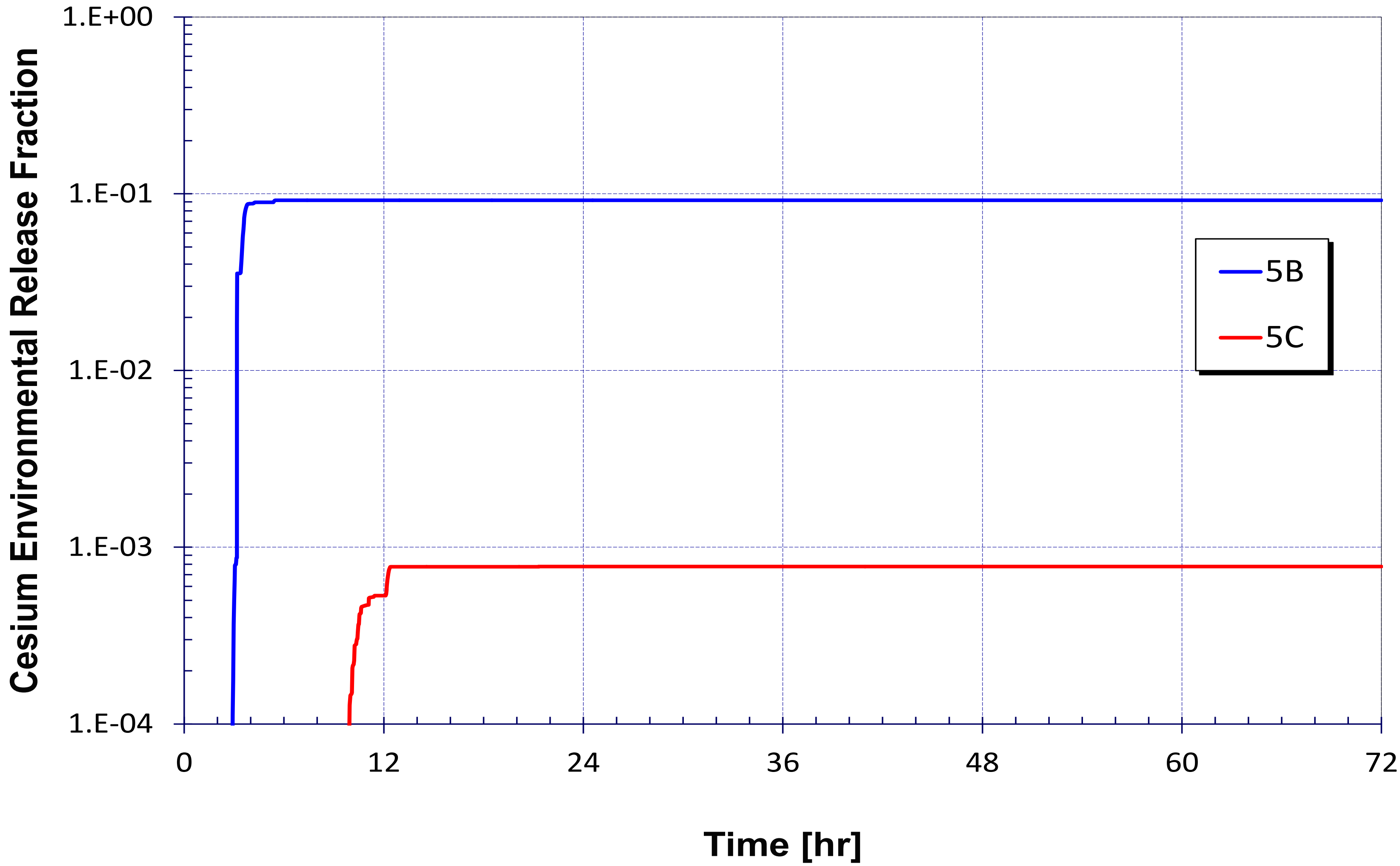

Figure 2-17: Cesium Releases for RC V-F-SC

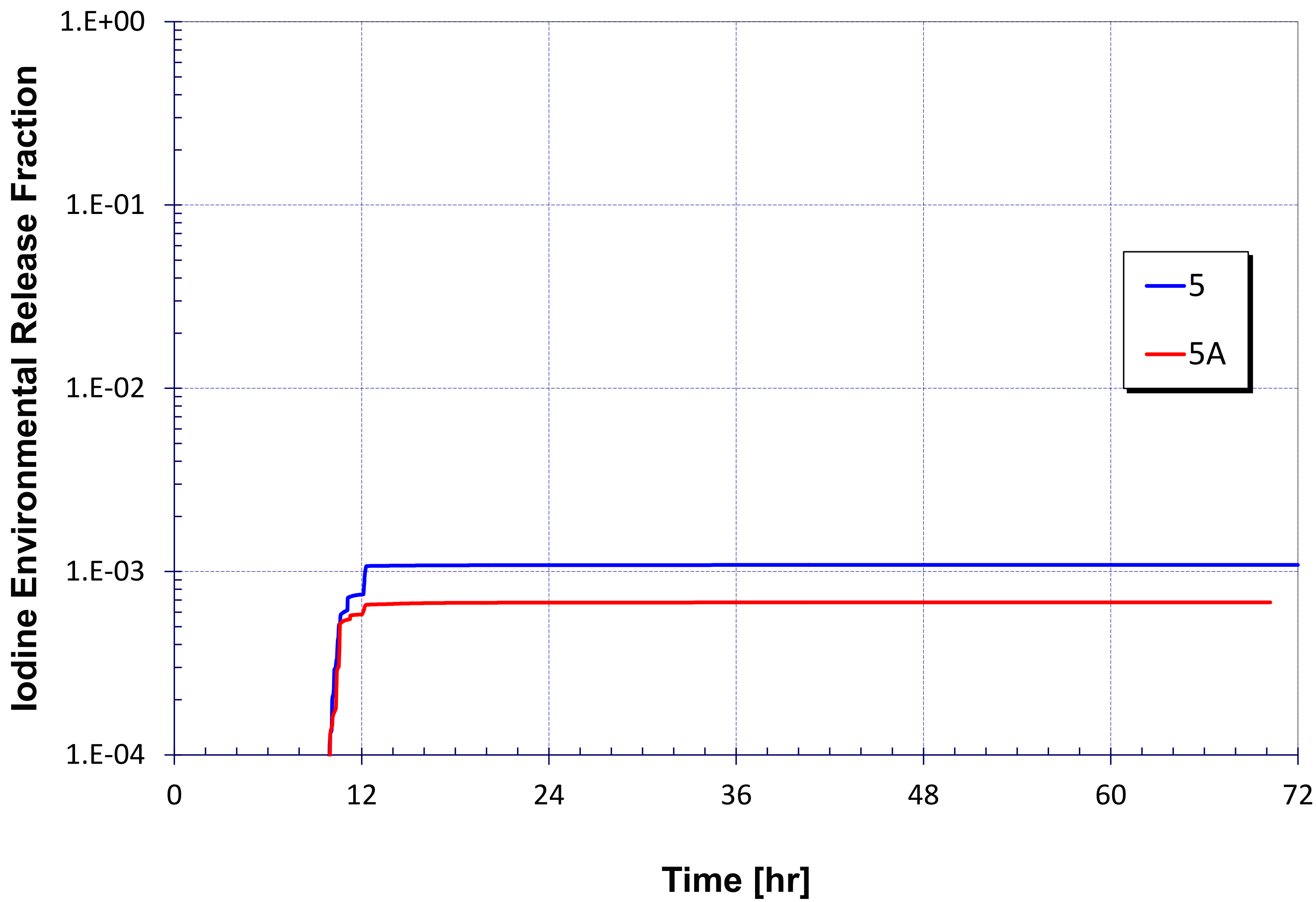

Figure 2-18: Iodine Releases for RC V

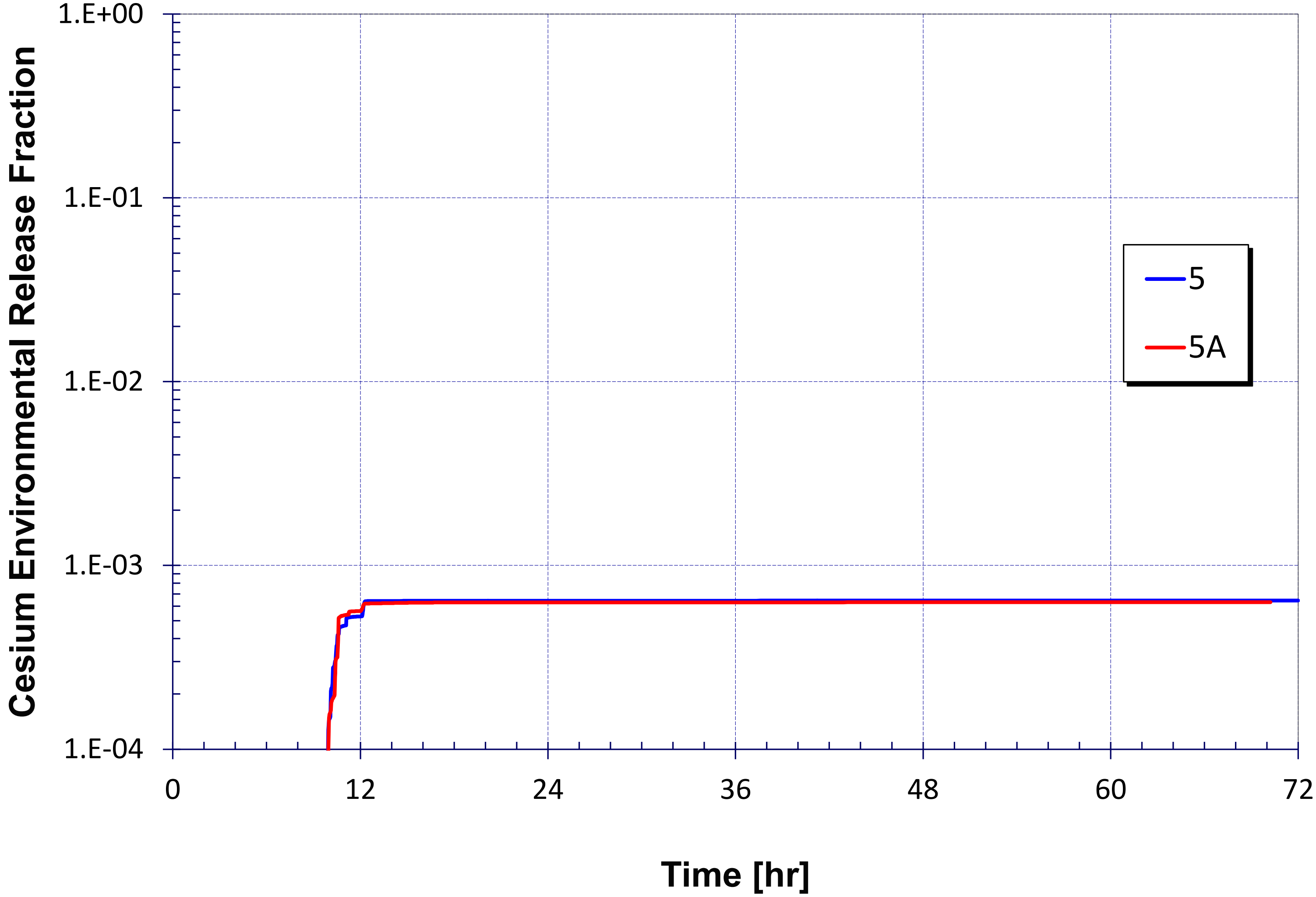

Figure 2-19: Cesium Releases for RC V

2.5.2.5. *Selection of release category SGTR-C representative source term*

Simulations for sequences 8, 8A, 8R1, and 8R2 all correspond to the SGTR-C release category. As shown in **Figure 2-21** and **Figure 2-22**, simulations 8 and 8R2 produced effectively the same source term (the recovery occurs too late to be of significance). Simulation 8A had similar release magnitudes but is significantly offset in time owing to longer auxiliary feedwater availability; however, there is a corresponding offset in time for the GE declaration (see **Table 2-20**). Simulation 8R1, on the other hand, has a dramatically lower source term owing to the effective early recovery action (this simulation is effectively an in-vessel recovery, but in-vessel recovery for bypass events is not considered in the modeling[23]). Simulation 8 (and therefore effectively 8R2 as well) was chosen as the representative source term, because it is the most conservative.

[23] In-vessel recovery for SGTRs (and ISLOCAs) are not treated in the CET modeling. They are treated for containment isolation failures, so one can get a sense for their relevance by looking at the results that show an ~8 percent contribution to 1-REL-CIF from in-vessel recovery cases. This suggests that in-vessel recovery may have a noticeable, but not significant, contribution to SGTR release categories (were it to have been modeled).

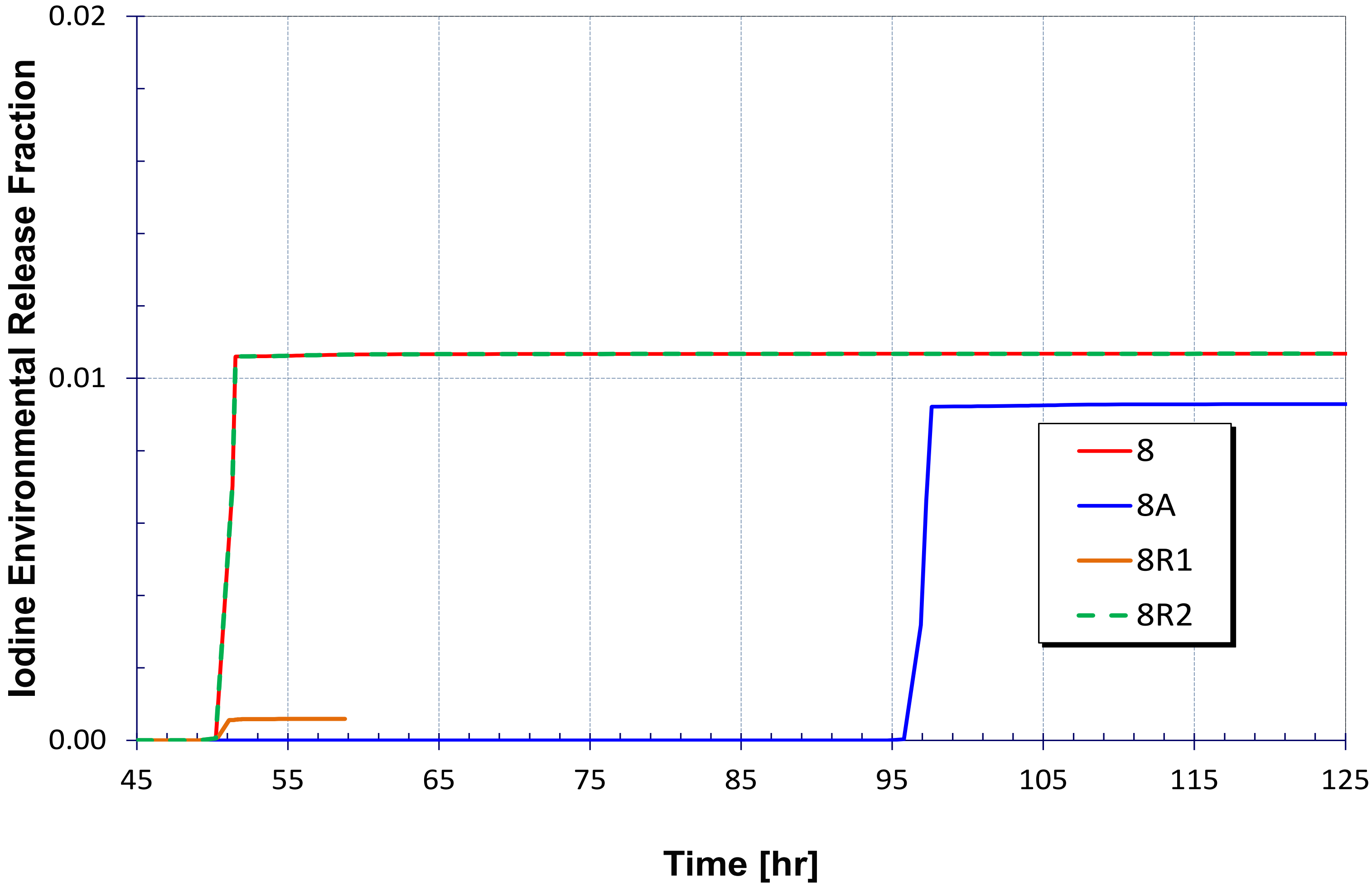

Figure 2-20: Iodine Releases for RC SGTR-C

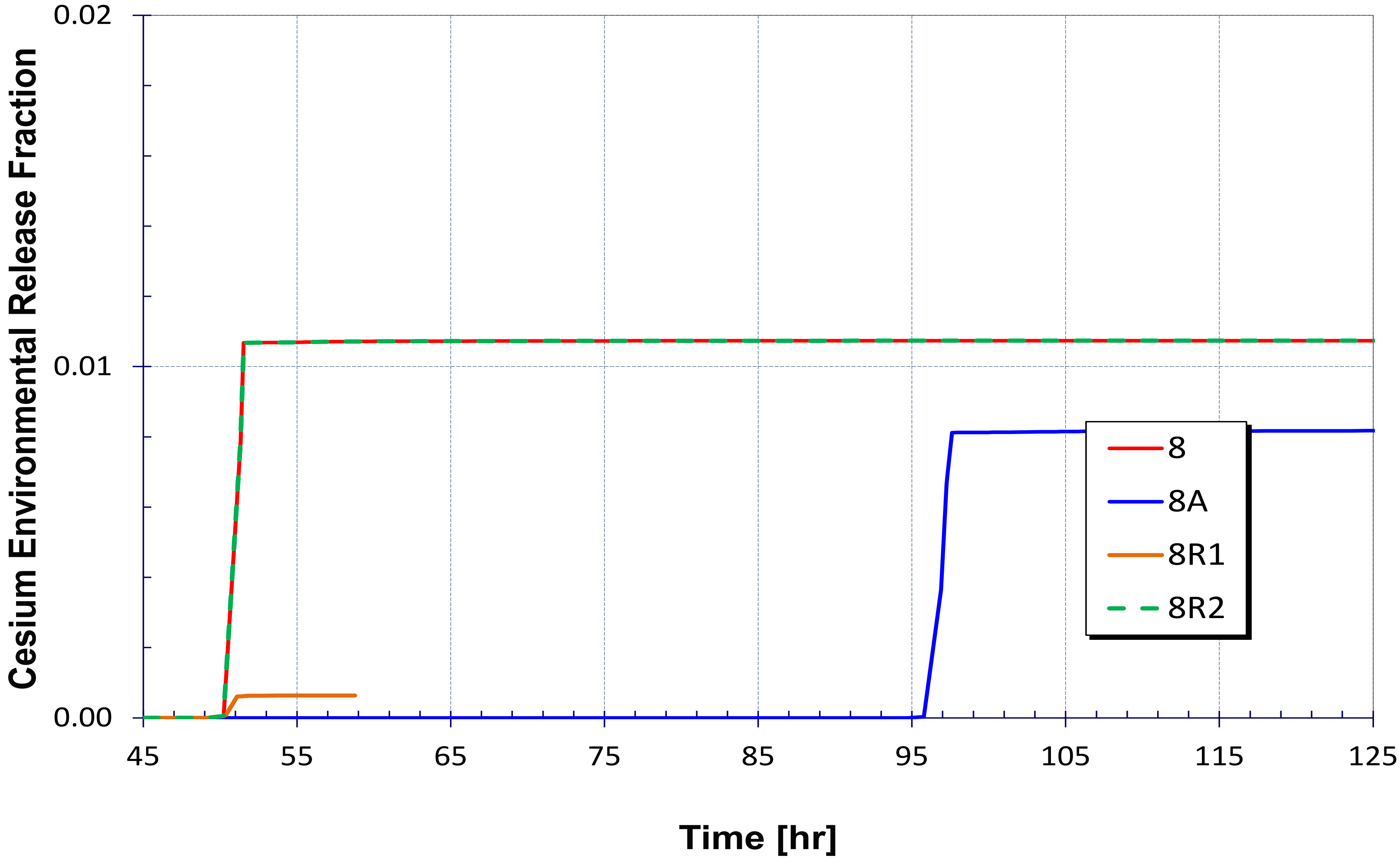

Figure 2-21: Cesium Releases for RC SGTR-C

2.5.2.6. *Selection of release category ISGTR (a.k.a., C-SGTR) representative source term*

For the severe accident induced-SGTR (ISGTR) cases studied using MELCOR, the loss of AFW is assumed to occur at 3 hours. Failure of AFW at time zero would be a lower-likelihood sequence that would result in both an earlier release and an earlier declaration of a General Emergency. However, this situation was not explored in the MELCOR analysis. This decision was based on an AC bus being taken out of service for maintenance in this accident sequence. The Limiting Condition for Operation for restoration is two hours, and the mid-point of this window was chosen for when the initiating event occurs. Battery charging is lost when the AC bus is taken out-of-service (based on an assumption that battery charging has not been aligned from another source), so batteries deplete 4 hours after the bus is taken out-of-service, which is 3 hours after the initiating event.

Two candidate source terms are available for ISGTR under these conditions, specifically source terms for Cases 3A2 and 3A3 (see **Figure 2-23** and **Figure 2-24** below). The environmental releases in Case 3A2 are roughly a factor of three higher than those from Case 3A3, owing to the suppression of subsequent hot leg failure in Case 3A2. These two cases demonstrate (but do not bound) the potential source term variations that can occur. The ISGTR research over the past 20 years indicates that subsequent hot leg failure is much more probable, while the possibility of it not occurring exists if the primary side depressurizes quickly (either due to the contemporaneous failure of a relief valve or multiple tube failures). Conversely, there are many uncertainties associated with ISGTR modeling (as discussed in Sections 4.1 and 4.3 of Appendix C), and the 3A2 and 3A3 results do not consider the possibility of multiple tube ruptures (though these are understood to be much less probable). In summary, a wide range of outcomes is possible (as evidenced by these two cases, and similarly evidenced by the preliminary results of the State-of-the-Art Reactor Consequence Analyses [SOARCA] Surry Uncertainty Analysis [SNL, 2016]). This range highlights some of the limitations in the Level 2 PRA modeling of ISGTR. As a conservative assumption, Case 3A2 was selected for the representative source term because it is the larger of the two source terms.

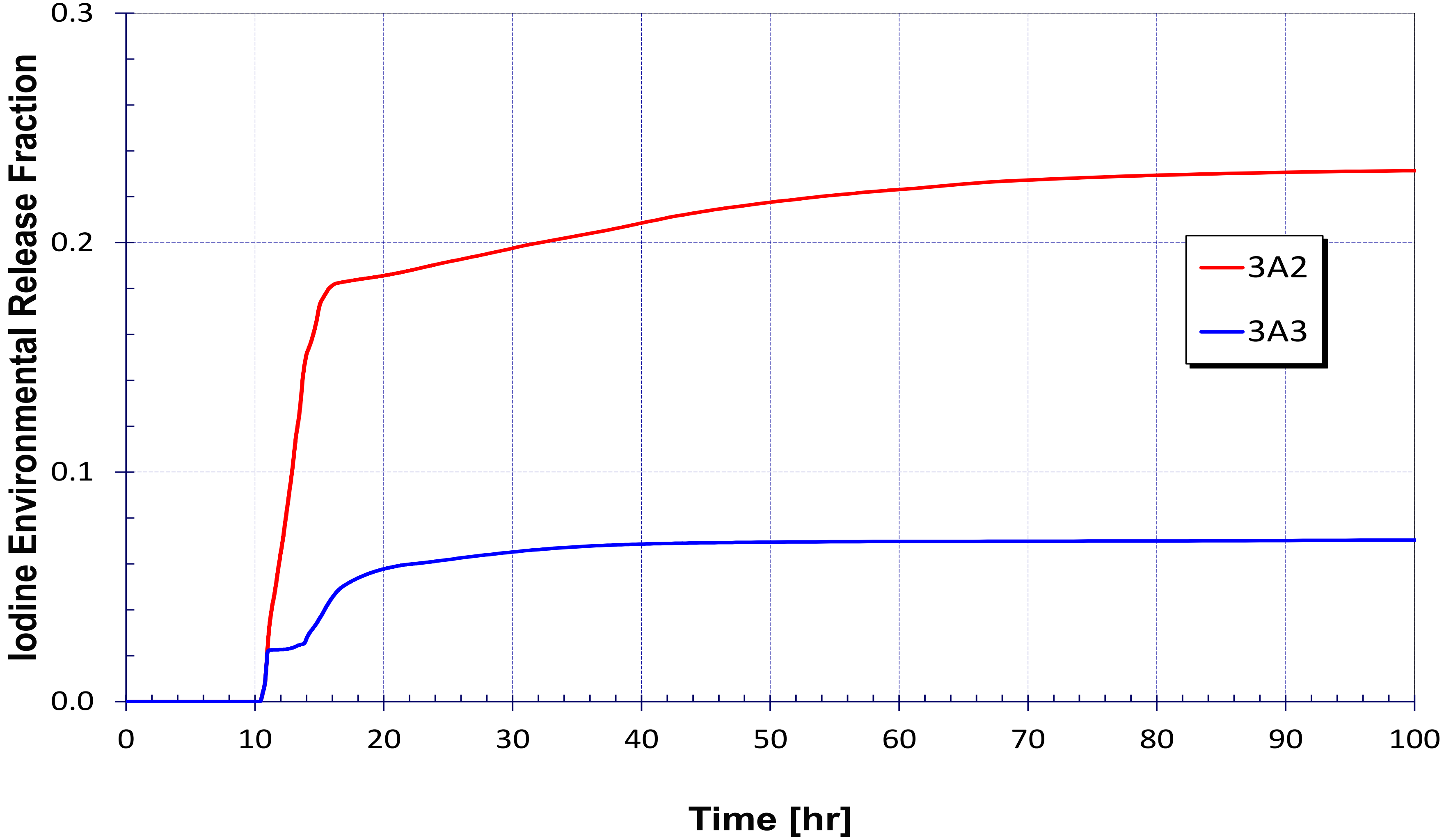

Figure 2-22: Iodine Releases for RC ISGTR (a.k.a., C-SGTR)

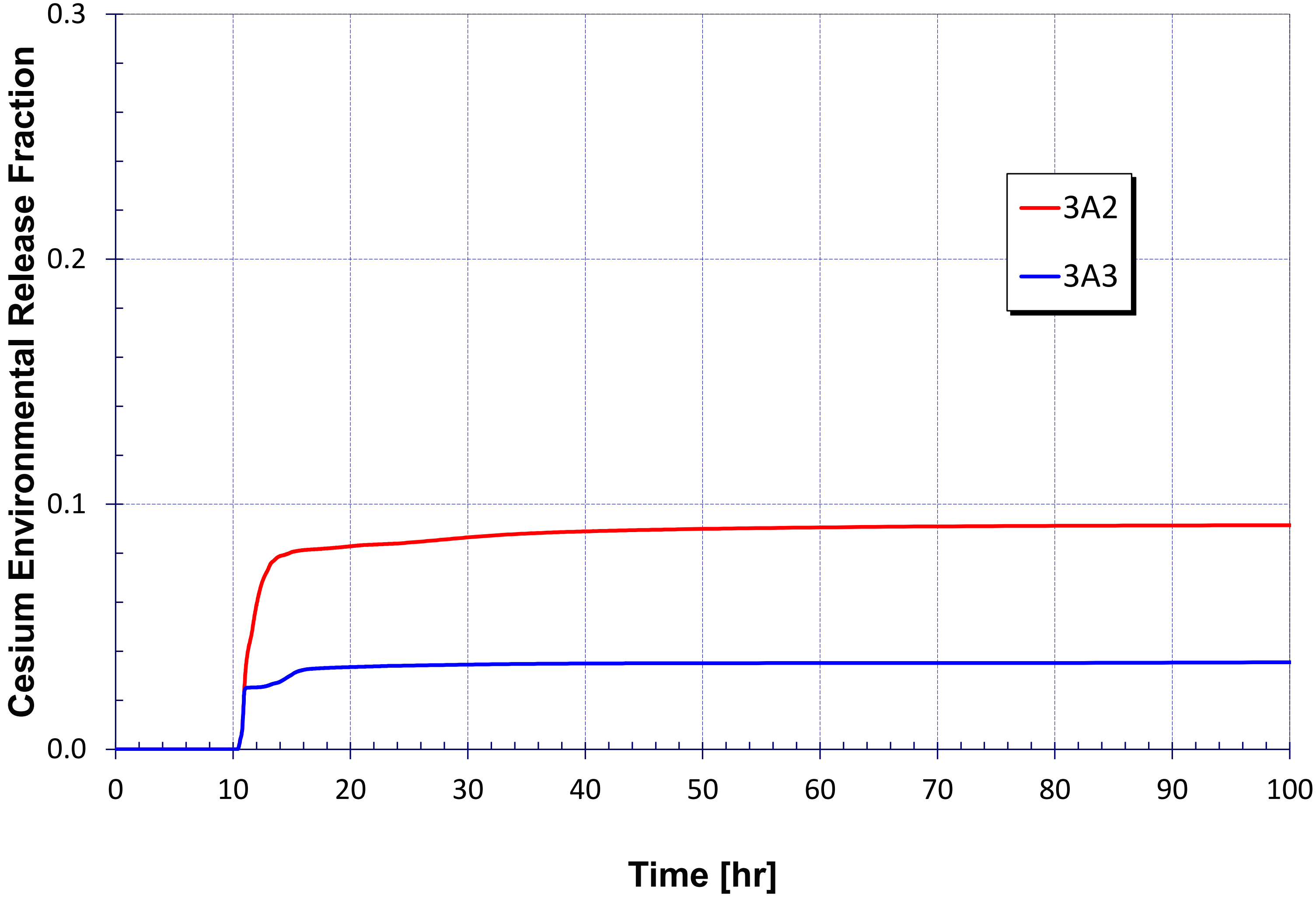

Figure 2-23: Cesium Releases for RC ISGTR (a.k.a., C-SGTR)

2.5.2.7. *Selection of release category ECF representative source term*

Simulations for Cases 2A, 6C, and 6D correspond to the early containment failure (ECF) release category. As shown in **Figure 2-25** and **Figure 2-26**, simulation 2A had a significantly higher source term, owing in large part to the fact that Cases 6C and 6D had containment cooling available. Conversely, Cases 6C and 6D have a shorter time window between the declaration of GE and the onset of radiological releases. Due to the dominance of station blackout accident sequences in the overall results, and the observation from preliminary Level 3 consequence analysis results about release magnitude importance versus timing (see Section 2.5.2.3 for discussion of this point), Case 2A is a more reasonable choice (and is also more conservative in terms of overall releases).

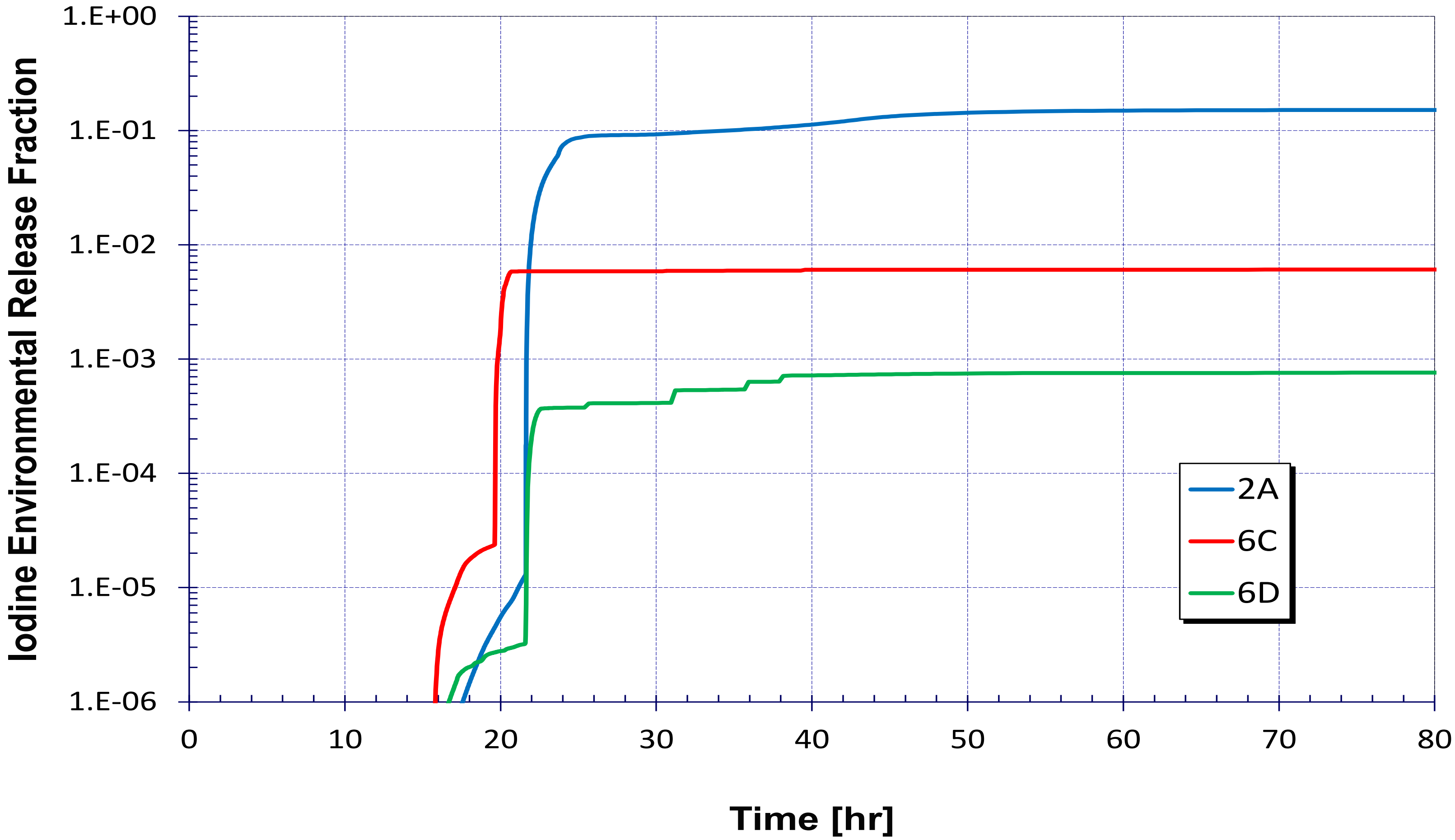

Figure 2-24: Iodine Releases for RC ECF

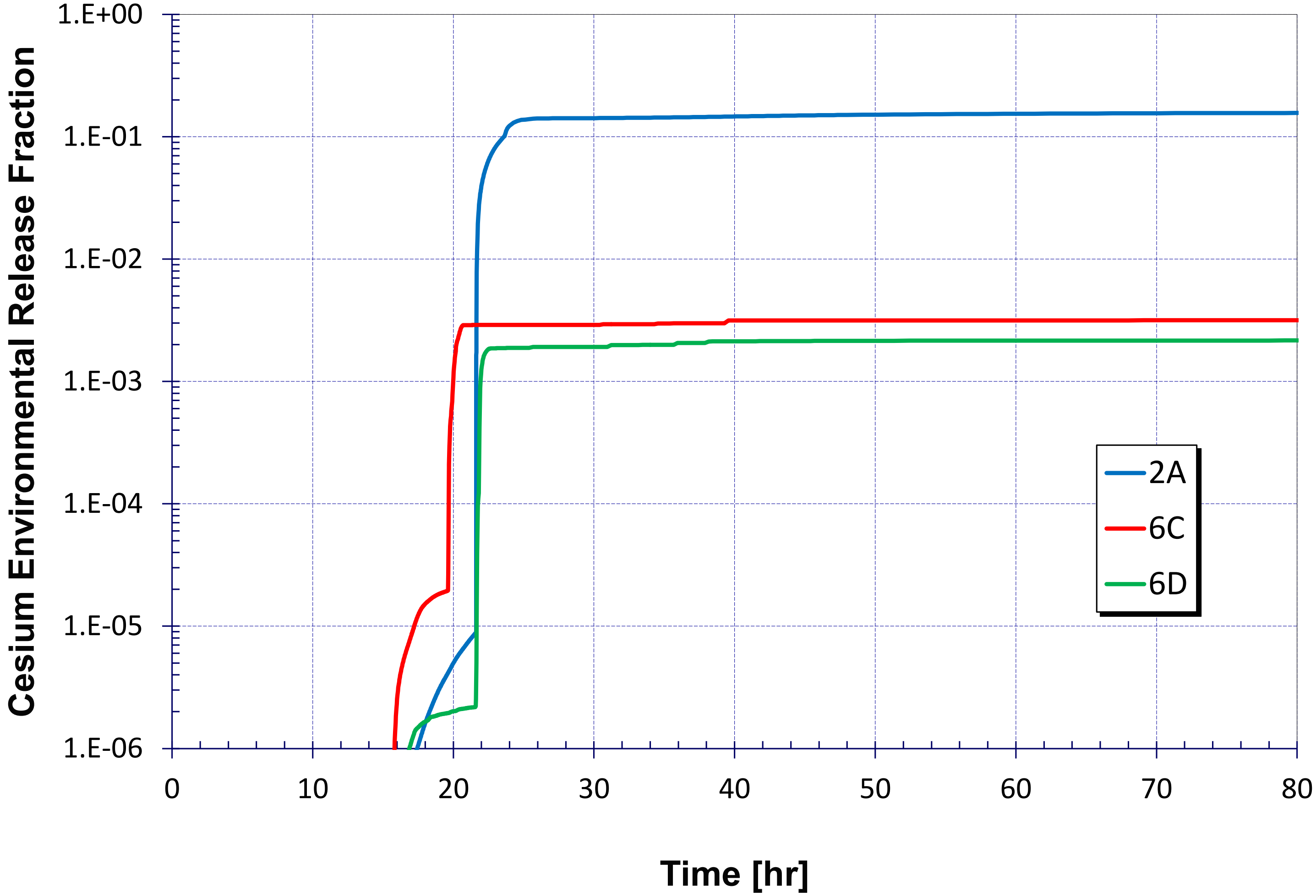

Figure 2-25: Cesium Releases for RC ECF

2.5.2.8. *Selection of release category LCF representative source term*

Figure 2-27 and **Figure 2-28** show the iodine and cesium release signatures for the candidate source terms in LCF. As can be seen, this release category covered a broad range of timings and release magnitudes. Several calculations that were performed resulted in predicted late containment failure with no spray system operation.

Note, the largest cesium release among LCF cases occurs in Case 1B2. This case involves a station blackout with failure of TDAFW at time zero and assumes that the pressurizer PORV sticks open due to excessive cycling. As expected, a large mass of fission product aerosol flows through the stuck-open PORV to the pressurizer relief tank (PRT). As noted in Appendix A of this report, it was later discovered that some simplifications for the PRT modeling led to unphysical behavior. In a sensitivity analysis documented in Sections 4.4 and 8.1.2 of Appendix C, this issue was demonstrated to substantially over-estimate the rate of re-vaporization (and subsequent environmental release) that occurs. For this reason, the 1B2 results were discounted for the purposes of selecting an LCF release category representative source term.

Of the remaining source terms, the 1-series station blackouts are of most overall relevance, given the large contribution of station blackout to CDF and release frequency. In all cases, the declaration of GE precedes the time of significant releases by many tens of hours. Cases 1A and 1B best represent the highest contributing accident progression sequence in that they involve no RCS pressure reduction associated with seal leakage or relief valve failure.

All of this leads to the observation that this release category has a myriad of similar contributors when considering timing, magnitude and frequency, and therefore most of the source terms are reasonable representatives. Case 1B was selected amongst these reasonable choices, having more relevance in terms of frequency contribution (as described above), while also having a larger release magnitude than Case 1A.

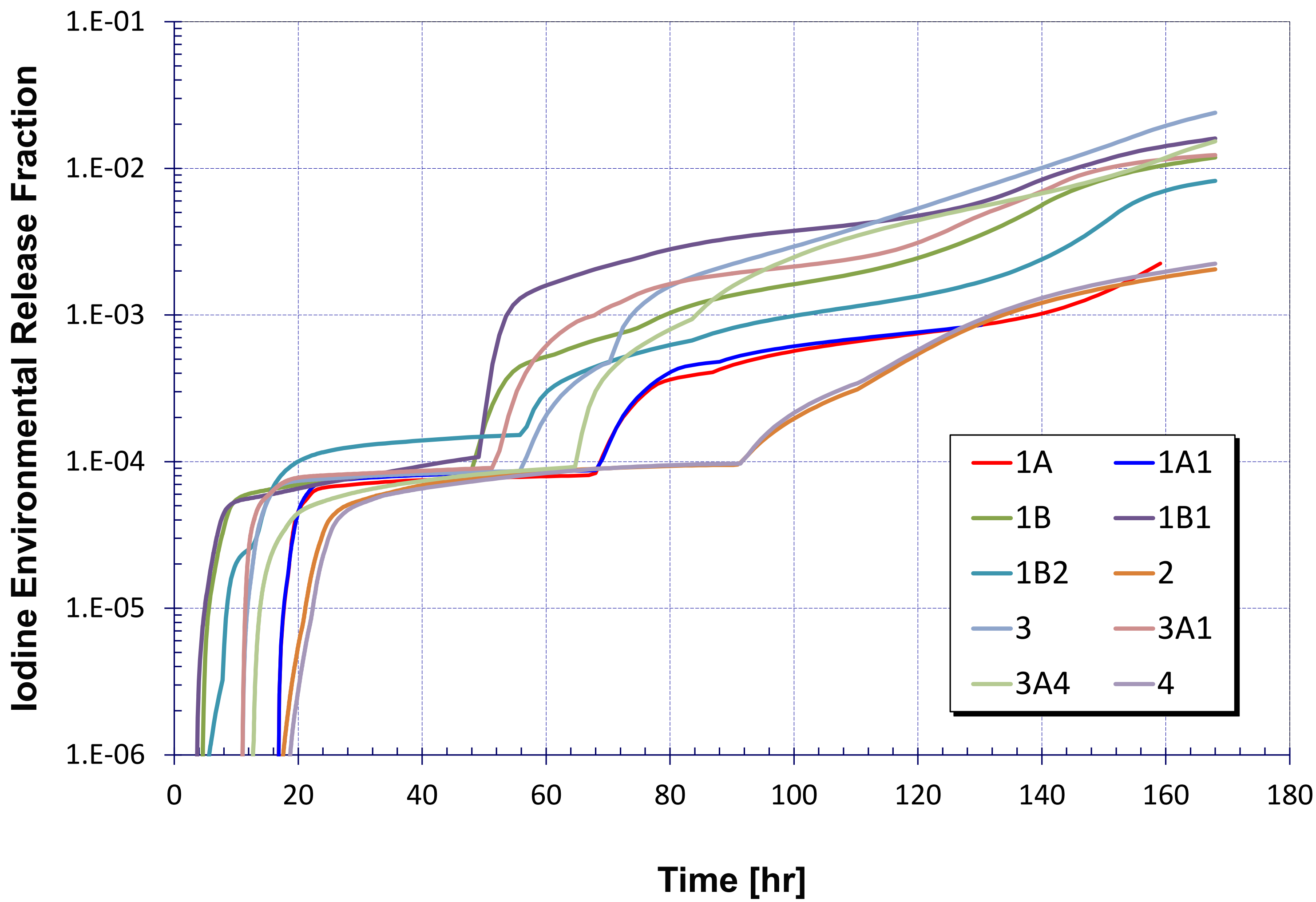

Figure 2-26: Iodine Releases for RC LCF

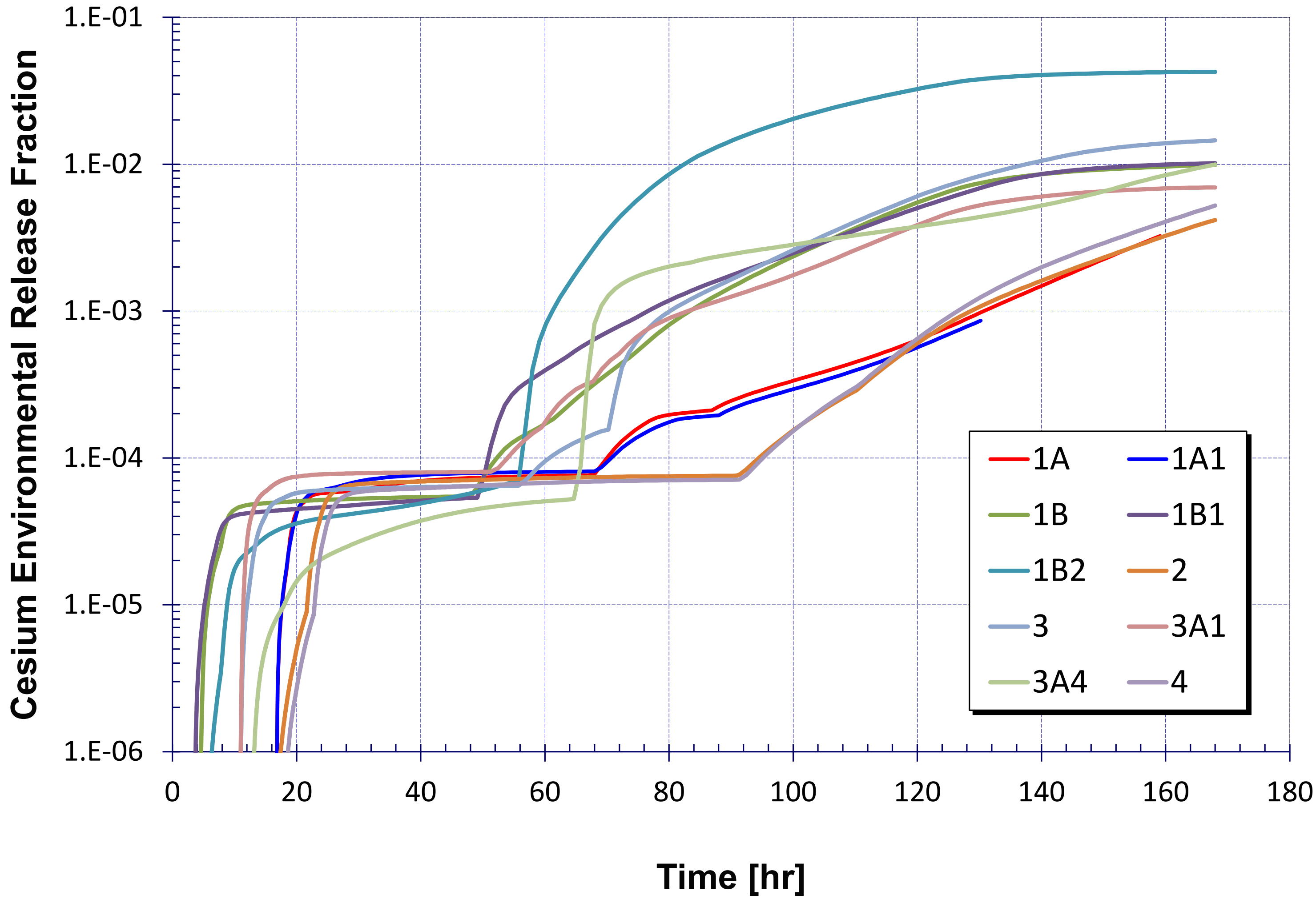

Figure 2-27: Cesium Releases for RC LCF

2.5.2.9. *Selection of release category BMT representative source term*

Cases 6, 6A, and 6B are candidates for the BMT source term. MELCOR calculates erosion depth, and this was used to estimate the time of BMT, but no actual change in the calculation occurs at this point. This is because MELCOR does not model actual basemat penetration, nor does it address aqueous releases into the ground. **Figure 2-29** and **Figure 2-30** show the source terms for these cases. Since Case 6A reflects a benevolent failure of operators not placing containment sprays in pull-to-lock as directed by procedures, and Case 6B suppresses combustion to providing boundary conditions for the stand-alone combustion analysis, Case 6 was selected as the representative source term. Note, this release category is not expected to have a noticeable contribution to the overall offsite consequence or risk results.

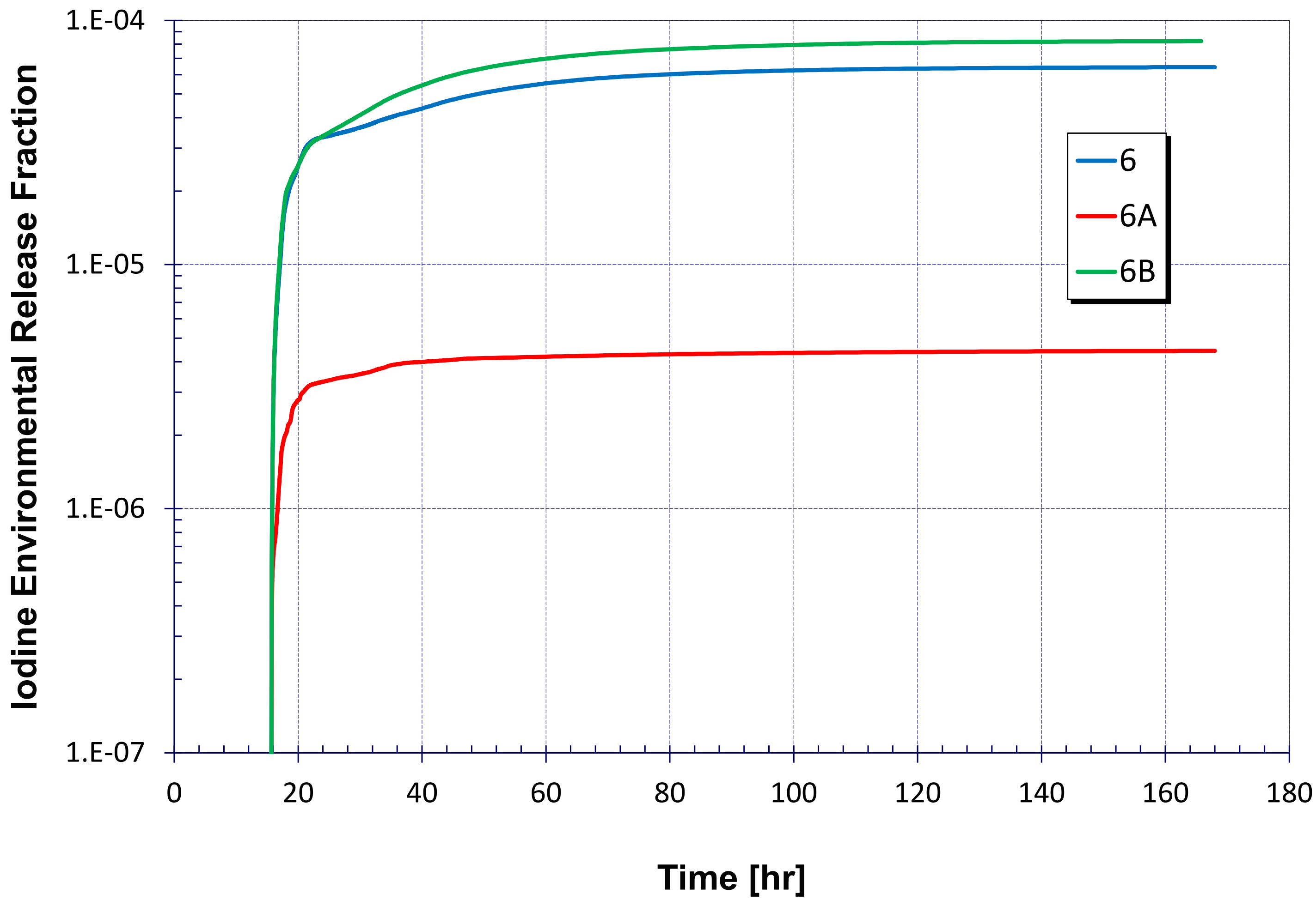

Figure 2-28: Iodine Releases for RC BMT

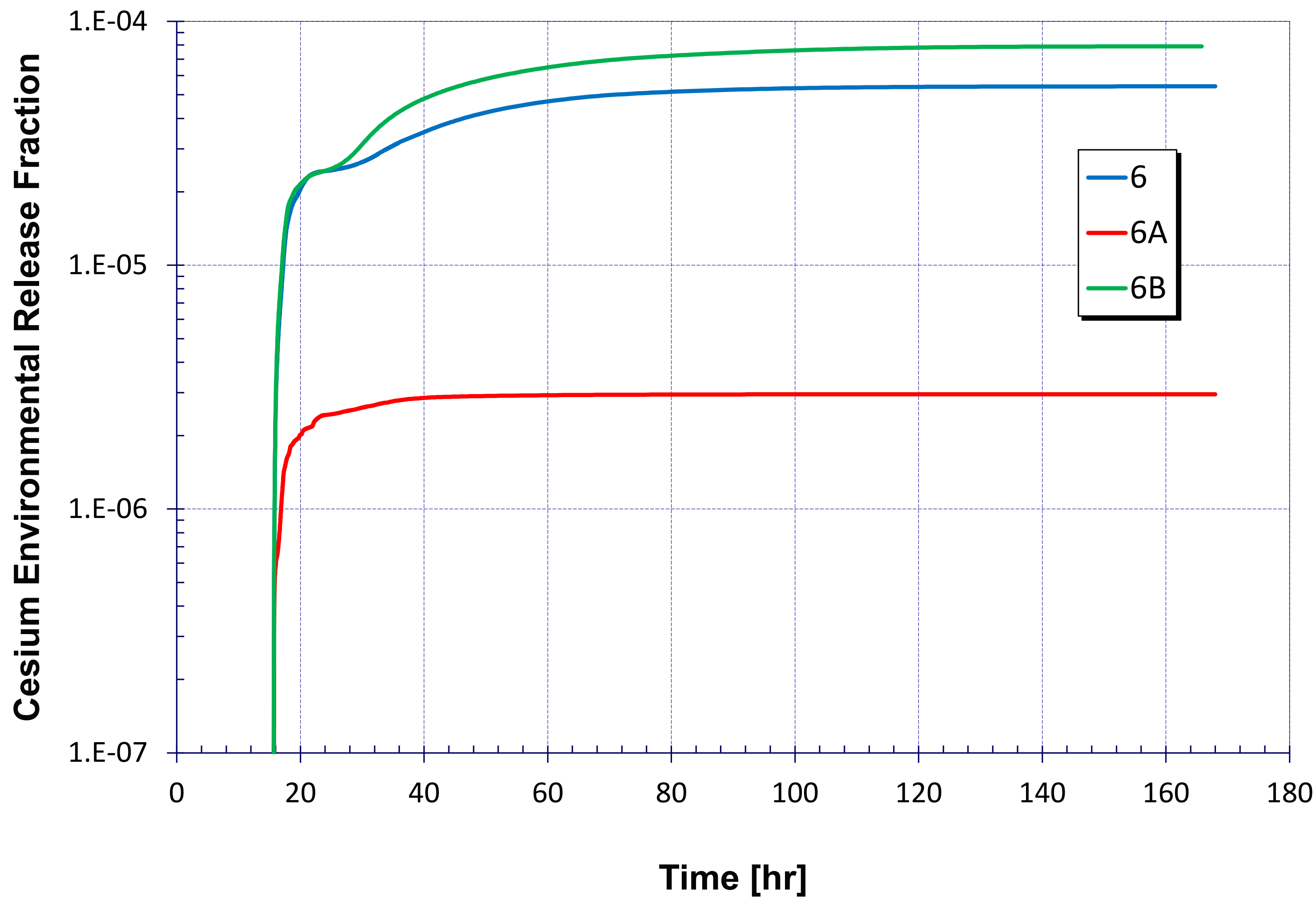

Figure 2-29: Cesium Releases for RC BMT

2.5.2.10. *Selection of release category NOCF representative source term*

Cases 1, 2R1, and 6R1 are candidates for the NOCF release category. **Figure 2-31** and **Figure 2-32** show the source terms for these cases. Case 6R1 experienced calculational problems that prevented it from being completed, while Case 1 is the situation with core damage dramatically delayed by continuous TDAFW operation during station blackout. The MELCOR cases are compared to the highest contributing accident progression sequences for the NOCF release category. Case 1 best reflects the highest contributing accident progression sequence (1-CET-001), which credits successful blind feeding of steam generators after battery depletion. Case 2R1 best reflects the second-highest contributing accident progression sequence (1-CET-002), which credits recovery actions to depressurize the RCS and allow for additional accumulator injection. This sequence includes successful in-vessel recovery and containment does not fail. Both sequences are important contributors to the overall release frequency (i.e., the second-highest and third-highest contributors to total release frequency). However, both sequences result in intact containment and do not contribute significantly to environmental releases. Thus, neither sequence is included in the discussion of significant accident progression sequences (see **Section 2.3.3** and **Table 2-10**). Both Case 1 and Case 2R1 represent important contributions to the total release frequency. Case 2R1 represents a significantly earlier release with a similar overall magnitude, and was selected as the representative source term. Note, this selection is not expected to have any notable effect on the overall offsite consequence or risk results (given the expected trivial contribution from intact containment releases).

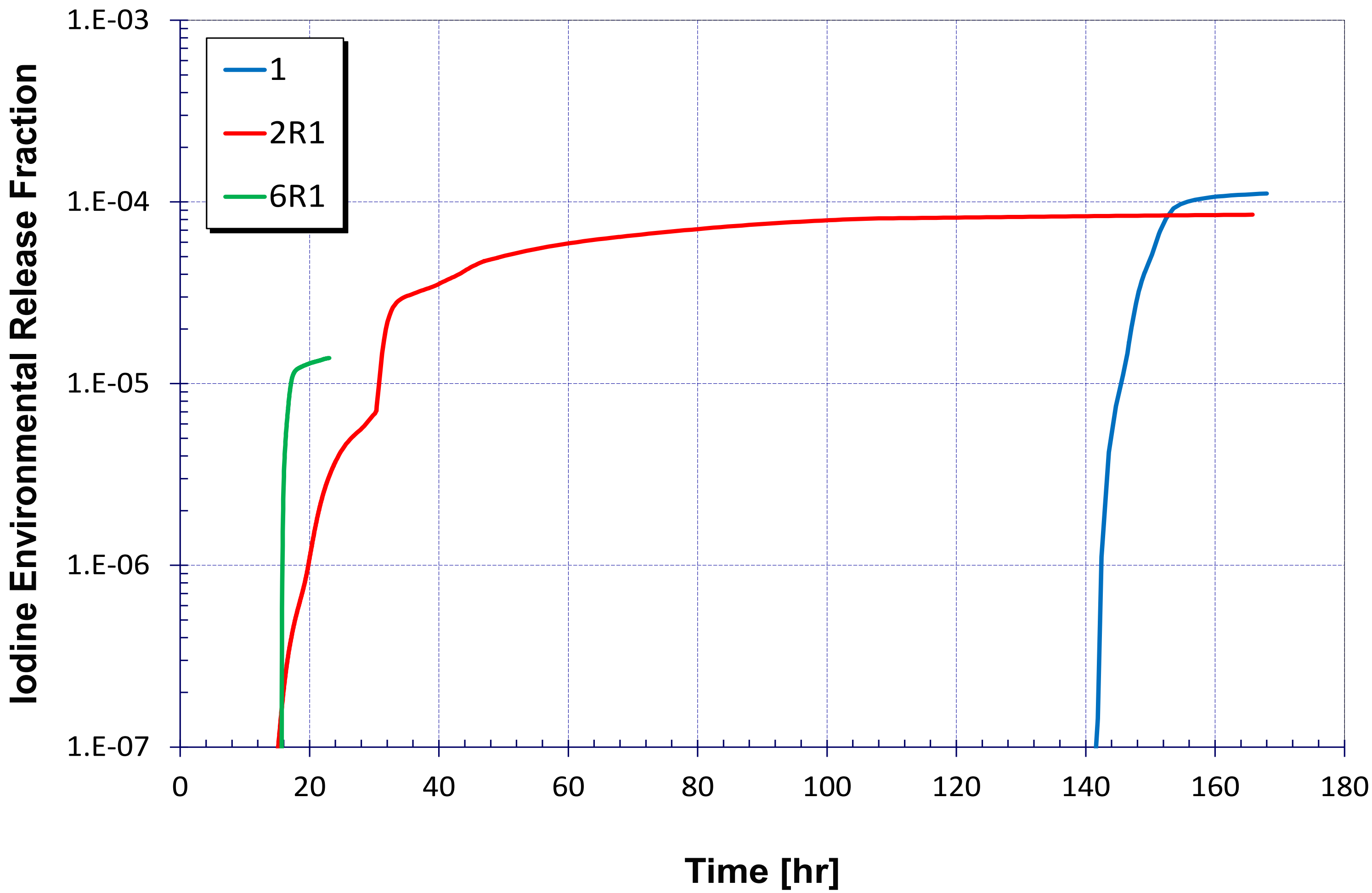

Figure 2-30: Iodine Releases for RC NOCF

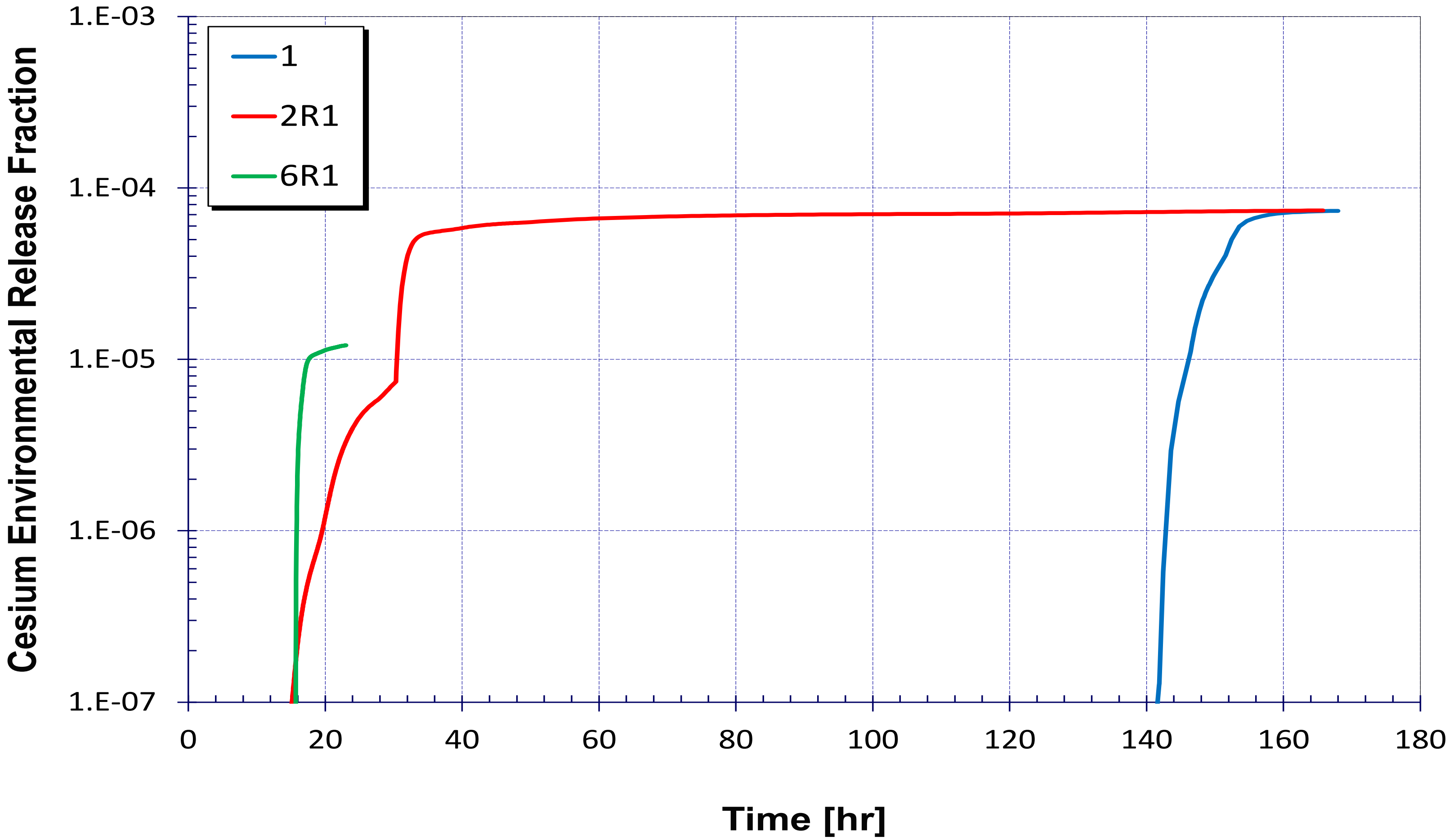

Figure 2-31: Cesium Releases for RC NOCF

2.5.2.11. *Other source terms*

For completeness in presenting the iodine and cesium releases graphically, **Figure 2-33** and **Figure 2-34** show the releases for all other cases (i.e., those cases that were the only candidate source term for their respective release category).

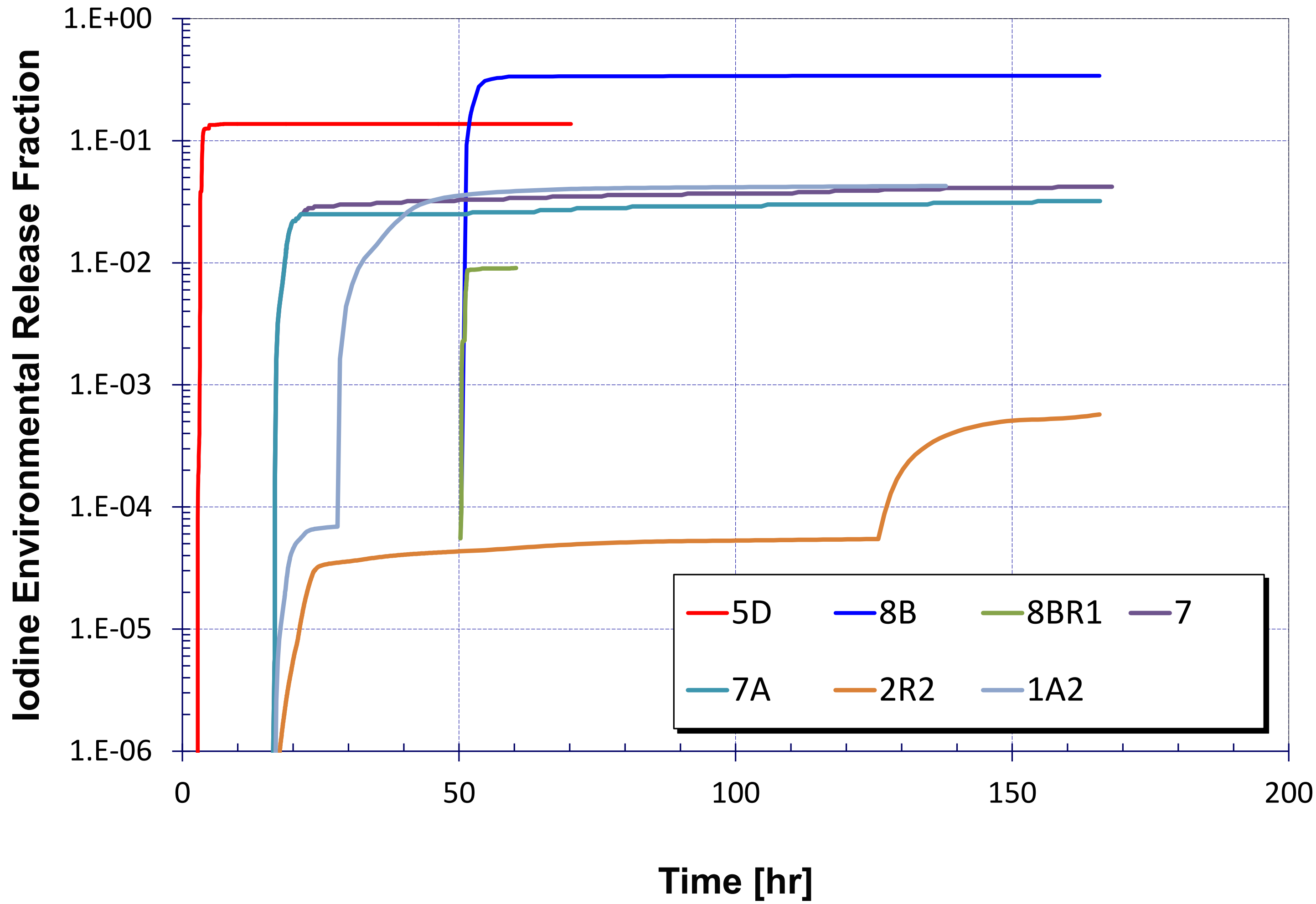

Figure 2-32: Iodine Releases for Other Cases

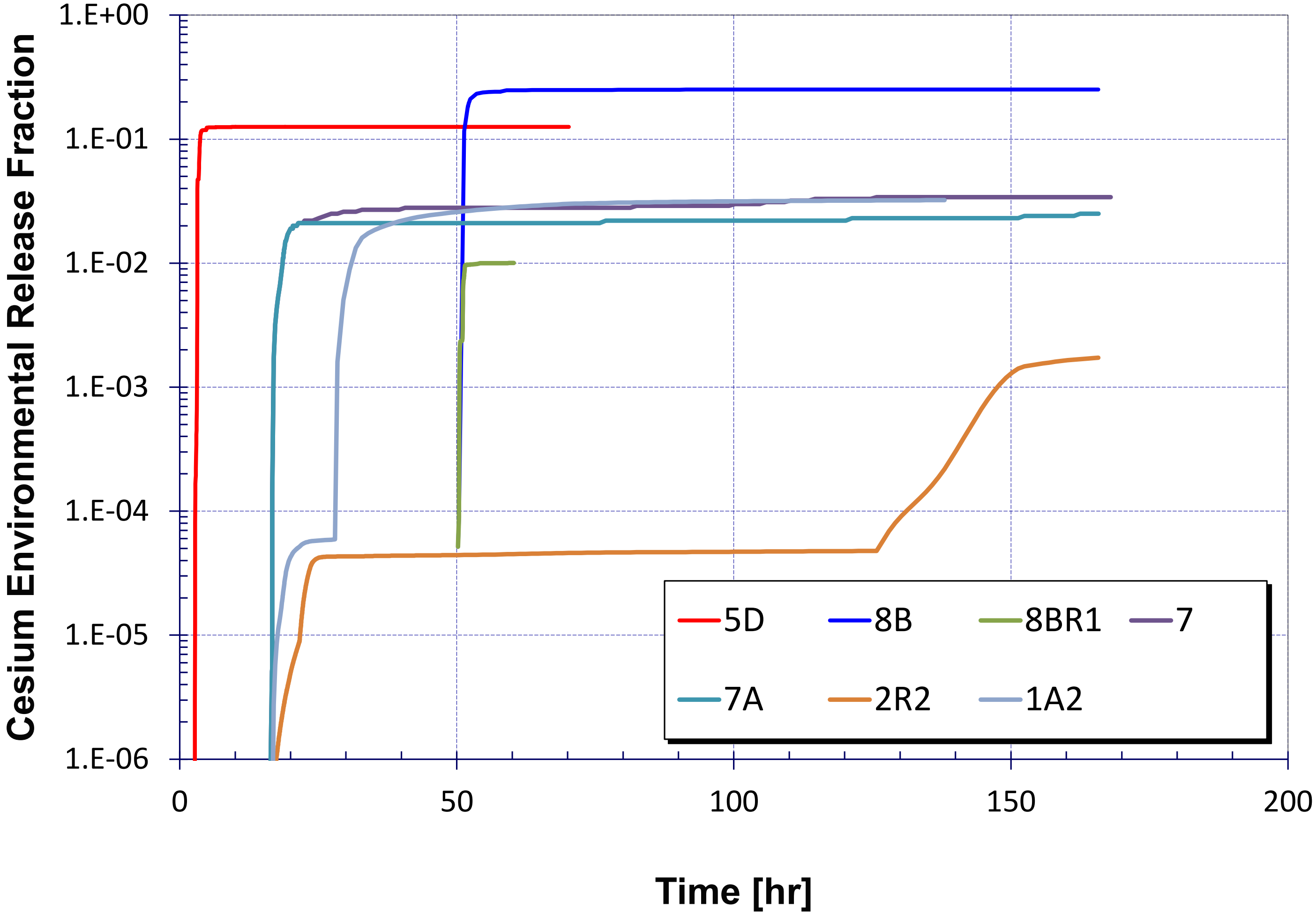

Figure 2-33: Cesium Releases for Other Cases

2.5.2.12. *Aerosol size distribution*

This section provides information about MELCOR's default treatment of aerosols. No aerosol size distribution is provided with the source terms in this document, but it is embedded in the set of outputs passed to MELMACCS. Aerosol dynamics in MELCOR are based on MAEROS, which is a multi-sectional, multicomponent model that determines the mass in each size bin, or section, for each type of aerosol mass, or component. Users may specify the minimum and maximum aerosol sizes, the number of aerosol size bins, and the number of components. (Note, MELCOR defaults are 0.1 μm and 50 μm, for minimum and maximum aerosol diameters, respectively; 10 size bins; and two components – one for water vapor, and the other for all other radionuclide classes.) Individual section boundaries are calculated from the user-input values so that the ratio of the upper and lower bound diameter of each section is the same. The distribution of aerosol mass within a section is treated as constant with respect to the logarithm of particle mass. The particle size distribution changes due to aerosol sources from fuel and MCCI release models, or user-specified sources; condensation and evaporation of water and fission product vapors to and from particles; particle agglomeration, deposition on surfaces, or settling through flow paths into lower control volumes; advection of particles between control volumes due to bulk flows; and removal of aerosols by engineered safety features, such as filter trapping, pool scrubbing, and spray washout. Aerosol particles move between sections and between control volumes, or are removed from the control volume atmosphere entirely, because of the above processes. Code output includes information about the size distribution of aerosols advected through user-specified flow paths.

2.5.2.13. *Release points*

The MELCOR output files used by MELMACCS include release point information (in that radiological releases are broken down by the MELCOR flow path through which they are entering the environment). The relevant release path characterization information is captured later in **Section 2.7**. The MELCOR model treats most potential pathways that can lead to environmental radiological release. Two exceptions to this are:

- *Steam jet air ejectors* – If a SGTR or ISGTR leads to significant contamination of the steam lines, and the MSIV on the ruptured steam generator is open during and following core damage, and the condenser continues to be in operation, and the radiological material is not effectively scrubbed by the water and steam in the SG / steamline / condenser, then radiological material can enter the environment through the steam jet air ejectors. Since the release frequency for all SGTR and ISGTRs is very low, and since the additional factors listed above would have to occur, this is not viewed as a significant modeling limitation.
- TDAFW pump turbine discharge – If a SGTR or ISGTR leads to significant contamination of the steam generators, and TDAFW supply from the ruptured steam generator is placed in operation during or following core damage, then radiological material can enter the environment though the turbine exhaust discharge point (ground level). Note, the steam supply from the ruptured SG can only exist if the ruptured SG has water, so some scrubbing would be expected. Also note, the TDAFW pump turbine can be fed from either SG #1 or SG #2 without a procedurally established preference, but with procedural directions for the operators to isolate TDAFW steam flow from a ruptured SG. Since the release frequency for all SGTR and ISGTRs is very low, and due to the additional factors listed above, this is not viewed as a significant modeling limitation.

2.5.3. Step 3 – Consideration of uncertainties in the source term development

This step is analogous to the final step under the probabilistic treatment technical element (i.e., **Section 2.4.7**), and the treatment of these uncertainties is covered in Appendix C in order to provide an integrated treatment of accident progression and source term uncertainties. Whereas **Section 2.4.7** focused on the release category frequency uncertainty, this section focuses on the source term uncertainty.

In developing the MELCOR cases, the analysts made many modeling choices that impact the source terms. The modeling approach reflects the best current knowledge of severe accident progression. Nevertheless, these modeling choices introduce uncertainty in the model results. The sensitivity analyses discussed in Appendix C explore the impacts of alternative modeling choices. It should be understood that many of the limitations that are described in terms of MELCOR modeling, are actually limitations in the current state-of-knowledge.

The following source term-related high-level observations are made:

- The compilation of MELCOR-related model uncertainty sensitivity analyses discussed in Appendix C gives some indications that the baseline MELCOR results used to develop the Level 2 PRA (and to define the representative source terms) may exhibit a general tendency of under-predicting iodine releases and over-predicting cesium releases (relative to the central tendency suggested by those results). However, given the limited number of results and the general expected correlation between iodine and cesium releases, this is not judged to be a robust conclusion.
- Given the results of the sensitivity analyses, as well as the insights gained from past and ongoing severe accident studies, the central tendency of the cumulative MELCOR release fractions can reasonably be expected to vary within a factor of three (for those values greater than ~1 percent, as discussed further in Appendix C). **Figure 2-35** and **Figure 2-36** provide a graphical representation of the spread in the model uncertainty results. In these figures, the x-axis corresponds to the individual MELCOR sensitivity analyses performed in Appendix C, the left y-axis (and the bars) provides the cumulative release fraction from each simulation, and the right y-axis (and the points/line) provides the ratio of the sensitivity's cumulative release fraction to that of the associated base case.
- When comparing the various radiological releases (by release category) from the sensitivity cases to the pre-existing cases in Appendix B used in developing the Level 2 PRA, a mix of outcomes is observed. In other words, in some cases the model uncertainty sensitivity analysis results show a broader spread, whereas in others they do not.

Section 5 of Appendix C provides more detail on the above points and goes on to highlight several individual modeling uncertainties of note.

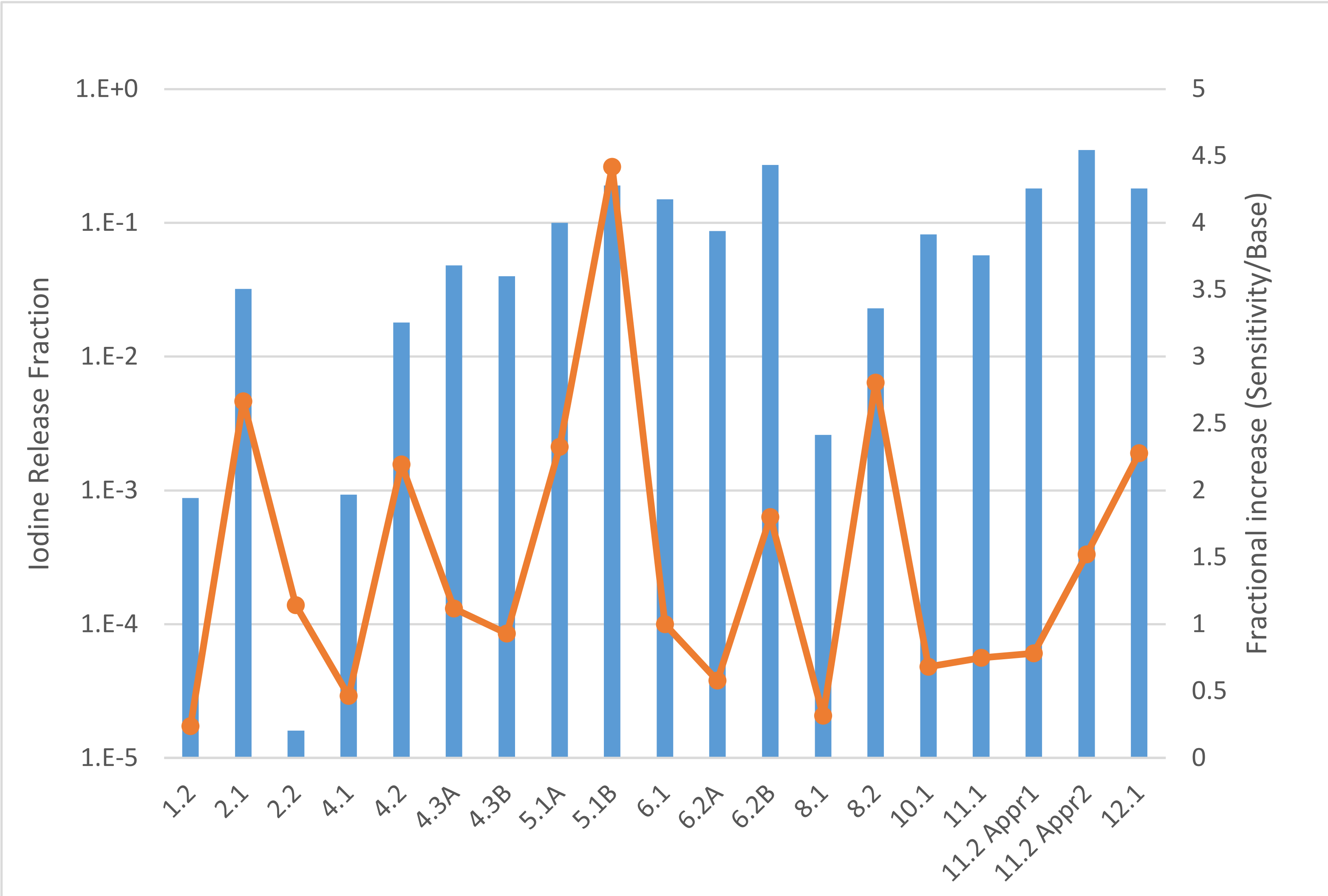

Figure 2-34: Iodine Comparison from the Treatment of Uncertainty

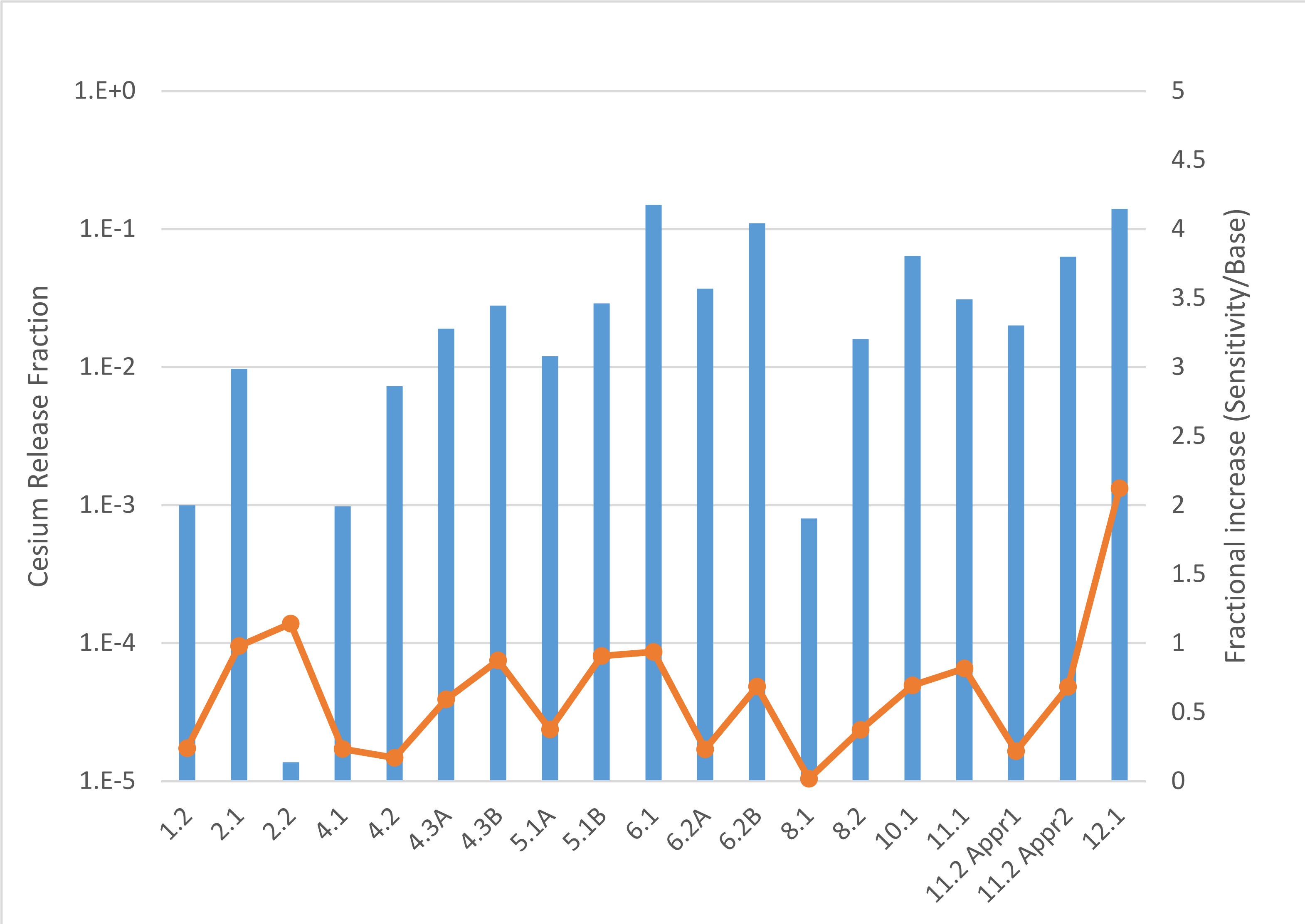

Figure 2-35: Cesium Comparison from the Treatment of Uncertainty

2.6. Evaluation and Presentation of Results

2.6.1. Consolidation of results

Section 2.3.3 and Section 2.5.2 provide the deterministic results from MELCOR, and also assign specific source terms to each of the radiological release categories. Section 2.4.6 provides release category frequencies obtained from the probabilistic model. The presentation of the combination of these results required a decision regarding when each accident sequence would be successfully terminated, and what metrics would be used to condense these results. The following project-specific risk metric definitions were developed:

- **LERF (early injuries)**: release categories are defined to contribute to large early release frequency (LERF) if their representative source term has a warning time (based on iodine release exceeding 1 percent) less than 20 hours simultaneous with the cumulative iodine release fraction being greater than 4 percent
- **LERF (early fatalities)**: release categories are defined to contribute to this alternative LERF definition if their representative source term has a warning time (based on iodine release exceeding 1 percent) less than 3.5 hours simultaneous with the cumulative iodine release fraction being greater than 4 percent
- **LRF**: release categories are defined to contribute to large release frequency (LRF) if they include containment bypass or containment failure, excluding those where fission product scrubbing (or other mechanisms) result in a source term comparable to, or smaller than, the remainder of the (intact containment) source terms
- **CCFP**: conditional containment failure probability (CCFP) is defined as the ratio of the release categories involving a failed or bypassed containment to the overall release frequency

Table 2-23 provides the breakdown of these risk surrogates for different truncation times. As can be seen in the table (and as expected), LERF is insensitive to the selected truncation time, while LRF and CCFP are very sensitive to this choice.

2.6.2. Model convergence

To assess whether the model has converged based on the Level 2 PRA standard, one has to specify which release categories are significant. The Standard provides the following definition for a significant radionuclide release category:

> One of the set of radionuclide release categories contributing to LRF/LERF or to the overall radionuclide release frequency that, when rank-ordered by decreasing frequency, sum to 95% of the LRF/LERF or overall release frequency (excluding design basis leakage RCs) or individually contribute more than 1% of LRF/LERF or 5% of the overall release frequency.

Note that in assessing the significant release categories, the definition of LERF considers both potential for early injuries and early fatalities consistent with the two LERF definitions described above. For the results presented in the proceeding sub-sections, the following release categories meet this definition:

- 1-REL-CIF
- 1-REL-ECF
- 1-REL-ICF-BURN
- 1-REL-ISGTR
- 1-REL-LCF
- 1-REL-LCF-SC

- 1-REL-SGTR-O
- 1-REL-V-F
- 1-REL-V-F-SC

The model has been solved at three truncation levels to demonstrate how these (and other) release categories are affected by the choice of a 10^{-11}/yr truncation, and the results are shown in **Table 2-24**. The example given in the Standard for proving convergence is that "successive reductions in truncation value of one decade result in decreasing changes, and the final change is less than 5%." This example criterion is not met, in that 1-REL-CIF, 1-REL-ECF, and 1-REL-SGTR-O each show a change greater than 5% when going to 10^{-12}/yr. Nevertheless, this convergence study also demonstrated that truncation limits greater than 10^{-11}/yr are not practical for model use, given the long quantification time to produce the 10^{-12}/yr solution, and the model could only be solved in small pieces (1 to 3 initiators at a time). However, **Table 2-24** demonstrates that at a truncation of 10^{-11}/yr, the fractional (or percentage) contributions of the release categories to the overall release frequency are very stable, with a higher truncation limit only affecting these to the tenth of a percent. For this reason, 10^{-11}/yr is a reasonable compromise between precision and quantification time, so long as it is understood that changes in results in the tenths of a percent range may be convergence-related.

2.6.3. Importances

As part of the Level 2 PRA model quantification, Level 1 and 2 PRA basic event importance rankings were developed for all of the release categories. A synopsis of theses importance results is provided below.

- ISLOCAs (1-REL-V-F, 1-REL-V, 1-REL-V-F-SC): The highest-ranking importances for ISLOCA release categories are exclusively Level 1 PRA basic events, with the exception of Level 2 events associated with break submergence (1-L2-BE-ISLOCASUBM-SM, 1-L2-BE-ISLOCASUBM-LRG), which have high Fussell-Vesely (F-V) rankings for 1-REL-V and 1-REL-V-F, respectively. The dearth of Level 2 basic events is a function of the relatively simple Level 2 modeling for ISLOCAs (few operator actions, system responses, and phenomena are considered).
- SGTRs before CD (1-REL-SGTR-O, 1-REL-SGTR-O-SC, 1-REL-SGTR-C): The only two Level 2 basic events that appear in the high-ranking importance measures are 1-L2-OP-SCG1-2 and 1-L2-OP-SCG1-3 (which are failure to carry out SG feed-and-bleed mitigation prior to and following vessel breach). The Level 1 PRA basic events that are high-ranking are those associated with the SGTR initiating event, events that are significant for IE-SGTR core damage (e.g., consequential LOOP with failure of switchyard equipment), ATWS-related events (due to their potential for leading to PI-SGTR), and secondary-side break events (for the same reason).
- ISGTRs (1-REL-ISGTR): In the current input model, ISGTR is a possibility only for high/dry/low core damage scenarios. The most important event for this release category (at least from a ratio importance measure perspective) is 1-L2-BE-INDSGTR-HDL (conditional probability of SGTR before hot leg failure in a high/dry/low scenario). Other Level 1 and Level 2 basic events appearing in the lists are factors which would affect the probability of a high/dry/low core damage sequence, such as failure to extend TDAFW and failure to depressurize the RCS during core damage by dumping steam. Level 1 events are dominated by LOOP initiators, electrical system failures, and NSCW failures, all of which contribute strongly to the probability of a high-pressure core damage sequence.
- Containment isolation failures (1-REL-CIF, 1-REL-CIF-SC): As mentioned previously, the error in the 1-CET assignment of sequence #74 resulted in most 1-REL-CIF-SC cut sets winding up in 1-REL-CIF. Only one cut set was above truncation from the other 1-

REL-CIF-SC sequences, and so the importances only reflect that one cut set (and thus produce erratic results). For 1-REL-CIF, no post-core damage events are high-ranking, though a number of bridge tree events do appear. Prime amongst these is 1-L2TEAR, which has also been found elsewhere to dominate the likelihood of containment isolation failure. There are additionally some air-operated valve failures from the containment isolation system (CIS) model. Otherwise, the high-ranking events are predominantly SBO and NSCW-related events, because these are the initiators that dominate the CDF, and thus dominate the 1-REL-CIF frequency, since 1-REL-CIF is comprised essentially of Level 1 PRA cut sets with independent isolation failure appended.

- Early containment failures (1-REL-ECF): A number of Level 2 basic events show up as high-ranking for either F-V or risk achievement worth (RAW). These include (1) the likelihood of an in-vessel steam explosion failing containment at low RCS pressure (high-ranking for both importance measure types) or higher RCS pressure (RAW only), (2) the likelihood of a combustion event failing containment at the time of vessel breach (both), (3) the likelihood of extending TDAFW indefinitely (F-V only), (4) the likelihood of a combustion event prior to vessel breach (F-V only), and (5) the likelihood of in-vessel recovery (F-V only). The remainder of the high-ranking events are inherited Level 1 PRA basic events. These are generally SBO and NSCW-related (as expected due to their CDF dominance), although XLOCA and MLOCA initiating events also have high RAW importance.
- Intermediate combustion-induced failures (1-REL-ICF-BURN, 1-REL-ICF-BURN-SC): High-ranking F-V importances have a large number of Level 2 basic events, including extension of TDAFW, operator actions to spray containment after vessel breach, the myriad of basic events associated with hydrogen combustion during different time frames (including deflagration and detonation-induced failures well after vessel breach), induced hot leg failure, and in-vessel recovery likelihood. Conversely, high-ranking RAW importances are exclusively inherited Level 1 PRA (pre-core-damage) events.
- Late containment failures (1-REL-LCF, 1-REL-LCF-SC): High-ranking F-V importances have a large number of Level 2 basic events, including extension of TDAFW, operator actions to carry out SAG-1 prior to vessel breach, operator actions to spray containment after vessel breach, the myriad of basic events associated with hydrogen combustion during different time frames, induced hot leg failure, containment overpressure versus basemat melt-through likelihood, and in-vessel recovery likelihood. Conversely, high-ranking RAW importances are exclusively inherited Level 1 PRA (pre-core-damage) events.
- Basemat melt-through (1-REL-BMT): High-ranking F-V importances have a large number of Level 2 basic events, including operator action to carry out SAG-1 prior to vessel breach, a handful of basic events associated with hydrogen combustion during different time frames, in-vessel recovery likelihood, and the likelihood of basemat melt-through given containment heat removal. Conversely, high-ranking RAW importances are exclusively inherited Level 1 PRA (pre-core-damage) events. Note, the MLOCA initiating event is high-ranking for both importance measure types.
- Intact containment (1-REL-NOCF): The high-ranking F-V importances are predominantly Level 1 PRA events that are important to core damage frequency. The two exceptions are events associated with combustion events in containment prior to vessel breach. These events' importances are a function of their pervasiveness (their outcome does not affect containment integrity and they appear in far more 1-REL-NOCF cut sets than virtually any other events).

The above synopsis does not reflect the fact that release categories do not contribute equally to risk. As such, the Level 1 and Level 2 basic events with the highest F-V importance (i.e., those with a F-V importance measure greater than 0.1) for any of the significant release categories[24] were combined into a single listing of those events that are most important to Level 2 PRA risk surrogates (see **Table 2-25**)[25]. The following classes of events are represented:

- Numerous Level 1 PRA initiating events
- Level 1 PRA electrical distribution events
- Level 1 PRA NSCW-related events
- Level 1 PRA ATWS events germane to PI-SGTR
- Level 1 PRA RCP stage 2 seal failure event
- Level 1 PRA ISLOCA events
- Two additional Level 1 PRA SGTR events
- Five Level 1 PRA operator actions
- The CIS event associated with a pre-existing or maintenance-related CIS failure
- Indefinite extension of TDAFW during relevant SBO events
- Level 2 PRA hot leg and SG tube creep failure events
- Level 2 PRA in-vessel recovery event
- Numerous Level 2 PRA combustion-related events
- Level 2 PRA in-vessel steam explosion event
- Level 2 PRA late containment over-pressure failure without containment heat removal
- Level 2 PRA ISLOCA break submergence for large ISLOCAs
- Four of the modeled Level 2 operator actions

[24] The selection of significant release categories is based on the definition of significant radionuclide release category from the ASME/ANS Level 2 PRA Trial Use and Pilot Application standard (ASME, 2014). A significant radionuclide release category is defined as one of the set of radionuclide release categories contributing to LRF/LERF or to the overall radionuclide release frequency that, when rank-ordered by decreasing frequency, sum to 95 percent of the LRF/LERF or overall release frequency (excluding design basis leakage RCs) or individually contribute more than 1 percent of LRF/LERF or 5 percent of the overall release frequency.

[25] The table does not include success events, since the importance measures for these events do not comport with traditional PRA importance measures that focus on failure events, and their meaning for success events is not intuitively obvious. In addition, it is believed that the more meaningful insights to be obtained through evaluation of importance measures will be manifested through the importance measures for the corresponding failure events that occur in the significant release categories.

Table 2-23: Risk Surrogates Presented for Different Accident Termination Times

Release Category (RC)	Percent contribution to overall release frequency	Time at which airborne radiological releases are terminated								
		SAMG entry + 36 hours			SAMG entry + 60 hours			7 days after event initiation		
		LERF[1, 2]	LRF[2]	CCFP[2]	LERF[1, 2]	LRF[2]	CCFP[2]	LERF[1, 2]	LRF[2]	CCFP[2]
1-REL-BMT	1%	-	-	-	-	-	-	-	-	1%
1-REL-CIF	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-CIF-SC	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-ECF	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-ICF-BURN	12%	-	12%	12%	-	12%	12%	-	12%	12%
1-REL-ICF-BURN-SC	4%	-	-	4%	-	-	4%	-	-	4%
1-REL-ISGTR	1%	1%	1%	1%	1%	1%	1%	1%	1%	1%
1-REL-LCF	42%	-	-	-	-	-[3]	42%	-	42%	42%
1-REL-LCF-SC	5%	-	-	-	-	-	-	-	5%	5%
1-REL-NOCF	35%	-	-	-	-	-	-	-	-	-
1-REL-SGTR-C	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-SGTR-O	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-SGTR-O-SC	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-V	<1%	-	<1%	<1%	-	<1%	<1%	-	<1%	<1%
1-REL-V-F	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%
1-REL-V-F-SC	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%	<1%
Total	100%	1%	14%	18%	1%	14%	60%	1%	61%	65%

1 Relative to the definitions presented earlier, this was the LERF value based on the potential for early fatalities (as opposed to early injuries). The latter included additional release categories but was numerically extremely similar due to the very low contribution of those additional release categories.

2 These columns simply show which release categories apply to the subject metric, and when they apply, the release category's percentage contribution was repeated in order to show how the risk metric value was calculated.

3 The definition of LRF for the L3PRA project includes release category 1-REL-LCF. However, while containment fails prior to "SAMG entry plus 60 hours" for this release category, the resulting radiological release occurs slowly over a long period of time (see Figures 2-27 and 2-28). Therefore, this release category does not contribute to LRF (i.e., does not meet the quantitative definition of LRF used in this project) for the case where radiological releases are terminated at "SAMG entry plus 60 hours."

Table 2-24: Results of Model Convergence Study

	1E-10/yr				1E-11/yr				1E-12/yr			
	Value	% Diff	Cut sets	Fractional	Value	% Diff	Cut sets	Fractional	Value	% Diff	Cut sets	Fractional
Total =	6.67E-05	-	29,533	1.000	7.04E-05	5.6%	141,750	1.000	7.21E-05	2.4%	601,149	1.000
BMT	6.46E-07	-	860	0.010	8.15E-07	26.1%	6,237	0.012	9.34E-07	14.7%	36,363	0.013
CIF[1]	4.79E-08	-	111	0.001	6.54E-08	36.3%	693	0.001	7.53E-08	15.3%	4,075	0.001
CIF-SC[1]	0.00E+00	-	0	0.000	1.11E-11	-	1	0.000	1.11E-11	0.0%	1	0.000
ECF	2.61E-09	-	11	0.000	6.56E-09	150.8%	165	0.000	9.56E-09	45.8%	1,227	0.000
ICF-BURN	8.10E-06	-	4,661	0.121	8.74E-06	7.9%	23,114	0.124	9.01E-06	3.1%	102,156	0.125
ICF-BURN-SC	1.99E-06	-	2,725	0.030	2.47E-06	24.3%	15,835	0.035	2.73E-06	10.3%	72,772	0.038
ISGTR	5.28E-07	-	409	0.008	5.79E-07	9.7%	2,062	0.008	6.02E-07	4.0%	9,529	0.008
LCF	2.82E-05	-	11,877	0.423	2.94E-05	4.4%	52,524	0.418	2.99E-05	1.6%	202,262	0.415
LCF-SC	3.05E-06	-	1,961	0.046	3.31E-06	8.5%	9,010	0.047	3.41E-06	3.0%	36,376	0.047
NOCF	2.36E-05	-	6,408	0.353	2.44E-05	3.4%	29,519	0.346	2.48E-05	1.6%	122,522	0.344
SGTR-C	3.03E-08	-	66	0.000	3.69E-08	21.9%	352	0.001	4.19E-08	13.5%	2,130	0.001
SGTR-O	1.53E-08	-	64	0.000	2.10E-08	37.5%	286	0.000	2.41E-08	14.8%	1,240	0.000
SGTR-O-SC	1.89E-07	-	278	0.003	2.16E-07	14.6%	1,108	0.003	2.31E-07	7.0%	6,104	0.003
V	3.83E-09	-	20	0.000	9.71E-09	153.7%	264	0.000	1.27E-08	30.9%	1,380	0.000
V-F	9.17E-08	-	39	0.001	9.76E-08	6.5%	282	0.001	1.01E-07	3.4%	1,464	0.001
V-F-SC	2.21E-07	-	43	0.003	2.27E-07	2.9%	298	0.003	2.31E-07	1.5%	1,548	0.003

[1] CET sequence number 74 should have been assigned to the 1-REL-CIF-SC release category rather than the 1-REL-CIF release category, since in-vessel recovery occurs. The change would raise the 1-REL-CIF-SC frequency to 5x10-9/yr, while reducing the 1-REL-CIF frequency to ~5.8x10-8/yr (the 1-CET-074 sequence frequency is ~5x10-9/yr), discounting any minimization that would occur. This would not affect the release categories' percent contribution (<0.1% and 0.1%, respectively).

Table 2-25: Highest Ranking F-V Events from Significant Release Categories*

Event	Description
1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC BREAKERS AA205 & BA301 TO OPEN
1-EPS-DGN-FR-G4001___	DG1A FAILS TO RUN BY RANDOM CAUSE (24 HR MISSION TIME)
1-EPS-DGN-FR-G4002___	DG1B FAILS TO RUN BY RANDOM CAUSE (24 HR MISSION TIME)
1-EPS-SEQ-CF-FOAB	SEQUENCERS FAIL FROM COMMON CAUSE TO OPERATE
1-IE-ISL-RHR-CLI-A	RHR COLD LEG INJECTION TRAIN A ISOLATION
1-IE-ISL-RHR-CLI-B	RHR COLD LEG INJECTION TRAIN B ISOLATION
1-IE-ISL-RHR-HLS	RHR HOT LEG SUCTION ISOLATION
1-IE-LOCHS	LOSS OF CONDENSER HEAT SINK
1-IE-LOMFW	LOSS OF MAIN FEED WATER
1-IE-LONSCW	LOSS OF NSCW
1-IE-LOOPGR	LOSS OF OFFSITE POWER (GRID- RELATED)
1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELA TED)
1-IE-MLOCA	MEDIUM LOCA
1-IE-RHR-MOV-CO-HV8701A	RHR SUCTION MOV HV8701A TRANSFERS OPEN (ISLOCA INITIATOR)
1-IE-RHR-MOV-CO-HV8701A	RHR SUCTION MOV HV8701A TRANSFERS OPEN (ISLOCA INITIATOR)
1-IE-RHR-MOV-CO-HV8701B	RHR SUCTION MOV HV8701B TRANSFERS OPEN (ISLOCA INITIATOR)
1-IE-RHR-MOV-CO-HV8702A	RHR SUCTION MOV HV8702A TRANSFERS OPEN (ISLOCA INITIATOR)
1-IE-RHR-MOV-CO-HV8702B	RHR SUCTION MOV HV8702B TRANSFERS OPEN (ISLOCA INITIATOR)
1-IE-RHR-MOV-RP-HV8701B	RHR SUCTION MOV HV8701B (ISLOCA INITIATOR)
1-IE-RHR-MOV-RP-HV8702B	RHR SUCTION MOV HV8702B (ISLOCA INITIATOR)
1-IE-SGTR	SGTR
1-IE-SSBO	SECONDARY SIDE BREAK OUTSIDE OF MSIVs
1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF
1-L2-BE-CONTOP-NCHR	Containment Overpressure Failure Late (No CHR)
1-L2-BE-H2CF-E-GEN	Global Deflagration Fails Containment At/Around VB
1-L2-BE-H2CF-L-NACNPB	Late CF from Burn (without AC, without prior burn)
1-L2-BE-H2CF-L-NACPB	Late CF from Burn (without AC, with prior burn)
1-L2-BE-H2CF-L-NCHRPB	Late CF from Burn (without CHR, with prior burn)
1-L2-BE-H2DET-L-CHR	Late Detonation with CHR
1-L2-BE-H2IGN-E-PB	Combustion in Containment at VB given Prior Burn
1-L2-BE-H2IGNSRC-E-AC	No Ignition Source in Containment at VB, with Power
1-L2-BE-H2IGNSRC-E-NAC	No Ignition Source in Containment at VB, without AC Power
1-L2-BE-H2IGNSRC-L-AC	Ignition Source in Containment Late, with Power
1-L2-BE-H2IGNSRC-L-NAC	Ignition Source in Containment Late, without AC Power
1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
1-L2-BE-H2IGN-VE-GEN	Combustion in Containment before VB (General)
1-L2-BE-INDHLF-MP	Induced Hot Leg Failure (Intermediate pressure)

Table 2-25: Highest Ranking F-V Events from Significant Release Categories*

Event	Description
1-L2-BE-INDSGTR-HDL	Induced SGTR given High/Dry/Low
1-L2-BE-ISLOCASUBM-LRG	ISLOCA Break Not Submerged or Significantly Scrubbed for Large ISLOCAs
1-L2-BE-IVREC	No in-vessel retention, Vessel Breach Occurs
1-L2-BE-IVSE-LP	In-vessel steam explosion (IVSE) Fails Containment (Low RCS Pressure)
1-L2-BE-MANUALTDAFW-GEN	Failure of Manual Extension of TDAFW in SBO
1-L2-OP-SAG1	Operator Fails to Carry Out SAG-1 (Open 2 ARVs and Feed SGs)
1-L2-OP-SCG1-1	Operator Fails to Carry Out SCG-1 (Spray Containment w/ Firewater)
1-L2-OP-SCG1-2	Operator Fails to Carry Out SCG-1 (F&B SGs)
1-L2-OP-SCG1-3	Operator Fails to Carry Out SCG-1 (F&B SGs - Late)
1-L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS
1-NO-UET2-NOPORV-BLK	NOT IN UET - ATWT
1-OAB_SI-------H	OPERATOR FAILS TO BLEED & FEED - SI AVAILABLE
1-OA-ORS-------H	OPERATOR FAILS TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
1-OAR_HPML-----H	OPERATOR FAILS TO ESTABLISH HIGH PRESSURE RECIRCULATION - MLOCA
1-OEP-VCF-LP-CLOPL	CONSEQUENTIAL LOSS OF OFFSITE POWER - LOCA
1-OEP-VCF-LP-CLOPT	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
1-OEP-XHE-XL-NR02HGR	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 2 HOURS (GRID-RELATED)
1-OEP-XHE-XL-NR02HWR	OPERATOR FAILS TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
1-RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS
1-RHR-MOV-CO-HV8701A_	RHR SUCTION MOV HV8701A TRANSFERS OPEN
1-RHR-MOV-CO-HV8701B_	RHR SUCTION MOV HV8701A TRANSFERS OPEN
1-RHR-MOV-CO-HV8702A_	RHR SUCTION MOV HV8702A TRANSFERS OPEN
1-RHR-MOV-CO-HV8702B_	RHR SUCTION MOV HV8702B TRANSFERS OPEN
1-RHR-MOV-OO-HV8809A-HDP	LP CL INJ MOV HV8809A FAILS TO CLOSE WITH HIGH DIFFERENTIAL PRESSURE
1-RHR-MOV-RP-HV8701A-CON	RHR SUCTION MOV HV8701A FAILS (CONDITIONAL)
1-RHR-MOV-RP-HV8702A-CON	RHR SUCTION MOV HV8702A FAILS (CONDITIONAL)
1-RPS-BME-CF-RTBAB	CCF RTB-A AND RTB-B (MECHANICAL)
1-RPS-ROD-CF-RCCAS	CCF 10 OR MORE RCCAS FAIL TO DROP
1-RPS-XHE-XE-NSGNL	OPERATOR FAILS TO RESPOND WITH NO RPS SIGNAL PRESENT
1-SGTR2	SGTR IS IN SG 2
1-SGTR3	SGTR IS IN SG 3
1-UET2-NOPORV-BLK	UET-ATWT

* Listed alphanumerically (not by relative importance)

2.6.4. Comparison to past studies

There are two fundamental outputs to a Level 2 PRA: the release category source terms and the release category frequencies. In Section 4 of Appendix D, there is an extensive comparison of the MELCOR analyses used to generate the release category source terms to other relevant information sources. This comparison finds:

- The current results are similar to the reference plant's IPE source term results, with major differences stemming from: (i) larger source terms for some accident sequences, due to the longer accident termination time used in the L3PRA project; (ii) a larger range of source terms for station blackout, due to the inclusion of hydrogen-induced containment failure; (iii) generally higher "non-volatile" releases owing to volatilization of molybdenum during sustained MCCI; and (iv) lower ISLOCA source terms owing to treatment of additional fission product retention mechanisms.
- The current results are comparable to the Surry SOARCA results in NUREG-1935, with major differences stemming from a combination of: (i) differences in plant design, (ii) differences in phenomenological and system modeling assumptions, and (iii) fundamental differences in the scope of the studies and the assessment technologies employed (PRA versus consequence analysis).
- The MELCOR analyses are similar to comparable MAAP analyses for the studied cases.

2.6.5. Outcomes

The following is a synopsis of the major outcomes from the work described in this report:

O1. A Level 2 PRA model for postulated at-power internal event and flood[26] accidents has been developed that covers a broad spectrum of severe accident sequences, and which considers a variety of different systems, phenomena, and operator actions. This model is directly integrated with the corresponding Level 1 PRA, including the ability to inspect cutsets, importance measures, and sensitivities/uncertainties in an integrated fashion.

O2. The probabilistic model has a strong basis in underlying deterministic analysis, most notably using MELCOR, but also using other analytical tools where appropriate. It also benefits from Level 2 HRA methods development performed specifically for this project. Finally, parameter and model uncertainty were extensively investigated.

O3. The majority (>99 percent) of potential accidents were found to not contribute to the potential for a large early release. This was caused by the very low likelihood of initiator-driven containment bypass or early impairment (ISLOCAs, SGTRs, containment isolation failures), and the low likelihood of severe-accident induced containment bypass or early impairment (induced SGTRs, energetic failure of containment at the time of vessel breach).

O4. Combustion-induced failure of containment hours (or tens of hours) after vessel breach, following sustained MCCI, was found to have an important contribution to the results. This outcome is tied to the decision to model combustion as a stochastic rather than deterministic process.[27]

O5. When accident sequences are carried out to longer timeframes (e.g., 7 days), without credit for successful mitigative actions beyond the initial core damage and vessel breach

[26] Due to the small contribution from internal flooding, the PRA does not attempt to capture some aspects specific to flooding (e.g., accessibility for taking local actions during an internal flooding event), and caution is advised if using the model to draw flooding-specific insights.

[27] In many studies, combustion is treated as a deterministic process, wherein assumptions about auto-ignition limit the magnitude of combustible gas concentrations that are simulated. The results here are generally consistent with past studies that have predicted higher late combustion-related containment failure probabilities when modeling the process as stochastic.

response, the majority of accidents ultimately leads to containment failure. Though radiological releases are generally lower in these cases than for the bypass, early containment impairment, and combustion-induced failure cases, they are still significant. This results in high LRF[28] and CCFP estimates for these “untruncated” results.

O6. Regarding the above long-term modeling, sensitivity analysis “MU 3.2” in Section 4.3 of Appendix C illustrates the following useful observations:

a. Consideration of longer-term equipment and offsite power recovery would not affect LERF.

b. The greatest reduction in LRF would occur from recoveries related to controlling containment pressure, with additional benefit seen if this were combined with preventing combustion.

c. The greatest reduction in CCFP would occur from combined recoveries to control containment pressure and prevent basemat melt-through (either one by itself would not be sufficient), with additional benefit seen if this is combined with controlling combustion.

2.7. Level 2/3 PRA Interface

The Level 2/3 PRA interface consolidates the release category information in a format conducive for use by the Level 3 PRA analysts. The objective of this subtask is to catalogue the characteristics of each release category (release frequency, time-dependent chemical class-specific release fractions, sequence information sufficient for establishing declaration of emergency action levels, release energy and elevation, and aerosol size distributions).

The initial inventories of each radionuclide group at the time of the accident are provided in **Table 2-26**, for convenience. Middle-of-cycle characteristics were used in the Level 2 analysis. The Level 3 PRA analysis will use the radioisotope-specific activities in the actual SCALE datasets.

Table 2-26: Initial Inventories

MELCOR RN Class	Initial Inventory (kg)		
	Beginning-of-cycle	Middle-of-cycle	End-of-cycle
XE	219.1	386.4	546.1
CS	5.676	9.952	13.86
BA	92.48	164.2	225.5
I2	0	0	0
TE	20.21	35.69	50.67
RU	137.5	247.9	371.9
MO	120.0	209.8	295.8
CE	650.8	1127	1475
LA	291.2	514.5	733.7
UO2	79,870	78,580	77,440
CD	2.938	5.607	9.402
AG	4.236	7.694	11.90
CSI	16.61	29.58	42.16
CSM	146.8	257.3	358.4

[28] LRF is not a risk metric used for operating United States reactors. Nevertheless, LRF has been tabulated in this study to provide additional context because (like CDF) it is a risk surrogate for long-term offsite consequences.

Table 2-27 provides a synopsis of the release category results for use in the Level 3 PRA analysis. The recommended representative source term for each release category is bolded. The more complete set of needed information relative to releases is provided in the associated MELCOR output files. Those MELCOR output files include fission product release fractions as a function of time, by chemical class and aerosol size bin, and per flow path. Energy content is also included. **Table 2-28** provides a characterization of all relevant release paths.

The following are some factors to be considered in the Level 3 PRA analysis related to the use of the MELCOR results:

1. Time zero in MELCOR simulations correspond to the initial upset condition, which may or may not correspond to the time of reactor trip (depending on whether the upset condition caused an immediate trip). This should be considered in setting the initial radionuclide activities in MACCS, such that MACCS decay of the source term starts at the correct time (see **Table 2-21**).
2. MELCOR simulations have deliberately suppressed containment rupture at the time basemat melt-through is predicted, since there is no mechanistic model in MELCOR for this situation.

Table 2-27: Release Category Summary Table for the Level 3 PRA Team

Release Category	Frequency (/rcy)[1]	MELCOR Modeling Case	Core Damage[2] (hr)	GE (hr)	Major Release[3] (hr)	Cumul. Iodine Release	Cumul. Cesium Release
1-REL-V-F	1×10^{-7}	5D	2.8	1.25	3.2	1.4E-1	1.3E-1
1-REL-V-F-SC	2×10^{-7}	5B	2.8	1.25	3.2	1.2E-1	9.2E-2
		5C	9.5	7.5	10	1.5E-3	7.8E-4
1-REL-V	$<7\times10^{-8}$	5	9.5	7.5	10	1.1E-3	6.4E-4
		5A	9.5	7.5	10	6.8E-4	6.3E-4
1-REL-SGTR-O	$<7\times10^{-8}$	8B	50	47	51	3.4E-1	2.5E-1
1-REL-SGTR-O-SC	2×10^{-7}	8BR1	50	47	51	9.1E-3	1.0E-2
1-REL-SGTR-C	$<7\times10^{-8}$	8	50	47	52	1.2E-2	1.1E-2
		8A	96	90	97	9.4E-3	8.3E-3
		8R1	50	47	Never	5.9E-4	6.4E-4
		8R2	50	47	52	1.1E-2	1.1E-2
1-REL-ISGTR (a.k.a., C-SGTR)	6×10^{-7}	3A2	11	8	11	2.3E-1	9.2E-2
		3A3	11	8	11	7.6E-2	3.8E-2
1-REL-CIF	$<7\times10^{-8}$	7	16	3	18	4.2E-2	3.4E-2
1-REL-CIF-SC	$<7\times10^{-8}$	7A	16	3	18	3.3E-2	2.5E-2
1-REL-ECF	$<7\times10^{-8}$	2A	15	8	22	1.5E-1	1.6E-1
		6C	15	13	40	6.1E-3	3.2E-3
		6D	15	13	74	7.7E-4	2.2E-3
1-REL-LCF	2.9×10^{-5}	1A	16	3	75	3.7E-3	4.2E-3
		1A1	16	3	76	2.7E-3	3.8E-3
		1B	3.9	3	55	1.2E-2	9.9E-3
		1B1	3.0	2.5	56	1.6E-2	1.0E-2
		1B2	3.3	3	64	8.2E-3	4.3E-2
		2	15	8	99	2.0E-3	4.2E-3
		3	11	8	62	2.4E-2	1.5E-2
		3A1	11	8	58	1.2E-2	6.9E-3
		3A4	11	8	71	1.5E-2	9.9E-3
		4	15	17	99	2.2E-3	5.2E-3
1-REL-LCF-SC	3×10^{-6}	2R2	15	8	128	6.0E-4	1.7E-3
1-REL-ICF-BURN	9×10^{-6}	1A2	16	3	28	4.3E-2	3.2E-2
1-REL-ICF-BURN-SC	2×10^{-6}	1A2 (truncated at ~28 hrs)	16	3	28	6.9E-5	5.9E-5
1-REL-BMT	8×10^{-7}	6	15	13	Never	6.4E-5	5.4E-5
		6A	15	13	Never	4.4E-6	3.0E-6
		6B	15	13	Never	8.2E-5	7.9E-5
1-REL-NOCF	2.4×10^{-5}	1	139	3	Never	1.1E-4	7.4E-5
		2R1	15	8	Never	8.5E-5	7.4E-5
		6R1	15	13	Never	1.4E-5	1.2E-5
Total[4]	**7.0×10^{-5}**						

[1] Values less than 0.1 percent of total release frequency are listed as being less than 7×10^{-8}/rcy (0.1 percent of total release frequency), so as not to over-state the accuracy of the model. All frequencies represent the longest accident termination time considered (i.e., 3 days for ISLOCAs and 7 days for other accidents).

[2] This represents the time at which peak nodal clad temperature exceeds 1204 degrees Celsius (2200 degrees Fahrenheit).

[3] This is based on the time at which integral environmental xenon class release exceeds 10 percent, or iodine class release exceeds 1 percent (whichever comes first) to provide a sense of the delta between the initial gap release from the hottest group of fuel assemblies versus the time at which a notable fraction of the noble gasses or volatiles reaches the boundary with the environment.

[4] Some inflation is expected in comparing the total release frequency to core damage frequency. Potential non-minimal cut sets that are reduced at the core damage level can be mapped into different release category sequences and would not be reduced in the release category quantification. Large failure probabilities and treatment of success terms can also influence the quantification accuracy.

Table 2-28: Release path characterization

Containment Failure or Bypass Mechanism	Release Path Description	Applicable Release Categories
Normal leakage	The release path is modeled with multiple flow paths distributed throughout the containment shell above grade with the total flow area adjusted to match the maximum allowable containment leakage flow rate.	1-REL-NOCF 1-REL-BMT[1]
Overpressure failure due to energetic phenomena	The release path is modeled as an opening at the junction of the containment wall to the basemat leading to the tendon gallery. The flow area is equivalent to an 8-inch diameter opening. The tendon gallery connects to the environment, as described in Appendix D.	1-REL-ECF 1-REL-ICF-BURN 1-REL-ICF-BURN-SC
Overpressure failure due to gradual overpressure	The release path is modeled as an opening at the junction of the containment wall to the basemat leading to the tendon gallery. The flow area is variable depending on containment pressure, reaching a maximum of one square feet, or approximately 13.5-inch diameter. The tendon gallery connects to the environment, as described in Appendix D.	1-REL-LCF 1-REL-LCF-SC
Containment isolation failure	The release path is modeled as an opening in the lower containment volume leading directly to the environment. The flow area is equivalent to an 2-inch diameter opening.	1-REL-CIF 1-REL-CIF-SC
Steam generator tube rupture	The release path is modeled as a tube break resulting in primary to secondary side flow in the loop 1 steam generator with a total area of 0.29 square inch. The release to the environment is through cycling open the atmospheric relief valve and assumed secondary side leakage (0.5 square inch per steam generator) caused by cycling of secondary side valves and presumed failure to completely re-seat.	1-REL-SGTR-C 1-REL-SGTR-O 1-REL-SGTR-O-SC
Thermally-induced steam generator tube rupture	The release path is modeled as a tube break resulting in primary to secondary side flow in the loop 2 steam generator with a total area of 0.67 square inch. The release to the environment is through cycling open the atmospheric relief valve (when operating power is available) and safety valves, and assumed secondary side leakage (0.5 square inch per steam generator) caused by cycling of secondary	1-REL-ISGTR

Table 2-28: Release path characterization

Containment Failure or Bypass Mechanism	Release Path Description	Applicable Release Categories
	side valves and presumed failure to completely re-seat.	
Interfacing systems LOCA (auxiliary building intact)	The release path is modeled as a 4-inch diameter break in the RHR system located in the auxiliary building and connected to RCS loop 4 hot leg. The release to the environment is through a filtered exhaust as part of the buildings filtration and exhaust system. Leakage to the environment is also modeled, but the exhaust system maintains the building below atmospheric pressure causing inflow from the leakage pathways.	1-REL-V
Interfacing systems LOCA (auxiliary building failed)	The release path is modeled as a 8-inch diameter break in the RHR system located in the auxiliary building and connected to RCS loop 4 hot leg. The auxiliary building pressure exceeds the building failure pressure resulting in an unfiltered opening to the environment.	1-REL-V-F 1-REL-V-F-SC

[1] Release category 1-REL-BMT includes containment failure due to radial erosion of the reactor cavity wall. However, the releases to the environment for airborne releases are dominated by the normal containment leakage pathway. Only airborne releases are passed to the offsite consequence analysis.

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Appendix A: Model Errors and Enhancements

This Appendix contains known model errors and potential model enhancements, and in concert with the uncertainty analysis, the items listed in **Section 1.2**, and the outcomes discussed in **Section 2.6**, provides perspective on the limitations of the Level 2 PRA. The following tables group entries by the associated technical elements, using the following identifiers (from the Level 2 PRA standard):

- L12 – Level 1/2 PRA Interface
- CP – Containment Capacity Analysis
- SA – Severe Accident Progression Analysis
- PT – Probabilistic Treatment of Accident Progression
- ST – Radiological Source Term Analysis
- ER – Evaluation and Presentation of Results
- L3 – Level 2/3 PRA Interface

Table A-1 provides a list of known modeling errors that were not corrected, due to the point at which they were discovered during the model development relative to the amount of re-work that would be required to correct them, along with an assessment of the acceptability of their impact.

Table A-1: Known Modeling Errors

Description	Expected Impact	Level of Effort	Notes
L12 – The modeling assumes that safety-related batteries will deplete after 4 hours, without consideration of load shedding; in retrospect, shedding of non-essential loads is required to extend battery life from 2 to 4 hours.	Low	Medium, in light of the HRA work and re-quantification	The impact of this is limited by the lack of offsite power recovery after 2 hours, the lack of credit for post-core damage operator actions during station blackout conditions, and the high HEP for blind feeding using TDAFW during station blackout conditions.
SA – The pressure relief tank (PRT) has an adiabatic boundary condition in the MELCOR model that will affect its heatup, and the subsequent re-vaporization of fission products should the PRT dry out.	Low	Low to correct; High to re-run all analyses	The basis for assessing the impact as low is provided in Sections 4.4 and 8.1.4 of Appendix C and has to do with the fact that the PRT only dries out in a couple of the MELCOR analyses.

Table A-1: Known Modeling Errors

Description	Expected Impact	Level of Effort	Notes
SA – A discrepancy between the default CORSOR Booth release rate model for cesium and the final SOARCA best practices [Ross, 2014] was observed in the version of the MELCOR code used in the L3PRA project. In a later revision of the code, sensitivity coefficients were modified to be consistent with the final SOARCA best practice report. The SOARCA best practice values predict a faster release rate, particularly at lower temperatures.	Low	Low to correct; High to re-run all analyses	The basis for assessing the impact as low is from two offline sensitivity calculations. These showed small changes in release fractions of volatile fission products to the environment, which were within the model uncertainty.
SA – Several model corrections and numerical improvements to how MELCOR treats quenching were developed and implemented that have significantly improved the robustness of the code for reflood conditions and were not reflected in the MELCOR version used for the Level 3 PRA Project. In addition, the Lipinski dryout model was also incorrectly applied above the core support plate with upward flow of coolant and would lead to problems when the occurrence of particulate debris, along with intact components, would stop convective heat removal from the intact components in a core cell. These code revisions can lead to variations in accident progression and the output parameters (e.g., such as in-vessel hydrogen production).	Medium	High to re-run all analyses with new version of MELCOR	The uncertainty brought on by these corrections is tied to the issues of in-vessel hydrogen generation and accumulator injection considered in Section 4.4 of Appendix C.
PT – CET sequence number 74 should be assigned to the 1-REL-CIF-SC release category rather than the 1-REL-CIF release category, since in-vessel recovery occurs.	Large for 1-REL-CIF-SC frequency; negligible otherwise	Low to correct; Medium to re-quantify and update documentation	The change would raise 1-REL-CIF-SC frequency by a couple of orders of magnitude, but it would still be a very small contributor (<0.1% of total release frequency), and still smaller than the more risk-significant 1-REL-CIF (0.1%). In the current quantification, this sequence has a frequency of 5×10^{-9}/rcy.
PT – The uncertainty distribution for 1-L2-BE-RCP480GPM-DEP was given an error factor of 10 (see Table 2.8 of Appendix C), but per the specified scheme it should have been given an error factor of 5.	Small	Low to correct; Low to re-quantify and update documentation	It's unlikely that this change would tangibly manifest in the uncertainty propagation.

Table A-2 identifies potential model enhancements that were not included in the L3PRA project because they were not expected to have a significant impact on the model results and, as such, the costs associated with including them were not justified.

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
L12 – The binning of consequential LOOPs that further degrade to SBO events is hard-wired into the ACCTYPE linkage rules (since the Level 1 modeling addresses consequential LOOP with fault tree logic rather than event tree logic), and is based on a specific set of Level 1 results from a preliminary version of the model, and thus does not conform to the current Level 1 model (for both the baseline model and sensitivity analyses).	Variable	High	The impact depends on specifically what fraction of the consequential LOOPs resulting in SBO events are not captured by the current Level 1/2 linkage rules. The specific impacts have not been further explored. Improving the situation requires significant overhaul of the Level 1 PRA and the Level 1/2 PRA interface.
L12 – Since offsite power recovery is handled at the event tree level in the Level 1 PRA, the bridge tree (containment systems) are not aware of instances where AC power has been recovered, leading to instances where containment sprays and fan coolers are treated as unavailable, despite being potentially available (i.e., the fault tree logic for containment sprays and fan coolers will result in these systems being modeled as failed for cut sets that include loss of AC power, even if offsite power is subsequently recovered).	Variable	High	This issue is not straight-forward to resolve in the current modeling construct. However, it is greatly mitigated by the dominance of an operator action to restore systems (thus the cause of core damage despite power recovery), such that treatment of the containment systems as unavailable is typically the correct assumption.

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
L12 – The credit for indefinite (up to 5 days) extension of blind feeding SGs using TDAFW during certain slow-developing SBO sequences uses a simplified, parametric approach. A reference HEP value of 0.3 for blind-feeding SGs post-battery depletion was considered in the Level 1 PRA (though ultimately not credited). For the Level 2 PRA, the reference HEP was scaled to account for the extended time that is predicted for core damage beyond the 24-hour mission time considered in the Level 1 PRA. For this scenario, the MELCOR analysis predicts core damage at approximately 140 hours. The HEP was also scaled to account for the reduced sensitivity to under-filling or over-filling for the later period of the accident progression, where decay heat loads have decreased. The HEP was scaled by the ratio of the estimated TDAFW flow rate requirements for the pre-24-hour and post-24-hour periods. The resulting HEP used in the Level 2 PRA for extended blind feeding of the SGs is 0.65.	Medium	Medium, in light of the HRA work and re-quantification	Applying this credit for extending blind feeding SGs has the following impacts: (a) it provides significant credit for avoiding containment failure, and (b) it counteracted the lack of SBO HRA credit post-core damage. Additional development to improve the basis of this reliability estimate would improve the accuracy of the results.
L12 – Modeling of containment isolation failures could be improved by additional effort associated with the screening/inclusion of failure paths and further discretization of the assumed isolation failure size.	Medium	Low, in that alternative failures have undergone some analysis; High to implement additional failure sizes	More on the assumptions related to isolation failure modeling can be found in Appendix D in Section 2, “Treatment of Bypass and Unisolated Containment Events with Low Frequency,” and Section 6, “Containment Leakage, Effective Size Under Normal and Accident Conditions.” Appendix C provides a sensitivity analysis in Section 4.12.

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
CP – The assumption that gradual over-pressure failure occurs in to the tendon gallery (as opposed to the auxiliary building) has a large effect on the characterization of the long-term accident response, and the cumulative radiological releases. Related to this, the precise nature of the failure (e.g., geometry of the liner tear, extent of concrete fracturing and resulting surface area) has the potential to enhance fission product deposition (and therefore reduce environmental releases), which is not accounted for in this study. Reduction in the uncertainty associated with the containment failure path detailed characterization would improve the usefulness of the results for developing insights on long-term accident management and contaminated water mitigation.	Variable	High	This issue is further discussed in Appendix D, Section 20, "Tendon Gallery Release Pathway."
SA – The treatment of equipment survivability and instrument degradation in the Level 2 PRA model for the L3PRA project is fairly simplistic owing to limited availability of relevant plant-specific information, the level-of-detail of accident management modeling, and the level of project resources.	Low	Medium	The potential impact on the results is not well-understood, though it generally should be limited by the degree of HRA credit given and the dominance of SBO initiators. This is a topical area that would benefit from additional investigation. This issue interrelates with limitations in the HRA modeling, for the aspects related to instrumentation availability and usage.
PT – Some of the phenomenological evaluations (most notably combustion and vessel thrust) were not re-computed as part of the Level 2 PRA update, due to limitations on the available contracting resources at the time the work would have been done.	High for combustion; Low for vessel thrust	Medium	The potential impact on combustion is further explored in Section 4.7 of Appendix C.

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
PT (Logic model) – The basic event descriptions for 1-L2-BE-H2IGNSRC-E-AC, 1-L2-BE-H2IGNSRC-E-NAC, 1-L2-BE-PZRVSTUCK-PORV, and 1-L2-BE-PZRVSTUCK-SRV are misleading, by virtue of being stated in a way that makes it sound like the detrimental event did not occur when in fact it did.	None	Low to correct; Medium to re-quantify and update documentation	To be consistent with the general paradigm (and the analogous other basic events), these descriptions should be reversed (i.e., made into positive statements) such that their appearance in a cutset (as other than a success term) will state the failure that did occur.
PT (Logic model) – The basic event description for 1-L2-BE-CCI-DISP, "MCCI occurs - Debris is not dispersed despite HPME," is misleading in that it applies to both high-pressure and intermediate pressure situations, by virtue of its reliance on 1-L2-VBMODE which applies to both high/intermediate pressure.	None	Low to correct; Medium to re-quantify and update documentation	
PT (Logic model) - The treatment of IVREC in 1-CET may result in cut sets (and certainly results in sequences) where operator-action to depressurize is credited in the Level 1 PRA (denoting that either accumulators dumped or were isolated), and then the accumulators are re-credited here as arresting core damage.	Moderate	High	Addressing this limitation would require revising the PDS binning or directly relying on Level 1 sequence information in the IVREC linkage rules. In-vessel recovery has a notable but not dominant effect on the results (it would tend to decrease NOCF and increase several other release categories proportionately).

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
PT (Logic model) – During high/dry RCS conditions, the logic model assumes that the SGs are depressurized (i.e., high/dry/low), which will tend to inflate the estimation of TI-SGTR.	Low	Medium	There are a handful of reasons why the SGs are expected to be depressurized by the point in the accident when TI-SGTR is of concern: • The SGs are unable to hold pressure because they have boiled dry due to insufficient feedwater or they are no longer steaming • The secondary-side has been deliberately depressurized by operator action prior to core damage • A secondary-side valve has stuck-open Implementing these various conditions would be complex, and the expected benefit is small. For simplicity, it is assumed in the L3PRA project that secondary-side pressure is always low by this point in the accident.
PT (Logic model) – Presently the model does not account for the higher probability of an ignition source prior to vessel breach during station blackouts if the hot leg experiences creep rupture (top event 1-L2-H2IGNSRC-VE).	Medium	Medium	Addressing this limitation would require changes to the model structure, since the top event in question precedes the determination of hot leg creep rupture (which exists in a subsequent DET). The main effect would be to consume some hydrogen early (though not enough to fail containment), and thus potentially reduce the amount present later.
PT (Logic model) – There are instances where the logic model assumes that alternate feedwater is feasible, in implementing the results of the HRA. The Level 1/2 interface does not preserve information that is only accounted for in Level 1 fault trees because of the mechanics of using linkage rules (that can only access sequence information).	Low	High	This is not expected to have a significant effect, because the Level 2 HEPs are already very high and alternate feedwater would be feasible for a broad range of non-SBO conditions. Resolving this limitation would be very difficult, and perhaps practically impossible with the current modeling tools.
PT (Uncertainty) – There are a handful of sensitivity analyses that were not completed in Appendix C because of their complexity or computational burden, relative to other competing demands.	Low	Medium	These sensitivity studies would provide additional information about model uncertainty but are not expected to have a fundamental impact on the overall understanding of the Level 2 PRA's uncertainty.

Table A-2: Model Enhancements Excluded Due to Time and Resource Limitations

Description	Expected Impact	Level of Effort	Notes
ST – The early GE declaration (due to SBO conditions) for the ICF-BURN source term (1A2) causes it not to meet the LERF criteria, whereas it might otherwise (for at least the early injuries version). Given that non-SBO cutsets contribute significantly to this release category (e.g., 4 of the top 10 cutsets), some additional LERF contribution could reside within this release category.	Medium	High	Resolving limitations of the mapping of cutsets to release categories would be difficult. Further refining the cutset contributions may not be practical.
L3 – Related to an over-arching issue of the simulation end-time selection, there may be value in performing the consequence analysis by processing one or more release categories for the three different simulation truncation times considered in this report, to further illustrate the effect that this modeling decision has on the predicted results.	Variable	High	There are practical and resource constraints that will limit the level of effort that can be applied to the consequence analysis to address this issue.
Several analysis areas contribute to overall model uncertainty. In some cases, the assessments are limited by the state-of-practice in the area, the availability of plant-specific information, or the availability of qualified staff to work on these assessments. Examples of such areas include: severe accident-induced ISLOCAs, containment isolation failures that could result in non-bypass events degrading to bypass events, re-criticality during reflood (which was not explicitly modeled), and late combustion-induced containment failures.	Variable	High	Several of these issues are discussed further in Appendix C, which describes the treatment of uncertainty. Further work in investigating these issues would improve the confidence in the findings of this study and the relative importance of the issues.

Table A-3 identifies potential model enhancements that may apply more broadly to PRAs in general (beyond the Level 3 PRA project) and/or require additional research or discussion with the PRA technical community.

Table A-3: Model Enhancements Requiring Evolution of the State-of-Practice

Description	Expected Impact	Level of Effort	Notes
L12 – The scheme used for PDS binning (i.e., detailed linkage rules) achieved the primary motivators for its adoption over other means (i.e., an integrated Level 1 and Level 2 PRA model that was developed and executed similarly to how Level 1 PRA SAPHIRE models are developed and executed), but comes at the expense of being onerous to develop and difficult to maintain (any Level 1 sequence logic change has the potential for breaking this linkage).	Low	High	Efforts to further automate this process would greatly assist model maintenance and future upgrade.
SA/PT - Effects of inadvertent criticality	Low	High	The current study does not include consideration of inadvertent criticality during core reflood, which is only of relevance for situations where the core is reflooded with unborated water following significant heatup of the fuel (to the point of melting poison material) but prior to core relocation.[29] Based on the results of the HRA/PRA, the only situations where in-vessel recovery is currently credited in this study are a subset of the cases where operator action to depressurize an otherwise medium or high-pressure sequence leads to cold-leg accumulator injection (borated water) and subsequent secondary-side decay heat removal (Top Event 1-L2-IVREC). Even there, the failure probability is currently very high owing to phenomenological considerations related to coolability. This issue is discussed further in prior studies (e.g., [Wagner, 2012]).
SA/PT - Additional quantitative equipment survivability assessment	Medium	High	This issue is discussed further in **Section 2.3.6**.

[29] The current general accident management approach is to inject water regardless of re-criticality potential, though this potential might affect the flow rate used. Re-criticality is a possibility for ATWS and non-ATWS accidents.

Table A-3: Model Enhancements Requiring Evolution of the State-of-Practice

Description	Expected Impact	Level of Effort	Notes
PT – The HRA (especially the consideration of human actions in SBO conditions), the lack of consideration of repair and recovery of equipment (including restoration of AC power after battery depletion), and the treatment of accident termination/truncation are limited by the state-of-practice and general modeling scope.	Variable	High	Additional work in these areas, particularly for SBO conditions, could produce additional realism in the model.
PT – The probabilistic model is extremely computationally intensive. There were challenges in solving he model primarily due to the large number of accident sequnces in the fully linked Level 1/2 model. The model was solved in a piecewise fashion by grouping initiating events in small groups and combining the dominant results. The reduced quantification in most instances captured at least the top 99.94% of CDF, but the process required roughly a week to perform. The size of the model challenged the limitations of the software, and periodic software updates were required to expand model size capacity and correct software errors to ensure stability of the results.	Low	High	Further investigation is needed to reduce this computational burden. The quantification process has improved with software updates, but additional improvements could be made. One possible approach could be to disable the CET sequences that have been demonstrated to have little or no contribution, thus significantly reducing the number of sequences to be quantified.
PT – The modeling of relief valve performance under core damage conditions, and the applicability of valve test data to PRA reliability estimates, are areas of ongoing deliberation.	Low	Medium	This refers in large part to the ongoing work at NRC under the Peach Bottom, Surry, and Sequoyah SOARCA uncertainty analyses. The impact could be significant for accident sequences, but is expected to be low overall, in part due to the low estimated contribution of TI-SGTR and HPME effects.
PT – Human reliability analysis is not considered for post-core damage station blackout conditions, due to the lack of a reliable means of predicting SAMG-driven plant response in situations where no instrumentation is available.	Variable	High	This also relates to the lack of credit for restoration of AC or DC power following their loss (if after switchyard battery depletion). See Section 4.3 of Appendix C to this report for more context.

Table A-3: Model Enhancements Requiring Evolution of the State-of-Practice

Description	Expected Impact	Level of Effort	Notes
ST - Impacts of aqueous releases	Low based on reference plant information	High	This is a fundamental scope limitation of this study and is felt to have low impact due to (i) the historical perspective on airborne releases dominating public health impacts and (ii) the large standoff distance between the nuclear island and the adjacent body of water. Site layout vulnerabilities in this regard (e.g., ingress into, and transit through, the circulating water tunnels) have not been investigated.

Appendix B: Description of Case Studies for the Reactor, At-Power, Level 2 PRA for Internal Events and Floods

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Introduction

As described in **Section 2.1.5** of the main body of this report, steps were taken to place the tens of thousands of cutsets into bins of a smaller group (hundreds) of plant damage states (PDSs). From each of the significant PDSs, one (or a few) "representative" sequences were selected for deterministic treatment in the Level 2 probabilistic risk assessment (PRA). These sequences are surrogates for the numerous cutsets that, while carried forward in the integrated probabilistic model, are not explicitly modeled by deterministic accident progression analysis. Therefore, it is important that they reflect the general characteristics of the numerous cutsets. The role of the representative sequences is discussed further in **Section 2.1.2** of the main body of this report.

While the representative sequences are intended to encompass the individual traits associated with the Level 1 PRA cutsets, across the full scope of a Level 2 PRA, it is often difficult to anticipate the effects of an assumption. Therefore, judgment was necessarily a part of representative sequence selection. In some cases, variability was addressed through the identification of new sources of model uncertainty (see **Section 2.4.7** of the main body of this report) or through the identification of sensitivities. PDSs were selected for deterministic evaluation if they:

- Comprise an important portion of PDS frequency
- Were of potentially high conditional consequence based on projection of release magnitude or timing
- Illustrated or yielded data or phenomenological insights (e.g., combustible gas accumulation or reactor coolant system (RCS) piping creep rupture behavior) into items of interest to the containment event tree (CET) modeling.

The representative sequences are comprised of various station blackout sequences, dual-train electrical transients, loss of nuclear service cooling water sequences, and interfacing system LOCAs. Except for interfacing system LOCAs (which remain of interest due to being bypass events), these contributors remain important to the PDS results.

There were eight base representative sequences identified, with a total of 22 additional permutations and 6 recovery cases simulated to fill all information needs related to development of the probabilistic model and population of release category representative source terms. The full list of cases associated with the representative sequence selection is provided in **Table 0-1** below.

Table 0-1: Representative Sequence and Sensitivity/Recovery Case Descriptions

Case #	Description
1	Station Blackout with 21 gpm per Reactor Coolant Pump Seal loss of coolant accident (LOCA), Indefinite Auxiliary Feedwater, and Rapid Depressurization
1A	Base-Case station blackout (SBO) with Eventual Loss of auxiliary feedwater (AFW)
1A1	Base-Case SBO with Eventual Loss of AFW, and Suppressed Deflagrations
1A2	Base-Case SBO with Eventual Loss of AFW and Late Combustion-Induced Containment Failure
1B	Base-Case SBO with Initial Loss of AFW, and No Depressurization
1B1	Base-Case SBO with Initial Loss of AFW, No Depressurization, and 182 gpm per reactor coolant pump (RCP) seal LOCA
1B2	Base-Case SBO with Initial Loss of AFW, No Depressurization, and Stuck-Open power-operated relief valve (PORV)

Table 0-1: Representative Sequence and Sensitivity/Recovery Case Descriptions

Case #	Description
2	Transient Induced by Total Loss of Nuclear Service Cooling Water, 182 gpm per RCP Seal LOCA, Auxiliary Feedwater and Controlled Depressurization
2R1	Base-Case with severe accident mitigation guideline (SAMG) prompted Additional Secondary-Side Cooldown During Core Damage
2R2	Base-Case with SAMG-prompted Firewater-based Containment Spray Following Vessel Breach
2A	Base-Case with Containment Failure Forced at the Time of Vessel Breach
3	Transient Initiated by Loss of Main Feedwater, Auxiliary Feedwater Lost at 3 Hours, and emergency core cooling system (ECCS) Unavailable
3A1	High-Pressure Transient, with Instrument Tube Failure
3A2	High-Pressure Transient, with Induced Rupture of Steam Generator Tubes (SGTs)
3A3	High-Pressure Transient, with Induced Ruptures of SGTs and Hot Leg Nozzle
3A4	High-Pressure Transient, with All Induced RCS Failure Paths Disabled
4	Transient Induced by Electrical Distribution and nuclear service cooling water (NSCW) Failures, 182 gpm per RCP Seal LOCA, Auxiliary Feedwater, and Controlled Depressurization
5	Interfacing System Loss of Coolant Accident (ISLOCA) with Submerged Break
5A	Base-Case ISLOCA but with Uncovered Break
5B	Base-Case ISLOCA but with Double-Ended Eight-Inch Break
5C	Base-Case ISLOCA with Plugging of piping penetration area filtration and exhaust system (PPAFES) Filters
5D	Base-Case ISLOCA but with Double-Ended Eight-Inch Uncovered Break
6	Transient Initiated by Loss of Offsite Power, Auxiliary Feedwater Lost at 6 Hours, ECCS Available, and Containment Cooling Available
6R1	Base-Case with SAMG-prompted Low Pressure Injection Initiated During Core Damage
6A	Base-Case with Containment Sprays Actuating After Core Damage
6B	Base-Case with Deflagrations Suppressed
6C	Base Case with Containment Failure Forced at the Time of Vessel Breach
6D	Case 6A with Containment Failure Forced at the Time of Vessel Breach
7	Station Blackout with 21 gpm per RCP Seal LOCA, Auxiliary Feedwater Lost at 4 hours, and Containment Isolation Failure
7A	Base Case with Portable Pump Injection through Containment Spray Lines
8	Un-isolated Steam Generator Tube Rupture with Auxiliary Feedwater
8R1	Base-Case with SAMG-prompted Flooding of Ruptured steam generator (SG) During Core Damage
8R2	Base-Case with SAMG-prompted Flooding of Ruptured SG Following Vessel Breach
8A	Base-Case SGTR with AFW Supplied to Affected Steam Generator
8B	Base-Case SGTR with Stuck-Open Relief Valve in Affected Steam Generator
8BR1	Case 8B with SAMG-prompted Flooding of Ruptured SG During Core Damage

1. Scenario 1: Station Blackout with 21 gpm per Reactor Coolant Pump Seal LOCA, Indefinite Auxiliary Feedwater, and Rapid Depressurization

The scenario is initiated by a station blackout. Reactor cooling pump (RCP) seal leaks begin at the start of the transient with nominal rate of 21 gpm per pump. The turbine-driven auxiliary feedwater (AFW) pump is available at all times, and the condensate storage tank (CST) is credited with unlimited refills, if required. Rapid depressurization of the steam generators begins at 30 minutes. At this time, only the steam generator of the pressurizer loop is depressurized. To accomplish the depressurization, the atmospheric relief valves (ARVs) in the pressurizer-loop steam generator is held fully open until the steam generator pressure falls below 300 psig. At all later times, the ARV is cycled (fully open or fully closed), with deadband, between setpoints of about 315 psig (assumed) and 300 psig. At 35 minutes, the same procedure is initiated in steam generator 2. At 45 minutes, the same procedure is initiated in steam generator 3. At 50 minutes, the same procedure is initiated in steam generator 4. The staggering of ARV opening times mimics the local actions required. Like all the scenarios considered in this project, except for Scenario 7, the containment is initially isolated. (Scenario 7, on the other hand, is identical to Scenario 1, except for the initial failure of the containment to be isolated and a limitation on the duration of AFW.) **Table 1-1** provides the details of the conditions applicable to this accident scenario, and includes the definition of several sensitivity cases.

Table 1-1: Specific Conditions Applicable to Accident Scenario 1

Description	Station Blackout. Initiated by "sunny day[1]" loss of offsite power (LOOP), with independent common cause failure (CCF) to open of both switchyard alternate current (AC) breakers.
Reactor	Reactor trips at time zero due to loss of power.
RCPs	Pumps trip at time zero due to loss of power.
Break	RCP seal leaks of size 21 gpm/RCP from the start of the accident.
Pressurizer Power Operated Relief Valves (PORVs)	Both PORVs are permitted to cycle normally while battery power is available (first 4 hours), after which they fail closed. All three safety valves (SVs) are permitted to cycle normally throughout the accident.
ECCS	None.
Feed Water	Turbine-driven AFW functions for the duration of the scenario under manual control following battery depletion, or until CST water is exhausted. CST refill (or swap to alternate CST) is credited, if relevant.
Steam Generator Valves and Steam Line	ARVs cannot cycle automatically due to loss of electrical power (although they are available for manual use in secondary-side depressurization; see below). All main steam safety valves (MSSVs) are permitted to cycle normally. Main steam isolation valves (MSIVs) close at time zero on loss of power/instrument air.
Containment Sprays/Coolers	None.

1 This term is used here to denote a plant-centered, switchyard-centered, or grid-related LOOP, as opposed to a weather-related LOOP. In terms of the MELCOR analysis they are all the same, but in terms of any credited local operations actions or the offsite consequence modeling, they may not be.

Table 1-1: Specific Conditions Applicable to Accident Scenario 1

Containment Isolation	Isolated.
Operator Actions	SG ARVs are used to depressurize all SGs to between 200-300 psig[2]; valve(s) are (locally) opened at t = 30 minutes and t = 35 minutes for the first two ARVs, and t = 45 minutes and t = 50 minutes for the final two, if used. Manual feedwater control to maintain SG normal range (NR) level > 10%
Sensitivity Cases	1A: Same event, but turbine-drive auxiliary feedwater (TDAFW) trips on over-speed upon loss of direct current (dc) power, and operators fail to manually operate TDAFW thereafter; ARVs close on loss of dc power and remain closed. 1A1: Disable burns in the containment, to provide boundary conditions for any separate consideration of hydrogen combustion using assumptions other than MELCOR's defaults. 1A2: Same as 1A, but containment is assumed to fail due to a late combustion event. 1B: Same event, but TDAFW is totally unavailable from the start of the accident due to independent mechanical failure. Depressurization as noted under "Operator Actions" is not attempted given unavailability of feedwater. 1B1: Same event as in 1B, but a 182 gpm/RCP seal leak develops at t=13 minutes. 1B2: Same event as in 1B, but with the assumption that a pressurizer safety valve fails open on either the first cycle in which liquid flows through the valve or the 251st cycle in which only gas flows through it.

1.1 Sensitivity Case 1A

In the Base Case, after the rapid depressurization ends at about 1 hour, each ARV opens and closes indefinitely to maintain the corresponding steam generator at pressures between 300 and 315 psig. Also, AFW is supplied continuously throughout the accident as demanded to maintain the water level in all steam generators. These conditions are reconsidered for sensitivity Case 1A: after 4 hours AFW is assumed to be unavailable, and all ARVs are closed when steam generator water level falls below 10% narrow range. Otherwise, Case 1A is the same as the Base Case.

1.1.1 Sensitivity Case 1A1

Case 1A1 is the same as Case 1A except that deflagrations are globally suppressed. Since in Case 1A there are no deflagrations before hot leg creep rupture, the results of Cases 1A and 1A1 are identical up to and including that event.

[2] This human failure event has a low likelihood of failure in the L3PRA project "R01" Level-1 model (1-OPR-XHE-XM-RSSDEP; HEP = 1E-3) and there are no assumed equipment failures that would prevent its success here.

1.1.2 Sensitivity Case 1A2

Case 1A2 is the same as Case 1A except that containment is assumed to fail at 28 hours. Note that containment steam concentration is between 55% and 60% at this time, so MELCOR considers containment to be steam inerted (because steam concentration is greater than 55%). However, there is uncertainty in both the flammability limits and the calculated steam concentration in containment, so hydrogen combustion around 28 hours is not unreasonable, given the composition of the containment atmosphere.

1.2 Sensitivity Case 1B

Case 1B is like Case 1, except: AFW is unavailable at all times, depressurization of the steam generators does not occur, and at all times the steam generators are relieved by SRVs, not ARVs.

1.2.1 Sensitivity Case 1B1

In Case 1B1, the RCP seal leakage area is assumed to increase at 13 minutes, such that the per-pump leakage rate reaches 182 gpm at normal primary system pressure. Otherwise, Case 1B1 is the same as Case 1B.

1.2.2 Sensitivity Case 1B2

This case is identical to Case 1B except that one pressurizer PORV is assumed to stick-open at 2.0 hours, at the time of initial flow of liquid water through the valve.

2. Scenario 2: Transient Induced by Total Loss of Nuclear Service Cooling Water, 182 gpm per RCP Seal LOCA, Auxiliary Feedwater and Controlled Depressurization

The scenario is initiated by loss of the Nuclear Service Cooling Water (NSCW) system. The model does not represent the NSCW system directly; however, it is assumed that all fan coolers stop running at time zero. Otherwise, a normal full-power situation holds until 5 minutes. Manual reactor scram and manual trip of the RCPs occur at 5 minutes into the accident. ECCS and containment sprays are not available as of 5 minutes. By scenario definition, RCP seal leaks begin at 43 minutes (30-minute heatup of auxiliary component cooling water [ACCW] plus 13-minute WOG 2000 delay) with nominal rate of 182 gpm per pump. Normal charging (with letdown, acting during times between reactor scram and ECCS signal to control the pressurizer level to 25% of indicated range), previously available, becomes unavailable at 30 minutes. Two motor-driven AFW (MDAFW) pumps are available at all times, and the CST is credited with unlimited refills if required. Controlled depressurization (via the steam generators) begins 30 minutes after the ECCS signal. (ECCS itself is unavailable.) The depressurization aims at 55.56 K/hr rate of reduction in the average temperature of the liquid in the hot and cold legs (T_{avg}) by means of the ARVs in the four steam generators, whose pressure, however, is prevented to be reduced below 200 psig. **Table 2-1** provides the details of the conditions applicable to this accident scenario.

Table 2-1: Specific Conditions Applicable to Accident Scenario 2

Description	RCP Seal LOCA with Feed Water. Initiated by loss of NSCW (CCF of all 6 pumps), with developing RCP seal leak.
Reactor	Manual trip (see “Operator Actions” below).
RCPs	Manual trip (see “Operator Actions” below).
Break	RCP seal leaks of size 182 gpm per pump, developing at t = 43 minutes (following a period of NSCW heatup).
Pressurizer PORVs	Both PORVs and all three SVs are permitted to cycle normally throughout the accident. No feed and bleed.
ECCS	None (while initially available, ECCS becomes unavailable prior to actuation due to NSCW heatup).
Feed Water	Both motor-driven AFW pumps are permitted to inject to all four SGs in level-control mode until the CST is emptied. CST refill is assumed successful, if relevant.
SG Valves and Steam Line	ARVs and MSSVs are permitted to cycle normally throughout the accident. MSIVs close automatically based on Engineered Safety Feature (ESF) logic.
Containment Sprays/Coolers	None.
Containment Isolation	Isolated.

Table 2-1: Specific Conditions Applicable to Accident Scenario 2

Operator Actions	Manual trip of reactors and RCPs at t = 5 minutes Maintain SG NR level between 10% and 65% Maintain pressurizer level at 25% using the normal charging pump[3] and letdown. No other action prior to RCP seal failure, based on efforts to restore NSCW equipment A controlled cooldown (~100F/hr) commences ~30 minutes after seal leak-induced safety injection (SI) using steam dumps or ARVs[4]. When core exit thermocouple (CETC) reach 711F, an emergency cooldown begins while maintaining SG pressure > 200 psig.
Other	The normal charging pump becomes unavailable at t=2 hours due to the loss of NSCW.
Sensitivity Cases	2A: Containment fails at the time of vessel breach

2.1 Recovery Case 2R1

This scenario is initiated by loss of nuclear service cooling water, which results in a loss of ECCS and containment heat removal essentially at time zero and RCP seal leaks of 182 gpm per pump at 43 minutes. Operators take action to depressurize the steam generators to 200 psig. Feedwater is available throughout the accident, but the loss of inventory through the RCP seals and the lack of makeup from the ECCS pumps eventually results in core damage. Strategies for recovery follow Severe Accident Management Guidelines (SAMGs) as shown in **Figure 2-1**.

Before vessel failure, the highest priority strategy is SAG-2, "Depressurize RCS." Operators have several options for this strategy, the most viable of which is to fully open the steam generator relief valves. Alternative actions include opening the pressurizer PORVs or the vessel head vent valves, but both actions would release hydrogen and fission products into containment. Opening the steam generator ARVs alleviates this concern.

Although the steam generators are already at 200 psig when operators fully open the ARVs, this recovery action has a noticeable effect on the accident progression. RCS pressure immediately decreases, which allows additional accumulator inventory to inject into the RCS. In the Base Case, approximately half of the accumulator inventory injected before lower head failure. This provides some cooling for the fuel and delays vessel failure by approximately 9 hours. The action also limits containment pressurization because the steam generators remove heat from containment through the steam generator walls. This heat transfer is much greater in the recovery calculation than in the Base Case because the steam generators are saturated at a lower pressure, and the open ARVs provide a bleed path to maintain that low pressure.

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3 In this case the normal charging pump would become unavailable at 30 minutes upon effective loss of NSCW/ACCW.

4 This is a rough estimate based on procedural steps and transitions; cooldown can include temporarily opening a PORV if necessary to perpetuate RCS depressurization, but, with the lack of ECCS, it is unlikely that the associated subcooling margin requirement would be met.

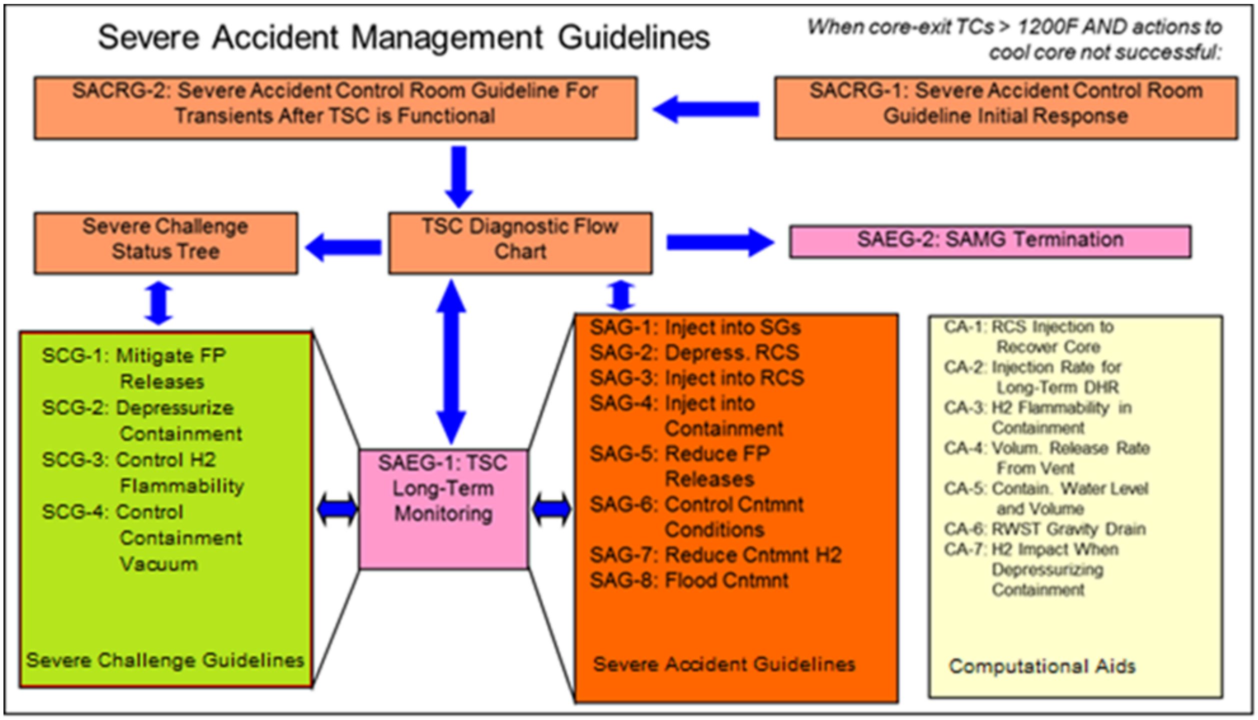

Figure 2-1: SAMG Structure

2.2 Recovery Case 2R2

This scenario is initiated by loss of nuclear service cooling water, which results in a loss of ECCS and containment heat removal essentially at time zero and RCP seal leaks of 182 gpm per pump at 43 minutes. Operators take action to depressurize the steam generators to 200 psig. Feedwater is available throughout the accident, but the loss of inventory through the RCP seals and the lack of makeup from the ECCS pumps eventually results in core damage.

After vessel failure, the highest priority strategy is SCG-1, Mitigate Fission Product Releases. In this case, releases are through normal containment leakage, so the best way to mitigate releases is to condense steam and decrease containment pressure using containment cooling systems. Containment fan coolers and spray pumps are unavailable because of the accident initiator, but the portable spray pump can be used to inject through the containment spray lines, per the relevant strategy in the Extensive Damage Mitigation Guidelines (EDMGs). This action is modeled here. Specifically, operators are assumed to begin spraying containment 2 hours after vessel failure at a rate of 350 gpm, and to stop the portable pump after injecting 600,000 gallons into containment. The volume of water injected into containment is approximately equal to the volume of water in the fire water storage tanks.

2.3 Sensitivity Case 2A

Case 2A is a sensitivity case to Case 2. In Case 2A, the containment is forced to fail (i.e., the containment failure flow path is manually forced opened) at the time of vessel breach. Otherwise Case 2A is identical to Case 2. The containment failure is modeled differently in this case than in cases where containment fails after slow pressurization due to molten core concrete interaction (MCCI). Here, a 324 cm^2 flow area leads from near the containment floor to the tendon gallery, which itself communicates with the environment and the Auxiliary Building through large flow areas. The flow area is larger in this case due to the assumed failure mode in this case (e.g. hydrogen combustion, ex-vessel steam explosion, vessel rocketing). Note that this same containment failure area is also used for Cases 1A2 and 6C.

3. Scenario 3: Transient Initiated by Loss of Main Feedwater, Auxiliary Feedwater Lost at 3 Hours, and ECCS Unavailable

The scenario is initiated by various electrical losses with consequences of reactor scram, main feedwater trip, and MSIV closure at time zero. Turbine-driven auxiliary feedwater is supplied until 3 hours; motor-driven auxiliary feedwater is never available. Normal charging (with letdown, acting during times between scram and ECCS signal to control the pressurizer level to 25%), previously available, becomes unavailable at 3 hours. ECCS and containment sprays are unavailable as of time zero, and all fan cooler activity stops at time zero. By scenario definition, RCP seal leaks begin at 3 hours 13 minutes at a rate of 21 gpm per pump. **Table 3-1** provides the details of the conditions applicable to this accident scenario.

Table 3-1: Specific Conditions Applicable to Accident Scenario 3

Description	High-Pressure Transient. Initiated by loss of one DC bus 1BD1, with 4.16 kV bus 1AA02 in maintenance (assumed here to have occurred 1 hour prior to the independent DC bus failure). Note that taking the 4.16 kV bus 1AA02 offline will lead to failure of DC bus 1AD1 and 1CD1 in 4 hours (3 hours after the independent DC bus failure), following battery depletion[5].
Reactor	Reactor trips at time zero due to closure of the main steam and main feed isolation valves upon loss of dc bus 1BD1.
RCPs	Pumps trip according to normal system logic or on cavitation condition, or once SG NR level drops below 10% in all SGs and AFW has been lost.
Break	RCP seal leaks of size 21 gpm/RCP starting at 3 hours and 13 minutes (i.e., 13 minutes after the second train of NSCW and ACCW are lost).
Pressurizer PORVs	PORV fails closed from the start of the scenario due to loss of DC bus 1BD1. The second PORV fails closed 3 hours later, as batteries deplete on DC bus 1AD1. After this time, there is no PORV function and SVs are depended upon for primary-side pressure relief. All three SVs are permitted to cycle normally. No feed and bleed.
ECCS	None. AC bus 1AA02 being in maintenance takes out Train A, while failure of DC bus 1BD1 fails Train B.
Feed Water	Turbine-driven AFW to all four SGs occurs during the first 3 hours of the accident, although steam supply to the pump's turbine from SG 1 fails due to loss of DC bus 1BD1. Loss of DC bus 1BD1 also takes out 1 of the 2 MDAFW trains. Due to the AC bus being in maintenance at time zero, the second MDAFW is also unavailable. After 3 hours, battery depletion of bus 1AD1 fails the second TDAFW steam supply valve from SG 2, resulting in loss of all AFW. While available, feed continues in level-control mode or until the CST is empty.
SG Valves and Steam Line	ARVs from SG 2 and SG 3 fail from the start of the accident due to loss of DC bus 1BD1. ARVs from SG 1 and SG 4 remain available for the first 3 hours (until the failure of DC bus 1AD1), after which they also fail due to battery depletion.

5 Having bus 1AA02 in maintenance places the unit in a 2-hour LCO and starts the nominal 4-hour clock on the associated DC buses powered by 1AA02 (based on an assumption that battery charging is not aligned from another source). Notionally, what is captured in this scenario is bus 1BD1 failing an hour after 1AA02 was taken out for maintenance, along with an assumption that the 1AA02 bus could not be restored to service afterward. This is simply one representation of numerous electrical distribution failure combinations that would have the same (or similar) Level 1 cutsets.

Table 3-1: Specific Conditions Applicable to Accident Scenario 3

	MSSVs are permitted to cycle normally throughout the accident. The four MSIVs of train B fail closed at the start of the accident due to loss of DC bus 1BD1. Steam dump to condenser is unavailable for the entire accident due to MSIV closure at time zero.
Containment Sprays/ Coolers	None. AC bus 1AA02 being in maintenance takes out Train A, while failure of DC bus 1BD1 fails Train B.
Containment Isolation	Isolated.
Operator Actions	Maintain SG NR level between 10% and 65% Maintain pressurizer level at 25% using the normal charging pump (while available) and letdown[6]. Trip RCPs if SG NR level drops below 10% in all SGs coincident with loss of all feedwater. Operators are specifically assumed not to initiate feed and bleed or perform an emergency depressurization given the lack of ECCS, and the lack of significant RCS leakage.
Other	The normal charging pump is available until t = 3 hours, unless an SI signal occurs prior to that time.
Sensitivity Cases	3A: Studies of induced RCS failures, including: 3A1: Induced ruptures of instrumentation tubes. 3A2: Induced ruptures of SGTs. 3A3: Induced ruptures of hot-leg and SGTs. 3A4: Suppressed creep rupture

3.1 Sensitivity Case 3A1

Case 3A1 considers failures of the in-core instrumentation tubes inside the core region. Otherwise, Case 3A1 is identical to Case 3.

3.2 Sensitivity Case 3A2

Case 3A2 considers induced rupture of steam generator tubes and suppresses hot leg creep rupture. The steam generator tubes (SGTs) are calculated not to be challenged by high-temperature during the entire in-vessel phase of the accident. Hot leg creep rupture occurs at 10.9 hours in Case 3, and at that time the creep damage parameters for the SGTs of all loops are all about 0.015. (Components are assumed to fail when the damage index is greater than 1.) However, the calculation of the creep damage parameters for SGTs does not include tube-to-tube temperature variations, which cause some tubes to exceed the temperature of the average tube, which is the quantity directly predicted by the model. Also neglected up to now is the effect of any pre-existing damage that may be present in some SGTs. These effects are accounted for by a stand-alone consequential steam generator tube rupture (C-SGTR)

6 The normal charging pump (unlike the 2 ECCS charging pumps) is powered by a non-1E AC source and will be available so long as ACCW is available (which in this case will be the first 3 hours). Note that an SI signal, if one occurred, would trip the normal charging pump, and it would require operator action to restart it.

calculator. The calculator estimates that there is some small (i.e. a few percent) probability that a tube would rupture in Case 3. For this reason, C-SGTR is considered for Case 3A2.

Case 3A2 includes the representation of induced rupture of the SGTs in loop 2. Based on the C-SGTR calculator results obtained during an earlier revision of the PRA model, the tubes were predicted to rupture 50 minutes before the predicted time of hot leg creep rupture. In the associated earlier MELCOR simulation there was a larger delta-time between the onset of core damage and hot leg creep rupture. Applying the same 50-minute interval to the current simulation results in SGTR occurring after the onset of significant core heatup but prior to any actual fuel damage (i.e., prior to the SGTs reaching very high temperatures); nevertheless, this time interval was still used. For this reason, it is important to keep in mind that the timing of SGTR in this simulation is somewhat arbitrary. The arguably early SGT failure, in addition to the larger SG leakage considered in the 3-series Cases, exaggerates the primary-side depressurization, which likely impacts the cumulative creep damage calculated for the hot leg nozzle and surge lines. Hot leg creep rupture is disabled for Case 3A2 and forced for Case 3A3. Otherwise, Case 3A2 is the same as the Base Case.

For this case MELCOR makes no special effort to account for enhanced aerosol retention in the secondary side of a dry steam generator. The results simply use the deposition mechanisms available in MELCOR (i.e. gravitational settling, Brownian diffusion, thermophoresis, and diffusiophoresis). However, it is expected that aerosols from a broken steam generator tube would impinge on nearby tubes, along the tubes, and at flow discontinuities caused by the tube support plates and steam separators and dryers. These mechanisms would increase deposition in – and thus decrease environmental releases from – the steam generator secondary side beyond what is calculated by MELCOR.

Deposition in the secondary side of a dry steam generator has been studied experimentally in the ARTIST program (Lind, 2012). Researchers at Sandia National Laboratories have developed models to calculate the decontamination factors at the tube support plates and in the steam generator separators and dryers based on the experimental results (hereafter referred to as the Powers model). The same decontamination factor model was used for the draft State of the Art Reactor Consequence Analysis (SOARCA) uncertainty analysis for the Surry Power Station (SNL, 2016). Variables in the Powers model include the gas superficial velocity in the steam generator secondary side, the particle density, the gas viscosity, and the number of steam generator stages (i.e. the number of support plates) above the break. The overall decontamination factor (DF) is then

$$DF_{all} = DF_{dryer} DF_{stage}^{N_{stages}}$$

where DF_{dryer} is the decontamination factor in the dryers and separators, DF_{stage} is the decontamination factor in a single stage, and N_{stages} is the number of stages. The DF is a function of particle size, so there is a DF for each aerosol size bin. The L3PRA project uses the MELCOR default of 10 size bins.

The models were implemented as MELCOR control functions for the Surry uncertainty analysis (UA) project (SNL, 2016). (The control function input for this model was provided to the authors of this report by the Surry UA project team.) For various reasons,[7] the secondary-side deposition models were not included in the MELCOR input for the L3PRA project. Instead, the models have been used to calculate decontamination factors from the MELCOR output. This allows the end user to vary the model input parameters (i.e. the number of stages, the particle density, and the gas viscosity; the gas superficial velocity is calculated by MELCOR).

DFs have been calculated for case 3A2 using the same values as the Surry UA (SNL, 2016) for the particle density (2650 kg/m^3) and the gas viscosity (1.78E-5 Pa-s) and the number of stages in the reference plant's steam generators. DFs range from 1.3 for iodine to 11.1 for rubidium.[8]

Caution must be used in applying these DFs to the results. The DFs are calculated based on the aerosol flowing through the ruptured tube(s), whereas the environmental releases represent material flowing through the steam generator SRVs or the assumed secondary side leakage path. The composition and size distribution of aerosol flowing through the ruptured tube(s) to the secondary side is not the same as the composition and size distribution that enters the environment because MELCOR calculates agglomeration and deposition using its built-in models. Accounting for the enhanced deposition would reduce overall aerosol concentration and decrease the aerosol mass median diameter.[9] This would reduce MELCOR's calculated agglomeration and deposition in the secondary side, which are clearly important based on the secondary side retentions. Secondary side retention is typically on the same order of magnitude as the environmental releases.

An alternative is to apply the DFs to the break flow paths and to then assume that anything that gets into the secondary side gets into the environment (i.e. treat the C-SGTR flow paths as the release paths and apply the DFs to them). However, as the previous paragraph states, this approach is overly conservative because it ignores other deposition mechanisms as well as changes in the size distribution due to agglomeration of aerosol particles and condensation of fission product vapors.

Note that for case 3A2, for most classes the size distribution is similar between what comes through the break and what goes to the environment. This suggests that applying the DFs directly to the environmental releases may be a reasonable approximation: it may be non-conservative with respect to the overall release, but at least it does not have a major effect on the size distribution. The exceptions are Cs and CsI; the environmental release is skewed towards the larger size bins compared to the distribution that flows through the break. This is because a large fraction of the total mass of these species is in vapor form when it flows through the break, so much of this vapor condenses onto existing particles and causes the size distribution to shift towards the larger bins between the time at which the material enters the steam generator and the time at which it is released to the environment. Even still, applying the DFs for these species is a reasonable approach, at least compared to the alternatives. This will

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7 For example, the ARTIST tests used a scaled representation of a steam generator design different from the reference plant's steam generators. The steam generators are conceptually the same, but there are differences in geometry that could have a significant impact on aerosol deposition. There are also considerable uncertainties introduced in extrapolating the results of a scaled experiment to a full-scale plant accident situation.

8 The calculated DFs include the mass of fission product vapors. The Powers model only affects aerosols. This is particularly important for cesium iodide because a substantial portion of the release is in vapor form. This helps explain why cesium and iodine DFs are on the lower end of the calculated spectrum.

9 As mentioned, the DF is a function of particle size. Almost all of the material in the largest size bins (100s of microns) is predicted to be deposited in the steam generator. In contrast, almost none of the material in the smallest size bins (submicron) is expected to be removed. Thus, there is a shift in the aerosol size distribution.

likely result in slightly non-conservative results because it is likely over-estimating the retention in the SGs for the lower aerosol concentrations that would be present had MELCOR explicitly included the enhanced deposition model, but the error is probably less than the uncertainty in the results.

3.3 Sensitivity Case 3A3

Case 3A3 considers induced rupture of SGTs, followed by induced rupture of the hot legs nozzles. Like Case 3A2, it supposes that steam generator tubes rupture 50 minutes before the time of hot leg creep rupture in Case 3, and that the break flow area is 4.3 cm^2. Unlike Case 3A2, the hot leg is assumed to fail at the same time as in Case 3 (i.e. 50 minutes after C-SGTR). The timing of the SGTR and the number of tubes involved are based on an external probabilistic calculation that considered hottest-tube effects and effects of pre-existing damage. The timing of SGTR is somewhat arbitrary and occurs early in the core degradation. These factors, along with the enhanced SG natural leakage modeled, exaggerate the primary-side depressurization such that the assumed time of hot leg creep rupture in this calculation well precedes the hot leg nozzle creep rupture index reaching unity. The Powers model was applied to calculate retention at the steam generator tube support plates and in the steam separators and dryers for Case 3A3. Overall, DFs for this case are comparable to those calculated for Case 3A2.

3.4 Sensitivity Case 3A4

Case 3A4 suppresses rupture paths because of creep damage parameters. Therefore, hot leg creep rupture (along with surge line and steam generator tube rupture) is not allowed. Otherwise, Case 3A4 is identical to Case 3.

4. Scenario 4: Transient Induced by Electrical Distribution and NSCW Failures, 182 gpm per RCP Seal LOCA, Auxiliary Feedwater, and Controlled Depressurization

The scenario is initiated by loss of one 4.16 kV bus with NSCW tower fans out for maintenance. The model does not represent these losses directly, except that all fan coolers stop running at time zero. Otherwise, a normal full-power situation holds until 2 hours. At 2 hours (the assumed time for NSCW to heatup and cause associated heatup of the CCW and ACCW systems) a manual scram occurs, normal charging/letdown are terminated, the RCPs are manually tripped, and ECCS and containment sprays become unavailable. By scenario definition, RCP seal leaks begin at 2 hours 13 minutes at rate 182 gpm per pump. One MDAFW pump and the TDAFW pump are available at all times, and the CST is credited with unlimited refills as required. Controlled depressurization begins 30 minutes after the ECCS signal. (ECCS itself is unavailable). The depressurization aims at 55.56 K/hr rate of decrease of the average temperature of the liquid in the hot and cold legs (T_{avg}) by means of the ARVs of the four SGs, whose pressure, however, may not fall below 200 psig . At 4 hours, one pressurizer PORV, the steam dump valves, and two SG ARVs lose the ability to open. In the case of the ARVs, only their capability to cycle on pressure is lost: they remain available for the SG cooldown. **Table 4-1** provides the details of the conditions applicable to this accident scenario.

Table 4-1: Specific Conditions Applicable to Accident Scenario 4[10]

Description	Depressurized RCP Seal LOCA with Feed Water. Initiated by loss of one 4.16 kV bus A, all NSCW tower fans out for maintenance, and a developing RCP seal leak.
Reactor	Manual trip at t=2 hours.
RCPs	Per Step 4 of AOP 18021-C, Loss of Nuclear Service Cooling Water System, operators should trip the RCPs if neither NSCW train can be placed into service. Thus, RCPs should be tripped at 2 hours for this scenario.
Break	RCP seal leaks of size 182 gpm per pump, developing 13 minutes after the functional loss of NSCW (i.e., t=2 hours and 13 minutes).
Pressurizer PORVs	One PORV fails closed 4 hours after the start of the accident due to battery depletion following loss of a 4.16 kV bus. The remaining PORV, and all three SVs, are permitted to cycle normally throughout the accident. No feed and bleed.
ECCS	Train A is lost at time zero (due to the ac bus failure). Train B is lost prior to actuation due to the NSCW failure.
Feed Water	One motor-driven AFW pump is taken out due to loss of a 4.16 kV bus. The other motor-driven AFW pump, and the turbine-driven AFW pump, are permitted to inject to all four SGs in level-control mode until the CST is emptied. If relevant, CST refill is credited.

10 Notionally, what this scenario represents is a time zero failure of 1 train of AC power, coincident with all NSCW fans already out for maintenance. The failure of 1 train of NSCW pumps at time zero (due to the AC failure) is assumed to perpetuate NSCW heatup already underway due to the fans being out for maintenance. For the first 2 hours, the operators are assumed to continue at-power operation and to focus on repair of the AC power failure, while NSCW is also presumed to still be operable. At 2 hours, NSCW is assumed to have heated up to temperatures affecting ACCW and CCW, and the operators are assumed to trip the reactor due to both the NSCW issues and a Tech Spec requirement on electrical distribution. A seal LOCA occurs 13 minutes later, and the DC buses affected by the time zero AC bus failure deplete at t = 4 hours.

Table 4-1: Specific Conditions Applicable to Accident Scenario 4[10]

SG Valves and Steam Line	ARVs from SG 1 and SG 4 are unavailable after 4 hours due to loss of DC buses fed by the 4.16 kV bus, The other two ARVs, and all MSSVs, are permitted to cycle normally. MSIVs closure occurs at 4 hours due to battery depletion, or earlier automatically based on ESF logic. Steam dump to condenser is assumed to function normally until isolation, or until they fail 4 hours into the accident from battery depletion[11].
Containment Sprays/Coolers	None.
Containment Isolation	Isolated.
Operator Actions	Maintain SG NR level between 10% and 65% A controlled cooldown (~100F/hr) commences ~30 minutes after seal leak-induced SI using steam dumps or ARVs[12]. When CETC reach 711F, an emergency cooldown begins per FR-C.1, but maintaining SG pressure > 200 psig.
Other	The normal charging pump becomes unavailable at t=2 hours due to the loss of NSCW.
Sensitivity Cases	None

11 Steam dump control power is provided by control panel 1AD11, which is powered by dc bus 1AD1, which is in turn powered by AC bus 1AA02.

12 This is a rough estimate based on procedural steps and transitions; cooldown can include temporarily opening a PORV if necessary to perpetuate RCS depressurization, but with the lack of ECCS it is unlikely that the associated subcooling margin requirement would be met.

5. Scenario 5: Interfacing System Loss of Coolant Accident (ISLOCA) with Submerged Break

The scenario is initiated by a break in the residual heat removal (RHR) system in the Auxiliary Building, connected to the hot leg modeled by CV 610 (upper half of the upstream section of the hot leg of loop 4 in the model's numbering scheme – this loop does not contain the pressurizer). The flow area of the break, modeled by flow path FL 994, is 20.3 cm^2, or the area of a circle of 2-inch diameter. For the Revision 7 of the MELCOR model, offline testing indicates that a 2.5-inch break is the critical size below which the Auxiliary Building does not fail during initial blowdown and above which it does (acknowledging that there is uncertainty in the prediction of this value given the simplified Auxiliary Building treatment in the MELCOR model and the imprecise knowledge on building structural capacity). The model includes two control volumes and two heat structures to model the in-containment and ex-containment sections of the RHR pipe between the hot leg and the break. The piping and the break location are modeled in such a way that precludes (back-side) cooling of these heat structures by a pool of water in containment or in the Auxiliary Building. No attempt has been made to model turbulent deposition in the RHR piping. Other deposition mechanisms (i.e. gravitational settling, Brownian diffusion, thermophoresis, and diffusiophoresis) are modeled in the RHR piping, but little deposition is expected given the short residence times of the carrier gas in the piping and the lack of pipe cooling. The break is to CV 982, which models the next-to-lowest level of the Auxiliary Building. The elevation of FL 994, the volume/altitude table of CV 982, and the opening height range of FL 011 (i.e., the drain from CV 982 to the building's lowest level) are all specified so that about 1 m^3 of liquid in CV 982 puts the water level approximately at the level of the break; about 101 m^3 of liquid in CV 982 puts the water level approximately at 5 m above the break; and any liquid added to CV 982, in the case that it already contains about 101 m^3, spills over via FL 011 into the building's lowest level, whose volume is large. All modeled systems are available. Suction for the ECCS pumps is taken from the RWST until it is drawn down to Lo-3 instead of the usual Lo-2 (since conditions for sump switchover will not be met). Control of ECCS injection is included, such that all ECCS pumps are stopped for pressurizer level above 25 m (the normal value is about 20 m); pumps are not allowed to restart until the level falls again to below 21 m. These control setpoints were assumed. In this scenario, the five of six ECCS pumps and the RWST function normally, in that they are not directly affected by the break, and the liquid pumped by the five pumps reaches the cold legs as designed. The other pump – the RHR pump in the ruptured RHR line – is assumed to fail due to the break location. **Table 5-1** provides the details of the conditions applicable to this accident scenario, and defines several sensitivity cases.

Table 5-1: Specific Conditions Applicable to Accident Scenario 5

Description	ISLOCA. Initiated by a break outside containment and upstream of the RHR pump, from hot leg 1 (Train A) – this is loop 4 in the model nomenclature and is not the pressurizer loop, with an assumed pipe rupture diameter of 2 inches.[13] (single-ended); this break location results in blowdown of the RCS (through hot leg 1), and gravity drain of the RWST (if relative pressures and timing dictate); Train B is unaffected because the break is upstream of a check valve, which is in turn upstream of the cross-connect line.

[13] There is a very broad range of rupture sizes, locations, and corresponding system flow paths all subsumed within any of the Level-1 ISLOCA cutsets. For the Base Case, the break size was chosen to be on the larger end of the size range that does not overpressurize and fail the auxiliary building during the initial blowdown. (Scoping

Table 5-1: Specific Conditions Applicable to Accident Scenario 5

Reactor	Reactor trips according to normal Reactor Trip System (RTS) logic.
RCPs	Pumps trip according to normal system logic or on cavitation assumptions.
Break	2-inch equivalent diameter break in one hot leg to the Auxiliary Building. It is assumed that the break will quickly be submerged due to the layout of the RHR pump room[14].
Pressurizer PORVs	Both PORVs, and all three SVs, are permitted to cycle normally (although they should not be demanded in this scenario).
ECCS	All divisions of ECCS are permitted to inject from the RWST normally following generation of the automatic SI signal, except for the RHR pump in the broken RHR line, which is assumed to fail.[15] Injection continues until ECCS pumps are isolated from the RWST at a narrow-range RWST level of 8%. However, most injected water will be lost out the break into the Auxiliary Building[16] (Train A RHR pump, Centrifugal Charging Pump [CCP] and SI[17], and likewise water from all Train B pumps, having passed through the RCS first). In consequence, the containment sump level should be too low to support ECCS recirculation.
Feed Water	Both motor-driven AFW pumps and the turbine-driven AFW pump are permitted to feed all four SGs in level-control mode until the CST is empty. The course of this scenario should not be significantly impacted by the presence of feed water.
SG Valves and Steam Line	All ARVs and MSSVs are permitted to cycle normally. MSIV closure is governed by normal plant logic. Steam dump is also assumed to function normally. The RCS should be at low pressure due to the size of the break, and no depressurization due to secondary-side cooldown is credited.
Containment Sprays/Coolers	Both containment spray trains, and all containment coolers, are available and function according to their normal actuation logic. However, since no significant containment pressurization is expected, these systems may not be demanded and in any case should not affect the course of the accident significantly.
Containment Isolation	Isolated (except for the break in the RCS from the hot leg to the Auxiliary Building).
Operator Action	Maintain SG NR level between 10% and 65% No other pre-core damage actions modeled. The break is not isolable, and procedures do not appear to direct closure of the valve between the RWST and the break location.

calculations revealed that the building would remain under the assumed failure pressure of 1.1 bar for breaks up to 2.5 inches in diameter.) A more limiting combination is selected for a sensitivity study.

14 The RHR heat exchanger and RHR pump rooms are assumed to have normally closed floor drains and would be expected to flood to at least the 1-story level (penetrations below this height are sealed).

15 Note that both RHR pumps were enabled in the input for this run, whereas one pump should have been disabled. However, neither injected because pressure did not fall below the pump shutoff head.

16 Part of the blowdown from the RCS passes through the RHR relief valve to the pressurizer relief tank. The RHR relief valve opens when pressure in the RHR pump suction piping exceeds 450 psig. After the PRT rupture disk bursts, some water will spill into containment. However, the amount of water in the sump is insufficient to permit ECCS recirculation.

17 In injection mode, high-head safety injection is independent of RHR via normal valve alignment and check valves.

Table 5-1: Specific Conditions Applicable to Accident Scenario 5

Other	n/a
Sensitivity Cases	5A: Same as above, but the break does not submerge, representing a case where the floor drain is not isolated, or the rupture is in a branch line not located in the RHR pump compartment. 5B: Initiated by a break outside containment, downstream of the RHR pump, and proximate to the RHR heat exchanger, from hot leg 1 (Train A), with an assumed pipe rupture diameter of 8 inches (double-ended); this break location results in blowdown of the RCS (through hot leg 1), and gravity drain of the RWST (if relative pressures and timing dictate[18]). Train B is affected because the break is downstream of the check valve that would prevent communication with the normally-open cross-connect line. Due to the pressure differential between the RCS and the break location, RHR B flow is assumed to go to the break. 5C: Investigates retained mass limits of PPAFES filters. 5D: Same as 5B, but the break does not submerge.

5.1 Sensitivity Case 5A

Case 5A considers an uncovered break. As discussed for Case 5, a covered break is obtained for the Base Case by specifying the elevation of the bottom of the opening of FL 011 (which represents the drain from auxiliary building Level C to Level D) high enough that the break water can stand about 5 m above the break elevation before it must spill over to the building's lowest level. In Case 5A, the bottom of FL 011 is made flush with the floor of CV 982 (i.e., Level C). This geometry leaves the break opening well up in the gas phase of CV 982, with the result that radionuclide scrubbing in the pool should not occur. Otherwise, Case 5A is the same as the Base Case.

5.2 Sensitivity Case 5B

Case 5B considers a different location and size for the break. As discussed for Case 5, in the Base Case five of the six ECCS pumps[19] and the RWST function normally, in that they are not directly affected by the break, and the water pumped by the five ECCS pumps reaches the cold legs as designed. (Note that the RCS pressure did not decrease below the RHR pump shutoff head until after RWST depletion.) This circumstance is reconsidered in Case 5B. The Base Case models the break as a single-ended 2-inch break from the RHR pump suction line to the Auxiliary Building (Level C, modeled by CV 982). The break from the RHR line is modeled by FL 994 as in the Base Case, except that the flow area now corresponds to an 8-inch break. (In particular, as before the break is covered: FL 994 is submerged by about 5 m whenever CV 982 contains more than about 101 m^3 of water.) The other end of the break discharges water that spills directly from the RWST to the Auxiliary Building, as driven by gravity and by one RHR pump. This end is modeled by control functions and a mass source to CV 982. Any real interaction between the ends of the break is ignored. An estimate for the gravity-driven flow *w* can be found from the Bernoulli equation:

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[18] Since the rupture point is downstream of the RHR pump, the initial pressurization of the piping through the pump is assumed to damage the pump such that it never runs.

[19] The exception is the RHR pump in the broken RHR line, which is assumed to fail.

$$w = \rho A \sqrt{2g\Delta h}$$

that gives 822 kg/s with density of the RWST water ρ = 958.35 kg/ m^3, flow area A = 0.03243 m^2 corresponding to an 8-inch break, gravitational acceleration g = 9.8 m/s^2, and height of the RWST above the break Δh = 35.66 m corresponding to 245 ft – 128 ft. This value is likely an over-estimate since it neglects all losses. The scenario, therefore, adopts half this value, 411 kg/s, for the gravity-driven flow from the RWST directly to the Auxiliary Building. Although this water passes the operating RHR pump, this rate exceeds the pump's runout rate of 284 kg/s. Therefore, the pump is not credited to add any driving head. Due to minor technicalities of the modeling, the break flow from the RWST direct to the Auxiliary Building begins at time zero at rate 411-284 = 127 kg/s. At about 6 minutes the head seen by the RHR pump, calculated for simplicity as if the pump were operating normally between the RWST and the RPV, falls below the deadheading value (~1.3 MPa), and the break flow direct to the building increases. The so-reckoned head falls below the runout value at about 7 minutes, and from then until RWST depletion (Lo-3, at 1.2 hours) the direct break flow rate from the RWST to the Auxiliary Building is 411 kg/s. The direct break stops when the RWST attains Lo-3, as does the successful injection by the charging and SI pumps (whose water, however, spills to the Auxiliary Building from the broken hot leg after traversing the vessel once). In Case 5B, the other RHR pump is assumed to be inoperable from the start of the transient as a consequence of the break. Apart from these changes, Case 5B is the same as the Base Case.

5.3 Sensitivity Case 5C

Case 5C considers filter plugging in the Piping Penetration Area Filtration and Exhaust System (PPAFES). As discussed for Case 5, the Base Case takes no account of any limitation of the amount of mass that the filters can retain. Case 5C models filter plugging as an abrupt event in which the exhaust fans will trip when the back pressure offered by the filters increases above some limit due to the retained aerosol mass.

5.4 Sensitivity Case 5D

Case 5D is a sensitivity case to Case 5B. In Case 5B, the ISLOCA break is an eight-inch double-ended break situated to quickly become covered by liquid. In Case 5D, the elevation of the floor drain of Level C in the Auxiliary Building is lowered so that the liquid quickly drains away, exposing the break. (The elevation manipulations are the same as have been described for Case 5A.) Otherwise, Case 5D is the same as Case 5B. In particular, the Auxiliary Building fails during the early blowdown, implying failure of the Piping Penetration Area Filtration and Exhaust System (PPAFES).

6. Scenario 6: Transient Initiated by Loss of Offsite Power, Auxiliary Feedwater Lost at 6 Hours, ECCS Available, and Containment Cooling Available

Representative Scenario 6 represents a situation in which there has been a loss of 1 train of AC power, loss of all feedwater, and failure of operators to initiate feed and bleed. There are several considerations that complicate the translation of the notional PRA sequence to a specific set of deterministic boundary conditions, and some of these are discussed here:

- The failure-to-run of an emergency diesel generator (EDG) following the LOOP initiator requires the prescription of a failure timing that is not directly supported by data, and which in turn affects the complete loss of feedwater and thus the accident progression. Here, a failure timing of 2 hours was assumed.
- The MELCOR model does not include treatment of the pressurizer heaters, and this is generally not an important omission. However, in this sequence, it may cause an SI signal on low-pressurizer pressure that does not otherwise occur.
- At one point in this accident, the primary system may approach the point of being water solid if an early SI occurs (or in fact become water solid). The timing of this occurrence may affect the operator response, since procedures have control of pressurizer level as a continuous action, as well as direction to secure ECCS depending on secondary-side conditions (which will be changing during the course of the accident due to the delayed failure of AFW).
- The diagnosis and implementation of feed and bleed is guided by the Heat Sink Critical Safety Function Status Tree (CSFST) (see **Figure 6-1**), cued by low SG level. The human error probability is inherently influenced by the limited time available (tens of minutes) to start bleed and feed to prevent core damage, based on Level 1 PRA success criteria and sequence timing analyses for conditions that are more limiting (failure of AFW at time zero) than what occurs in this scenario.
- Also, the Level 1 PRA does not separately query later operator actions because they would not (by the Level 1 PRA's definition) avert core damage. Most notably, the Level 1 PRA does not query the use of ECCS in response to the Core Cooling CSFST (see **Figure 6-2**) orange and red paths, because it would not be expected to change the determination that core damage occurs, based in part on the aforementioned, more restrictive set of success criteria and sequence timing assumptions.
- The above leads to a situation in which any number of assumption combinations are plausible within the bounds of the Level 1 PRA accident sequence, with many combinations leading to situations that are overly artificial. More plausible combinations, taking the delayed (2-hour) failure of the diesel generator as a given, include:
  - An SI signal does occur early in the accident, and the operators secure ECCS to prevent overfilling the pressurizer. Later, upon the delayed loss of AFW, operators fail to initiate bleed and feed. Hours later, when the orange path conditions on the Core Cooling CSFST (see **Figure 6-2**) are reached (which would occur based on Reactor Vessel Level Indication System (RVLIS) < 41% and CETC > 711F), operators re-start ECCS. MELCOR would potentially predict that core damage is averted in this situation, based on similar calculations run for other plants. (Note that the yellow path on Core Cooling CSFST would be reached much earlier, would cue resumption of ECCS, and would very likely avert core damage. Red path on Core Cooling CSFST (see **Figure 6-2**) would be

reached shortly after the orange path conditions and might lead to a substantively similar outcome as the orange path case.)

- An SI signal does occur early, the operators never take action to secure ECCS, and the scenario develops with the charging pump injecting and the pressurizer PORVs cycling. For Representative Scenario 6, a side calculation was performed that indicated that this set of assumptions would lead to no core damage, as modeled by MELCOR. The side calculation assumed that the pressurizer PORVs and safety relief valves will never fail regardless of the number of cycles or the passing of liquid. Some details are given below.
- An SI signal either does not occur early or else does occur with subsequent operator action to secure ECCS to prevent overfilling the pressurizer. This is followed by failure to re-initiate ECCS in response to both bleed and feed cues and inadequate core cooling cues. This results in core damage and radiological release as modeled by MELCOR.

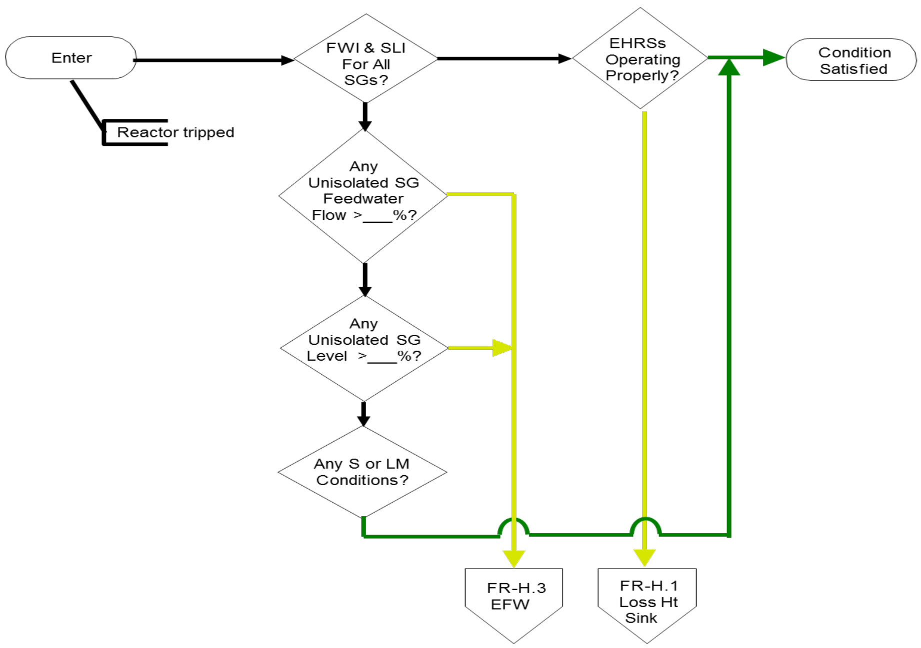

Figure 6-1: Heat Sink Critical Safety Function Status Tree

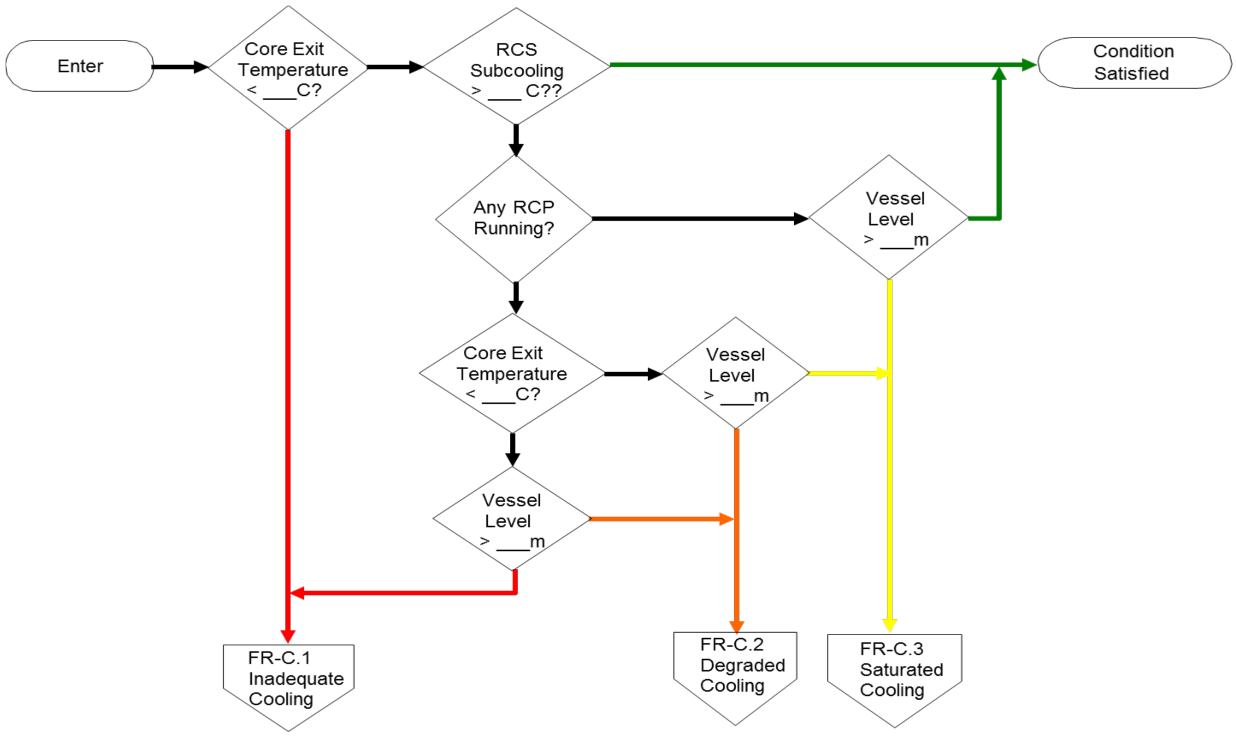

Figure 6-2: Core Cooling Critical Safety Function Status Tree

The third case adheres most strictly to the Level 1 PRA sequence definition and is the selected scenario. It is overly punitive in some senses, as outlined above, but should not be viewed as a worst-case representation (given the failure-to-run timing of the EDG and the potential advantage of having the ECCS available in injection mode following the onset of core damage).

The scenario is initiated by loss of offsite power with consequences of reactor trip, main feedwater trip, RCP trip, and MSIV closure at time zero. Initial unavailability of one of two trains of MDAFW plus an independent failure of one of two diesel generators at 2 hours, and eventual battery depletion, result in unavailability of auxiliary feedwater after 6 hours. Normal charging and letdown are unavailable as of at time zero. ECCS, containment sprays, and fan coolers are available, but each of these is reduced to half the normal capability at 2 hours (i.e., after 2 hours, one ECCS train, one sprays train, and four fan coolers remain available). Both pressurizer PORVs are available until 6 hours. After that, only one PORV is available. There are no leaks or breaks. **Table 6-1** provides the details of the conditions applicable to this accident scenario.

Table 6-1: Specific Conditions Applicable to Accident Scenario 6

Description	Transient with ECCS and Containment Cooling. Initiated by LOOP with failure to run of DG A, AFW MDP B in maintenance, and operator failure to feed and bleed. A 2-hour failure-to-run time is assumed for DG A. Battery depletion for dc buses fed by Train A are assumed to occur 4 hours later (t = 6 hours). Note that failure of DG A will, following battery depletion, result in failure of MOV in the common steam supply to the turbine-driven AFW pump, resulting in its failure.
Reactor	Reactor trips at time zero due to LOOP.

Table 6-1: Specific Conditions Applicable to Accident Scenario 6

RCPs	RCPs trip at time zero due to LOOP.
Break	None.
Pressurizer PORVs	One PORV is permitted to cycle normally for the first 6 hours (i.e., until battery depletion following loss of offsite power and one DG at t = 2 hours). The other PORV, and all three SVs, are permitted to cycle normally for the duration of the accident. No feed and bleed.
ECCS	Train A of the ECCS is unavailable after 2 hours due to loss of offsite power and DG A. Train B of the ECCS is available throughout the accident, in both injection and sump recirculation modes, following automatic SI signal generation. Control of ECCS based on pressurizer level is assumed, but after once turning off ECCS on high pressurizer level, operators fail ever to re-start ECCS.
Feed Water	The B MDAFW pump fails at time zero due to independent failure. The A MDAFW pump fails at 2 hours when DG A fails. The TDAFW pump fails at 6 hours, when Train A batteries deplete. So, prior to 6 hours, adequate AFW exists to maintain SG level. After 6 hours, no FW is available.
SG Valves and Steam Line	Atmospheric Relief Valves (ARVs) cannot cycle automatically due to loss of instrument air[20]. All Main Steam Safety Valves (MSSVs) are permitted to cycle normally. MSIVs close at time zero on loss of power/instrument air.
Containment Sprays/ Coolers	For the first 2 hours, all containment cooling is available. After that time, one containment spray pump and 4 out of 8 containment coolers are permitted to automatically actuate and operate normally. Both containment spray pumps are available until failure of DG A at 2 hours. After 2 hours, the Train B containment spray pump is available. However, if containment sprays have not actuated before entry to the SAMGs, then the containment spray pumps are placed in pull-to-lock to prevent them from automatically actuating.
Containment Isolation	Isolated.
Operator Actions	It is implicitly assumed that operators restore RCP seal injection (by starting a centrifugal charging pump), though associated makeup capability (pre-SI) is not explicitly modeled. Maintain SG NR level between 10% and 65% Maintain pressurizer level using ECCS following receipt of the SI signal. But after once turning off ECCS on high pressurizer level, operators fail ever to re-start ECCS. Operators are specifically assumed not to initiate feed and bleed. Cooldown and depressurization activities are not modeled.
Other	The normal charging pump trips at time zero due to LOOP (it is powered by a non-1E AC source).
Sensitivity Cases	6A: Same scenario with containment sprays. 6B: Combustion is suppressed to provide flammability information for side studies 6C: Same scenario with early containment failure.

[20] This is based on the R01 Level 1 PRA model assumption that the ARVs are dependent on instrument air. This was later changed in the R02 Level 1 model (to remove this dependency). Under the latter assumption, two ARVs would have been available in this scenario, though this was not expected to have a major impact on the results.

Table 6-1: Specific Conditions Applicable to Accident Scenario 6

	6D: Same scenario with early containment failure and containment sprays

6.1 Recovery Case 6R1

This scenario is initiated by loss of offsite power with motor-driven AFW pump B in maintenance and failure to run of the train A diesel generator. The diesel is assumed to fail at 2 hours, causing motor-driven AFW pump A to fail. Train A batteries deplete 4 hours later, causing the loss of turbine-driven AFW – and thus all feedwater to the steam generators – at 6 hours. In this sequence, operators fail to initiate feed and bleed, which eventually leads to core damage. Containment heat removal systems are available in this scenario. ECCS is also available; however, this scenario assumes that operators would follow procedural steps to secure ECCS to prevent the pressurizer from going solid before the loss of AFW at 6 hours, and that operators would subsequently fail to restart ECCS pumps upon receiving degraded or inadequate core cooling indications in the core cooling critical safety function status tree.

At 14.7 hours, operators are assumed to enter SAG-1/SAG-2 activities and spend one hour determining that SAG-1 is not viable and SAG-2 is rendered moot by hot leg creep rupture. An additional 15 minutes for entry, 30 minutes for diagnosis, and 15 minutes for implementation of SAG-3 are assumed for a total delay of 2 hours (at 16.7 hours) for the initiation of one train of RHR.

6.2 Sensitivity Case 6A

In Case 6A, operators do not place containment sprays in pull-to-lock, so the Train B containment spray pump will actuate upon a high containment pressure signal. Otherwise Case 6A is the same as the Base Case.

6.3 Sensitivity Case 6B

Case 6B is a sensitivity case to Case 6. In Case 6B, deflagrations are globally suppressed at all times. This is done to support side studies related to hydrogen flammability. Otherwise Case 6B is identical to Case 6.

6.4 Sensitivity Case 6C

Case 6C is a sensitivity case to Case 6. In Case 6C, the containment is forced to fail at the time of vessel breach due to an assumed energetic event at that time. The failure size is set to 324 cm^2; this failure area is also used in Cases 1A2, 2A and 6D. Otherwise Case 6C is identical to Case 6.

6.5 Sensitivity Case 6D

Case 6D is a sensitivity case to Case 6A. In Case 6D, containment sprays are allowed to actuate when demanded (which occurs during a hydrogen burn that immediately follows hot leg creep rupture), while the containment is forced to fail at the time of vessel breach due to an assumed energetic event at that time. The failure size is set to 324 cm^2; this failure area is also used in Cases 1A2, 2A and 6C.

7. Scenario 7: Station Blackout with 21 gpm per RCP Seal LOCA, Auxiliary Feedwater Lost at 4 hours, and Containment Isolation Failure

Except for the loss of containment isolation at time zero, and the loss of all auxiliary feedwater at 4 hours, Scenario 7 is identical to the Base Case of Scenario 1 (in other words, Scenario 7 is the same as sensitivity Case 1A, but with the containment unisolated). **Table 7-1** provides the details of the conditions applicable to this accident scenario.

Table 7-1: Specific Conditions Applicable to Accident Scenario 7

Description	Un-isolated Station Blackout. Initiated by "sunny day[21]" LOOP, with independent common cause failure (CCF) to open of both switchyard AC breakers, and pre-existing leakage due to a containment isolation failure.
Reactor	Reactor trips at time zero due to loss of power.
RCPs	Pumps trip at time zero due to loss of power.
Break	RCP seal leaks of size 21 gpm/RCP from the start of the accident.
Pressurizer Power Operated Relief Valves (PORVs)	Both PORVs are permitted to cycle normally while battery power is available (first 4 hours), after which they fail closed. All three Safety Valves (SVs) are permitted to cycle normally throughout the accident.
ECCS	None.
Feed Water	Turbine-driven AFW functions until 4 hours, when it fails. CST refill (or swap to alternate CST) is credited, if relevant.
Steam Generator (SG) Valves and Steam Line	Atmospheric Relief Valves (ARVs) cannot cycle automatically due to loss of electrical power (although they are available for manual use in secondary-side depressurization; see below). All Main Steam Safety Valves (MSSVs) are permitted to cycle normally. MSIVs close at time zero on loss of power/instrument air.
Containment Sprays/Coolers	None.
Containment Isolation	Un-isolated (assumes a 2-inch equivalent diameter leakage from containment to the environment).
Operator Actions	SG ARVs are used to depressurize all SGs to between 200-300 psig[22]; valve(s) are (locally) opened at t = 30 minutes and t = 35 minutes for the first two ARVs, and t = 45 minutes and t = 50 minutes for the final two, if used. Valves are closed if SG NR level < 10% in all SGs. Manual feedwater control to maintain SG NR level > 10%

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21 This term is used here to denote a plant-centered, switchyard-centered, or grid-related LOOP, as opposed to a weather-related LOOP. In terms of the MELCOR analysis, they are all the same, but in terms of any credited local operator actions or the offsite consequence modeling, they may not be.

22 This human failure event has a low likelihood of failure in the L3 PRA Project "R01" Level-1 model (1-OPR-XHE-XM-RSSDEP; HEP = 1E-3), and there are no assumed equipment failures that would prevent its success here. For this action it is assumed that it takes 20 minutes to reach the relevant procedure steps, 10 minutes of travel time, and 7 minutes to open an additional valve in the same area (2 valves in each of 2 areas).

Table 7-1: Specific Conditions Applicable to Accident Scenario 7

Sensitivity Cases	7A: Portable pump used for containment spray following vessel breach.

7.1 Sensitivity Case 7A

Case 7A is a sensitivity case to Case 7. In Case 7A, operators are assumed to use the portable extensive damage mitigation guideline (EDMG) pump to spray containment beginning at the time of vessel breach. Specifically, operators are assumed to spray containment at a rate of 350 gpm, and to stop the portable pump after injecting 600,000 gallons to containment. The volume of water injected to containment is approximately equal to the volume of water in the fire water storage tanks. The assumed spray flow rate is reasonable given EDMG requirements and the head against which the pump must inject. This calculation is pursued here to ensure a source term exists for a scrubbed containment isolation failure. Other than the use of the EDMG pump to spray containment, Case 7A is identical to Case 7.

8. Scenario 8: Un-isolated Steam Generator Tube Rupture with Auxiliary Feedwater

The scenario is initiated by a double-ended break of one SGT, located in the pressurizer-loop SG, at elevation that corresponds with the top of a U-tube of average curvature radius. Whereas there are four candidate pressurizer-loop SGT control volumes that include this elevation, CV 873 and CV 853 are used as the "from" CVs for the two break flow paths. (CV 853 is the topmost, upstream, loop 1 SGT control volume that carries the normally-directed flow under conditions when countercurrent natural circulation is enforced. CV 873 is the topmost, downstream, loop 1 SGT control volume.) Both break flow paths go to CV 364, part of the boiler of the pressurizer-loop steam generator. The break flow area of each flow path is the cross-sectional area of one SGT. All AFW is available, but no AFW is supplied to the pressurizer-loop SG after 15 minutes. Refills of the CST are not considered. Operators close the dump valves and the MSIVs of all but the pressurizer-loop SG at 15 minutes. The MSIV of the pressurizer-loop SG is held open after 15 minutes, comprising the un-isolated aspect of the scenario. (The containment is isolated.) ECCS and containment sprays are unavailable, and fan coolers stop operating at the start of the transient. **Table 8-1** provides the details of the conditions applicable to this accident scenario, and defines several sensitivity cases.

Table 8-1: Specific Conditions Applicable to Accident Scenario 8

Description	Un-isolated SGTR with Feed Water. Initiated by SGTR, with independent common-cause failure to start of all NSCW fans and operator failure to isolate affected SG. Failure to isolate is assumed to take the form of failure to isolate ARVs, MSSVs, and MSIV from the affected SG. FW to the affected SG is assumed to be shut off successfully.
Reactor	Reactor tripped manually at 15 minutes since the start of the transient, or trips earlier automatically according to normal RTS logic.
RCPs	Pumps trip according to normal system logic.
Break	Double-ended break of one SG tube with leakage area equivalent to 2-tube diameter, assumed near the middle of the apex of the tube bundle in the SG of the pressurizer loop.
Pressurizer PORVs	Both PORVs and all three SRVs are permitted to cycle normally throughout the accident. No feed and bleed.
ECCS	None.
Feed Water	Both motor-driven AFW pumps and the turbine-driven AFW pump are available to feed the three unaffected SGs in level-control mode until the CST is empty. It is assumed that operators successfully stop feed to the affected SG (but see sensitivity Case 8A, below).
SG Valves and Steam Line	All ARVs and MSSVs (including from affected SG, due to failure of SG isolation) are permitted to cycle normally throughout the accident.
Containment Sprays/Coolers	None.
Containment Isolation	Isolated.
Operator Action	Operator action to close MSIV from affected SG fails. Operators then successfully close MSIVs and steam dump valves from unaffected SGs following EOPs. Assumed

Table 8-1: Specific Conditions Applicable to Accident Scenario 8

	manual time of MSIV closure from unaffected SGs is 15 minutes after the start of the accident. Assumed manual reactor trip at the same time if it has not already occurred automatically. No secondary-side cooldown or feed-and-bleed.
Sensitivity Cases	8A: Sensitivity case in which feed to the affected SG is assumed to be present (i.e., operators fail to shut off feed as part of isolation procedure or as part of routine SG level control). 8B: Sensitivity case in which the ARV of the affected SG is assumed to stick open at the first demand during core damage.

8.1 Recovery Case 8R1

This scenario is initiated by the spontaneous double-ended guillotine break of one steam generator tube near the top of the U-tube of average radius. ECCS is unavailable due to the independent (from the SGTR) common-cause failure of all NSCW fans. Operators fail to isolate the affected steam generator and to refill the condensate storage tank. The slow loss of inventory through the break eventually leads to core damage. A pre-vessel breach and post-vessel breach recovery case was performed. Additionally, a calculation has been performed that models a pre-vessel breach action for Scenario 8B, in which the relief valve on the faulted steam generator sticks open at the start of core damage.

Before vessel failure, the highest priority strategy is SCG-1, “Mitigate Fission Product Releases.” One of the options in SCG-1 Appendix B for mitigating releases from the steam generator is to isolate the affected steam generator; remote actions to accomplish this are assumed unavailable due to the explicit failure to do this in the Level 1 PRA sequence, while other remote actions are assumed untenable due to habitability concerns. Other options in SCG-1 Appendix B (e.g., filling the affected steam generator and dumping steam to the condenser) refer to the applicable strategies in SAG-1, “Inject into Steam Generators.” These actions are modeled in MELCOR by feeding the affected steam generator 1.25 hours after SAMG entry and by opening one bank of condenser steam dump valves once steam generator water level exceeds 38% narrow range. This calculation assumes that operators have taken action to refill the condensate storage tank or switch to the Unit’s alternate CST.

8.2 Recovery Case 8R2

This scenario is initiated by the spontaneous double-ended guillotine break of one steam generator tube near the top of the U-tube of average radius. ECCS is unavailable due to the independent (from the SGTR) common-cause failure of all NSCW fans. Operators fail to isolate the affected steam generator and to refill the condensate storage tank. The slow loss of inventory through the break eventually leads to core damage. A pre-vessel breach and post-vessel breach recovery case was performed. Additionally, a calculation has been performed that models a pre-vessel breach action for Scenario 8B, in which the relief valve on the faulted steam generator sticks open at the start of core damage.

Highest and second highest priority strategies after vessel breach are the same as before vessel breach, i.e. SCG-1 and SAG-1. The post-vessel breach case assumes operators take the same actions as in the pre-vessel breach case, but in this case, operators take action 1.25 hours after vessel failure.

8.3 Sensitivity Case 8A

In Case 8A, auxiliary feedwater is supplied to the affected steam generator as required to maintain the water level. Otherwise Case 8A is identical to the Base Case.

8.4 Sensitivity Case 8B

In Case 8B, the ARV in the affected steam generator sticks open at 50.4 hours, following the first ARV demand after the onset of core damage. Otherwise Case 8B is identical to the Base Case.

8.4.1 Recovery Case 8BR1

An additional calculation has been performed in which operators feed the ruptured steam generator and depressurize using steam dumps before vessel breach, to provide a source term for steam generator tube rupture with a stuck-open relief valve and scrubbed releases. The system response in this case is similar to the response in Case 8R1, except here the RCS is depressurizing due to the stuck-open ARV when operators begin taking action to feed the faulted steam generator. Opening one bank of steam dump valves depressurizes the system much more rapidly than the single ARV, which causes the accumulators to dump. Similar to Case 8R1, accumulator injection quenches debris in the lower plenum and intact fuel in the core. Note that the accumulators also inject before vessel breach in Scenario 8B, but accumulator injection is too late to prevent lower head failure.

9. References

Lind, 2012 Lind, T., and A. Dehbi, "The Final Summary Report of the ARTIST II Project Severe Accident Tests," TM-42-11-25; ARTIST-95-11, Paul Scherrer Institute, 2012.

SNL, 2016 Sandia National Laboratories, "State of the Art Reactor Consequences Analysis Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station (Draft Report)," United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, (January 2016). [ML15224A001]

Appendix C: Treatment of Uncertainty for the Reactor, At-Power, Level 2 PRA for Internal Events and Floods

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1. Introduction

Treatment of uncertainty within the Level 2 probabilistic risk assessment (PRA) portion of the Level 3 PRA (L3PRA) project is discussed in the main report for the reactor, at-power, Level 2 PRA for internal events and floods (NRC, 2019b), with the critical points being that two types of uncertainty are treated, as follows:

- Parameter uncertainty is defined as any uncertainty (aleatory or epistemic) that relates to a parameter in the PRA logic model that has a non-zero/non-unity value. This type of uncertainty is characterized through probability distributions about the basic event's point estimate, and propagated using standard PRA parameter uncertainty propagation techniques, to arrive at uncertainty distributions about the release categories' frequency.
- Model uncertainty is defined as any uncertainty (aleatory or epistemic) that relates to a parameter in the PRA logic model that has a zero/unity value, or that is a part of the underlying deterministic (e.g., MELCOR) analysis. This type of uncertainty is characterized through alternative treatments about the default modeling assumption and explored using sensitivity analysis. These sensitivity analyses can involve re-running the logic (SAPHIRE) model or MELCOR calculations and result in either alternative release category profiles or alternative source terms.

The approach used here does not maintain traditional boundaries between aleatory and epistemic uncertainty, but rather reflects limitations in the state-of-practice of uncertainty treatment, along with practicalities in how the overall PRA is constructed (as the combination of probabilistic and deterministic modeling). It ultimately fulfills the basic point of uncertainty treatment, which is to understand how sensitive the baseline results are to the modeling, and thus what range of results should be viewed as possible or probable.

Further, the approach here is to consider all parameter uncertainties in an integrated fashion, and to explore model uncertainties in groups (or categories). Currently, 16 categories are defined (see **Section 4**). In reality, accident analysis uncertainties are correlated/inter-related throughout the accident, so one should be cautious in thinking that this approach provides a comprehensive or holistic view of accident uncertainty. Once again, its limitations reflect the state-of-practice in uncertainty treatment and Level 2 PRA development.

Project-wide tenants regarding practical aspects of uncertainty treatment can be found in (Sancaktar, 2015). Meanwhile, **Table 1-1** provides the uncertainty-related supporting requirements from the Level 2 PRA Standard (ASME, 2014).

Table 1-1: Level 2 PRA Standard Supporting Requirements Related to Uncertainty

SR	Capability Category I	Capability Category II	Capability Category III
L1 – Level 1/Level 2 PRA Interface – Accident Sequence Grouping			
L1-B7	ENSURE that the grouping process into PDS [plant damage state] (or other interface issues) does not result in screening out (i.e., prematurely truncating) accident sequences that are important in the characterization of the radionuclide release (e.g., significant radionuclide release categories) or sequences that defeat all or most containment mitigation measures [see ER-C1].	TRANSFER the total CDF [core damage frequency] from the Level 1 PRA to the Level 2 PRA [see Note (10)].	TRANSFER the total CDF from the Level 1 PRA to the Level 2 PRA including the uncertainty distributions on the Level 1 PRA cut sets/sequences [see Note (10)].
CP – Containment Capacity Analysis			
CP-D1	IDENTIFY sources of parameter uncertainty, modeling uncertainty, and assumptions used in the deterministic analysis of containment failure [see Note (8) for examples].		
CP-D2	CHARACTERIZE the uncertainty range in thresholds for containment failure using engineering judgment [see Note (7)]	CHARACTERIZE the uncertainty in containment failure criteria in the form of a probability density function (fragility curve) [see Note (7)].	
CP-D3	CHARACTERIZE the uncertainty range in the final opening size of containment failure using engineering judgment.	CHARACTERIZE the uncertainty range in the final opening size of containment to permit a characterization of uncertainties in applications using structured sensitivity analysis.	
CP-D4	For each source of model uncertainty and related assumption identified in CP-D1, CHARACTERIZE how the containment strength or resistance to failure is affected [see Note (8) for examples].		
CP-E5	DOCUMENT the technical basis for the location and opening size (or leak rate) resulting from each failure mechanism and the technical basis for the probabilities used to characterize uncertainty.		
CP-E6	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in CP-D1 through CP-D4).		
SA - Severe Accident Progression Analysis			
SA-B1	LIST assumptions and sources of uncertainty used in performing deterministic calculations.		
SA-E2	IDENTIFY input parameters particular to the modeling tool selected in SA-C1 that reflect the uncertain models or assumptions defined in SA-E1.	DEFINE variations in input parameters particular to the modeling tool selected in SA-C1 that reflect the uncertain models or assumptions defined in SA-E1 [see Note (2)].	

Table 1-1: Level 2 PRA Standard Supporting Requirements Related to Uncertainty

SR	Capability Category I	Capability Category II	Capability Category III
SA-E3	For significant accident progression sequences, CHARACTERIZE the effects of uncertainties associated with input parameters.	For significant accident progression sequences with uncertain models or assumptions, PERFORM sensitivity analyses to evaluate the effects of uncertainties associated with calculation input parameters.	PERFORM sensitivity analyses to evaluate the effects of uncertainties associated with calculation input parameters.
SA-E5	For each source of model uncertainty and related assumption identified in SA-E4, CHARACTERIZE how the accident progression analysis results are affected.		
SA-F3	DOCUMENT alternative modeling assumptions and/or values of uncertain input parameters used in sensitivity and/or uncertainty analysis.		
SA-F4	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in SA-E3 and SA-E5).		
PT - Probabilistic Treatment of Accident Progression and Source Terms			
PT-B1	SELECT a method (e.g., expert judgment, parametric analysis) for defining numerical values of probability to reflect epistemic (modeling) uncertainty in phenomenological events [see Note (11)].		
PT-B10	CHARACTERIZE the uncertainty range for branching probabilities (split fractions) using engineering judgment.	CHARACTERIZE the uncertainty range for branching probabilities (split fractions) to permit a characterization of uncertainties in applications using structured sensitivity analysis.	CHARACTERIZE the uncertainty distribution of branching probabilities (split fractions) to permit the propagation of uncertainty under PT-E6.

Table 1-1: Level 2 PRA Standard Supporting Requirements Related to Uncertainty

SR	Capability Category I	Capability Category II	Capability Category III
PT-E6	CHARACTERIZE the uncertainty interval for the frequency of RC(s) that represent the largest and earliest releases of radionuclides to the environment. STATE a basis for the estimate consistent with the characterization of parameter uncertainties (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE-C15 from the ASME/ANS RA-Sa-2009 [1] and PT-B10).	CHARACTERIZE the frequency uncertainty interval for each RC. ESTIMATE the uncertainty intervals associated with parameter uncertainties, taking into account the state-of-knowledge correlation (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE- C15 from ASME/ANS RA-Sa-2009 [1] and PT-B10).	PROPAGATE parameter uncertainties (see DA-D3,HR-D6, HR-G8, and IE-C15 from the ASME/ANS RA-Sa-2009 [1]) using a Monte Carlo approach or other comparable means for each of the radionuclide categories. PROPAGATE parametric uncertainties in such a way that the state-of-knowledge correlation between event probabilities is taken into account from Level 1 PRA analysis through the end of the Level 2 PRA analysis (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE- C15 from ASME/ANS RA-Sa-2009 [1] and, PT-B10).
PT-E8	IDENTIFY sources of model uncertainty in the probabilistic treatment of severe accident progression [see Note (15)].		
PT-E9	CHARACTERIZE sources of model uncertainty in the probabilistic treatment of severe accident progression. For example, for each assumption and source of model uncertainty, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, changes to radionuclide release frequency, magnitude, or timing, or introduction of a new initiating event).		
PT-F9	DOCUMENT the characterization of the sources of model uncertainty and assumptions (as identified in PT-E8).		
PT-F12	DOCUMENT the characterization of uncertainties on RC consistent with PT-E6.		DOCUMENT percentile values from uncertainty propagation results obtained in PT-E6 to discuss confidence level (only applicable to parameter uncertainties and those model uncertainties explicitly characterized by a probability distribution).
	ST - Source Term Analysis		
ST-C1	IDENTIFY uncertain parameters influencing source terms for the representative sequences [see ST-C3].		
ST-C2	CHARACTERIZE the source of model uncertainty in the source term analysis [see Note (5)].		
ST-C3	EVALUATE effects of uncertainties based on selected generic or existing analysis and provide qualitative discussion of impact of uncertainties on source terms [see Note (5)].	PERFORM plant-specific analysis to quantify the impact of uncertain parameters within credible and justified bounds [see Note (5)].	GENERATE probability density functions expressing the uncertainty in parameters defining the source terms, including the release magnitude and timing [see Note (5)].
ST-D2	DOCUMENT the characterization of the sources of model uncertainty and assumptions (as identified in ST-C2).		

Table 1-1: Level 2 PRA Standard Supporting Requirements Related to Uncertainty

SR	Capability Category I	Capability Category II	Capability Category III
ER - Evaluation and Presentation of Results			
ER-B1	CHARACTERIZE the method(s) used to characterize parametric uncertainty of the Level 2 PRA analysis and INCLUDE results of quantitative assessments of uncertainty (if any).		
L3 - Interface Between Level 2 PRA and Level 3 PRA			
L3-B3	DOCUMENT the specific sources of Level 2 PRA uncertainty that are different from those used in the Level 1 analysis. Include sources of source term uncertainty, any assumptions made in the source term characterization and release assumption, and limitations of the models used.		

2. Parameter Uncertainty Distribution Definitions

This section only addresses modeling elements unique to the Level 2 PRA. For example, the 1-CHR top event in the bridge tree uses the 1-CCU-x fault trees for each of the eight containment fan coolers. These same eight fan cooler fault trees are used in the Level 1 analysis, and their basic event parameter uncertainty has already been addressed through that process (and is carried forward into the Level 2 analysis). Therefore, they are not further discussed here.

The modeling elements unique to the Level 2 PRA were identified by the following procedure:

1. The Level 1 internal events and internal flood event trees were solved at a truncation level of $1x10^{-12}$/yr.
2. The results for the 1-CD and 1-CD-FLI end states were gathered at a truncation level of $1x10^{-12}$/yr.
3. The Level 1 internal events and internal flood event trees were solved through the 1-PDS-Q event tree at a truncation level of $1x10^{-12}$/yr. (Thus, basic events that only appear in cutsets below this truncation level are neglected.)
4. The results for the 384 PDS end states were gathered at a truncation level of $1x10^{-12}$/yr.
5. A comparison was performed of the basic events from the PDS cut set results against the basic events from the 1-CD and 1-CD-FLI cut set results to identify those basic events in the PDS results that did not have a match.
6. The PDS cut set results without a match to the 1-CD and 1-CD-FLI cut set results were tabulated and are shown in the Tables in **Sections 2.1**, **2.2**, and **2.3** below.

Because the containment system top events in the 1-PDS-Q event tree are the only elements that add new basic events to the Level 1 cut sets, the basic events in the PDS results that do not have a match to the 1-CD and 1-CD-FLI basic events are unique to the bridge tree results.

2.1 Bridge Tree SSC Events

The structures, systems, and component (SSC)-related events contained in the bridge tree that have non-zero/non-unity values are provided in **Table 2-1**, along with information about their uncertainty characterization. Except where otherwise noted, these uncertainty distributions were generated by the Level 1 PRA team as part of their efforts to develop parameter uncertainty distributions for Level 1 events. Section 7 of the report on the reactor, at-power, Level 1 PRA for internal events (NRC, 2019a) documents how some of the distributions were selected for different types of basic events, as well as information about the use of template events to define correlation classes for addressing the *state-of-knowledge correlation*[1] between basic events. The 2010 update to NUREG/CR-6928 provides the prior distributions for Bayesian estimation of plant-specific failure probabilities, and for direct use in cases where plant-specific values did not appear to be supported by the data. In cases where there is insufficient plant-specific or industry data to support use of other distributions, it is considered acceptable to use a constrained non-informative distribution. Additional information on parameter uncertainty derivation and estimation are provided in Section 4 of NUREG/CR-6928. As discussed in NUREG/CR-7039, Volume 2, SAPHIRE defines the beta distribution, beta(α, β), in terms of the parameters β and μ where μ, the mean value of the beta distribution, is equal to

[1] The state-of-knowledge correlation is also discussed further in Section 3 of this report.

$\alpha/(\alpha+\beta)$. Similarly, SAPHIRE defines the gamma distribution, $\Gamma(r)$, in terms of the parameters r and λ, where λ is equal to r/μ and μ is the mean value of the distribution.

2.2 Bridge Tree Human Reliability Events

The human reliability analysis (HRA)-related events contained in the bridge tree that have non-zero/non-unity values are provided in **Table 2-2**. Error factors for the human error probabilities (HEPs) are assigned following rules established by the Electric Power Research Institute (EPRI) HRA Calculator (EPRI, 2013a), which are an expanded version of rules originally found in THERP guidance (NRC, 1983). The THERP guidance recommended higher error factors for smaller HEPs to reflect the greater uncertainties associated with infrequently occurring events, and recommended that the nominal HEP be assumed to be the median of the lognormal distribution. Nevertheless, the HEP point estimate here is assumed to be the mean value, a lognormal distribution is prescribed, and the error factor is determined based on the magnitude of the mean as follows:

HEP < 0.001	Lognormal distribution	Mean = point estimate	Error Factor (EF) = 10
0.001 <= HEP <= 0.3	Lognormal distribution	Mean = point estimate	EF = 5
0.3 < HEP <= 0.6	Lognormal distribution	Mean = point estimate	EF = 3
0.6 < HEP <= 0.9	Lognormal distribution	Mean = point estimate	EF = 2
0.9 < HEP	Lognormal distribution	Mean = point estimate	EF = 1

For the usage in this report, use of error factors based on the median for the HEPs would not be expected to have much of an effect, particularly given the adjustments made to constrain the lower error factors (to limit sampling of probabilities greater than 1, which is discussed later in **Section 2.5**).

2.3 Bridge Tree Passive Failures and Phenomenological Events

The phenomenologically-related events contained in the bridge tree that have non-zero/non-unity values are provided in **Table 2-3**, along with information about correlation class and how uncertainty is estimated. Recall that the process used to generate this list uses solved cutsets, and thus events that do not appear in any cutsets above truncation are excluded. In particular, understand that asymmetries in Train A versus Train B probabilities (e.g., for loss of 4160V bus initiators) are the reason that Train B containment spray nozzle plugging appears in the list while Train A nozzle plugging does not.

2.4 Containment Event Tree and Decomposition Event Tree SSC Events

The SSC-related events contained in the Level 2 PRA model that have non-zero/non-unity values are provided in **Table 2-4**, along with their uncertainty distribution. **Table 2-5** provides additional context for the relief valve failure modeling in the Level 2 PRA, versus that used in the Level 1 PRA and the draft Surry State-of-the-Art Reactor Consequence Analysis (SOARCA) uncertainty analysis (SNL, 2016a). For reference, **Table 2-6** shows the number of simulated lifts of power-operated relieve valves (PORVs) and safety/relief valves (SRVs) are shown for each of the MELCOR case studies.

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

<table>
<tr><th>Event</th><th>Event description</th><th>BE Class</th><th>Mean</th><th>Corr. Class</th><th>Uncertainty distribution</th></tr>
<tr><td>1-CIS-AOV-OO-2626_27B-CC</td><td>AOV HV-2626B ANDAOV HV-2627B FAIL TO OPERATE (HARDWARE)</td><td>System</td><td>2.93E-05</td><td>See related input basic events</td><td rowspan="3">Compound event; Distributions are developed by propagating the uncertainty of the input basic events, as based on the applied event computation function. The input basic events 1-AVFAL (AOV fails to operate on demand) and QAOVFTOP2-2NS (common cause failure [CCF] multiplier AOV FTOP 2-2 non-staggered T) have beta uncertainty distribution types. The reference plant PRA model was the source of information for these events.

1-AVFAL QAOVFTOP2-2NS
β 2.105E+4 4.861E+1
μ 6.640E-4 4.410E-2</td></tr>
<tr><td>1-CIS-AOV-OO-HV28_29B-CC</td><td>AOV HV-2628B AND AOV HV-2629B FAIL TO OPERATE (HARDWARE)</td><td>System</td><td>2.93E-05</td><td>See related input basic events</td></tr>
<tr><td>1-CIS-AOV-OO-HV780781-CC</td><td>AOV HV-0780 AND AOV HV-0781 FAILTO OPERATE (HARDWARE)</td><td>System</td><td>2.93E-05</td><td>See related input basic events</td></tr>
</table>

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-CIS-AOV-OO-HV_0780_	AOV HV-0780 FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	Uses the ZT-AOV-OO template event, which has a beta distribution. This template was based on the 2010 Standardized Plant Analysis Risk (SPAR) model update and plant-specific data. ZT-AOV-OO β 3.350E+3 μ 6.310E-4
1-CIS-AOV-OO-HV_0781_	AOV HV-0781 FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	
1-CIS-AOV-OO-HV_2626B	AOV HV-2626B FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	
1-CIS-AOV-OO-HV_2627B	AOV HV-2627B FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	
1-CIS-AOV-OO-HV_2628B	AOV HV-2628B FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	
1-CIS-AOV-OO-HV_2629B	AOV HV-2629B FAILS TO CLOSE (HARDWARE)	System	6.31E-04	AOV-OO	
1-CSR-CKV-CC-008	CS PUMP B SUCTION CV 008 FAILS TO OPEN	System	1.07E-05	CKV-CC	Uses the ZT-CKV-CC template event, which has a beta distribution. This template incorporates data from the 2010 update to the Nuclear Regulatory Commission (NRC) parameter estimate component reliability data sheets. ZT-CKV-CC β 4.684E+4 μ 1.070E-5
1-CSR-CKV-CC-016	CS PUMP B DISCHARGE CV 016 FAILS TO OPEN	System	1.07E-05	CKV-CC	
1-CSR-ESFAS-A	FAILURE OF CONTAINMENT SPRAY ACTUATION - TRAIN A	System	1.00E-03	None	Distribution Type: Constrained Noninformative
1-CSR-ESFAS-B	FAILURE OF CONTAINMENT SPRAY ACTUATION - TRAIN B	System	1.00E-03	None	Distribution Type: Constrained Noninformative

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-CSR-MDP-CF-RUN	CS PUMPS FAIL FROM COMMON CAUSE TO RUN	System	1.34E-05	See related input basic events	CCF event; Distributions are developed by propagating the related total failure probability event uncertainty and alpha factor parameter uncertainty (beta-distributed) through the SAPHIRE CCF event computation. See 1-CSR-MDP-FR-A and B in this table for uncertainty characterization. The uncertainty distribution type for the alpha factors ZA-MDP-FR-02A01 and ZA-MDP-FR-02A02 is a beta distribution. These templates incorporate data from version 4.5.2010.3 of the NRC's CCF database. ZA-MDP-FR-02A01 ZA-MDP-FR-02A02 β 4.247E+0 1.173E+2 μ 9.651E-1 3.490E-2
1-CSR-MDP-CF-START	CS PUMPS FAIL FROM COMMON CAUSE TO START	System	4.88E-05	See related input basic events	CCF event; Distributions are developed by propagating the related total failure probability event uncertainty and alpha factor parameter uncertainty (beta-distributed) through the SAPHIRE CCF event computation. See 1-CSR-MDP-FS-A and B in this table for uncertainty characterization. The uncertainty distribution type for the alpha factors ZA-MDP-FS-02A01 and ZA-MDP-FS-02A02 is a beta distribution. These templates incorporate data from version 4.5.2010.3 of the NRC's CCF database. ZA-MDP-FS-02A01 ZA-MDP-FS-02A02 β 9.687E+0 3.774E+2 μ 9.750E-1 2.500E-2
1-CSR-MDP-FR-A	CS PUMP A FAILS TO RUN	System	1.98E-04	See related input basic events	Compound event; input parameters ZT-MDP-FR-E (MDP fails to run) and ZT-MDP-FR-L (Standby MDP fails to run) have beta and gamma uncertainty distribution types, respectively. This template is based on the 2010 SPAR model update and plant-specific data.

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-CSR-MDP-FR-B	CS PUMP B FAILS TO RUN	System	1.98E-04	See related input basic events	<u>ZT-MDP-FR-E</u> β 2.009E+4 μ 6.310E-4 <u>ZT-MDP-FR-L</u> r 1.780E+0 λ 2.540E-6
1-CSR-MDP-FS-A	CS PUMP A FAILS TO START	System	1.00E-03	MDP-FS-NS	Distribution is a beta, which is based on template event ZT-MDP-FS-NS. This template is based on the 2010 SPAR model update. <u>ZT-MDP-FS-NS</u> β 1.290E+4 μ 1.000E-3
1-CSR-MDP-FS-B	CS PUMP B FAILS TO START	System	1.00E-03	MDP-FS-NS	
1-CSR-MDP-MA-A	CS PUMP A IS IN MAINTENANCE	System	7.12E-03	MDP-TM(CSR)	Uses the ZT-MDP-TM(CSR) template event, which has a beta distribution. This template incorporates data from the 2010 update to the NRC parameter estimate component reliability data sheets. <u>ZT-MDP-TM(CSR)</u> β 1.290E+4 μ 7.124E-3
1-CSR-MDP-MA-B	CS PUMP B IS IN MAINTENANCE	System	7.12E-03	MDP-TM(CSR)	

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-CSR-MOV-CC-HV9001A	CS PUMP A DISCHARGE MOV HV9001A FAILS TO OPEN	System	3.53E-04	MOV-CC	Uses the ZT-MOV-CC template event, which has a beta distribution. This template is based on the 2010 SPAR model update and plant-specific data. ZT-MOV-CC β 5.400E+4 μ 3.530E-4
1-CSR-MOV-CC-HV9001B	CS PUMP B DISCHARGE MOV HV9001B FAILS TO OPEN	System	3.53E-04	MOV-CC	
1-CSR-MOV-CC-HV9002A	CS SUMP A MOV HV9002A FAILS TO OPEN	System	3.53E-04	MOV-CC	
1-CSR-MOV-CC-HV9002B	CS SUMP B MOV HV9002B FAILS TO OPEN	System	3.53E-04	MOV-CC	
1-CSR-MOV-CC-HV9003A	CS SUMP A MOV HV9003A FAILS TO OPEN	System	3.53E-04	MOV-CC	
1-CSR-MOV-CC-HV9003B	CS SUMP B MOV HV9003B FAILS TO OPEN	System	3.53E-04	MOV-CC	

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-CSR-MOV-CF-HV9001AB	CS PUMP DISCHARGE MOVs HV9001A & HV9001B FAIL FROM COMMON CAUSE TO OPEN	System	1.18E-05	See related input basic events	CCF event; Distributions are developed by propagating the related total failure probability event uncertainty and alpha factor parameter uncertainty (beta-distributed) through the SAPHIRE CCF event computation. See the entries for the basic events 1-CSR-MOV-CC-HV9001A and B, 1-CSR-MOV-CC-HV9002A and B, and 1-CSR-MOV-CC-HV9003A and B in this table for their respective uncertainty characterization. For each of these three CCF events, the alpha factors are represented by the same two template events ZA-MOV-CC-02A01 and ZA-MOV-CC-02A02, which both have beta distributions. ZA-MOV-CC-02A01 ZA-MOV-CC-02A02 β 5.280E+1 1.645E+0 μ 1.700E-2 9.808E-1
1-CSR-MOV-CF-HV9002A2B	CS SUMP MOVs HV9002A & HV9002B FAIL FROM COMMON CAUSE TO OPEN	System	1.18E-05	See related input basic events	
1-CSR-MOV-CF-HV9003A3B	CS SUMP MOVs HV9003A & HV9003B FAIL FROM COMMON CAUSE TO OPEN	System	1.18E-05	See related input basic events	
1-CSR-MOV-CF-HV9017AB	CS PUMP SUCTION MOVs HV9017A & HV9017B FAIL FROM COMMON CAUSE TO CLOSE	System	5.66E-06	See related input basic events	CCF event; Distributions are developed by propagating the related total failure probability event uncertainty and alpha factor parameter uncertainty (beta-distributed) through the SAPHIRE CCF event computation. See the entries for the basic events 1-CSR-MOV-OO-HV9017A and B in this table for their respective uncertainty characterization. The alpha factors are represented by the following template events, ZA-MOV-OO-02A01 and ZA-MOV-OO-02A02, which both have beta distributions. ZA-MOV-OO-02A01 ZA-MOV-CC-02A02 β 8.062E-1 9.900E+1 μ 9.919E-1 8.080E-3

Table 2-1: Parameter Uncertainties Related to Bridge Tree SSC Events

<table>
<tr><th>Event</th><th>Event description</th><th>BE Class</th><th>Mean</th><th>Corr. Class</th><th>Uncertainty distribution</th></tr>
<tr><td>1-CSR-MOV-OO-HV9017A</td><td>CS TRAIN A MOV HV9017A FAILS TO CLOSE</td><td>System</td><td>3.53E-04</td><td>MOV-OO</td><td rowspan="2">Uses the ZT-MOV-OO template event, which has a beta distribution. This template was sourced from the 2010 SPAR model update and includes plant-specific data.

ZT-MOV-OO
β 5.400E+4
μ 3.530E-4</td></tr>
<tr><td>1-CSR-MOV-OO-HV9017B</td><td>CS TRAIN B MOV HV9017B FAILS TO CLOSE</td><td>System</td><td>3.53E-04</td><td>MOV-OO</td></tr>
<tr><td>1-ESF-SSD-FC-A5161A_B</td><td>SAFEGUARDS DRIVER CIRCUIT FAILS</td><td>System</td><td>5.04E-04</td><td>None</td><td>This event has a beta distribution with β and μ equal to 9.920E+2 and 5.040E-4, respectively.</td></tr>
</table>

Table 2-2: Parameter Uncertainties Related to Bridge Tree HRA Events

Event	Event description	BE Class	Mean	Corr. Class	Uncertainty distribution
1-OA-CS-RECIRC	OPERATORS FAILS TO ALIGN CS RECIRCULATION	HRA	9.60E-03	None	Lognormal: Mean = 9.6E-3 and error factor (EF) = 5
1-OA-CS-RECIRC-LD	OPERATORS FAILS TO ALIGN CS RECIRCULATION	HRA	5.90E-02	None	Lognormal: Mean = 5.9E-2 and EF = 5
1-OA-OP-PHASE-AH	OPERATOR FAILS TO MANUALLY INITIATE PHASE A ISOLATION	HRA	3.00E-03	None	Lognormal: Mean = 3E-3 and EF = 5

Table 2-3: Parameter Uncertainties Related to Bridge Tree Phenomenological Events

Event	Event description	BE Class	Mean	Corr. Class	Thoughts on estimating uncertainty
1-L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS	Phenom	1.110E-3	None	Lognormal distribution with EF = 10 (see **Table 6-1**)
1-CSR-NZL-CF-TRNAB	CS TRAIN A & B NOZZLES PLUGGED DUE TO COMMON CAUSE	Phenom	1.74E-06	See related input basic events	CCF event; Distributions are developed by propagating the related total failure probability event uncertainty and alpha factor parameter uncertainty (beta-distributed) through the SAPHIRE CCF event computation. See the entry for the basic events 1-CSR-NZL-PG-TRNB in this table for its uncertainty characterization. 1-CSR-NZL-PG-TRNA is also used as an input for this event and it shares the same attributes as 1-CSR-NZL-PG-TRNB. The alpha factors are represented by the following template events, ZA-CCF-RATE-02A01 and ZA-CCF-RATE-02A02, which both have beta distributions. ZA-CCF-RATE-02A01 ZA-CCF-RATE-02A02 β 4.166E+1 1.067E+3 μ 9.624E-1 3.760E-2

Table 2-3: Parameter Uncertainties Related to Bridge Tree Phenomenological Events

Event	Event description	BE Class	Mean	Corr. Class	Thoughts on estimating uncertainty
1-CSR-NZL-PG-TRNB	CS TRAIN B NOZZLES FAIL DUE TO PLUGGING	Phenom	2.40E-05	NZL-PG	Uses the ZT-NZL-PG template event, which has a gamma distribution. This template incorporates data from the 2010 update to the NRC parameter estimate component reliability data sheets. ZT-NZL-PG r 3.000E-1 λ 1.000E-6

Table 2-4: Parameter Uncertainties Related to Containment Event Tree (CET)/Decomposition Event Tree (DET) SSC Events

Event	Event description	BE Class	Mean	Grouping	Uncertainty distribution
1-L2-BE-ABFANS-IND-FAIL	Independent or Induced Failure of Aux Bldg Ventilation	System	0.3	None	Histogram (see **Table 6-2**)
1-L2-BE-ARVSTUCK-SGTR	Air relief valve (ARV) or MSRV Stick or are Kept Open During Isolated Steam Generator Tube Rupture (SGTR)	System	0.21	Relief valve	Histogram (see **Table 6-3**)
1-L2-BE-PZRVSTUCK-PORV	Pressurizer power-operated relief valves (PORVs) Do Not Fail Open During CD	System	0.07	Relief valve	Histogram (see **Table 6-4**)
1-L2-BE-PZRVSTUCK-SRV	Pressurizer SRVs Do Not Fail Open During CD	System	0.08	Relief valve	Histogram (see **Table 6-5**)

Table 2-5: Comparison of Relief Valve Failure Modeling

	Event	Used In	Used For	Per Demand Failure Probability	Failure Probability	Failure Probability Per Demand Basis
L3PRA Project Level 1 PRA	1-RCS-PRV-OO-RV045*A_	1-PVC and 1-PVC-ATWS	Pressurizer PORV randomly fails to close (FTC) (anticipated transient without scram [ATWS] or non-ATWS)		1.46E-3 per valve, given a challenge[1]	NUREG/CR-7037 (INL, 2010), Table 30
	1-RCS-SWP-OO-V8010*__	1-PVC	Pressurizer safety relief valve (SRV) randomly FTC		7.32E-4 per valve, given a challenge[2]	2010 Update to the Parameter Estimation Component Reliability Data Sheets, Section 4.2 (INL, 2012)
	1-RCS-SWP-OO-V8010*_W	1-PVC-ATWS	Pressurizer SRV FTC after passing water during ATWS		1.0E-1 per valve	NUREG/CR-6928, Table 5-1
	1-RCS-PRV-CC-RV0455*_	1-PZR-L	Pressurizer PORV randomly fails to open (FTO) during ATWS		3.51E-3 per valve	Reference-Plant-Specific Template. Updated with: 2010 SPAR Update, PORV
	1-RCS-PRV-CF-RV546A__	1-PZR-L	Pressurizer PORVs FTO from CC		1.044E-4	R-Type CCF Event (Composed of Random Failure Basic Events); Alpha Factors
	1-IE-SSBI/1-IE-SSBO	1-FPI-SSBI/1-FPI-SSBO	Initiating event (IE) for all secondary-side break initiators		Subsumed in IE probability	
	1-MSS-ADV-OO-VPV30**_	1-SVC-ARV	Main steam ARV challenged and FTC		1.73E-3 per valve, given a challenge[3]	2010 Update to the Parameter Estimation Component Reliability Data Sheets; Section 4.3, Build Date: 07/19/2012 (INL, 2012)
	1-MSS-RMB-OO-PSV_30**	1-SVC-SRV	Main steam SRV challenged and FTC		7.32E-4 per valve, given a challenge[3]	2010 Update to the Parameter Estimation Component Reliability Data Sheets; Section 4.2, Build Date: 07/19/2012 (INL, 2012)

Table 2-5: Comparison of Relief Valve Failure Modeling

<table>
<tr><th></th><th>Event</th><th>Used In</th><th>Used For</th><th colspan="3">Per Demand Failure Probability</th><th>Failure Probability</th><th>Failure Probability Per Demand Basis</th></tr>
<tr><td>L3PRA Project Level 1 PRA</td><td>1-MSS-ADV-CC-VPV30**</td><td>1-SVC-ARV and 1-SVC-SRV</td><td>Main steam ARV FTO</td><td colspan="3"></td><td>5.56E-4</td><td>2010 Update to the Parameter Estimation Component Reliability Data Sheets; Section 4.3, Build Date: 07/19/2012 (INL, 2012)</td></tr>
<tr><td rowspan="5">L3PRA Level 2 PRA</td><td>1-L2-PZRVSTUCK-PORV</td><td>1-L2-DET-PRESVE</td><td>Pressurizer PORV FTC during CD</td><td colspan="3">~4x10^{-3} (initial), ~3x10^{-3} (subsequent)</td><td>0.07, based on 30 valve cycles and a PORV per lift failure probability</td><td>NUREG/CR-7037 (INL, 2010), Table 18</td></tr>
<tr><td>1-L2-PZRVSTUCK-SRV</td><td>1-L2-DET-PRESVE</td><td>Pressurizer SRV FTC during CD</td><td colspan="3">~3x10^{-2} (initial), ~3x10^{-3} (subsequent)</td><td>0.08, based on 30 valve cycles and a SRV per lift failure probability based on main steam safety valve data</td><td>NUREG/CR-7037 (INL, 2010), Table 20 for main steam system (MSS) Code Safety Valves</td></tr>
<tr><td rowspan="3">1-L2-BE-ARVSTUCK-SGTR</td><td rowspan="3">1-CET</td><td rowspan="3">Probability of stuck-open secondary relief valve during SGTR</td><th></th><th>Main Steamline ARV</th><th>Main Steamline SRV</th><td rowspan="3">0.21 (ARV) and 0.08 (SRV), based on 30 valve cycles and ARV/SRV initial/subsequent lift failure probabilities; 0.21 is used in the model</td><td rowspan="3">NUREG/CR-7037 (INL, 2010), Tables 18 and 20</td></tr>
<tr><td>Automatic Demand – Initial</td><td>~3x10^{-3}</td><td>~3x10^{-2}</td></tr>
<tr><td>Automatic Demand – Subsequent</td><td>~1x10^{-2}</td><td>~3x10^{-3}</td></tr>
<tr><td rowspan="5">Draft Surry SOARCA UA (SNL, 2016a)</td><td>SVLAMFTC</td><td rowspan="5">Sampled in the individual realizations</td><td>Pressurizer SRV stochastic FTC</td><td colspan="3">Beta distribution (α=17.5, β=756.5)</td><td>0.64</td><td rowspan="5">Draft Surry SOARCA UA's interpretation of the NUREG/CR-7037 (INL, 2010) results</td></tr>
<tr><td>SVLAMFTO</td><td>Pressurizer SRV stochastic FTO</td><td colspan="3">Beta distribution (α=0.5, β=773.5)</td><td>0.01[5]</td></tr>
<tr><td>SVWTR</td><td>Pressurizer SRV FTC due to passing water</td><td colspan="3">Beta distribution (α=0.5, β=4.5)</td><td>0.04</td></tr>
<tr><td>SVFAILT</td><td>Pressurizer SRV thermal FTC</td><td colspan="3">Temperature-based rather than cycling-based</td><td>0</td></tr>
<tr><td>SVLAMFTC-SG</td><td>Main steam SRV stochastic FTC</td><td colspan="3">Beta distribution (α=17.5, β=756.5)</td><td>0.95</td></tr>
</table>

1 For the PORVs, the probability of a challenge is separate, and is initiator-specific (see fault tree 1-PVC). Probabilities taken from Table 11 of NUREG/CR-7037 (INL, 2010).

[2] For pressurizer safety valves to be challenged, at least one PORV must fail.

[3] For steam generator (SG) ARVs to be challenged during a transient, the steam dumps must fail (or be unavailable). For the SG SRVs to be challenged, the applicable ARV must fail.

[4] Individual valves only, none as the system of three valves in parallel.

Table 2-6: Pressurizer Valve Counts from MELCOR Analyses

	# of pressurizer PORV lifts		# of pressurizer SRV lifts	
Case	Prior to core damage	After core damage [1]	Prior to core damage	After core damage [1]
1	0	0	0	0
1A	0	0	114	4
1A1	0	0	114	4
1A2	0	0	114	4
1B	226	14	0	16
1B1	46	0	0	0
1B2	62	0	0	0
2	0	0	0	0
2A	0	0	0	0
2R1	0	0	0	0
2R2	0	0	0	0
3	0	0	135	11
3A1	0	0	135	0
3A2	0	0	132	0
3A3	0	0	132	0
3A4	0	0	135	30
4	0	0	0	0
5	0	0	0	0
5A	0	0	0	0
5B	0	0	0	0
5C	0	0	0	0
5D	0	0	0	0
6	362	23	0	0
6A	362	24	0	0
6B	362	23	0	0
6C	362	23	0	0
6D	362	24	0	0
6R1	362	23	0	0
7	0	0	67	8
7A	0	0	67	8
8	0	0	0	0
8A	0	0	0	0
8B	0	0	0	0
8R1	0	0	0	0
8R2	0	0	0	0
8BR1	0	0	0	0

[1] Understand that once vessel breach has occurred, no further cycles can occur, given the intrinsic reactor coolant system (RCS) depressurization.

2.5 Containment Event Tree and Decomposition Event Tree Human Reliability Events

The HRA-related events contained in the Level 2 PRA model that have non-zero/non-unity values are provided in **Table 2-7**. Uncertainty distributions are used following the same rules as previously outlined in **Section 2.2**.

2.6 Containment Event Tree and Decomposition Event Tree Phenomenological Events

The phenomenologically-related events contained in the Level 2 PRA model that have non-zero/non-unity values are provided here. For these phenomenological events, default lognormal distributions were assigned based on the point estimate magnitude (with the point estimate being the mean), analogous to what is done for HEPs in **Sections 2.2 and 2.5** (repeated below). In several cases, these default distributions were then replaced by histograms, as identified in **Table 2-8** and expanded upon in **Section 6**.

Failure probability < 0.001	Lognormal distribution	Mean = point estimate	EF = 10
0.001 <= Failure probability <= 0.3	Lognormal distribution	Mean = point estimate	EF = 5
0.3 < Failure probability <= 0.6	Lognormal distribution	Mean = point estimate	EF = 3
0.6 < Failure probability <= 0.9	Lognormal distribution	Mean = point estimate	EF = 2
0.9 < Failure probability	Lognormal distribution	Mean = point estimate	EF = 1

The default scheme, although developed for human reliability purposes, was found to be appropriate for the spread of uncertainty in otherwise-uninformed phenomenological parameters. This treatment has the effect of reflecting the mean as higher than the median (i.e., average value is skewed toward more pessimistic failure probabilities), while conversely resulting in more random sampling below the mean. The authors considered using log-triangular distributions to promote more similarity between the mean/mode/median, but this is not a standard distribution form accepted by SAPHIRE.

There are two fundamental types of phenomenological failure probabilities in the model, those that are purely subjective (practitioner judgment) and those that are the result of underlying computations (e.g., ERPRA-BURN calculations). It is acknowledged that some will view these distributions as the uncertainty of an uncertainty (i.e., the split fraction itself is an uncertainty), particularly in the cases where the split fraction originates from a load-capacity interference concept using underlying calculations. Nevertheless, the assignment of a distribution on the split fraction is appropriate and acceptable, and representative of a degree-of-belief about the parameter estimation that aids in propagating the uncertainty of all input parameters in the logic model. In support of this argument, consider that the underlying models in some cases are simplified (e.g., ERPRA-BURN) and have inherent limitations (e.g., feedback effects) that are appropriate and acceptable, but not otherwise captured by the split fraction itself.

Table 2-8 shows the assigned basic event uncertainty distributions for each event. Each event is also assigned to a grouping category. The grouping is not used in the parameter uncertainty propagation but is used in the model uncertainty treatment discussed later in **Section 4**. The grouping does not mean that the basic events have correlated uncertainty distributions. In SAPHIRE, correlating two variables infers that they have identical distributions that are generated from the same data set. Since this is rarely, if ever, the case with the Level 2 phenomenological events, sampling correlation is not used.

Table 2-7: Parameter Uncertainties Related to CET/DET HRA Events

Event	Event description	Basic Event (BE) Class	Mean	Grouping	Uncertainty Distribution
1-L2-BE-MANUALTDAFW-GEN	Failure of Manual Extension of TDAFW in station blackout (SBO)	HRA	0.65	HRA	Lognormal: Mean = 0.65 and EF = 1.5*
1-L2-OP-SAG1	Operator Fails to Carry Out SAG-1 (Open 2 ARVs and Feed SGs)	HRA	0.4	HRA	Lognormal: Mean = 0.4 and EF = 2.4*
1-L2-OP-SAG2-1	Operator Fails to Carry Out SAG-2 (Open all ARVs - Not Depress)	HRA	0.07	HRA	Lognormal: Mean = 0.07 and EF = 5
1-L2-OP-SCG1-1	Operator Fails to Carry Out SCG-1 (Spray Containment w/ Firewater)	HRA	0.6	HRA	Lognormal: Mean = 0.6 and EF = 1.6*
1-L2-OP-SCG1-2	Operator Fails to Carry Out SCG-1 (F&B SGs)	HRA	0.1	HRA	Lognormal: Mean = 0.1 and EF = 5
1-L2-OP-SCG1-3	Operator Fails to Carry Out SCG-1 (F&B SGs - Late)	HRA	0.5	HRA	Lognormal: Mean = 0.5 and EF = 1.9*
1-L2-OP-SCG1-4	Operator Fails to Carry Out SCG-1 (Spray Containment w/ Cont. Spray System)	HRA	0.1	HRA	Lognormal: Mean = 0.1 and EF = 5

* Due to the relatively high mean failure probability of this basic event, the original error factor resulted in some Monte Carlo samples being discarded by SAPHIRE during the uncertainty analysis, because the sampled probability exceeded 1.0. As such, PRA practitioner judgment was used to adjust the error factor to minimize the number of discarded samples. In some cases, the adjusted error factor still resulted in a significant number of discarded samples. For these cases, a threshold value for the error factor was determined using an approach that preserves the mean value and anchors the 95th percentile of the distribution to a value of approximately 0.95. This replacement did not affect the point estimate calculations, which use the mean values.

Table 2-8: Parameter Uncertainties Related to CET/DET Phenomenological Events

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-ABFH2-FANS	Aux Bldg Failure due to Combustion (Ventilation)	1-L2-ABFH2	User judgment informed by MELCOR results	0.5	Interfacing systems loss of coolant accident (ISLOCA)	Histogram (see **Table 6-9**)
1-L2-BE-ABFH2-NOFANS	Aux Bldg Failure due to Combustion (No Ventilation)	1-L2-ABFH2	User judgment informed by MELCOR results	0.9	ISLOCA	Histogram (see **Table 6-10**)
1-L2-BE-BMT-CHR	Basemat melt through (BMT) Occurs Given containment heat removal (CHR) and No Preceding Over-Pressure Failure	1-L2-BMT	Adaptation of MELCOR results	0.87	Ablation	Lognormal: Mean = 0.87 and EF = 1.1*
1-L2-BE-BMT-NCHR	BMT Occurs Given No CHR and No Preceding Over-Pressure Failure	1-L2-BMT	Adaptation of MELCOR results	0.66	Ablation	Lognormal: Mean = 0.66 and EF = 1.5*
1-L2-BE-BMT-NCHRNFW	BMT Occurs Given No CHR No FW and No Preceding Over-Pressure Failure	1-L2-BMT	Adaptation of MELCOR results	0.92	Ablation	Lognormal: Mean = 0.92 and EF = 1.04*
1-L2-BE-CCI-DISP	Molten core concrete interaction (MCCI) occurs - Debris is not dispersed despite high-pressure melt ejection (HPME)	1-L2-MCCI	User judgment informed by past studies	0.01	None	Lognormal: Mean = 0.01 and EF = 5
1-L2-BE-CCI-FLABV	MCCI occurs - Top flooding does not quench debris	Not presently used	User judgment informed by past studies	0.9	Debris Cooling	Lognormal: Mean = 0.9 and EF = 1.05*
1-L2-BE-CCI-FLBEL	MCCI occurs - Existing cavity water does not quench debris	1-L2-MCCI	User judgment informed by past studies	0.9	Debris Cooling	Lognormal: Mean = 0.9 and EF = 1.05*

Table 2-8: Parameter Uncertainties Related to CET/DET Phenomenological Events

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-CONTOP-NCHR	Containment Overpressure Failure Late (No CHR)	1-L2-CONTOP-L	Adaptation of MELCOR results	0.99	None	No distribution
1-L2-BE-EVSE-GEN	Ex-Vessel Steam Explosion Occurs	1-L2-EVSE	User judgment informed by past studies	0.1	Other Energetic	Lognormal: Mean = 0.1 and EF = 5
1-L2-BE-H2CF-E-GEN	Global Deflagration Fails Containment At/Around vessel breach (VB)	1-L2-H2CF-E	ERPRA-BURN results	1.0E-4	Combustion	Lognormal: Mean = 1E-4 and EF = 10
1-L2-BE-H2CF-L-CHRNPB	Late containment failure (CF) from Burn (with CHR, without prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.74	Combustion	Lognormal: Mean = 0.74 and EF = 1.3*
1-L2-BE-H2CF-L-CHRPB	Late CF from Burn (with CHR, with prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.18	Combustion	Lognormal: Mean = 0.18 and EF = 5
1-L2-BE-H2CF-L-NACNPB	Late CF from Burn (without alternating current [AC], without prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.7	Combustion	Lognormal: Mean = 0.7 and EF = 1.4*
1-L2-BE-H2CF-L-NACPB	Late CF from Burn (without AC, with prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.62	Combustion	Lognormal: Mean = 0.62 and EF = 1.5*
1-L2-BE-H2CF-L-NCHRNPB	Late CF from Burn (without CHR, without prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.33	Combustion	Lognormal: Mean = 0.33 and EF = 2.9*
1-L2-BE-H2CF-L-NCHRPB	Late CF from Burn (without CHR, with prior burn)	1-L2-H2CF-L	ERPRA-BURN results	0.26	Combustion	Lognormal: Mean = 0.26 and EF = 3.8*
1-L2-BE-H2DET-L-CHR	Late Detonation with CHR	1-L2-H2DET-L	ERPRA-BURN results	0.66	Combustion	Lognormal: Mean = 0.66 and EF = 1.5*
1-L2-BE-H2IGN-E-PB	Combustion in Containment at VB given Prior Burn	1-L2-H2IGN-E	ERPRA-BURN results	0.999	Combustion	No distribution

Table 2-8: Parameter Uncertainties Related to CET/DET Phenomenological Events

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-H2IGN-VE-GEN	Combustion in Containment before VB (General)	1-L2-H2IGN-VE	ERPRA-BURN results	0.64	Combustion	Lognormal: Mean = 0.64 and EF = 1.5*
1-L2-BE-H2IGN-VE-SBO	Combustion in Containment before VB (SBO)	1-L2-H2IGN-VE	ERPRA-BURN results	0.24	Combustion	Lognormal: Mean = 0.24 and EF = 4.1*
1-L2-BE-H2IGNSRC-E-AC	Ignition Source in Containment at VB, with Power	1-L2-H2IGNSRC-E	User judgment	0.99	Combustion	No distribution
1-L2-BE-H2IGNSRC-E-NAC	Ignition Source in Containment at VB, without AC Power	1-L2-H2IGNSRC-E	User judgment	0.5	Combustion	Histogram (see **Table 6-6**)
1-L2-BE-H2IGNSRC-L-AC	Ignition Source in Containment Late, with Power	1-L2-H2IGNSRC-L	User judgment	0.99	Combustion	No distribution
1-L2-BE-H2IGNSRC-L-NAC	Ignition Source in Containment Late, without AC Power	1-L2-H2IGNSRC-L	User judgment	0.3	Combustion	Histogram (see **Table 6-7**)
1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power	1-L2-H2IGNSRC-VE	User judgment	0.99	Combustion	No distribution
1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power	1-L2-H2IGNSRC-VE	User judgment	0.1	Combustion	Histogram (see **Table 6-8**)
1-L2-BE-INDHLF-MP	Induced Hot Leg Failure (Intermediate pressure)	1-L2-INDHLF	User judgment informed by MELCOR results	0.5	Induced Piping Fail	Lognormal: Mean = 0.5 and EF = 2.1*
1-L2-BE-INDSGTR-HDL	Induced SGTR given High/Dry/Low	1-L2-INDSGTR	Consequential steam generator tube rupture (C-SGTR) calculator	0.02	Induced Piping Fail	Lognormal: Mean = 0.02 and EF = 5

Table 2-8: Parameter Uncertainties Related to CET/DET Phenomenological Events

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-ISLOCASUBM-LRG	ISLOCA Break Not Submerged or Significantly Scrubbed for Large ISLOCAs	1-L2-VSUBM-E	Expert elicitation and user judgment	0.29	ISLOCA	Histogram (see **Table 6-11**)
1-L2-BE-ISLOCASUBM-SM	ISLOCA Break Not Submerged or Significantly Scrubbed for Small ISLOCAs	1-L2-VSUBM-E	User judgment	0.5	ISLOCA	Histogram (see **Table 6-12**)
1-L2-BE-IVREC	No in-vessel retention (IVR), Vessel Breach Occurs	1-L2-IVREC	User judgment informed by past studies	0.5	None	Lognormal: Mean = 0.5 and EF = 2.1*
1-L2-BE-IVSE-LP	In-vessel steam explosion (IVSE) Fails Containment (Low RCS Pressure)	1-L2-IVSE	User judgment informed by past studies	2.0E-4	Other Energetic	Lognormal: Mean = 2E-4 and EF = 10
1-L2-BE-IVSE-NLP	IVSE Fails Containment (High or Inter RCS Press)	1-L2-IVSE	User judgment informed by past studies	1.0E-4	Other Energetic	Lognormal: Mean = 1E-4 and EF = 10
1-L2-BE-RCP480GPM-DEP	Reactor coolant pump (RCP) Leak Rate Sufficient to Cause Partial Depress. (SBO)	1-L2-SUM-BREAKSIZE	WOG 2000 model	2.5E-3	None	Lognormal: Mean = 2.5E-3 and EF = 10**
1-L2-BE-VROCK-HP	Vessel Thrust Fails Containment (High RCS Pressure)	1-L2-VROCK	Adaptation of ERPRA-ROCKET results	1.0E-4	Other Energetic	Lognormal: Mean = 1E-4 and EF = 10

* Due to the relatively high mean failure probability of this basic event, the original error factor resulted in some Monte Carlo samples being discarded by SAPHIRE during the uncertainty analysis, because the sampled probability exceeded 1.0. As such, PRA practitioner judgment was used to adjust the error factor to minimize the number of discarded samples. In some cases, the adjusted error factor still resulted in a significant number of discarded samples. For these cases, a threshold value for the error factor was determined using an approach that preserves the mean value and anchors the 95th percentile of the distribution to a value of approximately 0.95. This replacement did not affect the point estimate calculations, which use the mean values.

** An error factor of 5 should have been assigned, given the point estimate. This error has been recorded in Appendix A to the main body of this report.

3. Parameter Uncertainty Propagation and Results

The parameter uncertainty distributions presented in Section 2 were loaded into the SAPHIRE model. The uncertainty propagation was performed (release category by release category, and separately for all release categories at once) using SAPHIRE default options (Monte Carlo, 5000 samples, no specified random number seed). The results are shown in **Table 3-1**.

Table 3-1: Parameter Uncertainty Propagation Results by Release Category

All values are /year		**5th Percentile**	**Median**	**Mean**	**Point Estimate**	**95th Percentile**	**Standard Deviation**
Propagated One RC at a time	BMT	8.2E-08	4.7E-07	8.5E-07	8.2E-07	2.8E-06	1.4E-06
	CIF*	2.0E-09	1.8E-08	6.3E-08	6.5E-08	2.2E-07	2.0E-07
	CIF-SC*	3.2E-16	7.2E-13	1.1E-11	1.1E-11	4.4E-11	5.2E-11
	ECF	2.0E-10	2.0E-09	6.4E-09	6.5E-09	2.3E-08	2.4E-08
	ICF-BURN	1.3E-06	5.3E-06	8.2E-06	8.7E-06	2.3E-05	1.0E-05
	ICF-BURN-SC	4.2E-07	1.5E-06	2.4E-06	2.5E-06	6.8E-06	3.6E-06
	ISGTR	3.1E-08	2.5E-07	5.8E-07	5.8E-07	2.0E-06	2.3E-06
	LCF	5.9E-06	2.0E-05	2.8E-05	2.9E-05	7.5E-05	2.7E-05
	LCF-SC	2.4E-07	1.7E-06	3.4E-06	3.3E-06	1.1E-05	6.0E-06
	NOCF	5.0E-06	1.7E-05	2.6E-05	2.4E-05	7.2E-05	3.4E-05
	SGTR-C	1.7E-09	1.5E-08	3.7E-08	3.7E-08	1.4E-07	7.4E-08
	SGTR-O	1.2E-09	9.2E-09	2.0E-08	2.1E-08	7.2E-08	4.2E-08
	SGTR-O-SC	4.2E-08	1.4E-07	2.1E-07	2.2E-07	6.0E-07	2.4E-07
	V	4.4E-10	4.3E-09	9.7E-09	9.7E-09	3.5E-08	2.2E-08
	V-F	8.3E-09	6.5E-08	1.3E-07	9.8E-08	4.2E-07	2.0E-07
	V-F-SC	2.5E-08	1.7E-07	2.9E-07	2.3E-07	9.5E-07	3.8E-07
	SUM			7.0E-05	7.0E-05		
All-at-once		1.8E-05	5.0E-05	6.9E-05	7.0E-05	1.8E-04	7.2E-05

* CET sequence 74 should have been assigned to release category 1-REL-CIF-SC rather than 1-REL-CIF, since in-vessel recovery occurs. The change would raise the 1-REL-CIF-SC frequency to 5×10^{-9}/yr, while reducing the 1-REL-CIF frequency to ~5.8×10^{-8}/yr (the 1-CET-074 sequence frequency is ~5×10^{-9}/yr), discounting any minimization that would occur. This would not affect the release categories' percent contribution (<0.1% and 0.1%, respectively).

A subtlety regarding the tabulation of the uncertainty distributions for individual release categories is that they do not consider cross-category correlation (i.e., they are done in isolation of the effect that the sampled value would have on other release category frequencies). Further complicating the picture is that only certain types of parameter correlations are addressed in the sampling approach. The so-called *state-of-knowledge correlation* is addressed in the sampling approach used in the SAPHIRE. This correlation is addressed by assigning correlation classes to those basic event parameters that are derived from the same data set. The parameters that belong to the same correlation class use the exact same sampled value for each trial of the sample. However, there may be other correlations that exist among parameter uncertainty distributions. For example, there may be physical dependencies on accident phenomena that correlate the uncertainties of various parameters. These types of physical dependencies between parameters are not addressed in the sampling approach. As such, in the uncertainty propagation, various parameters are sampled independently, even if they do have some form of physical dependence (e.g., in reality the uncertainty of having an ignition source early during a station blackout would have some relationship to the uncertainty of having an ignition source later in that same station blackout). Additional study could be undertaken to understand and address the potential impacts of parameter dependencies on the uncertainty propagation; however, the current approach is consistent with the state-of-practice for treating uncertainty propagation for PRA results.

To provide additional information, uncertainty was also propagated through the combination of release categories comprising large early release frequency (LERF) and large release frequency (LRF), for different accident termination time assumptions. Again, cross-parameter correlation is not addressed. The results are provided in **Table 3-2**. Translating these frequencies to fractional contributions is not readily achievable, as it would require SAPHIRE to synchronously calculate the numerator (LERF or LRF set of release categories) and denominator (all release categories) for each trial, and then subsequently construct the distribution of the fractional value from those results, which it does not do.

Table 3-2: Parameter Uncertainty Propagation Results by Risk Surrogate

		Assumed Accident Termination Time*		
		36 hrs after severe accident mitigation guideline (SAMG) entry	**60 hrs after SAMG entry**	**7 days after initiator**
LERF (early fatalities)	5th percentile	1.5E-07	1.5E-07	1.5E-07
	Median	6.2E-07	6.2E-07	6.2E-07
	Mean	9.9E-07	9.9E-07	9.9E-07
	Point Estimate	9.0E-07	9.0E-07	9.0E-07
	95th Percentile	2.8E-06	2.8E-06	2.8E-06
LRF	5th percentile	1.9E-06	1.9E-06	9.8E-06
	Median	6.4E-06	6.4E-06	2.9E-05
	Mean	9.4E-06	9.4E-06	4.1E-05
	Point Estimate	1.0E-05	1.0E-05	4.3E-05
	95th Percentile	2.6E-05	2.6E-05	1.1E-04

* This refers to the time after SAMG entry (in the case of the first two categories), which can range quite a bit depending on the scenario. The third category is measured from the start of the accident, and always occurs well after the first two categories. In viewing these results, understand that limitations in the HRA and the phenomenological modeling make the longer-term results quite uncertain, and potentially pessimistic.

4. Model Uncertainty Alternative Treatment Definition

The following sub-sections decompose the Level 2 PRA model uncertainty into categories, identify the sources of the uncertainty within a given category, propose alternative treatments that can be used to explore this uncertainty in one or more sensitivity analyses, and provide the results of the sensitivity analyses that were performed. While many potential sources of model uncertainty are identified throughout the following subsections, not all sources of model uncertainty were evaluated quantitatively. The reduction of the identified uncertainties in each category to a couple of sensitivity studies reflects trade-offs between the importance of various issues, the degree to which they are adequately addressed in other studies, the extent to which they can readily and meaningfully be explored here, and resource constraints.

The tables listed below provide information on the identified sources of model uncertainty:

- Table 4-1: Identified Uncertainties in Plant Damage State Binning
- Table 4-4: Identified Uncertainties in Equipment Performance Modeling
- Table 4-6: Identified Uncertainties in Human Reliability Modeling
- Table 4-9: Uncertainties for In-Vessel Accident Progression Modeling
- Table 4-13: Identified Uncertainties in Induced RCS Component Failure Modeling
- Table 4-16: Uncertainties for Vessel Breach and Associated Energetic Event Modeling
- Table 4-19: Uncertainties for Combustible Gas Modeling
- Table 4-24: Uncertainties in Long-Term Containment Pressurization and Failure Modeling
- Table 4-28: Uncertainties for Ex-Vessel Coolability and MCCI Modeling
- Table 4-29: Uncertainties in ISLOCA Modeling
- Table 4-32: Uncertainties in SGTR and Induced SGTR Modeling
- Table 4-36: Uncertainties in Containment Isolation Failure Modeling
- Table 4-38: Uncertainties in Release Pathway Modeling
- Table 4-39: Uncertainties in Other Fission Product and Emergency Preparedness-Related Modeling
- Table 4-40: Uncertainties in Accident Termination Modeling
- Table 4-42: Uncertainties in MELCOR Solution Robustness

In viewing the tables below, many uncertainties are cast in terms of how they are modeled in MELCOR. In many ways, MELCOR serves as the NRC's repository for severe accident knowledge. It should be understood that many of the limitations that are described in terms of MELCOR modeling, are actually limitations in the current state-of-knowledge. Also, in some cases, the uncertainties are simply input required by the code (e.g., number of valve cycles prior to failure to close).

4.1 Plant Damage State Binning

4.1.1 Identified Uncertainties in Plant Damage State Binning

Table 4-1: Identified Uncertainties in Plant Damage State Binning

Item	Description	Other comments
Component/system mission times	The Level 1 PRA generally uses a 24-hour mission time when translating failure rate information to failure probabilities, as well as in screening out some modeling (e.g., lack of modeling of room cooling for containment spray pumps due to >24-hour heatup with failure probability for CCUs based on 24 hours of operation).	These systems may be credited in the Level 2 for longer time periods, particularly CCUs.
# of PDS bins	PDS binning necessarily causes the collapse of functionally similar cutsets into a combined representation, and the number of PDS bins must “reasonably” represent the variation in the Level 1 cutsets.	Examples include selection of a representative fail-to-run time (e.g., for TDAFW) or time-in-maintenance (e.g., for ac buses that affect battery depletion) EPRI 1016737 (EPRI, 2008) considers this to be a level-of-detail uncertainty
Specific PDS binning assumptions	The PDS binning was done in a manner to explicitly identify instances where sequence binning (accident type, SG cooling, refueling water storage tank [RWST] availability, and emergency core cooling system [ECCS] availability) was based on key underlying assumptions. Additional underlying assumptions (e.g., assumptions about consequential loss of offsite power [LOOP]-induced SBO) are also captured in the logic model report.	The linkage rules are tailored to facilitate investigating the effects of most of these assumptions.
Partial/degraded performance not credited in Level 1	Effects are missed for systems that come into play but do not fulfill their Level 1 mission.	This uncertainty may be effectively duplicative to other uncertainties identified above.
Treatment of PORVs and battery depletion	This could affect the time to battery depletion, which could in turn affect SBO scenarios (and partial loss of ac events that affect the direct current [dc] system).	The PORVs are solely solenoid powered, such that they would be a drain on dc power, if they cycled numerous times. This affect is not captured in the static battery depletion time assumed in the MELCOR model (and the Level 1 PRA). Conversely, the battery depletion time prescribed (4 hours for the safety-related dc batteries and 2 hours for the switchyard batteries) represent one possible (and potentially pessimistic) estimate.

Table 4-1: Identified Uncertainties in Plant Damage State Binning

Item	Description	Other comments
Effect of current CET modeling of long-term blind feeding	Since long-term blind feeding of SGs during SBO prior to core damage was moved from the Level 1 to the Level 2 PRA, and since it is captured in the first 1-CET top wherein success proceeds directly to NOCF, all other CET modeling (e.g., early containment failure modes, TISGTR) is not applied to this release frequency.	In reality, some portion of this sequence frequency would belong in other release categories, were the model to have passed the sequence through the remaining CET tops; whether the profile of this frequency would be similar to the profile of the remainder of the Level 2 results is debatable given the very long time frame involved (~5 days until core damage).

4.1.2 Alternative Treatment(s) of Uncertainties in Plant Damage State Binning

Some of the model uncertainties captured above are engrained in the model in a way that is not readily investigable, while others are more scrutable. Four sensitivity calculations were proposed:

MU-1.1A A re-quantification of the SAPHIRE model was performed in which all of the "indeterminate" linkage rules associated with the 4 PDS top events are assigned in the more pessimistic direction (i.e., if the availability of the equipment was ambiguous, then the equipment was assumed unavailable).

MU-1.1B A re-quantification of the SAPHIRE model was performed in which all of the "indeterminate" linkage rules associated with the 4 PDS top events are assigned in the more optimistic direction (i.e., if the availability of the equipment was ambiguous, then the equipment was assumed available).

MU-1.2 A MELCOR calculation was performed (based on Case 1A discussed in Section 1.1 of Appendix B) in which battery depletion occurs at 13 hours, leading to loss of the pressurizer PORVs and loss of TDAFW feed at that time. SG ARVs are closed when SG narrow range level falls below 10%, though it is noted that without dc power, SG level may not be known. This case will show both the effect of these timing assumptions on source term relative to MELCOR Cases 1, 1A, and 1B, as well as the timing of containment failure.

MU-1.3 Through simple hand calculations, it was shown how the baseline release category profile might differ if the frequency associated with CET sequence #1 (successful blind feeding for 5 days) was routed through the CET (i.e., accounted for early containment failure and TISGTR mechanisms), rather than directly to a NOCF end-state.

4.1.3 Sensitivity Analysis for Level 1 PRA Sequence Logic Related to SG Cooling, ECCS Availability or RWST Availability (MU-1.1A & B)

When constructing the PDS logic (and subsequent SAPHIRE linkage rules), several instances were identified where SG cooling, ECCS availability or RWST availability was ambiguous in the Level 1 PRA sequence logic. The purpose of this sensitivity analysis is to investigate the effects of these assumptions on the Level 2 results, by taking a more pessimistic or optimistic view about the relevant SSCs' availability. To do this, the PDS linkage rules were edited to assign all so-called indeterminate states to being unavailable (rather than the mix of availabilities and unavailabilities that exists in the base model). The manual PDS mapping was updated, the new SAPHIRE model was linked, and the resulting sequences were compared to the manual mapping to ensure that the new linkage rules were working as intended. The model was then re-quantified. The results are shown in **Table 4-2**. As can be seen, this change in the PDS logic did not have a notable effect on the contribution of any release categories. PDS binning necessitates important modeling assumptions and these modeling assumptions can influence the PRA results; no such effect was manifested through this sensitivity study. Note that this finding is somewhat inter-related with the broader aspects and limitations of the study, in that reduced availability of SG cooling or ECCS is of less relevance when the core damage frequency is dominated by station blackout and total loss of NSCW scenarios, and when the Level 2 PRA does not credit post-core-damage actions for station blackouts. Given the results obtained, looking at more optimistic assumptions (MU-1.1B) was not pursued as it was expected that it would demonstrate a similar lack of sensitivity.

Table 4-2: Results of Sensitivity MU-1.1

	Percent Contribution to Total Release Frequency		**Difference in Frequency**
Release Category	**Base Model**	**MU-1.1A**	0.10%
1-REL-BMT	1.2%	1.1%	0.00%
1-REL-CIF	0.1%	0.1%	0.00%
1-REL-CIF-SC	0.0%	0.0%	0.00%
1-REL-ECF	0.0%	0.0%	0.20%
1-REL-ICF-BURN	12.4%	12.2%	0.00%
1-REL-ICF-BURN-SC	3.5%	3.5%	0.00%
1-REL-ISGTR	0.8%	0.8%	0.00%
1-REL-LCF	41.8%	41.8%	0.00%
1-REL-LCF-SC	4.7%	4.7%	0.30%
1-REL-NOCF	34.6%	34.9%	0.00%
1-REL-SGTR-C	0.1%	0.1%	0.00%
1-REL-SGTR-O	0.0%	0.0%	0.00%
1-REL-SGTR-O-SC	0.3%	0.3%	0.00%
1-REL-V	0.0%	0.0%	0.00%
1-REL-V-F	0.1%	0.1%	0.00%
1-REL-V-F-SC	0.3%	0.3%	0.10%

4.1.4 Sensitivity Analysis for Safety-Related Battery Life Extension from 4 to 13 hours (MU-1.2)

Described in this section is a sensitivity study based upon Case 1A in which the safety-related battery life is extended from 4 to 13 hours. With battery depletion delayed by 9 hours, auxiliary feedwater (AFW) maintains the SG water level until 13 hours, keeping the core cooled. Following the loss of DC power and cooling, the accident progresses in much the same way as in Case 1A. Containment overpressure failure occurs at 85.5 hours, ~18 hours later than in Case 1A. Because of this delay in a failure flow path, the environmental release of iodine and cesium are less, as seen in **Figure 4-1** and **Figure 4-2,** but essentially linear after containment failure. This sensitivity suggests that earlier/later battery depletion time will have a somewhat linear impact on the timing of containment failure and environmental releases. Within the range of reasonable battery depletion times, this effect would not translate to changes in the release categorization (e.g., containment failure still occurs in the sensitivity study), except when an earlier simulation truncation time is applied.

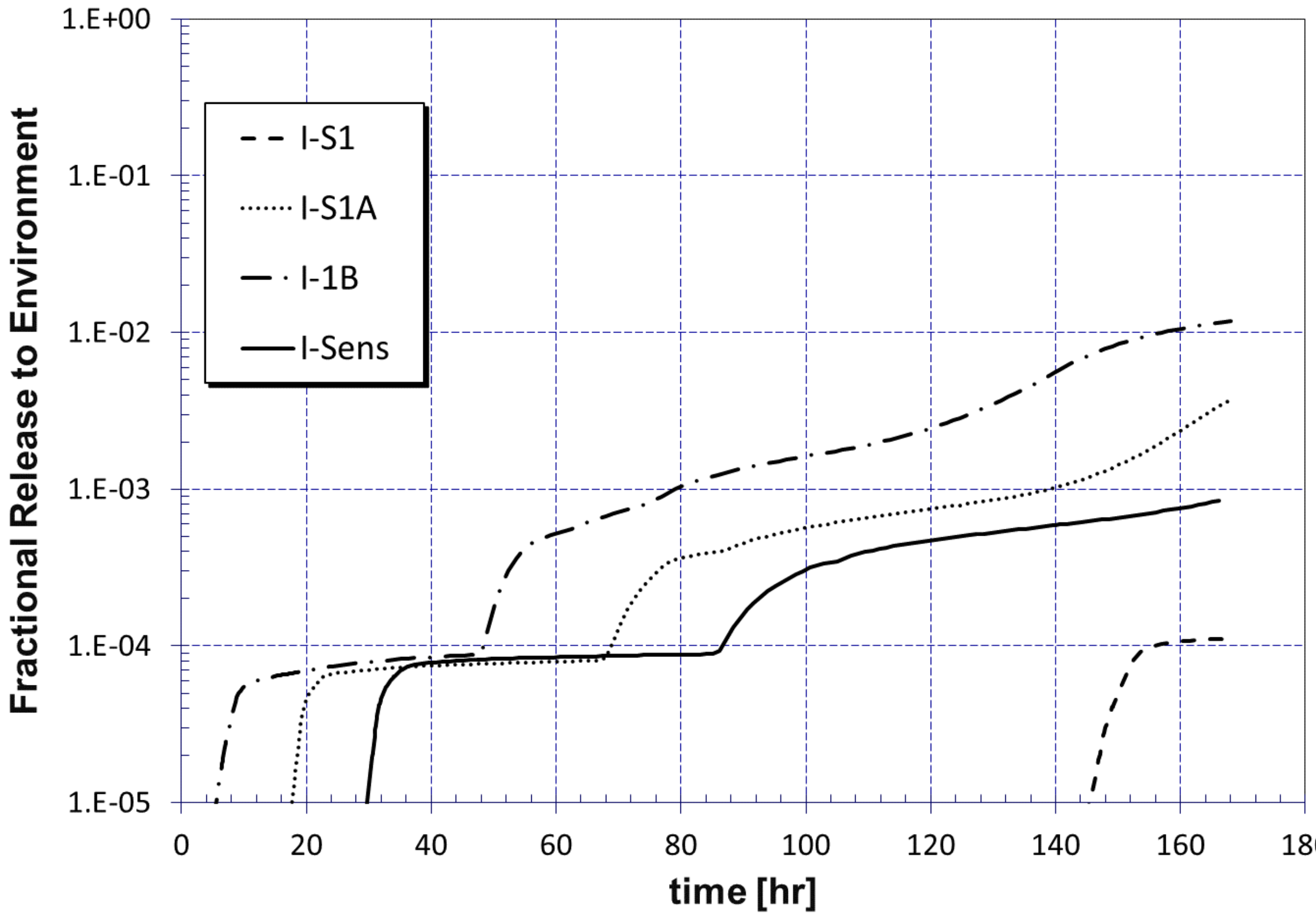


Figure 4-1: Fractional Release of Iodine to the Environment for Cases 1, 1A, 1B and MU-1.2

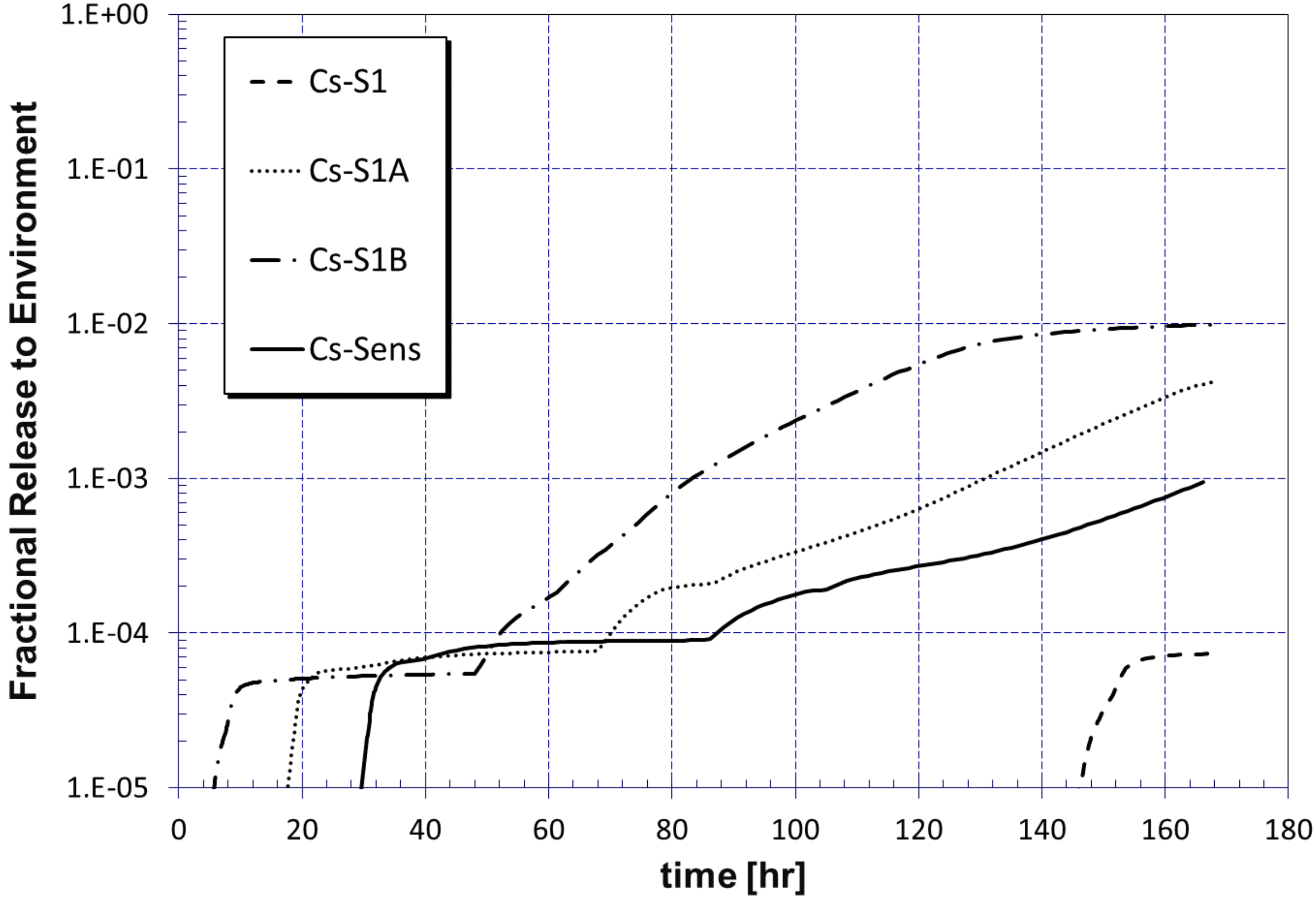


Figure 4-2: Fractional Release of Cesium to the Environment for Cases 1, 1A, 1B and MU-1.2

4.1.5 Sensitivity Analysis for Blind Feeding Steam Generators in Station Blackout Sequences (MU-1.3)

One of the modeling simplifications made in the Level 2 PRA involves station blackout sequences where extension of turbine-driven AFW following battery depletion (a.k.a., blind feeding of SGs), combined with a lack of elevated RCS leakage, extends the time to core damage by many days. Several factors contributed to this sequence being routed directly to an intact containment release category, rather than being routed through the main portion of the 1-CET tree. These included:

- An underlying MELCOR calculation that estimated that core damage would occur roughly 5 days after the initiating event, and that containment pressurization thereafter would be very slow (not approaching failure thresholds within 7 days after the initiator)
- A qualitative assertion that the energetic and high-temperature phenomena capable of failing containment (e.g., TISGTR) would be less challenging
- Recognition that overall uncertainty associated with this sequence of events is quite large
- A desire to reduce the already large computational burden associated with quantifying the Level 2 PRA model

To test the impact of this assumption, simple Excel-based manipulations were performed with the release category results. These manipulations rely on the baseline release category contribution profile, that the 1-CET-001 sequence comprises 15.1% of the overall release frequency, and how PDS-35-4 maps to various release categories (discounting the portion associated with 1-CET-001 going to 1-REL-NOCF). These manipulations result in an estimate of the release category contribution profile, were the 1-CET-001 cutsets to have been treated by the bulk of the CET modeling (i.e., treated like an SBO without indefinite feedwater and subject to containment failures or C-SGTR), and are shown in **Table 4-3**.

Table 4-3: Results of Sensitivity MU-1.3

	Percent Contribution to Total Release Frequency		Difference in Frequency
Release Category	**Base Model**	**MU-1.3**	
1-REL-BMT	1.2%	1.2%	0%
1-REL-CIF	0.1%	0.1%	0%
1-REL-CIF-SC	0.0%	0.0%	0%
1-REL-ECF	0.0%	0.0%	0%
1-REL-ICF-BURN	12.4%	12.4%	0%
1-REL-ICF-BURN-SC	3.5%	3.5%	0%
1-REL-ISGTR	0.8%	1.1%	0%
1-REL-LCF	41.8%	54.4%	13%
1-REL-LCF-SC	4.7%	4.7%	0%
1-REL-NOCF	34.6%	21.8%	13%
1-REL-SGTR-C	0.1%	0.1%	0%
1-REL-SGTR-O	0.0%	0.0%	0%
1-REL-SGTR-O-SC	0.3%	0.3%	0%
1-REL-V	0.0%	0.0%	0%
1-REL-V-F	0.1%	0.1%	0%
1-REL-V-F-SC	0.3%	0.3%	0%

As can be seen, the main effect of this sensitivity is to shift a large contribution from the intact containment release category to the late containment failure release category. This is logical and reflects the assumptions in the bulk of the CET regarding the rate of containment pressurization in the context of more quickly-evolving scenarios. As such, this shift is not truly applicable to the 1-CET-001 sequence (where core damage doesn't occur until 5 days after the initiating event). Separately, there is a small increase in ISGTR associated with the additional frequency of high-dry-low cutsets. The realism of this shift is debatable, in that the detailed RCS conditions many days after the initiating event may not be directly analogous to the same conditions for a more-quickly evolving SBO. The sensitivity has effectively shown the impact of this assumption, but overall, it is concluded that the baseline treatment of this issue is more defensible than the sensitivity, particularly given the other large uncertainties associated with these scenarios.

4.2 Equipment Performance Modeling

4.2.1 Identified Uncertainties in Equipment Performance Modeling

Table 4-4: Identified Uncertainties in Equipment Performance Modeling

Item	Description	Other comments
Tripping of RCPs on high voids	It is assumed in the pre-core damage portion of the MELCOR analyses that if RCPs have not been manually tripped or are unavailable, then they will trip at 10% void in the cold leg. This is a surrogate for pump cavitation damage or operator action in anticipation of damage.	MELCOR and MAAP modeling with significant voiding in the loops and RCPs running produces unphysical results (because there is no modeling of the physics involved). The pumps can be run under high void conditions (Westinghouse EOPs direct operators to start them as in FR-C.1 and the SAMGs if they are the only means of getting water in the core, by pushing a slug of water from the crossover leg), but the effects of running them under highly-voided conditions are not well-understood.
Equipment/instrument survivability for SAMG implementation	Affects the context for the HRA and the response of equipment that is credited	As an example, insufficient cable routing[2] information is available to address situations where the instrument is not damaged, but it's cabling elsewhere in the plant (or within containment) is.
Containment fan cooler survivability	The fan coolers are assumed to survive despite the potential for damage to the heat exchanger tubes from combustion pressure effects or deposition of radiological material.	This system is highlighted because its continued operation may have a strong effect on the results, by resulting in a higher frequency of basemat melt-through (by preventing long-term gradual over-pressure).

[2] Note that (EPRI, 2015) observes that environmental effects on electronics affects equipment performance/degradation more than environmental effects on the sensors or the pathways along which the signals are routed.

Table 4-4: Identified Uncertainties in Equipment Performance Modeling

Item	Description	Other comments
Reliability/availability of equipment beyond the Level 1 treatment	All equipment in the Level 1 PRA is subject to a 24-hour mission time; meanwhile, some equipment (e.g., the extensive damage mitigation guideline [EDMG] portable pump) does not have any reliability model applied.	While the Level 2 PRA notionally uses a 7-day mission time, most equipment is not relied on for that long, but in some cases equipment is relied on for more than 24 hours. Meanwhile, the absence of reliability modeling for SAMG-based mitigation equipment has been justified based on the very high HEPs associated with its use).
Clogging/damage issues when alternate water sources are used	Relates to sump clogging or pump damage when firewater (or other alternate sources) are used; this is a larger concern for high-pressure pumps due to them having tighter clearances and multi-stage design, and these uses are infrequent in the HRA and probabilistic model	See (Sampson, 1998) for an operating experience example
Hydrogen supply line in the auxiliary building not considered	Damage to this pipe from a combustion of accident-produced hydrogen could change the gas composition in the auxiliary building and thereby influence further damage and/or fission product releases	

4.2.2 Alternative Treatment(s) of Uncertainties in Equipment Performance Modeling

MU-2.1 A MELCOR calculation was performed (based on Case 6) in which containment fan coolers were rendered in-operable at the time of the first hydrogen combustion (around 15.7 hours) to demonstrate the impact of fan cooler survivability on the source term as compared to Cases 6, 6A, and 6B.

MU-2.2 A MELCOR calculation was performed (based on Case 6R1) in which the residual heat removal (RHR) injection rate was reduced from 1500 to 750 gpm when sourced by firewater at 18.3 hours. This is to show the impact of partial sump clogging on the key event timings and the source term relative to Case 6R1.

4.2.3 Sensitivity Analysis for Hot Leg Creep Rupture with Containment Fan Coolers (MU-2.1)

In Case 6, as discussed in Section 6 of Appendix B, creep rupture of the hot leg nozzle occurs at 15.7 hours, and the first hydrogen deflagration subsequently occurs in containment. For this sensitivity, containment fan coolers were assumed to be rendered inoperable at this time. Without this means of heat removal, the temperature in containment begins to rise (**Figure 4-3**). Pressure consequently also begins to increase, and containment over-pressurization occurs at 72 hours (**Figure 4-4**).

With fan coolers running, the amount of steam in Case 6 is kept relatively low and numerous deflagrations occur. In the sensitivity, however, after a sizeable deflagration at 27 hours, the containment becomes predominantly steam inerted (**Figure 4-5**). Even though hydrogen concentration increases slowly until containment failure and then levels off at a mole fraction around 0.6, no further deflagrations occur.

Table 4-5 gives the environmental release fractions for Cases 6, 6A, 6B, 6C and MU-2.1. Containment failure (apart from basemat melt-through) does not occur in the first three cases, so it is not surprising that the sensitivity case yields a much larger release. In the case of 6C, where containment is assumed to fail at the time of vessel breach, containment fan coolers are available to keep the pressure low; hence, the driving head for fission products out the failure pathway is much lower than that of the sensitivity case, where containment pressure is high. Thus, for many isotopes, the release is still larger in the sensitivity than Case 6C, despite the latter's earlier containment failure.

Therefore, this sensitivity suggests that susceptibility to combustion-induced damage of the containment fan coolers would tend to shift release category frequency from the BMT release category to the LCF release category, though this shift would be limited by the fraction of the frequency contribution coming from BMT cases that have containment heat removal (e.g., station blackout frequency would be unaffected). There could also be an effect wherein failure of containment cooling would shift the likelihood of combustion-induced containment failure (e.g., the singular but larger combustion seen in the sensitivity case). However, the effect on the release category profile of this is not discernible from the available results.

Table 4-5: Environmental Releases: 6-Series Cases Versus MU-2.1

Representative Element	S6	S6A	S6B	6C	MU-2.1
Xe	8.3E-03	9.1E-03	1.1E-02	4.0E-01	8.4E-01
Cs	5.4E-05	3.0E-06	7.9E-05	3.2E-03	9.7E-03
I	6.4E-05	4.4E-06	8.2E-05	6.1E-03	3.2E-02
Te	9.3E-05	4.4E-06	8.4E-05	5.5E-03	3.0E-03
Ba	4.0E-06	1.8E-07	5.4E-06	1.1E-03	2.6E-04
Ru	6.7E-07	5.4E-08	9.1E-07	9.8E-05	4.0E-05
Mo	7.1E-04	5.7E-06	1.1E-03	9.2E-04	3.8E-02
Ce	1.4E-07	3.6E-09	2.8E-07	7.9E-05	3.2E-06
La	1.1E-08	2.6E-10	2.0E-08	1.7E-06	4.5E-07
UO2	4.1E-07	1.3E-08	5.3E-07	1.6E-05	6.0E-05
Cd	2.2E-04	7.6E-06	2.5E-04	3.1E-03	1.7E-02
Ag	3.1E-04	1.1E-05	3.9E-04	3.1E-03	1.6E-02

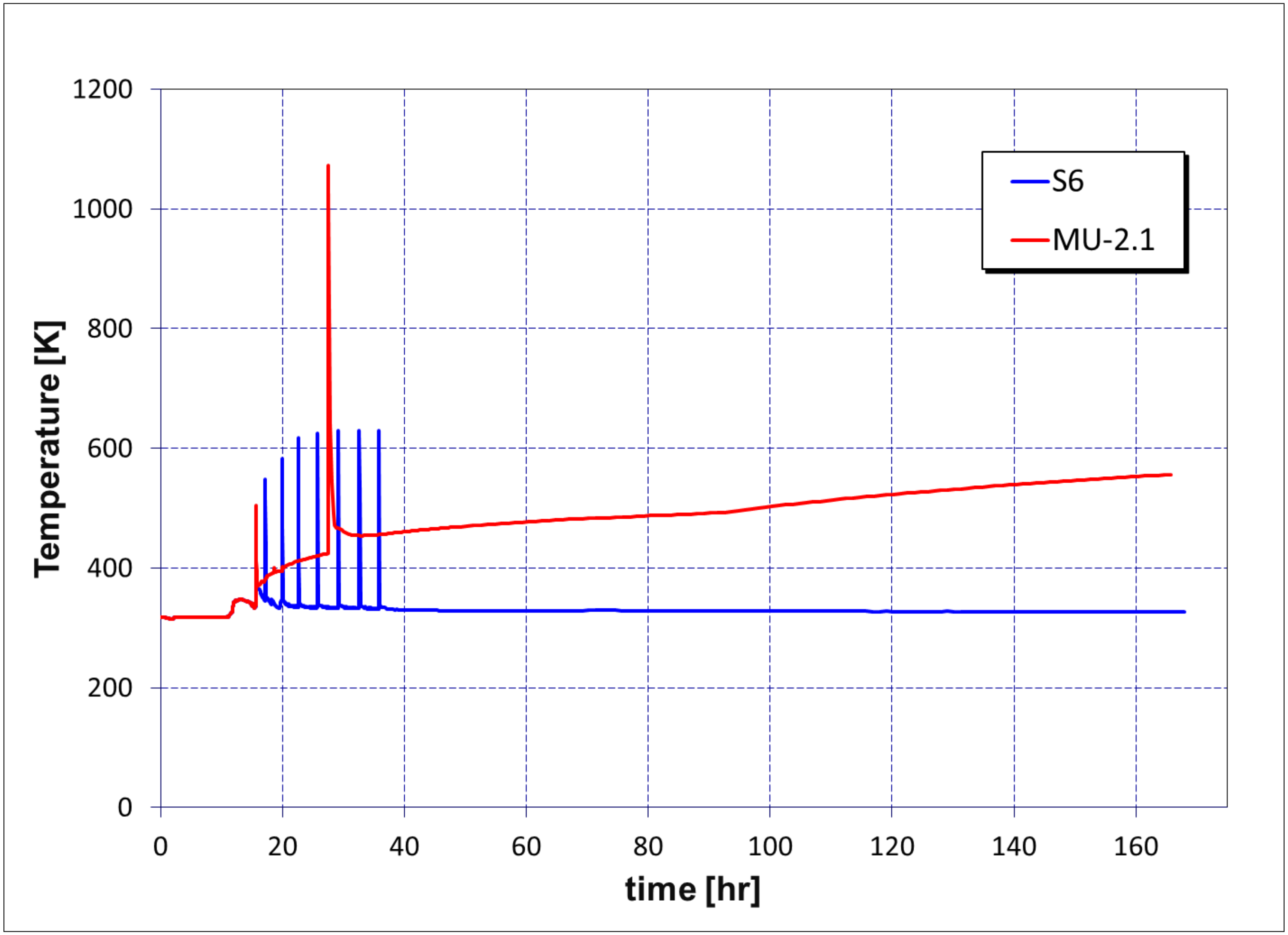


Figure 4-3: Containment Temperature in the S6 and the MU-2.1 Sensitivity Cases

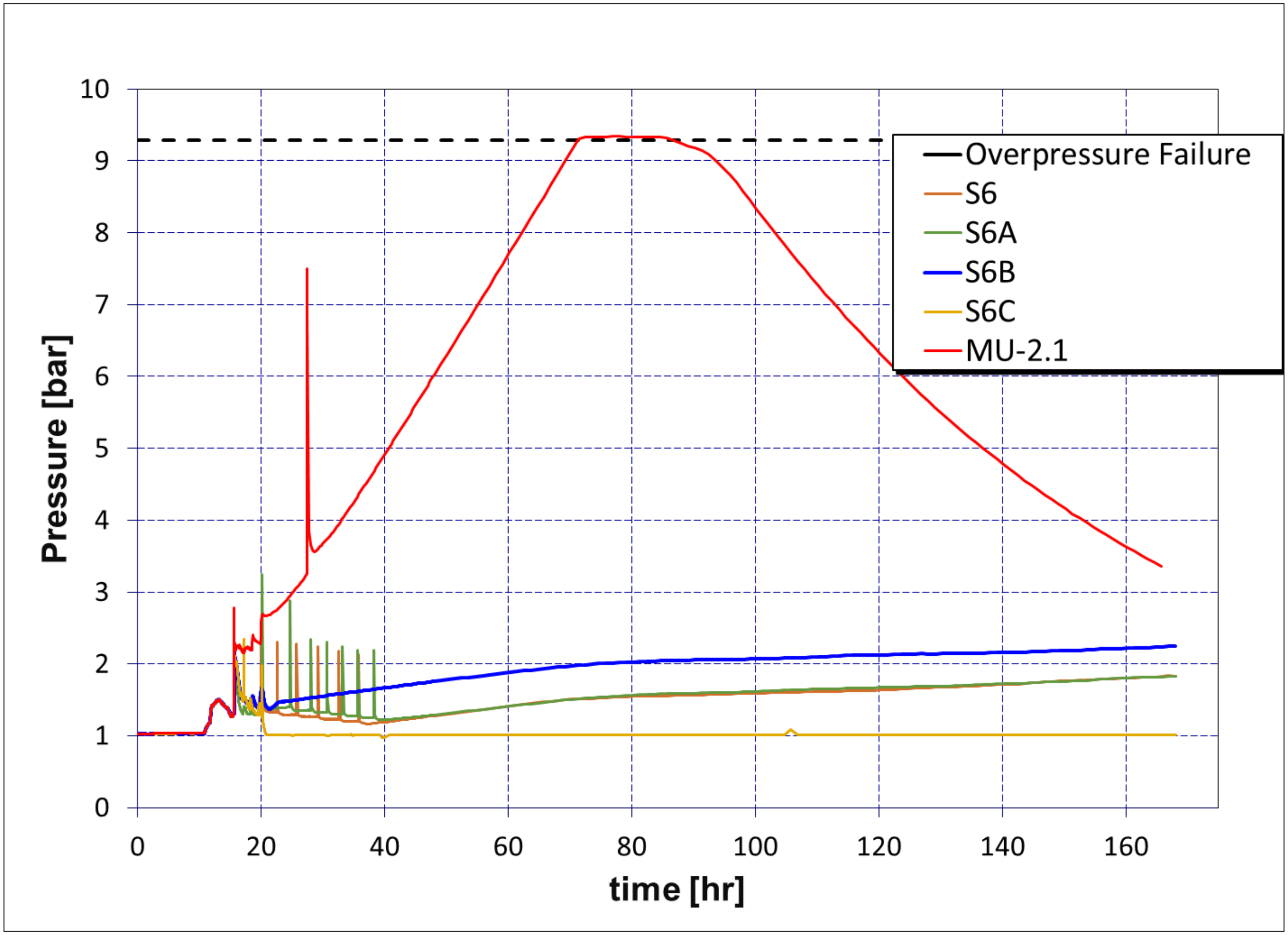


Figure 4-4: Containment Pressure in the 6-Series Cases and the MU-2.1 Sensitivity Case

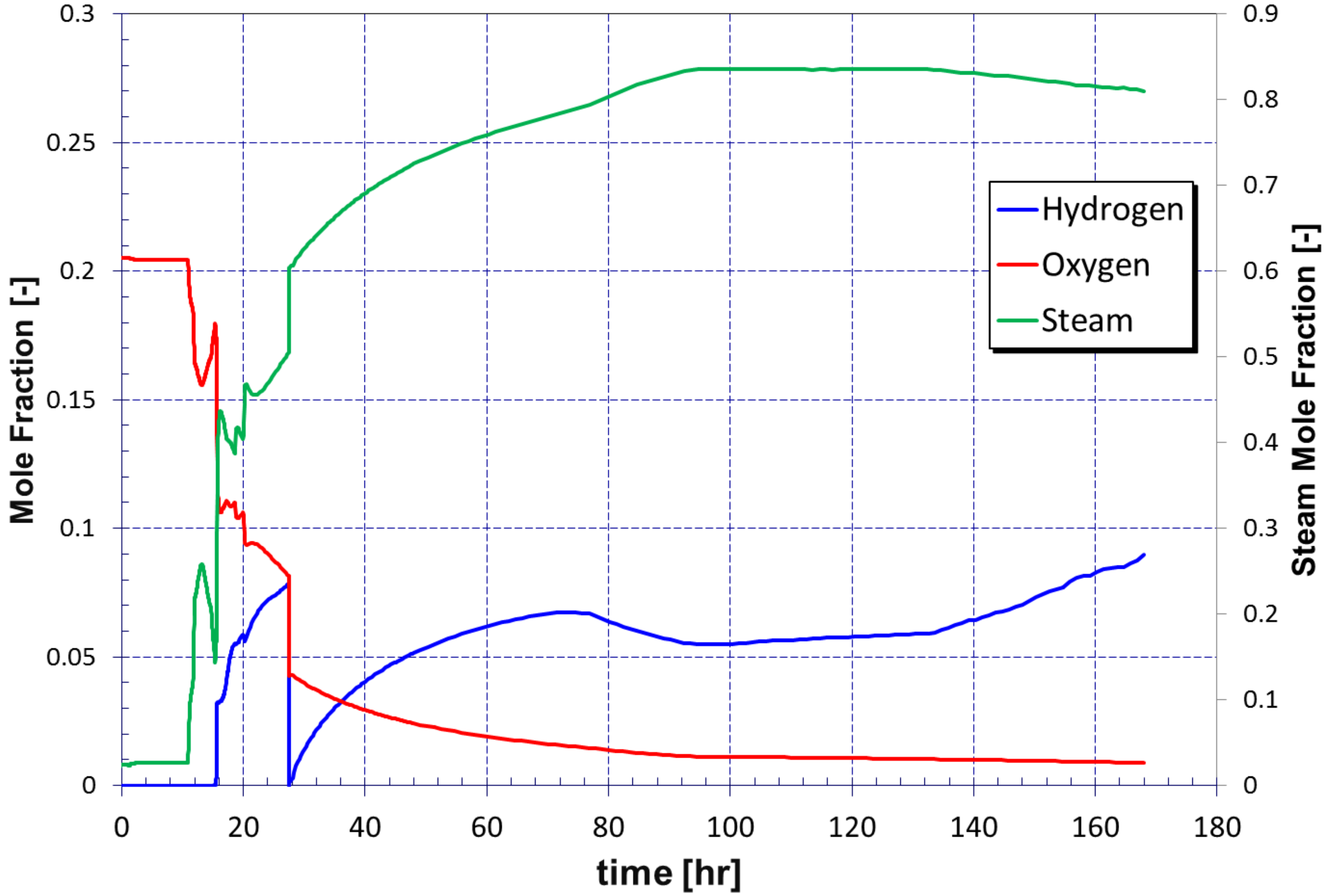


Figure 4-5: Mole fraction in the containment dome for sensitivity Case MU-2.1

4.2.4 Sensitivity Analysis for Recovery of RHR After Core Damage (MU-2.2)

Case 6R1, as discussed in Section 6.1 of Appendix B, assumes operators perform recovery actions after core damage, namely, a train of RHR is started. A sensitivity on this recovery case is described here in which a reduced flow rate of 750 gpm (rather than 1500 gpm) is assumed when the RWST refill begins. This is to explore the possible impact of clogging when an alternative source of water is introduced.

At 18.3 hours the flow rate is reduced to 750 gpm, as seen in **Figure 4-6**. Due to the reduced flow, the core support structure fails 14 minutes sooner at 19.9 hours, versus 20.1 hours in the base case. The level in the RCS is not as constant in the sensitivity case, but the core remains cooled and covered in both cases (**Figure 4-7**).

Figure 4-8 shows the environmental releases for these two simulations for which the difference is quite small at the conclusion of the base calculation (23.1 hours) and is not anticipated to grow larger, since core conditions have more or less stabilized in both cases.

This sensitivity demonstrates that the in-vessel recovery probability is not sensitive to the available injection flow rate, within the range of flow rates considered. Obviously, as the injection flow rate approaches lower values (due either to the use of alternative systems or clogging), a threshold effect will be observed wherein the vessel will breach. Note that this topic is the subject of SAMG Computational Aid 1 (CA-1).

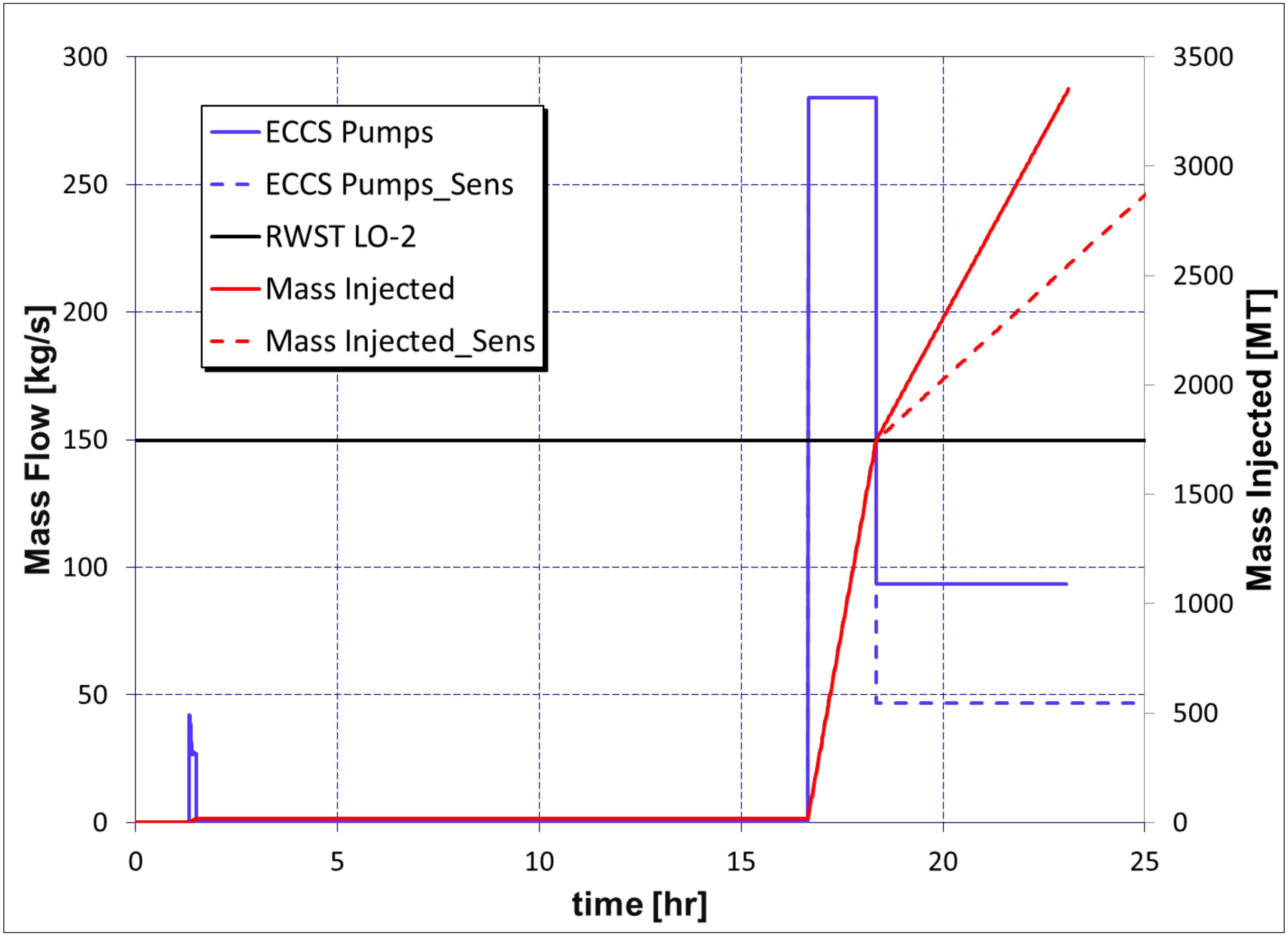


Figure 4-6: Liquid Drawn from the RWST by ECCS for Case 6R1 Base and Sensitivities

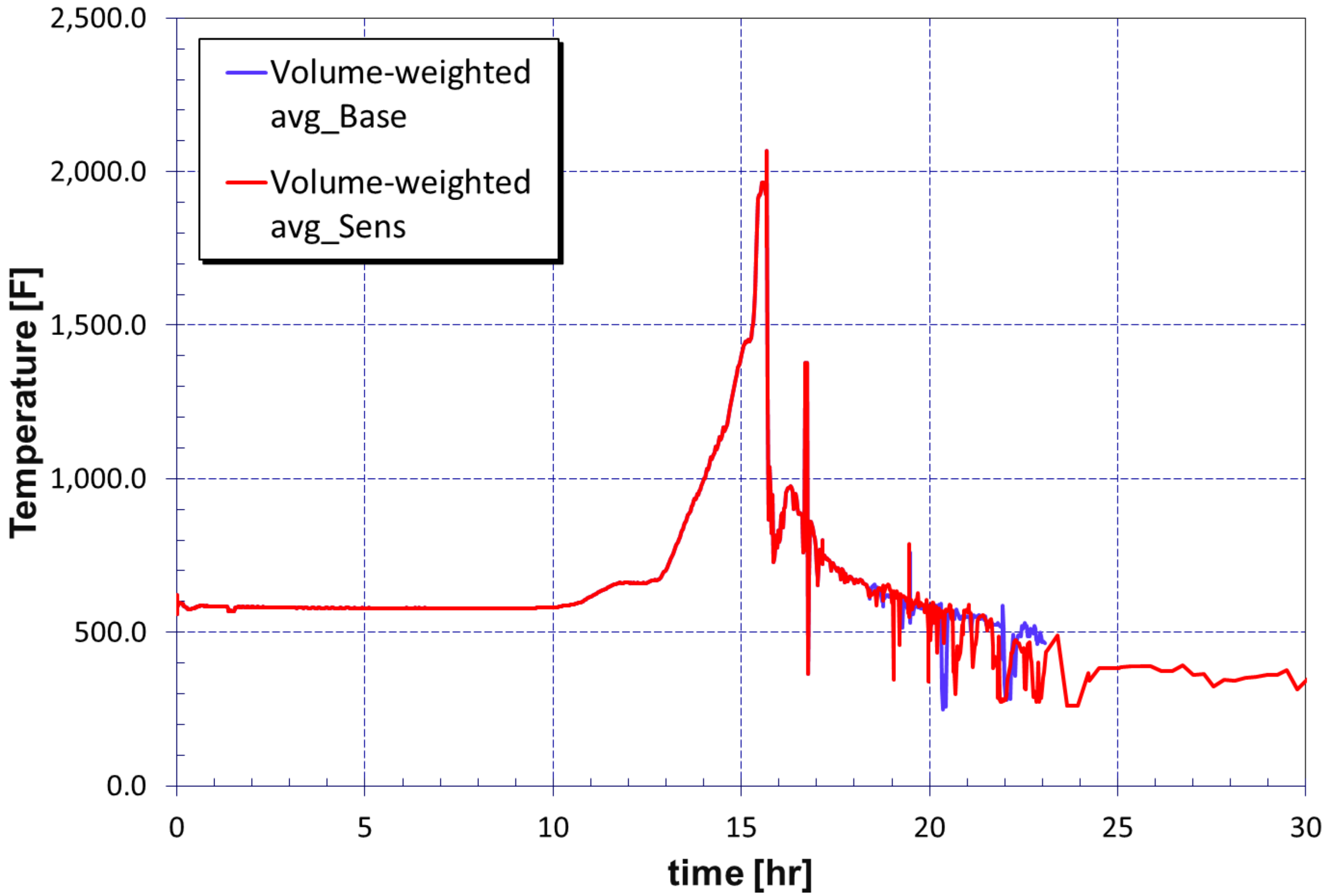


Figure 4-7: Average Core Exit Thermocouple Temperature for Case 6R1 and the Sensitivity

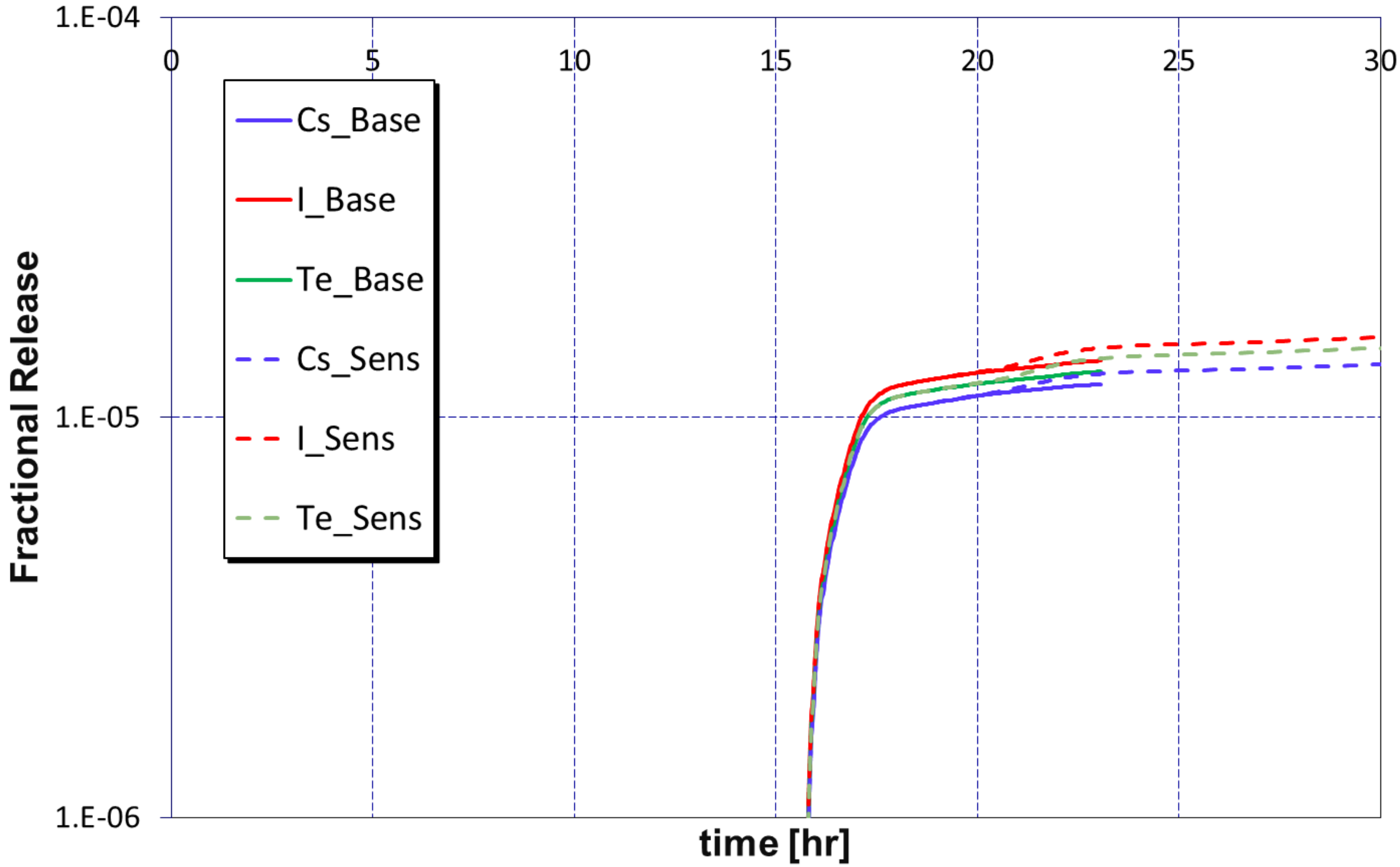


Figure 4-8: Environmental release for Case 6R1 base and sensitivity calculations

4.3 Human reliability modeling

4.3.1 Identified Uncertainties in Human Reliability Modeling

Table 4-6: Identified Uncertainties in Human Reliability Modeling

Item	Description	Other comments
Treatment of SAMG and other accident management guidance	The inclusion or exclusion of some or all accident management (most notably SAMG)-guided actions could affect accident progression and release characterization	These are treated in this project, and the uncertainties are mainly captured by other line items.
Modeling of operator actions during severe accidents	This refers to the inherent uncertainties in such modeling, and their impact on the identification of HFEs. The related impact on probability estimates would be captured by parameter uncertainty on the modeled HEPs.	This subsumes several specific concerns, such as modeling of the decision-making process, psychological effects, communications issues, etc., as discussed further in Section 2.4.4 of the main body of this report.
Limitations on the number of actions considered	Reality would likely be a more complex series of recovery actions that may be more effective at limiting releases than what is modeled	Generally, recovery modeling is focused on a single pre-vessel-breach action and single post-vessel-breach action, to accommodate resource and schedule limitations, and in light of the large uncertainties in the modeling. Actions such as containment venting are not captured due to their low priority. Venting may shift the release frequency of late containment failure downward, and at the same time shift the source term for no containment failure upward.
Lack of credit for post-core-damage actions during SBO	The HRA does not permit credit for actions during station blackout, due to the lack of instrumentation availability for guiding SAMG navigation.	This was a point of concern for both the Technical Advisory Group and the external peer review.
Automatic actuation of equipment not operating at the time of SAMG entry (a.k.a., pull-to-lock issue)	There are several cases in the PRA model where automatic actuation of containment sprays or ECCS equipment is not credited in a post-core-damage context. The reason for this is that SACRG-1 (the first guidance document the operators enter upon core damage) Step 2 has the operators place non-operating equipment (containment systems, ECCS, and feedwater pumps) into 'pull-to-lock' (i.e., they will not automatically actuate). Their automatic actuation due to operator failure to place them in pull-to-lock could have positive or negative impacts.	The most tangible instances of this in the current Level 2 model are: • 1-L2-IVREC • 1-L2-CSS-E in 1-L2-SCRUBE • 1-L2-CSS-L in 1-L2-SCRUBL • 1-L2-CAVWATER in 1-L2-SCRUBL • 1-L2-CCU-L in 1-L2-DET-CONTL (credit retained based on CCUs being in operation at time of SAMG entry)

Table 4-6: Identified Uncertainties in Human Reliability Modeling

Item	Description	Other comments
Modeling assumptions associated with habitability, including main control room habitability	This issue is discussed in detail in Section 9 of Appendix D; alternative assumptions could lead to more or less credit for success of operator actions post-core-damage, including actions to flood containment.	This is primarily thought to be an issue for bypass, containment isolation failure, or delayed containment failure events, in that operating reactor control room habitability designs include either design-basis accidents or the NUREG-1465 alternative source term (depending on the plant's licensing basis).
Modeling of SCG-1, SCG-3, and SAG-5 entry	Entry into these three guidelines is more uncertain because of the nature of their entry.	This relates to dose projection / field monitoring considerations in the case of SCG-1 and SAG-5, as described in Section 14 of Appendix D; separately, this relates to the use of the hydrogen Computational Aid and sampling line survivability in the case of SCG-3, as described in Appendix D.
Use of the recombiners as a deliberate ignition source	This action could have positive or negative impacts on radiological releases depending on the timing and scenario.	This is not a proceduralized action.
Modeling of offsite resources for accident management support	Affects credit given for resources in flooding the reactor cavity.	See Section 21 of Appendix D. Though identified as a model uncertainty here, this issue is addressed more systemically in the way that the overall PRA results are presented.

4.3.2 Alternative treatment(s) of Uncertainties in Human Reliability Modeling

MU-3.1A The SAPHIRE model was re-quantified with all Level 2 HEPs (i.e., those parameters in **Table 2-7**) set to 0.9.

MU-3.1B The SAPHIRE model was re-quantified with all Level 2 HEPs (i.e., those parameters in **Table 2-7**) reduced by one order of magnitude.

MU-3.2 The Level 2 HRA approach has deliberately excluded credit for operator actions following core damage during station blackout, and for actions in the long term (meaning roughly 6 hours or more after vessel breach) for all scenarios. This sensitivity was performed to show how varying levels of reliability of recovery for these longer-term actions (during station blackout and otherwise) would affect LERF, LRF, and conditional containment failure probability (CCFP).

4.3.3 Sensitivity Analysis for Human Reliability Modeling (MU-3.1A & B)

These sensitivity analyses involve adjusting all Level 2 post-core-damage HEPs either upward or downward to show their effect as a class of uncertainty. This involves those events in **Table 2-5**. Note that 1-L2-OP-H2CTL-L and 1-L2-OP-PRIDEPRES-VE do not appear in **Table 2-5**, nor are they adjusted here, because they are top events rather than basic events.

Table 4-7 provides a comparison of the sensitivity results to the base model. First, all Level 2 HEPs are increased to a failure probability of 0.9. This change has no discernible impact on the containment isolation failure and containment bypass-related release categories, demonstrating that the frequency of these types of events (but not necessarily the source terms) are not impacted by the Level 2 HRA, as would be expected (because these isolation failures and bypasses precede core damage, other than induced steam generator tube rupture [ISGTR]). However, there is a notable shift in release category frequency from the no containment failure release category to the combustion-related and late containment failure release categories. This is also as expected, most notably due to less reliable extension of TDAFW for relevant SBO sequences. The minor change in overall (gathered) release frequency reflects model convergence limitations, as discussed further elsewhere.

The second sensitivity involves reducing all the base Level 2 HEPs by an order of magnitude. Again, as can be seen in **Table 4-7**, there is no discernible impact on the containment isolation failure and containment bypass-related release categories. Here, the shift occurs from a handful of different release categories to the no containment failure and (to a lesser degree) the scrubbed late containment failure release category. The minor increase in the scrubbed combustion-induced failure release category, and the minor decrease in ISGTR are not viewed as particularly meaningful, given the precision of the model. In total, combustion-induced failure was reduced by 6% of total release frequency, late containment failure decreased by 21% of total release frequency, and the intact containment release category inherited this frequency. So as one would expect, greater success of operator actions does indeed lead to a more favorable release frequency profile.

Table 4-7: Results of MU-3.1 A&B Sensitivity

	Base Model	**MU-3.1A – All HEPs set to 0.9**	**MU-3.1B – All HEPs lowered by a factor of 10**
RC Fractional Contributions			
BMT	0.01	0.01	0.01
CIF	<0.01	<0.01	<0.01
CIF-SC	<0.01	<0.01	<0.01
ECF	<0.01	<0.01	<0.01
ICF-BURN	0.12	0.18	0.05
ICF-BURN-SC	0.04	0.01	0.05
ISGTR	0.01	0.01	<0.01
LCF	0.42	0.55	0.16
LCF-SC	0.05	0.01	0.10
NOCF	0.35	0.21	0.61
SGTR-C	<0.01	<0.01	<0.01
SGTR-O	<0.01	<0.01	<0.01
SGTR-O-SC	<0.01	<0.01	<0.01
V	<0.01	<0.01	<0.01
V-F	<0.01	<0.01	<0.01
V-F-SC	<0.01	<0.01	<0.01
Total Release Frequency	**7.04E-05**	**7.12E-05**	**7.03E-5**

4.3.4 Sensitivity Analysis for Operator Actions After Core Damage During an SBO and After Vessel Breach (MU-3.2)

The Level 2 HRA approach has deliberately excluded credit for operator actions following core damage during station blackout, and for actions in the long-term (meaning roughly 6 hours or more after vessel breach) for all scenarios. It is understood that operators would continue to take actions under station blackout conditions, and during these longer time-frames. Such actions could include restoration of dc power, restoration of ac power, containment venting, etc. Nevertheless, credit for such actions is beyond the scope of the Level 2 HRA approach used in this study to model the reliability of such actions, and generally beyond the state-of-practice in Level 2 PRA. The effect of excluding these assumptions is masked in most studies by the counter-acting effect of truncating the accident sequence time in the deterministic accident simulations and probabilistic accident modeling. That is not the case in this study because the accident trunctation time is extended to 7 days after the initiating event occurs.

This sensitivity shows how varying levels of reliability of recovery for these longer-term actions (during station blackout and otherwise) would affect LERF, LRF, and CCFP. The sensitivity relies on simple manipulations of the baseline release frequency results, and considers these actions at a generic, rather than scenario-specific, level. Further, the sensitivity assumes that (a) continuing actions will have an overall positive effect (i.e., the potential for actions to exacerbate the accident are not considered) and (b) actions fall into one of the following three categories:

- Actions that prevent significant combustion events in the intermediate and long term (named "$RF_{combust}$" here) by igniting at lower flammability levels, etc. – *this drives frequency from the "BURN" release categories to the "LCF" release categories*
- Actions that successfully control containment pressure through restoration of containment heat removal or containment venting (named "$RF_{pressure}$" here) – *this drives frequency from the "LCF" release categories to the "BMT" release category*
- Actions that flood the cavity with timing and flow rates that are sufficient to arrest basemat ablation prior to basemat failure (named "RF_{BMT}" here) – *this drives frequency from the "BMT" release category to the "NOCF" release category*

These actions (which can be thought of as recovery factors, represented by their failure likelihood) are applied in the above order, to the extent that they have over-lapping effects. The results are shown in **Table 4-8**, and the breakdown of release category contributions (by truncation time and by risk surrogate) in Section 2.4.6 of the main body of this report is useful in understanding these results. The effects are not always intuitive. Take for instance the case with $RF_{combust}$ = 0.1, $RF_{pressure}$ = 1, and RF_{BMT} = 1. The order-of-magnitude decrease in the 1-REL-ICF-BURN and 1-REL-ICF-BURN-SC shifts this frequency from those release categories to the 1-REL-LCF and 1-REL-LCF-SC release categories. None of these release categories is included for LERF, so LERF is unaffected. For LRF, both 1-REL-ICF-BURN and 1-REL-LCF contribute, so the shift in frequency between these release categories doesn't change LRF. Meanwhile, 1-REL-ICF-BURN-SC does not contribute to LRF, and thus its reduction is irrelevant (and would cause 1-REL-LCF-SC to increase). CCFP is similarly unaffected.

Table 4-8: Risk Surrogates Presented for Different Accident Termination Assumptions

Postulated Recovery Factors			Resulting Risk Surrogates		
$RF_{combust}$	$RF_{pressure}$	RF_{BMT}	LERF	LRF	CCFP
1	1	1	0.01	0.61	0.65
1	1	0.1	0.01	0.61	0.64
1	0.1	1	0.01	0.19	0.65
0.1	1	1	0.01	0.61	0.65
1	0.1	0.1	0.01	0.19	0.26
0.1	1	0.1	0.01	0.61	0.64
0.1	0.1	1	0.01	0.13	0.65
0.1	0.1	0.1	0.01	0.13	0.13

What this sensitivity suggests is that:

- None of these longer-term recoveries affects LERF (as would be expected);
- The greatest reduction in LRF occurs from recoveries related to controlling containment pressure, with additional benefit seen if this is combined with controlling combustion; and
- The greatest reduction in CCFP occurs from combined recoveries to control containment pressure and prevent basemat melt-through (either one by itself is not sufficient), with additional benefit seen if this is combined with controlling combustion.

These findings are consistent with the definitions of these risk surrogates (at least as defined in this study), and they demonstrate that reliability of recovery actions would need to be on the order of one decade for each of these three classes of recovery to counter-act the effects of longer accident simulation times.

4.4 In-Vessel Accident Progression Modeling

4.4.1 Identified Uncertainties for In-Vessel Accident Progression Modeling

Table 4-9: Uncertainties for In-Vessel Accident Progression Modeling

Item	Description	Other comments
Axial power profile	The axial power profile used in the L3PRA project MELCOR model was taken from the reference plant MAAP model. It is fairly flat and bottom-peaked.	Reference-plant-specific axial power profiles for EOC (end-of-cycle) and BOC (beginning-of-cycle) were used to support the shape of the MOC (middle-of-cycle) profile in the L3PRA project MELCOR analysis. The draft Surry (SNL, 2016a) and Sequoyah (Barr, 2016) SOARCA uncertainty studies specifically treat the broader issue of using BOC vs. MOC vs. EOC.
Accumulator injection vs. pressure	This could affect the timing of core damage and the estimation of containment conditions following hot leg creep rupture (high-pressure sequences) or vessel breach (low-pressure sequences).	In the current MELCOR analysis, only about 1/4 of the accumulator water is injected during SG cooldowns to 200/300 psig; this affects the timing of core damage, as well as the amount of water dumped later when the hot leg, steam generator tubes, or vessel fails.

Table 4-9: Uncertainties for In-Vessel Accident Progression Modeling

Item	Description	Other comments
Recriticality	This could have an important effect on accident recovery, but only for a small subset of scenarios (those where reflood coincides with melted poison material but still-standing fuel); this could increase the amount of short-lived radionuclides released.	This issue is briefly discussed generally in Appendix B of (EPRI, 2012a) and was investigated in the context of Fukushima in (EPRI, 2016).
Fuel failure modeling	This can affect timing of core relocation, as well as integral in-vessel hydrogen production.	The effect of fuel failure modeling parameters (e.g., Zircaloy metal breakout temperature, molten cladding drainage rate, and radial debris location time constants) has been investigated in several studies. The studies include dynamic Level 2 PRA work (LaChance, 2012), (SNL, 2013), and Fukushima-related analysis documented in (Denman, 2015) and (SNL 2016b). In the draft Surry (SNL 2016) and draft Sequoyah (Barr, 2016) SOARCA uncertainty analyses, the effect of eutectic interaction between UO_2 and ZrO_2, characterized by the melting temperature, was also considered as an uncertainty parameter. The above studies showed that fuel failure model parameter quantification has important influences on the magnitude of in-vessel hydrogen production, timing of key accident progression events such as reactor pressure vessel (RPV) lower head failure, and source term.
In-vessel hydrogen production	This can affect downstream combustion event effects.	It is worth noting that MELCOR can predict higher in-vessel hydrogen production than MAAP. The draft Surry (SNL, 2016a) and draft Sequoyah (Barr, 2016) uncertainty analysis (UA) results analyze in-vessel hydrogen production as an output metric and show a wide spread in total in-vessel production, and report regression analysis results indicating the most influential uncertain input parameters for in-vessel hydrogen production.
Debris porosity	This can affect in-vessel hydrogen production, RCS piping failures, and timing of lower head failure.	Uncertainties in the treatment of debris porosity and gradual melt relocations below the core plate affect local blockages and heat transfer to the water in the lower plenum (and thus availability of steam for oxidation and the heat transfer from the debris to the RPV upper plenum/RCS piping); this in turn affects the peak melt temperatures (and thus the timing of core plate failure and lower head failure).

Table 4-9: Uncertainties for In-Vessel Accident Progression Modeling

Item	Description	Other comments
Recovery of a degraded core	There is acknowledged uncertainty in capturing of the complex materials and thermal-hydraulics issues associated with this phenomenon. Note that this issue is already treated broadly as a parameter uncertainty.	There is a significant degree of uncertainty in this phenomenon, as well as in the human actions and system behaviors required to restore cooling. The timing of when the core is reflooded has a significant impact, with recovery being significantly more likely the earlier in core degradation that reflooding occurs. Other studies have varied widely in terms of the credit given. Some recovery calculations, as discussed in Appendix B, were predicted to result in in-vessel recovery, though code stability impacted the definiteness of this outcome. This issue was also investigated for a boiling water reactor in (Denman, 2015).
Pressurizer relief tank (PRT) modeling	Heat transfer from the PRT is not considered, in that the tank is modeled in such a way (namely use of an adiabatic wall) as to arbitrarily inhibit several forms of heat transfer that could have an impact on the rate of dryout of the PRT, including: (i) back-side convection, and (ii) back-side conduction when the tank is partially submerged from flooding of lower containment.	Since the PRT does not dry out and containment does not overfill to the point of completely submerging the PRT in any of the analyses (with the exception of Cases 1B2 and 3A4, which are addressed in **Section 6** and the sensitivity below), this is not thought to have a large effect here.

4.4.2 Alternative treatment(s) of Uncertainties for In-Vessel Accident Progression Modeling

MU-4.1 A MELCOR simulation was performed (based on Case 2 [see Section 2 of Appendix B]) with the accumulator nitrogen expansion modeled as isenthalpic rather than isentropic. This affects the pressure at which the full accumulator inventory is injected into the RCS (see **Section 7.1.1** for a more detailed explanation). The timing of core damage and the overall source term were compared to the results of Case 2 to understand the impact of accumulator injection versus RCS pressure.

MU-4.2 A MELCOR calculation was performed (based on Case 1B2 [see Section 1.2.2 of Appendix B]) in which convective heat transfer from the PRT to the containment atmosphere was considered no longer adiabatic. Also, pool scrubbing was enabled for each of the flowpaths entering the PRT. The rate of radionuclide volatilization and environmental release was compared to Cases 1B and 1B2 (see Section 1.2 of Appendix B).

MU-4.3A A MELCOR calculation was performed (based on Case 1A2 [see Section 1.1.2 of Appendix B]) where the Zircaloy breakout temperature was increased from 2400°K to 2450 °K and the eutectic temperature was increased from 2500 °K to 2585 °K.

These values represent the 90th percentile of distributions utilized in the draft Surry SOARCA UA project (see Figures 4-17 and 4-27 in [SNL, 2016a]).

MU-4.3B A MELCOR calculation was performed (based on Case 1A2) where the Zircaloy breakout temperature was decreased from 2400 °K to 2250 °K and the eutectic temperature was decreased from 2500 °K to 2375 °K. These values represent the 10th percentile of distributions utilized in the draft Surry SOARCA UA project (see Figures 4-17 and 4-27 in [SNL, 2016a]). The results, along with those of MU-4.3A, were compared with Case 1A2 to inform how these parameters impact hydrogen generation and event timing.

4.4.3 Sensitivity Analysis for Isenthalpic vs Isentropic Accumulator Nitrogen Expansion Model in MELCOR Case 2 (MU-4.1)

This sensitivity is a re-computation of Case 2 with the accumulator nitrogen expansion modeled as isenthalpic rather than isentropic (see **Section 7.1.1** for a more detailed explanation) to explore the impact of accumulator (ACC) injection rate versus RCS pressure in the timing of core damage and the source term.

At ~2 hours, ACCs begin to inject their contents. In the sensitivity case, this injection levels off at 3 hours to 20.4 m^3 per ACC, while in the base case it levels at 11.9 m^3, as seen in **Figure 4-9**. This has the positive effect of delaying core uncovery and vessel breach by 0.5 and 1.5 hours, respectively. However, with an increased volume of water in the RCS, zircaloy and steel oxidation is significantly greater (in-vessel hydrogen generation is 690 kg at the time of vessel breach versus 587 kg in the base case). Higher oxidation leads to in-vessel releases of volatile fission products (namely, Xe, Cs, I, and Te) that are nearly 13% greater than the base case. Most of this, however, is retained in the RCS, and the environmental release is slightly less than in Case 2 (**Figure 4-10**). Containment overpressure and basemat melt-through are delayed by several hours, which also aids in limiting the release (**Figure 4-11** and **Figure 4-12**).

This sensitivity shows assumptions about the accumulator injection flow rate (versus pressure) can have a notable impact on in-vessel accident progression (and, subsequently, ex-vessel accident progression), but based on the sensitivity, these differences do not appear to significantly affect energetic events or cumulative radiological releases.

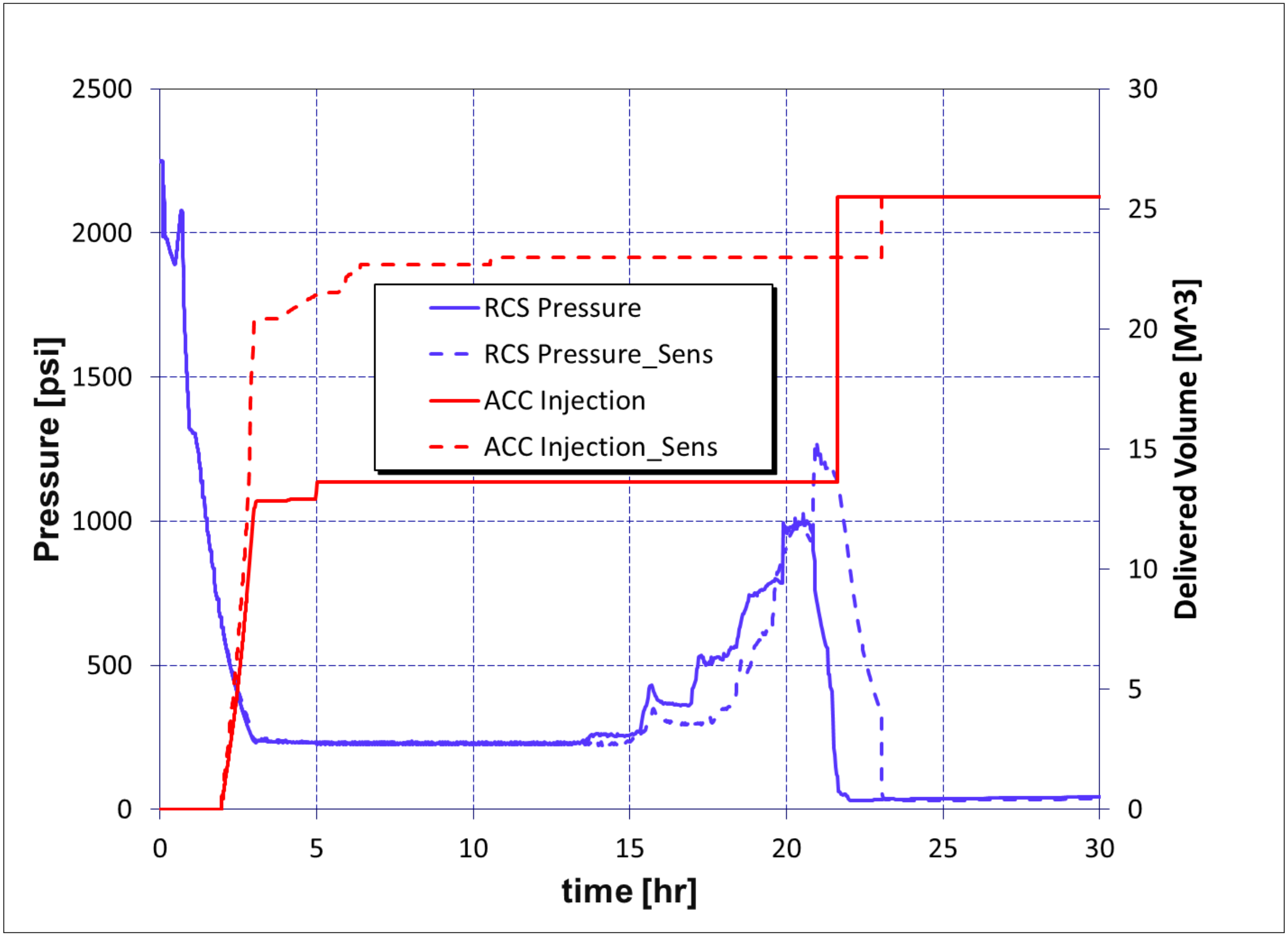


Figure 4-9: Accumulator Injection and RCS Pressure for MU-4.1

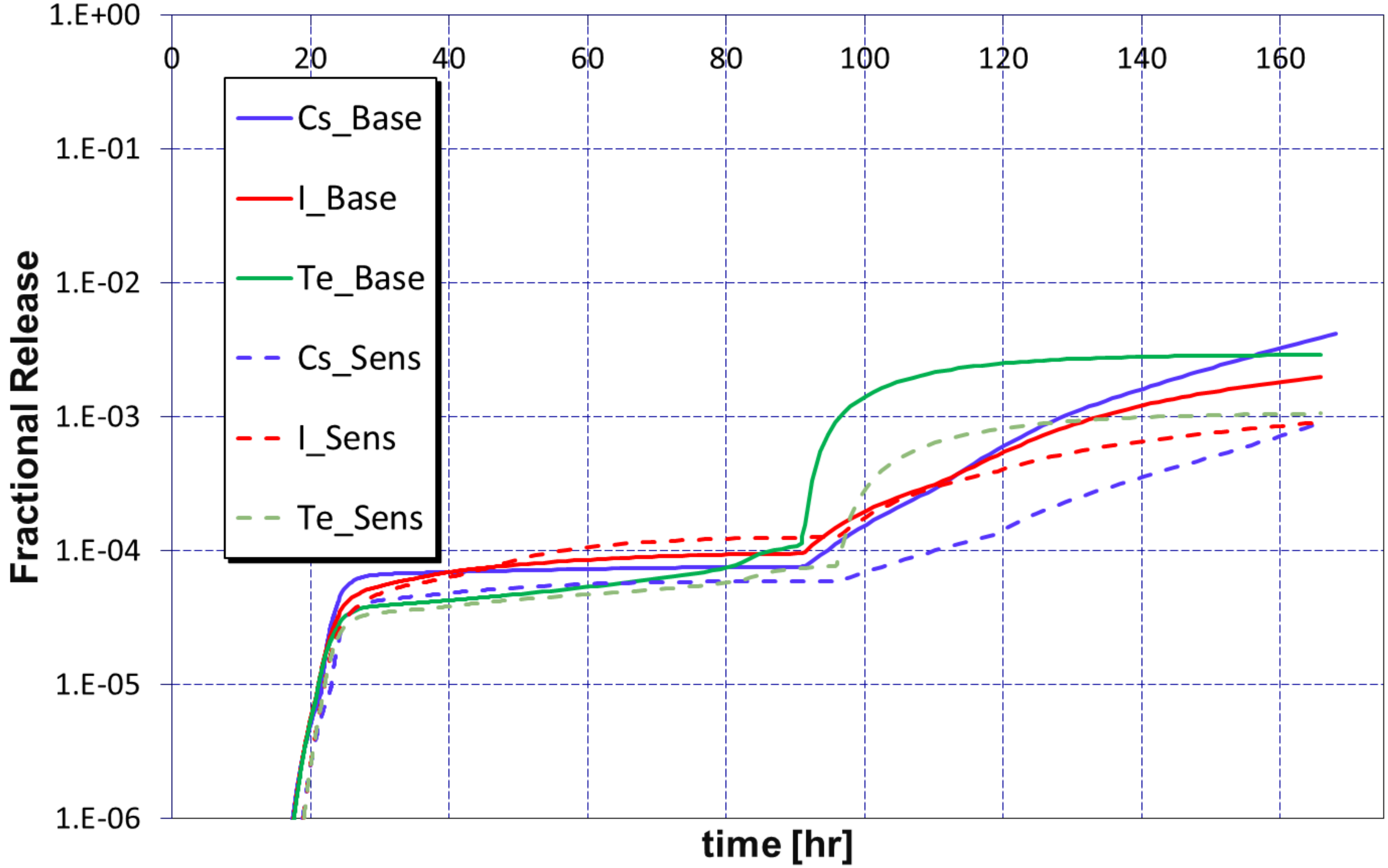


Figure 4-10: Environmental Release of Fission Products for MU-4.1

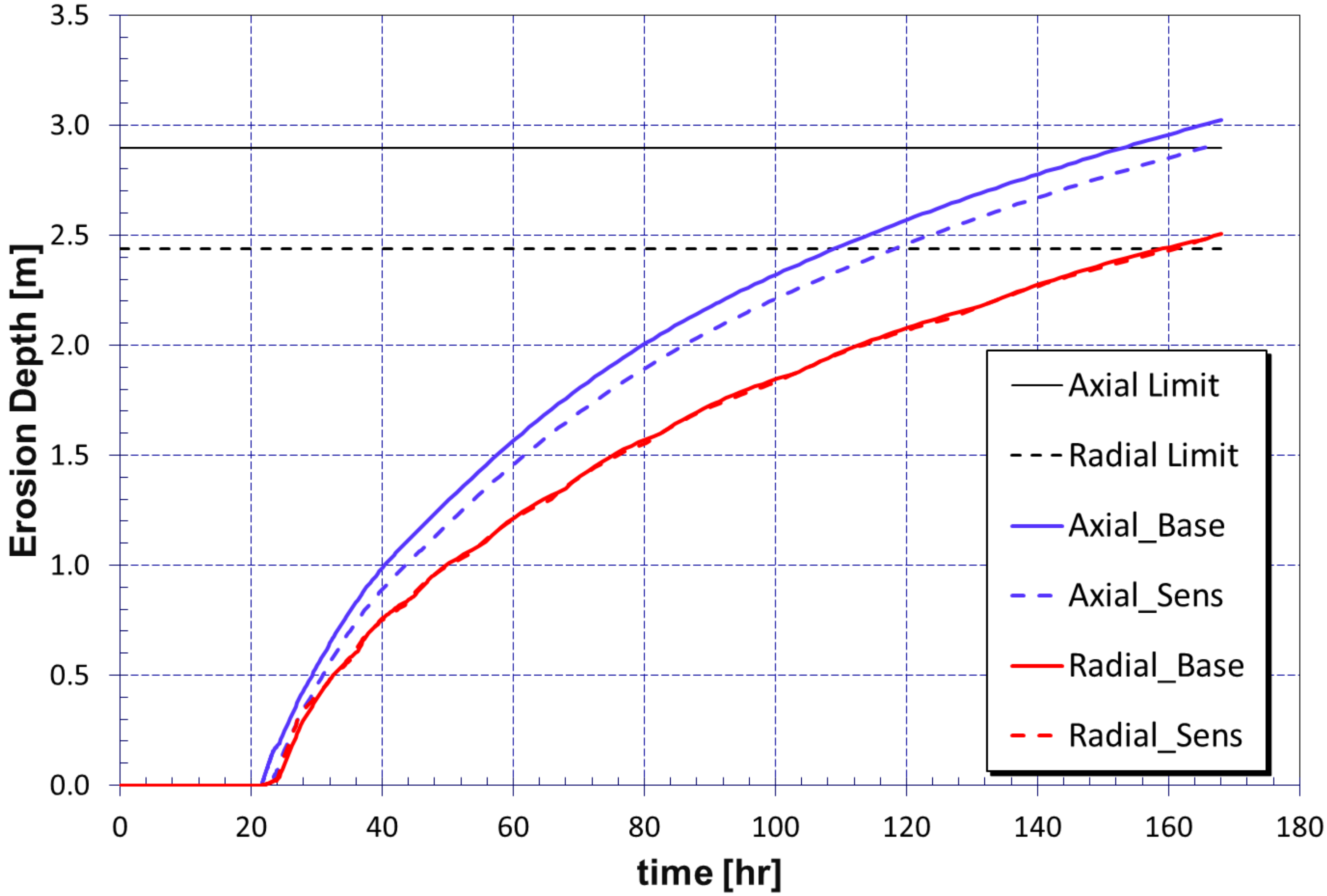


Figure 4-11: Basemat Melt-Through Timings for MU-4.1

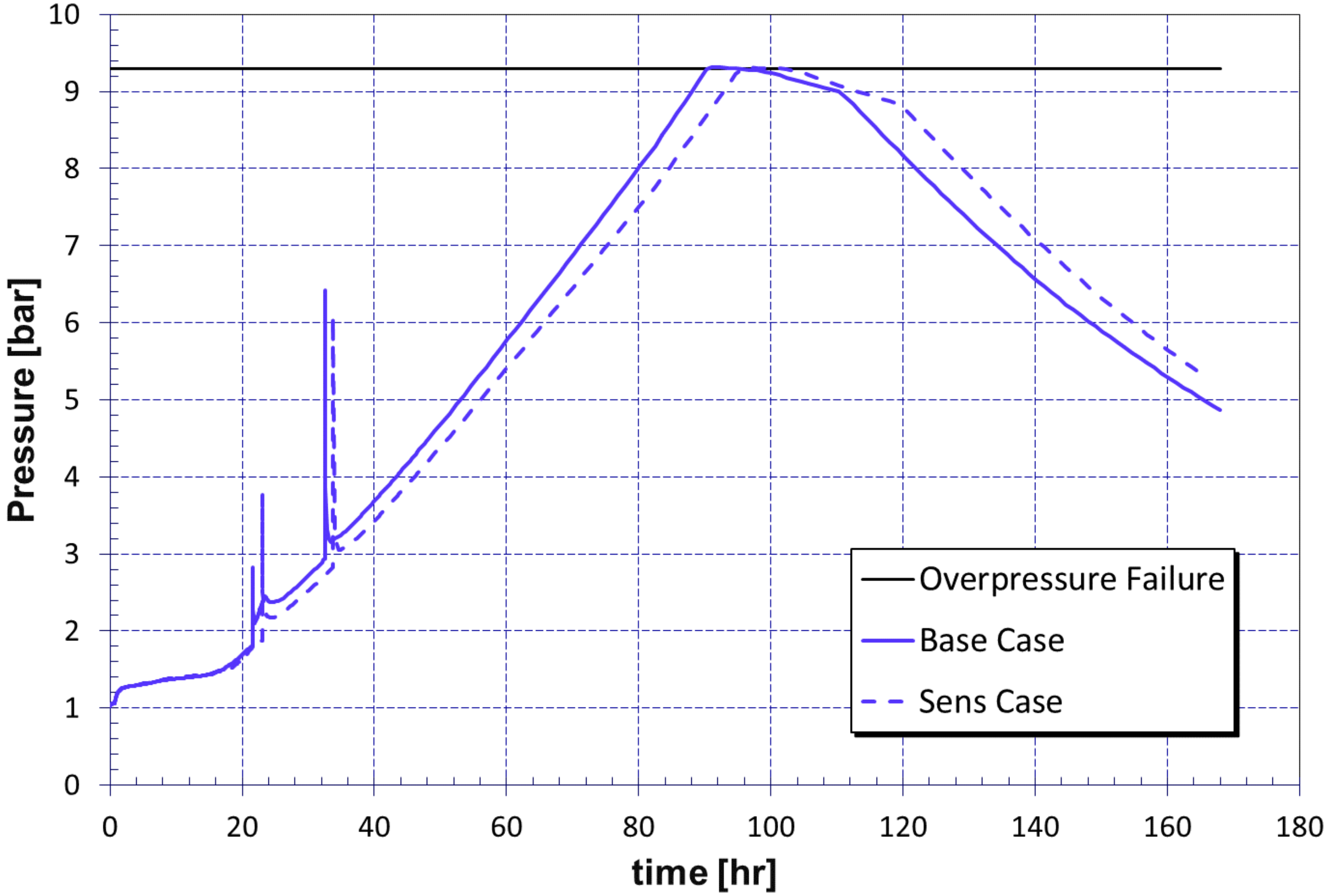


Figure 4-12: Containment Pressure for MU-4.1

4.4.4 Sensitivity Analysis for PRT Dry-Out with Non-Adiabatic Heat-Up Modeling (MU-4.2)

This sensitivity is based upon Case 1B2. This case, along with Case 3A4, are the only simulations in which the PRT dries out completely. With the PRT being modeled as adiabatic (i.e. insulated), the concern is that the tank overheats, volatilizing the retained fission products and causing a greater release to the environment. Note that this is not a major concern in Case 3A4 since the dryout occurs later in the simulation (86 hours) and little re-volatilization of the deposited radionuclides occurs. A more detailed discussion of this issue is given in Section 7.1.2.

For the sensitivity, the PRT model is modified in two significant ways. First, pool scrubbing (the SPARC model) is activated for the flowpaths entering the relief tank, modeled as a multi-hole sparger with area of 7.854×10^{-5} m^2. These parameters are adopted from the draft Surry SOARCA UA MELCOR deck (SNL, 2016a).

Second, the PRT is no longer modeled as being adiabatic (that is, insulated), an assumption that was adopted from the Byron MELCOR deck. This sensitivity attempts to characterize the impact of this misrepresentation of the PRT on this case.

The figures below show the mass of material retained on the PRT heat structure in time. Activating scrubbing to the PRT flowpaths leads to a modest increase in the retention of fission products in the tank. **Figure 4-13** and **Figure 4-14** show that the total mass of radionuclides retained in the PRT, and that subsequently deposit on the heat structure following PRT dryout, increases from 150 to 160 kg.

Figure 4-14 also demonstrates that re-volatilization of fission products does not occur after about 16 hours due to convective heat transfer from the PRT to the containment atmosphere. The temperature of the PRT heat structure is significantly less than the base case, as seen in **Figure 4-15**.

The fractional release of volatile fission products Cs, I, and Te to the environment is provided in **Figure 4-16**. Particularly in the case of cesium, re-volatilization in the PRT has a significant impact on the resulting release (4.3×10^{-2} when insulated versus 7.1×10^{-3} when cooling is allowed), since in the base case it is re-suspended after containment failure occurs.

This sensitivity shows that the PRT assumptions have a fairly large effect on the environmental releases for this case, with the base case over-predicting environmental releases with respect to this modeling error/assumption. As discussed, other cases are not expected to be sensitive to this issue (due to PRT dryout not occurring).

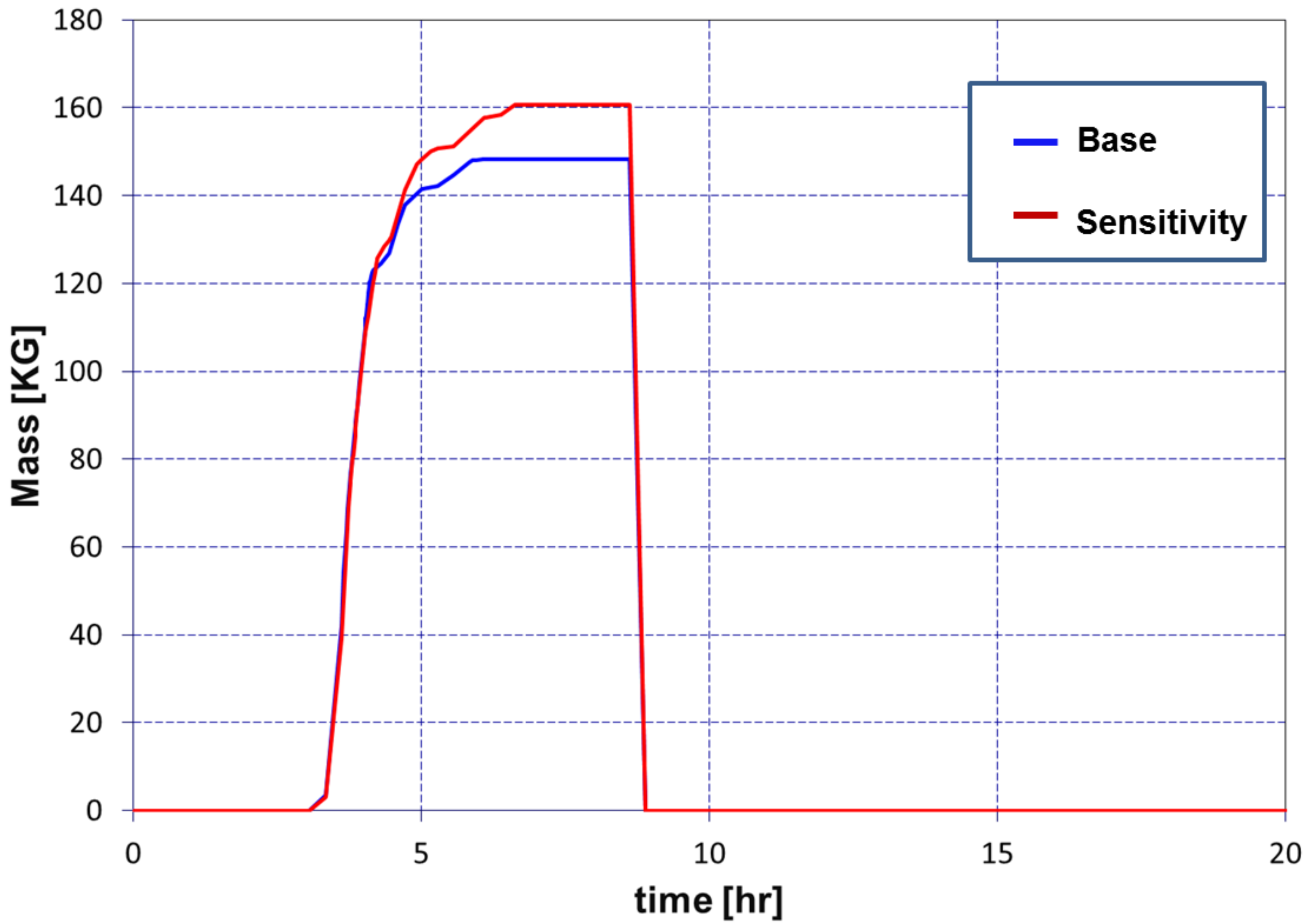


Figure 4-13: Mass of Radioactive Aerosol in the Liquid Phase in the PRT

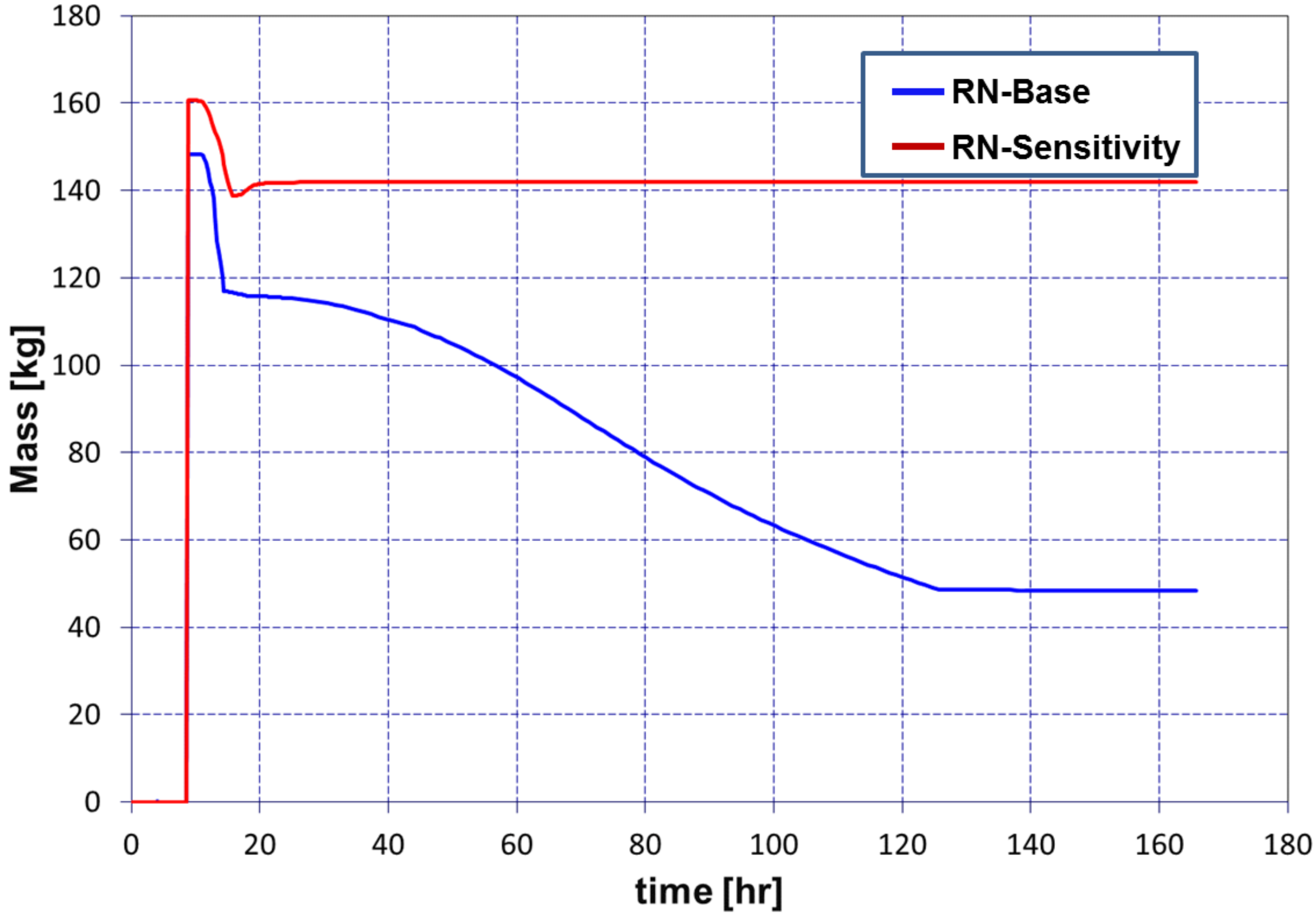


Figure 4-14: Mass of Radioactive Aerosol Deposited on the Inner Heat Structure of the PRT

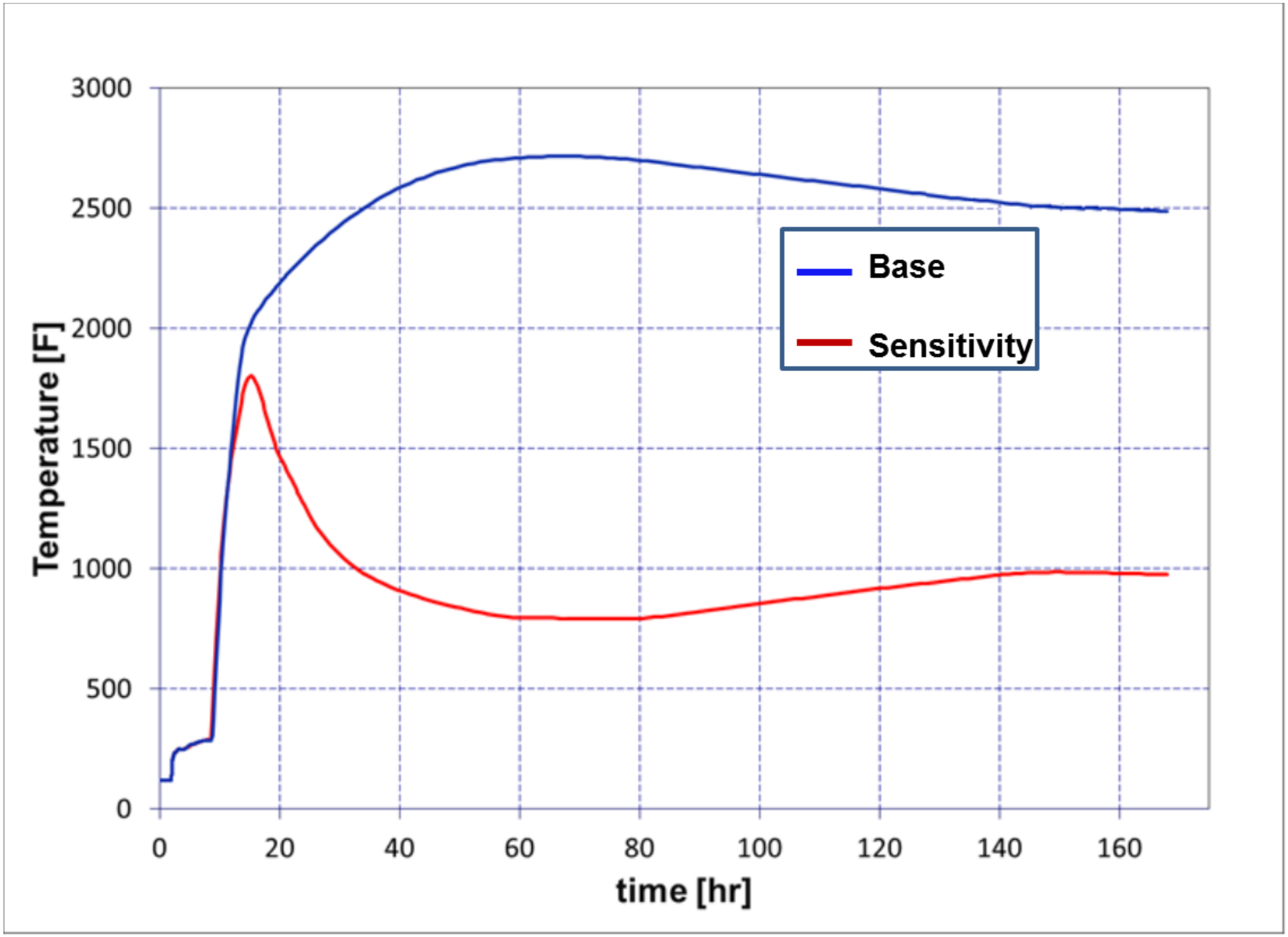


Figure 4-15: PRT Temperature for MU-4.2

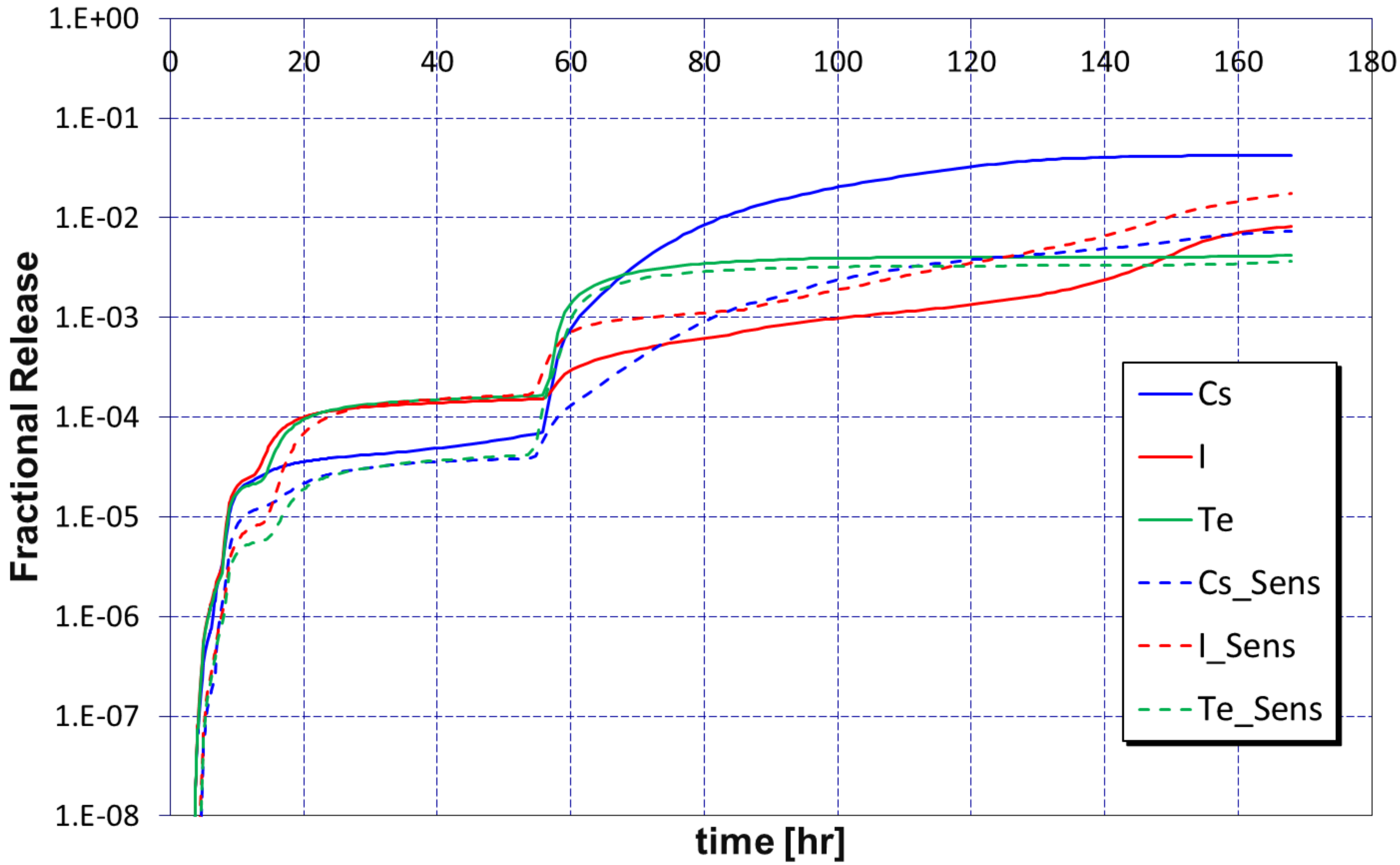


Figure 4-16: Fractional Release of Volatiles to The Environment

4.4.5 Sensitivity Analyses for Core Melt Progression (MU-4.3 A&B)

Two sensitivity calculations are described here, based on Case 1A2, exploring assumptions in core melt progression. Within the MELCOR code, the eutectic temperature parameter modifies the melting temperature of the UO_2/ZrO_2 eutectic for irradiated fuel. The breakaway temperature represents the maximum ZrO_2 temperature permitted to hold up molten zirconium in the fuel cladding. The SOARCA Surry UA contains figures describing the cumulative probability for each of these parameters (Figures 4-17 and 4-27 of [SNL, 2016a]). For the first sensitivity (MU-4.3A), both the eutectic and breakaway temperatures are increased to roughly the 90th percentile as read from these figures. For the second sensitivity (MU-4.3B), both temperatures are decreased to the 10th percentile. **Table 4-10** gives the breakout and eutectic temperatures being used in the base and sensitivity cases.

Table 4-10: Break-out and eutectic temperatures used in 1A2, MU-4.3A, and MU-4.3B

	Break-out Temperature [K]	Eutectic Temperature [K]
1A2 Base Case (MELCOR defaults)	2400	2500
MU-4.3A	2450	2585
MU-4.3B	2250	2375

The overarching impact of these changes is in the retention of fission products in the RCS. **Table 4-11** gives the retentions and releases of fission products. In the case of higher fuel failure temperatures (MU-4.3A), a larger portion of the radionuclides remain entrained in the RCS leading to a slightly diminished release to containment and ergo, the environment (**Figure** 4-17 and **Figure 4-18**). In the case of a lower fuel failure temperatures (MU-4.3B), the RCS retention is smaller than the other sensitivity and the base case because more fission products are released from the fuel sooner. However, the releases to the environment are comparable between the sensitivities. This is likely due to two things. First, these fission products have a greater amount of time to agglomerate and settle out. Second, the containment pressure is slightly less (~0.1 bar) at the time of containment failure, meaning less driving force for the airborne particles. The exception to this is iodine and tellurium in Case MU-4.3A, where an increased ex-vessel release results in a greater release to the environment despite the increased RCS retention.

Table 4-11: Fractional releases and retentions for Cases 1A2, MU-4.3A, and MU-4.3B

	Cesium			**Iodine**		
	1A2	**MU-4.3A**	**MU-4.3B**	**1A2**	**MU-4.3A**	**MU-4.3B**
In-vessel release	9.8E-01	9.8E-01	9.8E-01	9.6E-01	9.5E-01	9.6E-01
Ex-vessel release	1.9E-02	1.7E-02	1.7E-02	3.8E-02	4.5E-02	3.7E-02
RCS retention	3.3E-01	3.6E-01	3.3E-01	2.0E-01	2.1E-01	1.8E-01
Containment retention	5.9E-01	5.7E-01	5.9E-01	7.0E-01	6.8E-01	7.2E-01
Aux Building retention	4.8E-02	5.0E-02	4.9E-02	5.5E-02	6.0E-02	5.7E-02
Environment release	3.2E-02	1.9E-02	2.8E-02	4.3E-02	4.8E-02	4.0E-02

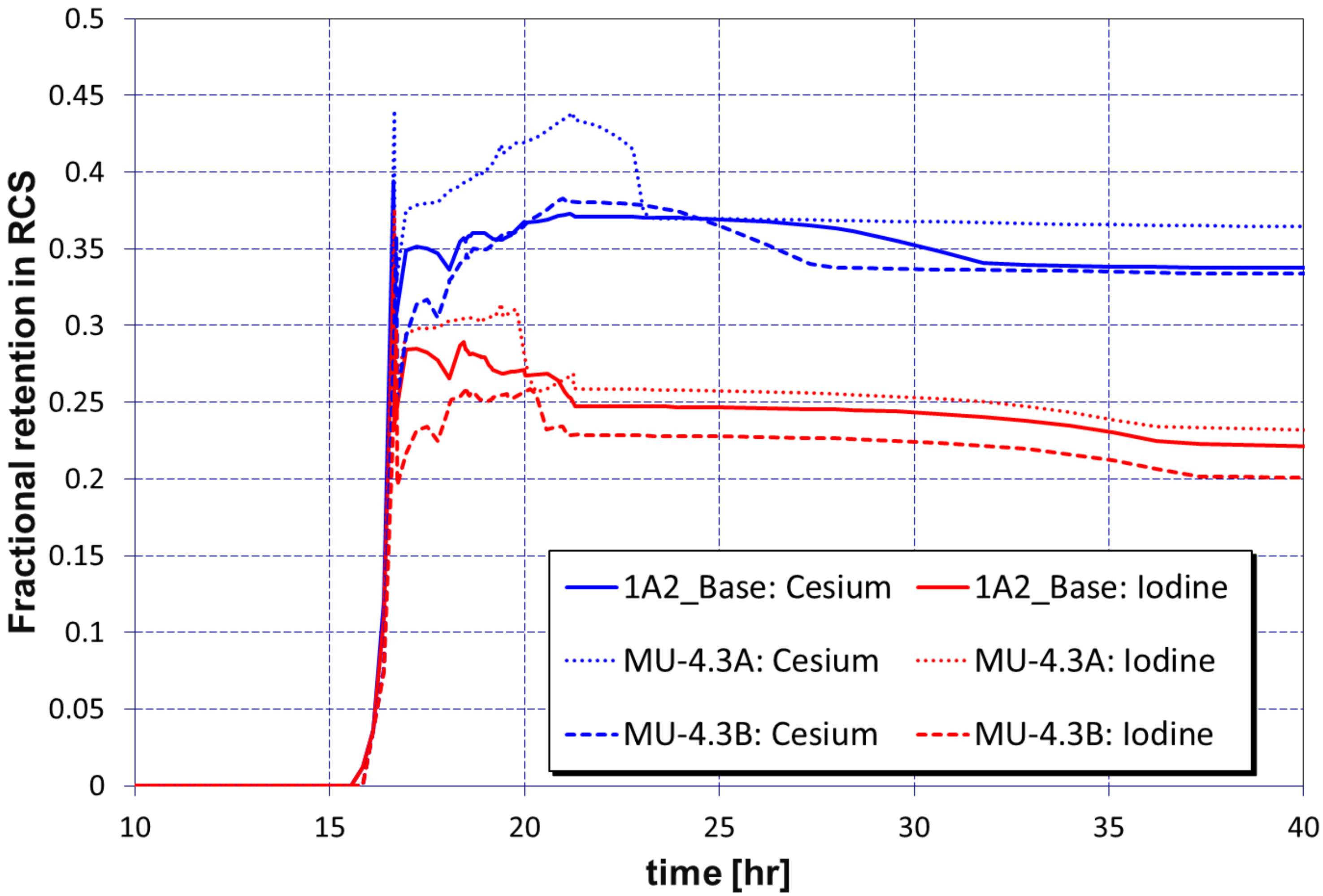


Figure 4-17: Retentions in the RCS of cesium and iodine for MU-4.3

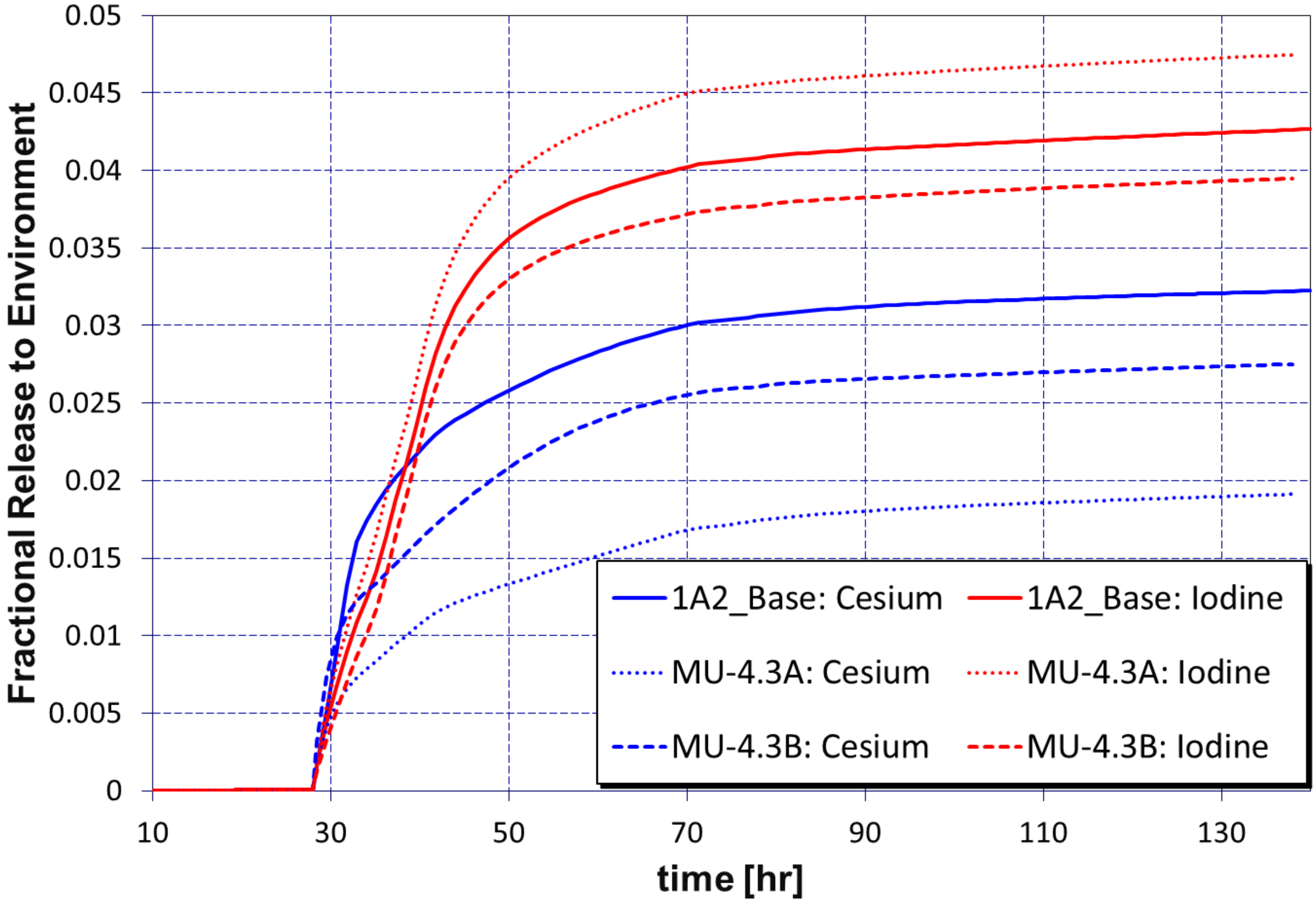


Figure 4-18: Environmental release of iodine and cesium for MU-4.3

The change in eutectic melt temperature has a measurable impact on the amount of hydrogen that is formed within the RCS and burns in containment. **Table 4-12** shows that with a small increase in the melt temperature, the amount of oxidation from Zircaloy/steam interaction increases, but the overall hydrogen generation is comparable to the base case. For MU-4.3B, significantly less hydrogen is generated within the RCS, resulting in reduced hydrogen burning within containment.

Table 4-12: Hydrogen generation for Cases 1A2, MU-4.3A, and MU-4.3B

Source		Hydrogen [kg]		
		Base Case	MU-4.3 A	MU-4.3 B
Generated In-vessel	Total	779	769	662
	From Zirc Oxidation	648	680	568
Generated Ex-vessel	-	4597	4581	4639
Burned in Containment	-	123	133	107

These two sensitivities show that the selected parameters have a measurable effect on in-vessel behavior, but the effect on cumulative environmental releases is modest.

4.5 Induced RCS Component Failure Modeling

4.5.1 Identified Uncertainties in Induced RCS Component Failure Modeling

Table 4-13: Identified Uncertainties in Induced RCS Component Failure Modeling

Item	Description	Other comments
Depressurization for Level 1 sequences where it is not queried	There are cases where the Level 1 PRA does not query operator action to depressurize the plant, because it does not affect the determination of core damage, although such action might be procedurally driven. This could have some impact on induced component failure for these sequences.	Note that this is less of a concern for the current model, because: (i) the Level 1 PRA has been upgraded to add additional top events in this regard, (ii) current MELCOR results suggest that the pressure at the time of in-vessel melt progression is less sensitive to such pre-core damage accidents, and (iii) the probability for induced hot leg failure at intermediate pressure acknowledges this uncertainty.
Stochastic and high-temperature seizure sticking of PORVs	In certain situations, this can have a significant effect on accident progression by changing fundamental RCS conditions (e.g., causing depressurization that reduces the likelihood of hot leg nozzle creep rupture).	Note that the effect of cycling failure on release category frequency (as opposed to source term) is conceptually covered by the propagation of parameter uncertainty in the PRA model. The draft Surry SOARCA UA (SNL, 2016a) considers this issue from the perspective of passing water, cycling, and high temperature seizure (the latter two are similar to the treatment of the issue in [SNL, 2013]).

Table 4-13: Identified Uncertainties in Induced RCS Component Failure Modeling

Item	Description	Other comments
Stochastic failure of secondary side relief valves	Particularly important for consequential SGTR (C-SGTR), but also relevant to other scenarios in terms of heat removal through SGs	Note that the effect of cycling failure on release category frequency (as opposed to source term) is conceptually covered by the propagation of parameter uncertainty in the PRA model. In the draft Surry SOARCA UA (SNL, 2016a), relief valve sticking and effects of secondary side depressurization (valve stuck open or main steam isolation valve (MSIV) leakage) are called out separately – here they are combined for C-SGTR.
Deterministic modeling of counter-current hot leg/SG flow	There are two basic approaches to this in MELCOR: (i) allow the code to calculate the buoyantly-driven flow based on an expected recirculation ratio or (ii) specify a Froude number and artificial pumps to match test and computational fluid dynamics (CFD) data. The modeling in both cases is tuned to CFD results and test data that were focused on high-pressure sequences but is applied across virtually all cases (it activates in all of the cases in Appendix B of this report except for the 8" ISLOCA cases).	As of Revision 4 and later, the MELCOR model takes the former of the two approaches described to the left (earlier versions used the latter approach). (Yuan, 2013a) provides some perspective on the differences observed (which were generally small). The modeling is expected to give reasonable results when applied at lower pressures, but this has not been specifically demonstrated in the MELCOR calculations for this project. Also, the current modeling predicts sufficient steam cooling so as to delay core damage until after complete core uncovery for high-pressure scenarios (most 1/3/6/7-series cases in Appendix B) exhibit this, while almost all of the remainder do not).
Modeling of instrument tube failure during core degradation	Can systemically affect RCS pressure following core damage, natural circulation effects, and early transport of hydrogen to the containment	This issue is discussed further in Section 5 of Appendix D and its effects can be seen in the 3-series cases in Appendix B.
Modeling of RCS induced component failures, including C-SGTR	There are potentially large model uncertainties associated with both the deterministic and probabilistic modeling of these issues. Assumptions regarding instrument tube failure during core degradation, SG tube flaw distributions, and secondary-side depressurization caused by leakage past isolation valves (etc.) can affect C-SGTR characterization. Secondary side depressurization (from all causes) and instrument tube failure can affect which non-SG RCS component fails first.	For the draft Surry SOARCA UA (SNL, 2016a), secondary side turbulent deposition during SRV blowdown, hot tube treatment, peak plume calculation, gas to structure radiation, enhanced natural circulation, counter-current flow limitations in the surge line, tube degradation multipliers, SG hot leg and mixing plenum ratios, and hot leg nozzle carbon steel safe end modeling are all identified. Other aspects are discussed in Section 5 of Appendix D.

4.5.2 Alternative Treatment(s) of Uncertainties in Induced RCS Component Failure Modeling

MU-5.1A A MELCOR simulation was performed (based on Case 1A2 [see Section 1.1.2 of Appendix B]) with an SRV sticking open at the time of its first cycle (around 12.6 hours).

MU-5.1B A MELCOR simulation was performed (based on Case 1A2) with an SRV sticking open at 14.5 hours. This case, along with MU-5.1A, demonstrate the effect of the failure assumptions on the source term relative to Cases 1 and 1A2.

4.5.3 Sensitivity Analyses for Effect of a Stuck Open Pressurizer Safety Relief Valve (MU-5.1 A&B)

In this pair of sensitivities based upon Case 1A2, the effect of the pressurizer SRV sticking open is explored with regards to the key event timings and fission product releases to the environment. In the first sensitivity, MU-5.1A, an SRV is assumed to stick open on its first demand at 12.6 hours. For MU-5.1B, an SRV sticks open around 14.5 hours,[3] at which point the SRV has cycled 99 times in the MELCOR simulation. For both cases, the PRT is expected to dry out and, as described in Section 7.1.2, if the PRT is modeled as insulated, it will reach un-reasonably high temperatures. For this reason, these sensitivities utilize the MELCOR deck from sensitivity Case MU-4.2 (see Section 4.4.2 - uninsulated PRT and pool scrubbing enabled).

A comparison is made in **Table 4-14** of the key event timings in the base and sensitivity cases. With an early failure of the SRV, the inventory of water in the RCS drops quickly and the fuel begins to be uncovered about half an hour sooner than in the base case. Complete core uncovery and oxidation are moved up in time. The same can be said of MU-5.1B, though to a lesser extent. The timing of vessel breach in both sensitivities, however, is not greatly affected.

Table 4-14: Key event timings (in hours) for MU-5.1

Event	MU-5.1A	MU-5.1B	1A2
SRV fails in the fully-open position	12.6	14.5	-
RPV water level reaches top of active fuel (TAF)	12.8	13.2	13.2
Core uncovery (bottom of active fuel [BAF])	13.8	14.6	14.8
Core oxidation	13.7	14.9	15.6
Exceed 2200	13.9	15.1	16.2
Hot leg nozzle creep rupture	-	-	16.6
PRT dryout	19.2	20.2	-
Vessel breach	21.1	21.2	21.2
Containment failure (assumed)	28.0	28.0	28.0

In both sensitivity cases, fission products released through the failed SRV are scrubbed in the PRT (making up the bulk of the “RCS” retentions seen in **Figure 4-19**). When the PRT dries out, these radionuclides deposit on the PRT surface, increasing the temperature of the heat structure (**Figure 4-20**), which causes the iodine and tellurium classes in particular to re-volatilize. By this time, containment failure has already occurred, and they are quickly released to the environment. The majority of the other fission products remain entrained in the PRT. In

[3] This time is chosen arbitrarily as a midpoint between the first sensitivity where the RCS depressurized at 12.6 hours from the stuck-open valve and the base case (1A2) where hot leg creep rupture causes the RCS to depressurize at 16.6 hours.

the base case of 1A2, the fission products are released to containment at the time of hot leg creep rupture (**Figure 4-21**). However, unlike the sensitivity cases, by the time containment failure occurs, the airborne fission products have had time to settle out and little environmental release occurs (**Figure 4-22** and **Table 4-15**).

As the sensitivities show, failure of the relief valve during core damage can significantly alter the time and nature of release to the containment. If the timing of this release adversely corresponds to the time of containment failure (such as the posited hydrogen combustion in Case 1A2), then the environmental release can also be significantly affected.

Table 4-15: Fractional releases to the environment in MU-5.1

Representative Element	A-Early	B-Late	1A2
Xe	9.9E-01	9.9E-01	9.9E-01
Cs	1.2E-02	2.9E-02	3.2E-02
I	1.0E-01	1.9E-01	4.3E-02
Te	1.1E-01	6.1E-02	4.3E-02
Ba	2.5E-03	6.8E-04	2.0E-03
Ru	0.0E+00	6.0E-05	6.2E-05
Mo	1.8E-01	1.1E-01	1.4E-01
Ce	0.0E+00	5.8E-06	3.2E-05
La	0.0E+00	3.8E-06	4.5E-06
UO2	0.0E+00	4.6E-05	5.3E-05
Cd	2.5E-02	3.8E-02	4.1E-02
Ag	4.3E-02	5.0E-02	5.7E-02

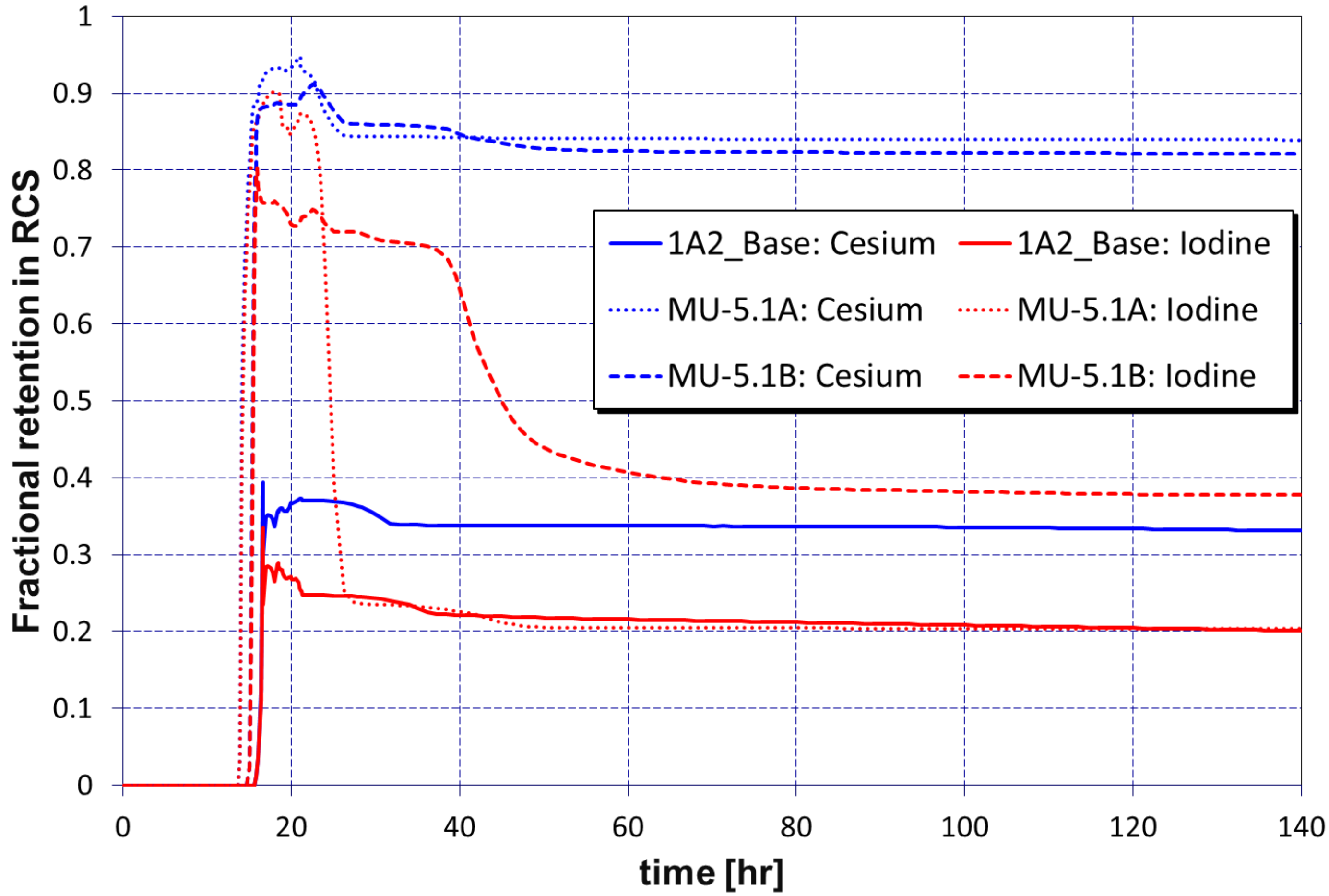


Figure 4-19: Fractional retention in the RCS[4] – Cesium and Iodine

[4] Fission products in the PRT are included in this category

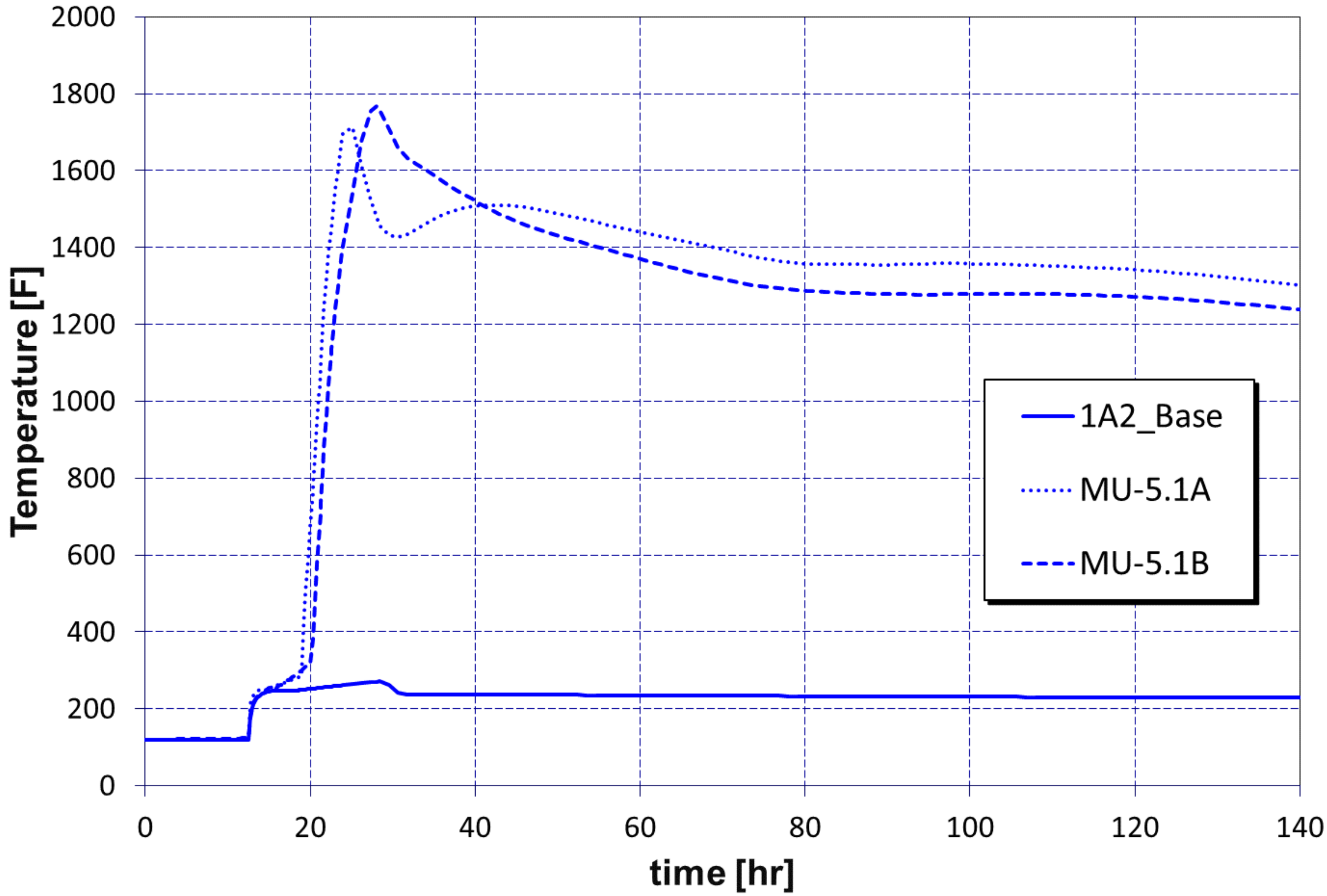


Figure 4-20: Temperature of the PRT Heat Structure in the Base and Sensitivity Cases

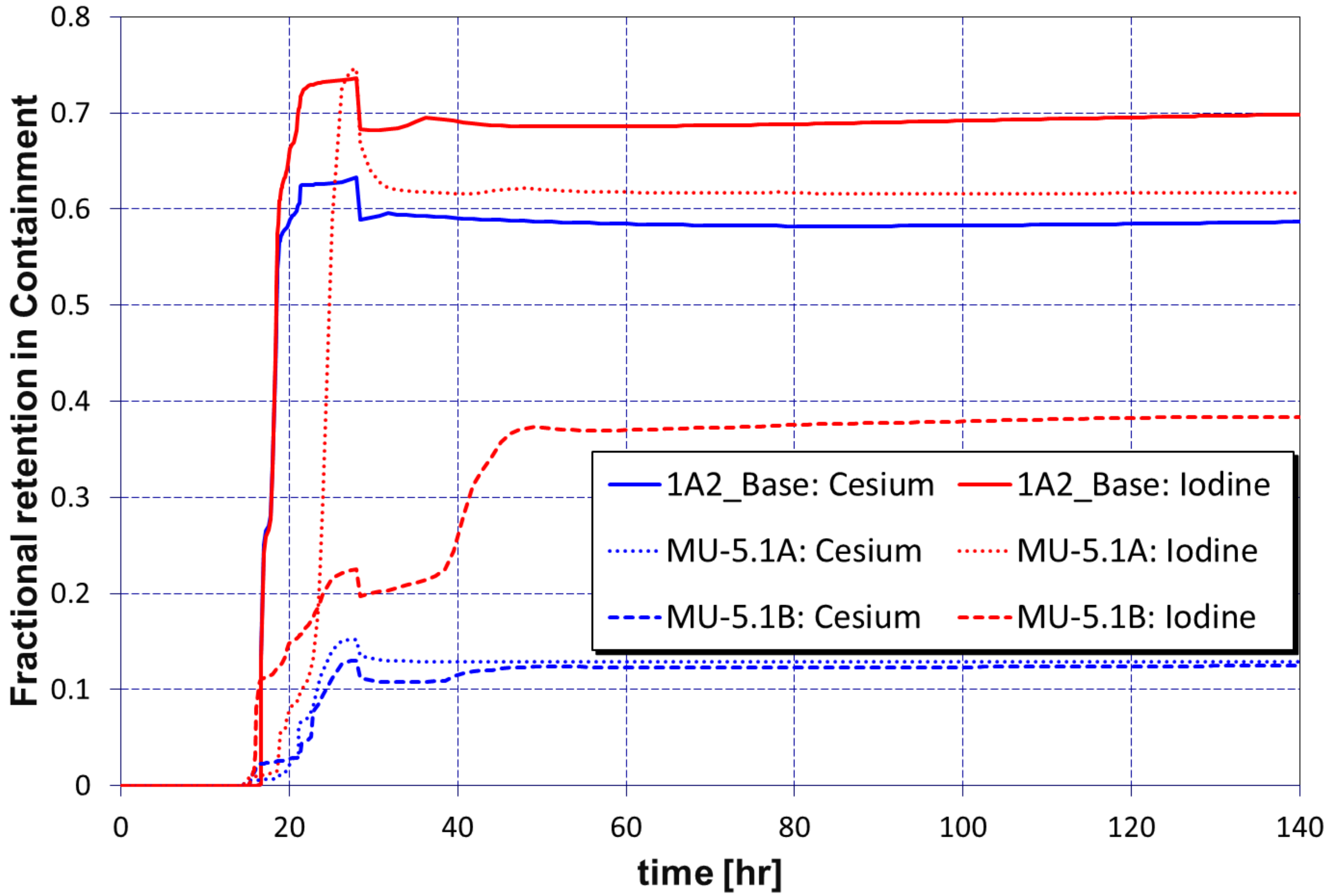


Figure 4-21: Retention in Containment – Cesium and Iodine

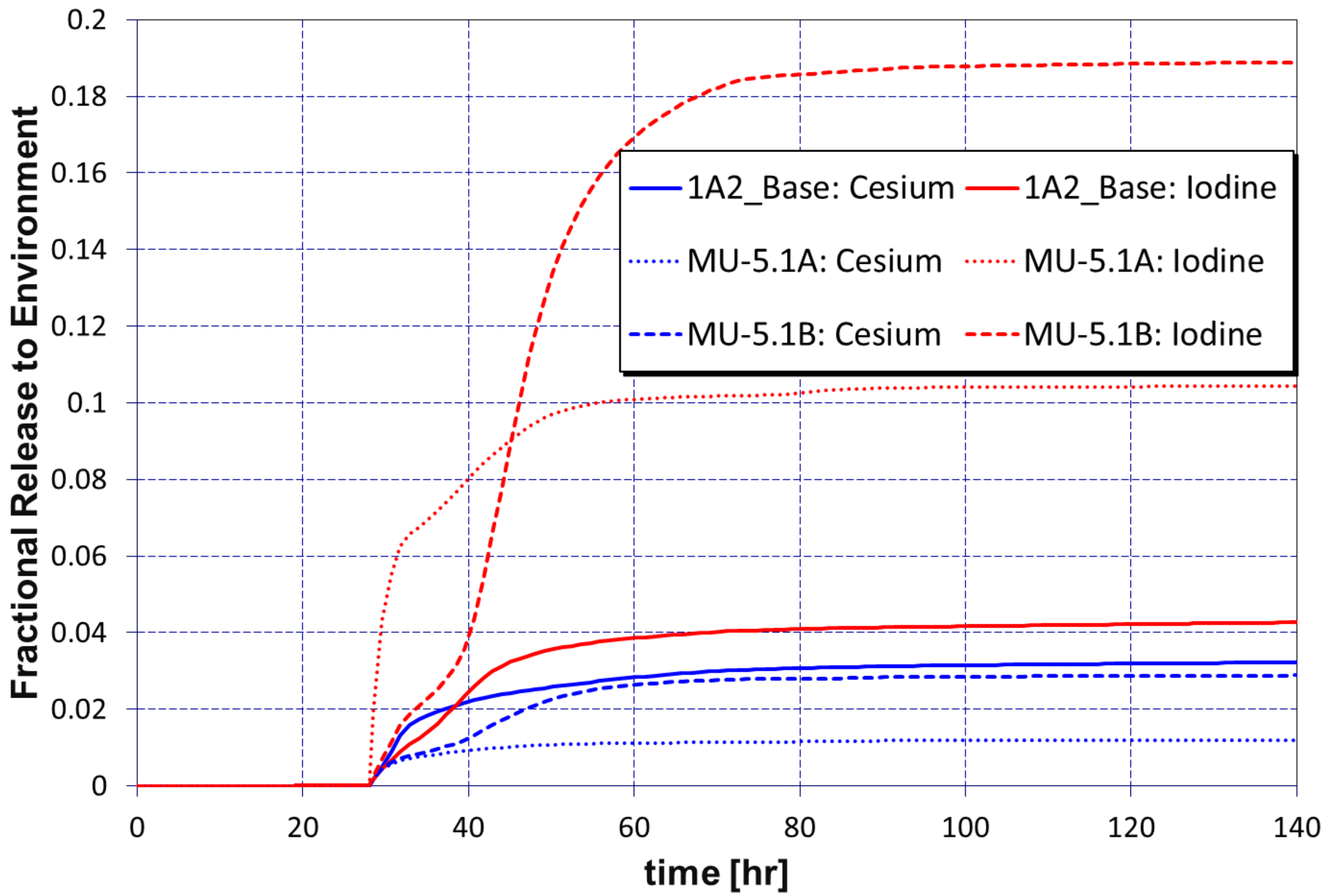


Figure 4-22: Environmental releases – Cesium and Iodine

4.6 Vessel Breach and Associated Energetic Event Modeling

4.6.1 Identified Uncertainties for Vessel Breach and Associated Energetic Event Modeling

Table 4-16: Uncertainties for Vessel Breach and Associated Energetic Event Modeling

Item	Description	Other comments
Spillover point between lower containment and the reactor cavity	Early leakage of water into the cavity could affect initial steaming rates and concrete ablation rates soon after vessel breach, though only for sequences where the RWST has been injected resulting in a partially-flooded lower containment.	This refers to leakage through sealed penetrations, as discussed further in Section 12 of Appendix D.
RPV lower head failure mechanisms	This will affect the accident progression downstream of vessel failure.	Different possible failure modes exist.
Vessel flange leak-by	This can lead to significant water intrusion in the cavity (and, in particular, prior to the time of vessel breach) for scenarios where containment spray has operated for an extended period of time and water would not have otherwise entered the cavity.	This issue is discussed from a design perspective in Section 12 of Appendix D.
Initial containment response to vessel failure	Some uncertainty exists in the pathways by which the cavity will communicate with various portions of lower containment.	This item is discussed in more detail in Section 12 of Appendix D.
Indirect mechanisms of containment failure	Energetic low-probability events could lead to containment failure (e.g., less-catastrophic vessel rocketing effects that lead to containment leakage due to stressed containment penetrations).	These events are formally considered, and either dispositioned as not credible or modeled. This line item simply captures any uncertainty in those estimates not otherwise captured by parameter uncertainty in their split fraction distributions. Note that many of the relevant items are already captured by other model uncertainty line items.
Direct containment heating (DCH) / high-pressure melt ejection (HPME)	DCH/HPME has the potential to lead to early containment failure, and conversely, to spread the melt into a coolable geometry.	This issue is covered in Section 7 of Appendix E.
Ex-vessel fuel-coolant interactions (steam explosions)	This could be a low-probability energetic failure event.	This issue is discussed in Section 12 of Appendix E. Note that containment failure due to ex-vessel steam explosion is currently disabled in the probabilistic model due to the extremely low estimated likelihood.

4.6.2 Alternative Treatment(s) of Uncertainties for Vessel Breach and Associated Energetic Event Modeling

MU-6.1 A MELCOR simulation was performed (based on Case 2A [see Section 2.3 of Appendix B]) where the failure mode is assumed to be a lower head penetration failure. The penetrations are modeled using the same input parameters as in the SOARCA Surry input model (NRC, 2013). It should be noted that penetration failure was not modeled in SOARCA as a mechanism of vessel failure, as discussed in NUREG/CR-7008, for a variety of reasons (including lack of an adequate penetration failure model). As in Case 2A, containment will be forced to fail at the time of penetration failure. The environmental source term will be compared between this and the base case.

MU-6.2A A MELCOR simulation was performed (based on Case 3A4 [see Section 3.4 of Appendix B]) with a small hole (2-inch diameter) forming in containment at the time of vessel breach, intended to reflect a tear in containment during a high pressure melt ejection (due to vessel rocketing, steam explosion, etc.).

MU-6.2B A MELCOR simulation was performed (based on Case 3A4) with a large hole (8-inch diameter) forming in containment at the time of vessel breach, intended to reflect a catastrophic failure during a high pressure melt ejection (due to vessel rocketing, steam explosion, etc.). This case, along with MU-6.2A, demonstrate the effect energetic low-probability events could have on the source term relative to Case 3.

4.6.3 Sensitivity Analysis for Pressure Vessel Melt-Through vs Penetration Failure (MU-6.1)

In this sensitivity, based upon Case 2A, the failure mode of the pressure vessel is changed from vessel melt-through to penetration failure. The five penetrations are adopted from the Surry SOARCA input model (NRC, 2013)―note, however, that this failure mode was not actually used in that study―with failure occurring at a penetration temperature of 1273°K. While the addition of these penetrations may affect the overall heat transfer to the vessel head, this effect is not thought to be significant.

Failure of the vessel occurs earlier in the sensitivity case with the first penetration reaching the defined failure temperature at around 20.1 hours in MU-6.1, a half hour sooner than lower head melt-through in Case 2A. The initial diameter of the failure opening is set at 0.123 m, but grows quickly as ablation of the hole occurs. The remaining penetrations fail at 20.6, 20.8, 20.9, and 21.0 hours, respectively. The RCS depressurizes at the time of the first failure and accumulators inject their remaining contents. While the water leaks quickly out of the vessel, it serves to cool the debris in the lower plenum and delay formation of significant molten material. MELCOR requires that a certain mass of molten material be present in the lower plenum before ejection can begin. This threshold is met at 21.3 hours and relocation to the cavity begins at this time. The debris is ejected at a somewhat slower rate initially due to the smaller hole size. However, with the growing hole diameter from ablation, by 23 hours roughly the same mass of debris has relocated to the cavity as in the base case.

Results of this sensitivity should be interpreted cautiously. Some of the phenomena described here are likely non-physical due to a high heat transfer coefficient to the penetrations at lower temperatures and the delayed ejection of core material due to the way MELCOR models relocation of debris to the cavity. The state of knowledge on penetration failure is itself limited, however.

Table 4-17 gives the release fraction of fission products to the environment and retention in the RCS at 140 hours and demonstrates that there is little impact on the release fractions and retentions when penetration failure is considered.

Table 4-17: Fractional RCS Retentions and Environmental Releases for MU-6.1

Representative Element	RCS		Environment	
	2A	**MU-6.1**	**2A**	**MU-6.1**
Xe	3.2E-4	3.8E-4	9.9E-1	1.0
Cs	3.6E-1	3.2E-1	1.6E-1	1.5E-1
I	3.1E-1	1.9E-1	1.5E-1	1.5E-1
Te	5.1E-1	2.0E-1	1.3E-1	1.8E-1
Ba	7.8E-3	2.4E-3	2.2E-3	3.6E-3
Ru	1.2E-2	5.7E-3	2.7E-4	1.7E-4
Mo	9.0E-2	8.3E-2	2.1E-1	2.2E-1
Ce	2.3E-7	1.4E-7	5.6E-6	8.0E-6
La	2.4E-7	1.4E-7	4.0E-6	3.1E-6
UO2	1.5E-3	7.2E-4	1.3E-4	1.3E-4
Cd	3.8E-1	2.6E-1	1.7E-1	1.8E-1
Ag	2.6E-1	2.3E-1	2.0E-1	2.1E-1

4.6.4 Sensitivity Analyses for Containment Breach Size (MU-6.2 A&B)

For this analysis, two calculations are presented. In the first (MU-6.2A), a small 20.3 cm^2 hole in containment is assumed to form at the time of vessel breach at 12.8 hours. For the second (MU-6.2B), a larger 324 cm^2 hole opens in containment at the time of vessel breach. These two cases are intended to explore the possible range of consequences that may result from high energy events during a high-pressure vessel breach (i.e. steam explosions, high pressure melt ejection, direct containment heating, and vessel rocketing leading to stressed containment penetrations). These sensitivities are based upon Case 3A4 – a high-pressure transient in which all induced RCS failure paths are disabled, resulting in a highly pressurized RCS at the time of vessel breach (see Section 3.4 of Appendix B).

Containment pressure (**Figure 4-23**) in the base case quickly increases at the time of vessel breach and reaches the containment overpressure set-point at 64.5 hours. The large failure path in the MU-6.2B sensitivity causes a rapid depressurization of containment (within 6 hours of vessel breach). In contrast, the smaller flow path of the MU-6.2A sensitivity keeps containment pressurized (still reaching 4.5 bar).

In the base and MU-6.2A calculations, the higher pressure of containment (and hence the breached RCS) leads to a greater retention of fission products – particularly iodine – in the RCS. **Figure 4-24** demonstrates that in Case MU-6.2B, however, the iodine does not remain retained in the RCS. This, in conjunction with the large flow path to the environment, leads to a large release of iodine to the environment (see **Figure 4-25**). The source terms for the three cases are compared in **Table 4-18**. Note that there is a significantly larger release of all fission products in both sensitivity cases, demonstrating that early containment failure is of greater environmental consequence than failure size, with early, large failure having the greatest impact.

The sensitivities show the expected trends, with low-probability events (and thus low-frequency PRA sequences) involving early energetic containment failure leading to significantly larger environmental releases than the situations where these early containment failure events do not occur.

Table 4-18: Environmental releases of fission products for MU-6.2

Representative Element	3A4	MU-6.2A	MU-6.2B
Xe	8.6E-1	9.9E-1	1.0
Cs	9.9E-3	3.7E-2	1.1E-1
I	1.5E-2	8.7E-2	2.7E-1
Te	8.3E-3	3.4E-2	1.2E-1
Ba	3.1E-4	5.6E-4	1.4E-3
Ru	7.8E-7	1.4E-5	7.5E-5
Mo	3.6E-2	6.8E-2	2.1E-1
Ce	8.1E-7	3.6E-6	6.1E-6
La	5.3E-7	1.6E-6	4.2E-6
UO2	5.9E-5	7.4E-5	8.1E-5
Cd	3.9E-2	5.9E-2	1.4E-1
Ag	2.7E-2	5.3E-2	1.7E-1

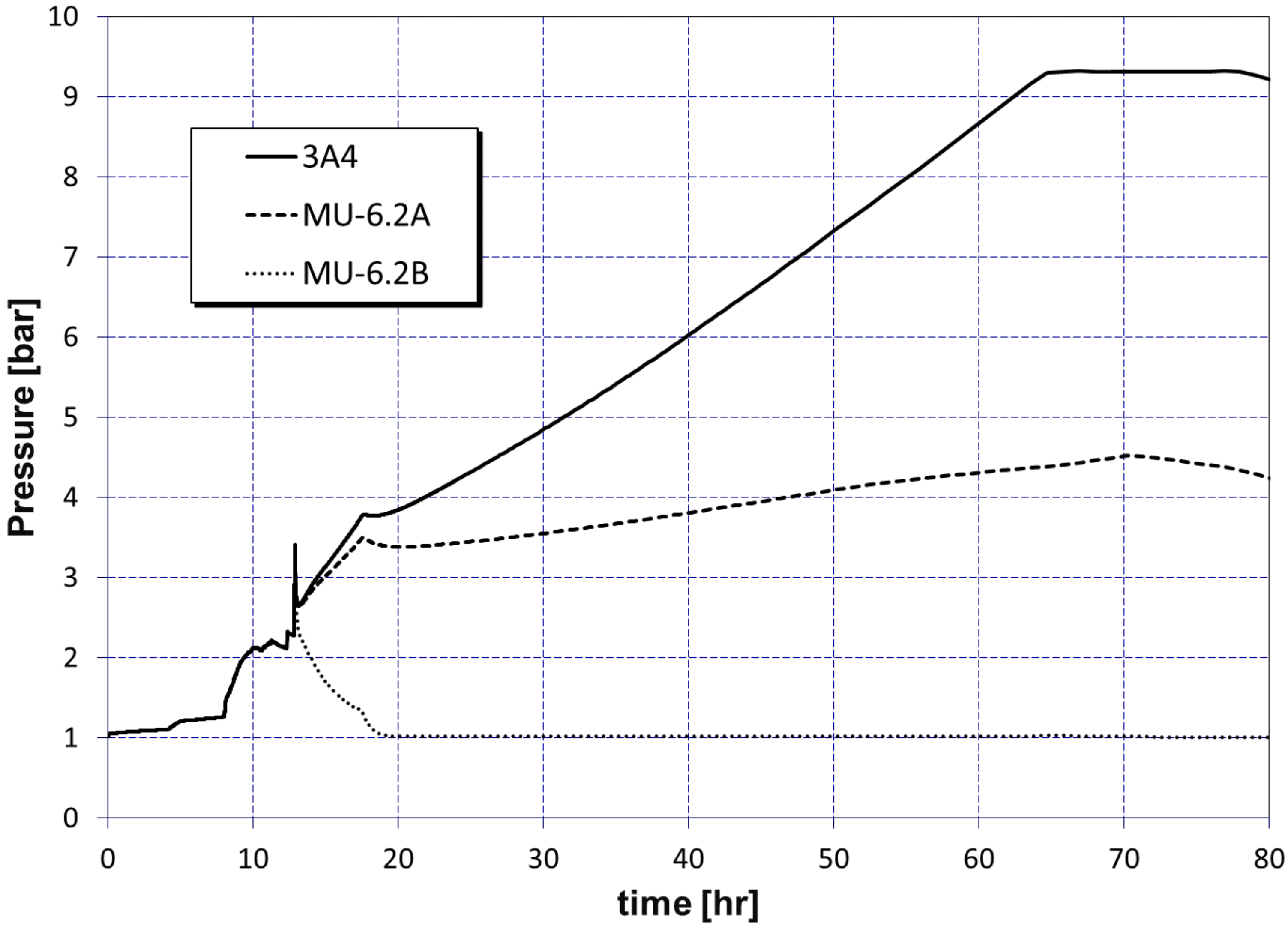


Figure 4-23: Containment pressure in the MU-6.2 base and sensitivity Cases

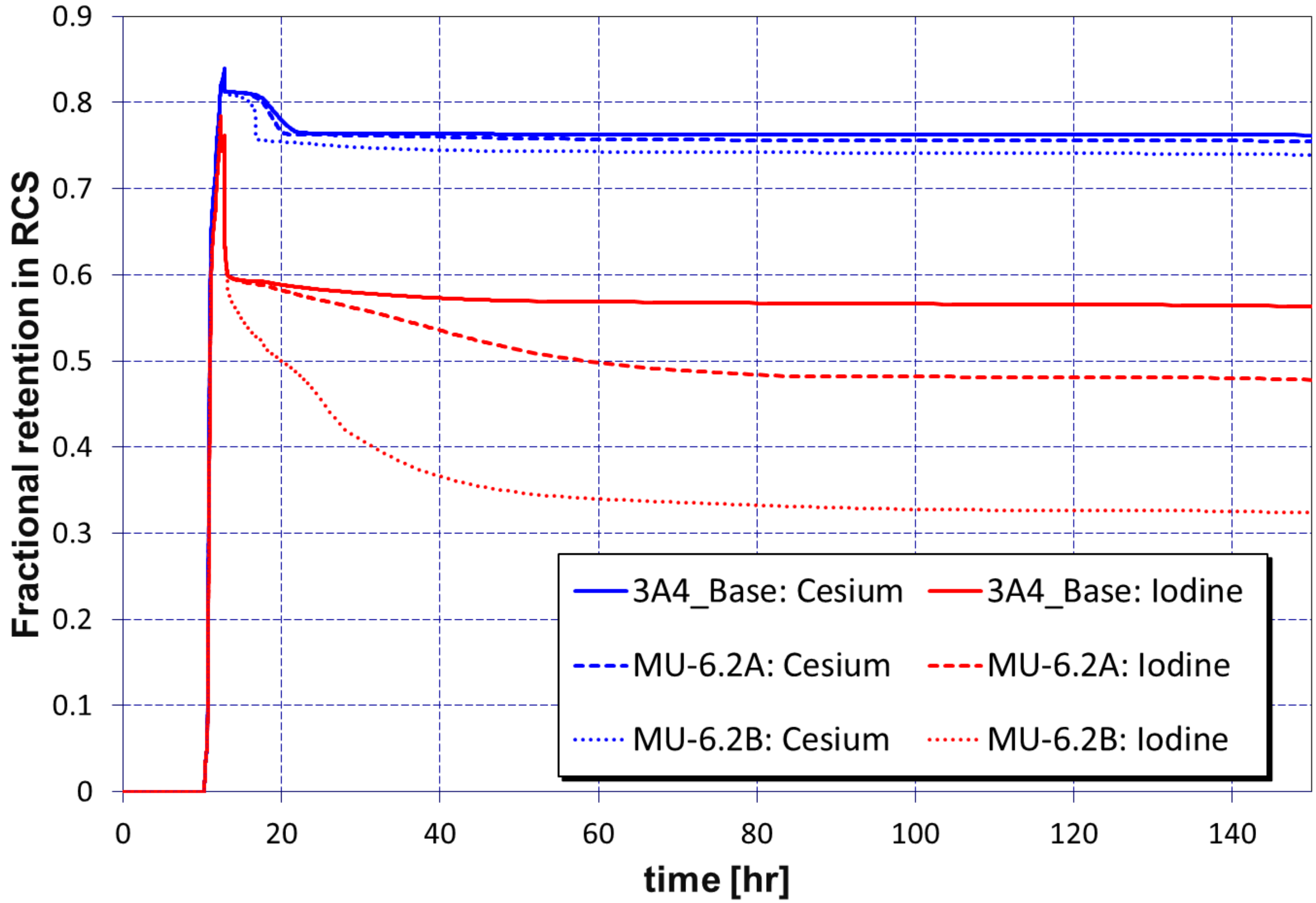


Figure 4-24: RCS retentions of Iodine and Cesium in the MU-6.2 base and sensitivity Cases

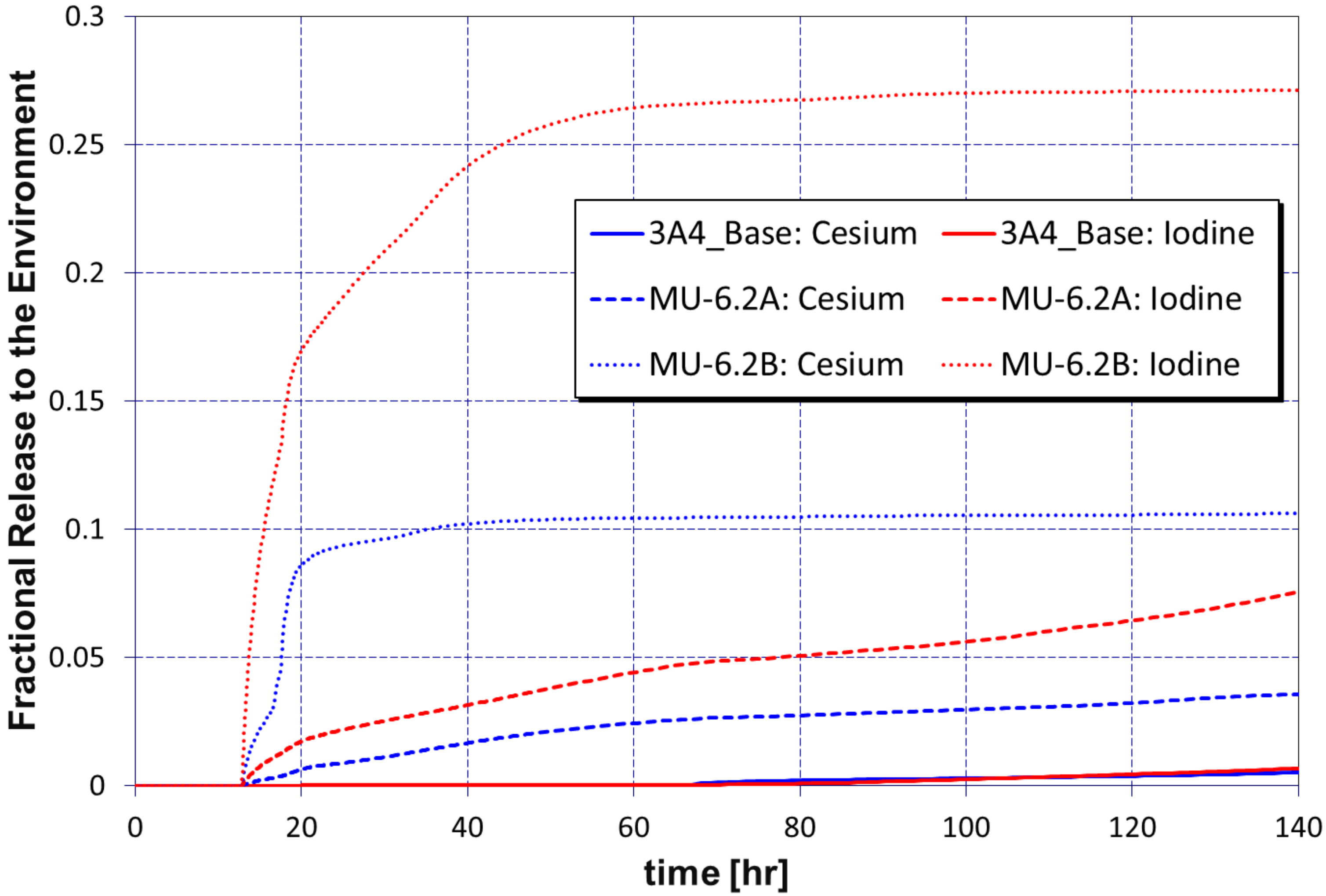


Figure 4-25: Environmental releases of Iodine and Cesium in the MU-6.2 base and sensitivity Cases

4.7 Combustible Gas Modeling

4.7.1 Identified Uncertainties for Combustible Gas Modeling

Table 4-19: Uncertainties for Combustible Gas Modeling

Item	Description	Other comments
In-vessel hydrogen production	This can affect downstream combustion event effects.	This item has been subsumed in Section 4.4 and is repeated here for the sake of cross-referencing.
Ex-vessel combustible gas production	This can affect the likelihood and severity of late burns.	The comparison of the L3PRA project station blackout results to the draft Surry SOARCA UA base case (SNL, 2016a) (see Section 4.3 of Appendix D) suggests a large difference in the cumulative production over equivalent time periods.
Dynamic load impact on containment failure mode	Large uncertainties may exist in both the deterministic and probabilistic modeling of containment response to energetic events (namely detonation of combustible gases).	More detail is offered in (EPRI, 2012b).
Energetic burning of hydrogen and other combustible gases	This is a source of uncertainty for containment failure / release characterization – note that this is intended to capture the model uncertainty aspect of this issue, recognizing that the logic model parameter uncertainties also reflect some of this uncertainty.	Additional commentary is provided in (EPRI, 2012b) and Section 8 of Appendix E.

4.7.2 Alternative Treatment(s) of Uncertainties for Combustible Gas Modeling

MU-7.1 A series of re-initiations of the MELCOR Case 6B (see Section 6.3 of Appendix B) were performed to evaluate alternative treatments for combustion of hydrogen and other combustible gases (allowing deflagrations to occur). The sensitivity cases vary the model inputs for different phases of the accident progression. The peak containment pressure is compared to the results of calculations estimating combustion-related loads. An analogous comparison was also performed with re-initiations of Case 1A1.

4.7.3 Sensitivity Analysis for Energetic Burning of Combustibles and Resulting Containment Failure (MU-7.1)

In developing the approach for modeling containment combustion events, there were differences in expert opinion about whether the combustion is most appropriately modeled as a deterministic process (e.g., as MELCOR typically does) or a stochastic process. For this study a stochastic process was used, which is described in Section 8 of Appendix E. This stochastic approach was used to develop parameters for the event tree model to represent containment failures from energetic combustion events, which are important contributors to the 1-REL-ICF-BURN release category frequency. A sensitivity analysis of this approach and comparisons to results that are shown from the MELCOR analysis are described below.

Calculations were performed using the ERPRA-BURN computer code to determine the probability of energetic burning of combustibles and resulting containment failure. The ERPRA-BURN computer code employs an approach consisting of a large number of adiabatic isochoric combustion calculations, integrated within a probabilistic, Monte Carlo framework that permits stochastic multiple combustion events during the severe accident scenario and consideration of uncertainties in various input parameters.

The duration of the accident was broken into three phases: prior to vessel breach (phase I), the time just after (typically within one hour of) vessel breach (phase II), and the period after vessel breach from the start of MCCI until containment failure or specified end time of the calculation (phase III). To complete the ERPRA-BURN calculations within a probabilistic framework, four sets of Monte Carlo combustion histories were simulated. The four sets of histories correspond to four sets of assumptions regarding the availability of an ignition source in the containment:

- An ignition source is available in phases I and III
- An ignition source is available in phases II and III
- An ignition source is available only in phase III
- An ignition source is available in phases I, II, and III

The results of each set are weighted by the net probability of their respective assumptions regarding the availability of an ignition source. The probabilities of existence of an ignition source used for the ERPRA-BURN analysis are: 0.99 for all phases for non-SBO secnarios, 0.1 for phase I of SBO scenarios, 0.5 for phase II of SBO scenarios, and 0.3 for phase III of SBO scenarios. As a result of performing the complete sets of Monte Carlo simulations of combustion history in the containment, in combination with the containment fragility, the net or conditional probabilities of combustion and containment failure in various phases of the accident can be determined.

Several ERPRA-BURN calculations were carried out for the two representative MELCOR scenarios:

- A general transient scenario (MELCOR Case 6B) with long-term containment heat removal available as a result of continued functioning of the containment sprays and fan coolers. Since AC power is available in this scenario, ignition source probabilities of 0.99 for all phases were assumed.
- An SBO (MELCOR Case 1A1) with loss of AFW and ARVs after a short duration. Since AC power is unavailable in this scenario, ignition source probabilities of 0.1, 0.5 and 0.3 were assumed for phases I, II and III, respectively.

Figure 4-26 shows that ERPRA-BURN predicts that the largest containment loads for Case 6B will occur during the period of MCCI when no earlier combustion events have occurred (i.e., “Phase III w/o Prior Burn”). In this ERPRA-BURN scenario, containment loads prior to vessel breach and during vessel breach (i.e., phases I and II) are negligible. However, the higher end of the loads due to combustion during phase III can challenge the containment integrity. The nominal ERPRA-BURN calculations predict a combined containment failure probability of 0.06 given ignition for Case 6B in phase III. The low probability of containment failure is due to the low pressure in the containment during phase III and the contribution of the fan coolers and sprays. The minimum and maximum containment pressure resulting from these calculations is given in **Table 4-20** below.

Table 4-20: Maximum and Minimum Pressures Taken from Figure 4-26

	ERPRA-BURN	
	MIN [bar]	**MAX [bar]**
Phase I	1.1	6.6
Phase II	1.9	6.8
Phase III	5.1	13.1

In Case 6 (see Section 6 of Appendix B), where burns are allowed, hydrogen concentration becomes quite high. The hydrogen concentration versus containment pressure at various MELCOR simulation times is super-imposed onto the SAMG computational aid for "Hydrogen Flammability in Containment" in **Figure 4-27** below. During phase III of the simulation the "Hydrogen Severe Challenge" region of this figure is entered. However, the concentration of oxygen is very low at this point due to the early burning (note the blue color indicating non-flammable conditions at that time). **Figure 4-28** shows a very high hydrogen concentration in the dome of containment but low oxygen concentration due to early burns.

In Case 6B, where combustion is suppressed, **Figure 4-29** and **Figure 4-30** show that conditions enter the "Hydrogen Severe Challenge" region but oxygen levels remain elevated. A burn during phase III would likely result in containment failure.

To demonstrate the likelihood of containment failure, MELCOR Case 6B was restarted at various times (see **Table 4-21**) with deflagrations enabled. The maximum pressure seen in the dome of containment is listed in **Table 4-21**. A comparison of these values to the cumulative probabilites predicted by ERPRA-BURN is given in the last column. Since this is comparing a point estimate to a somewhat normal probability density function (PDF) shape, the closer the corresponding cummulative probability is to 0.5, the better the agreement (i.e., the point estimate is close to the mean as well as the median).

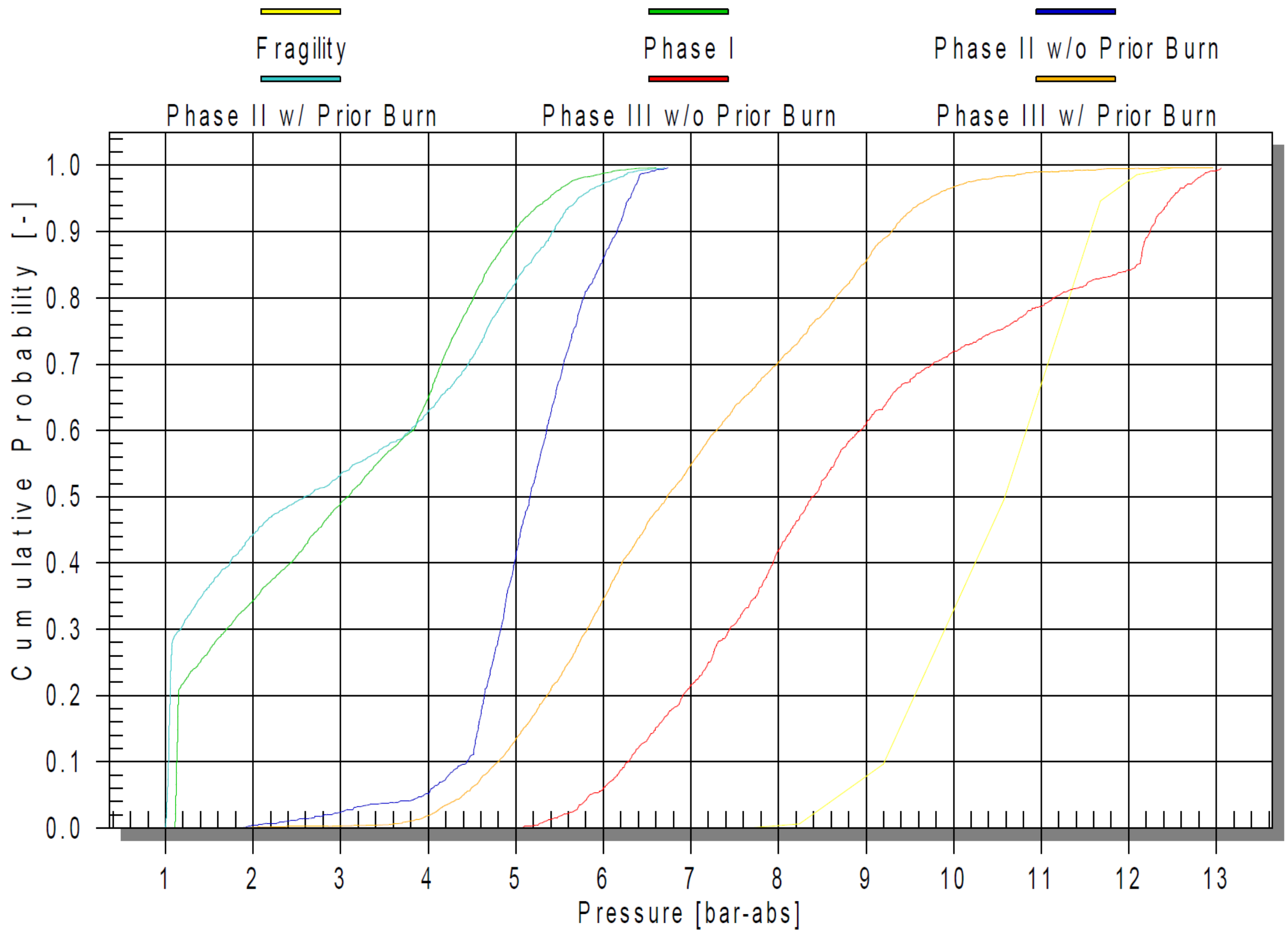


Figure 4-26: ERPRA-BURN peak global deflagration pressure loads for Case 6B

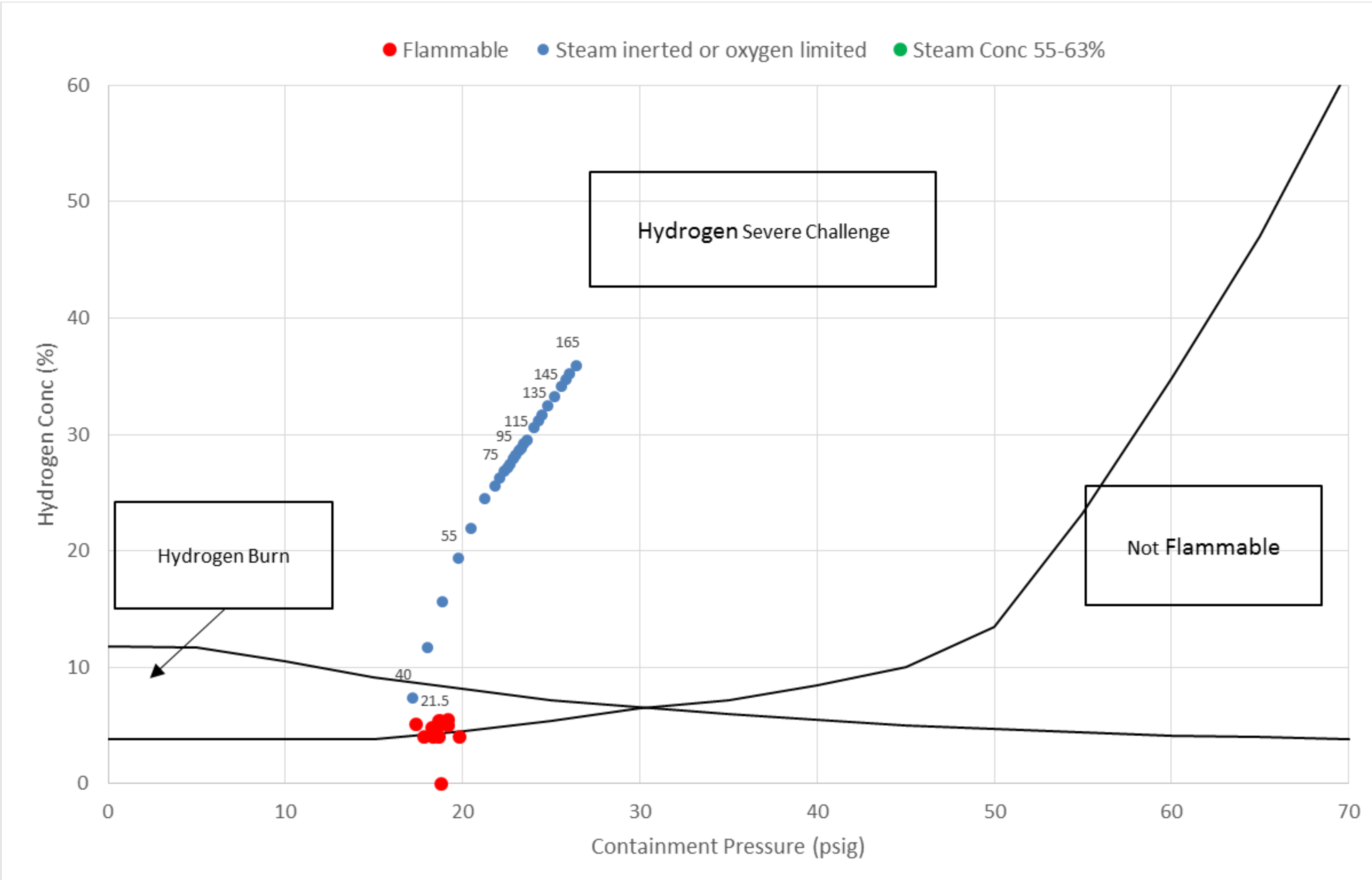


Figure 4-27: Containment Conditions for Case 6 Superimposed onto the SAMG Computational Aid

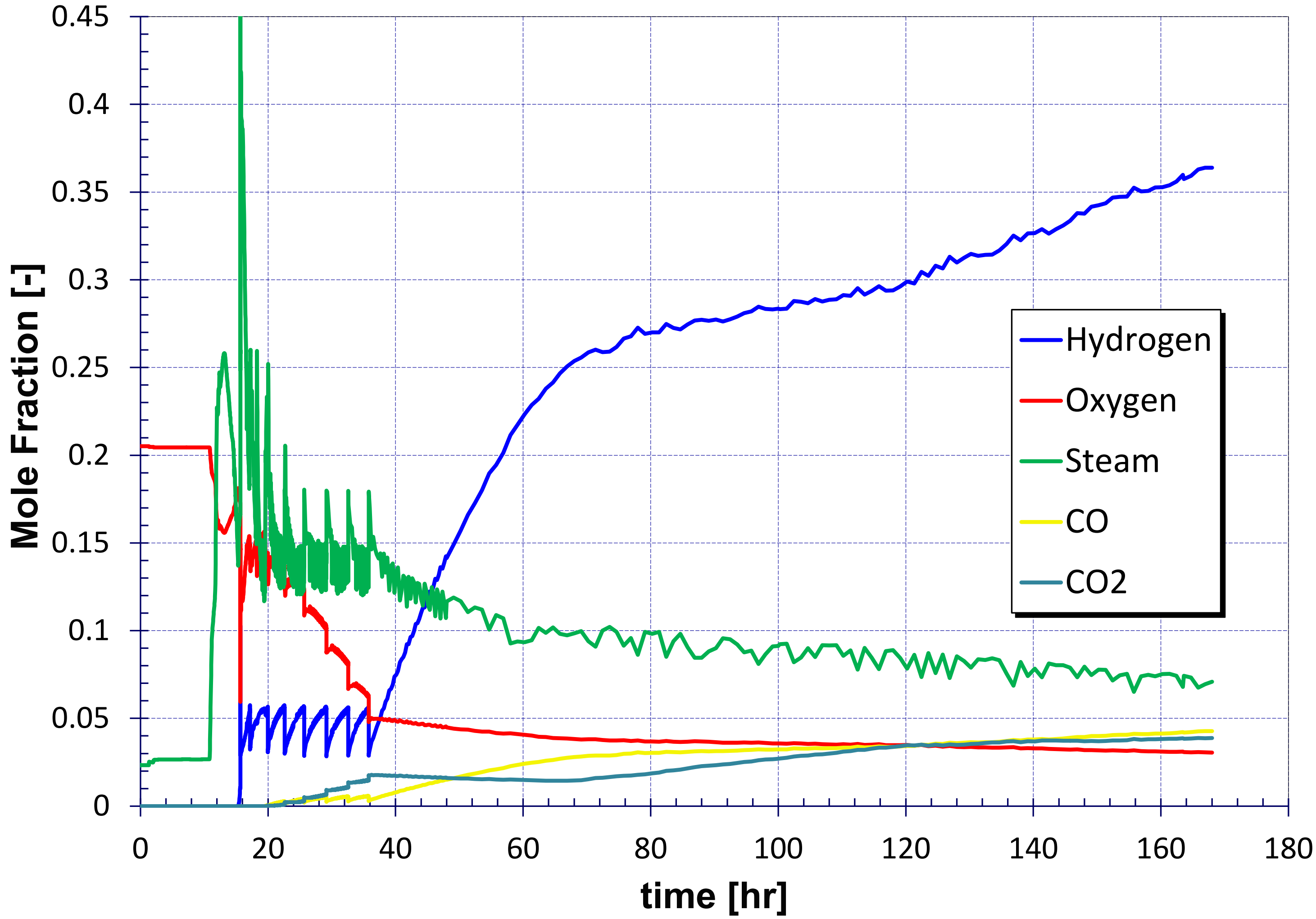


Figure 4-28: Concentrations Within the Containment Dome for Case 6

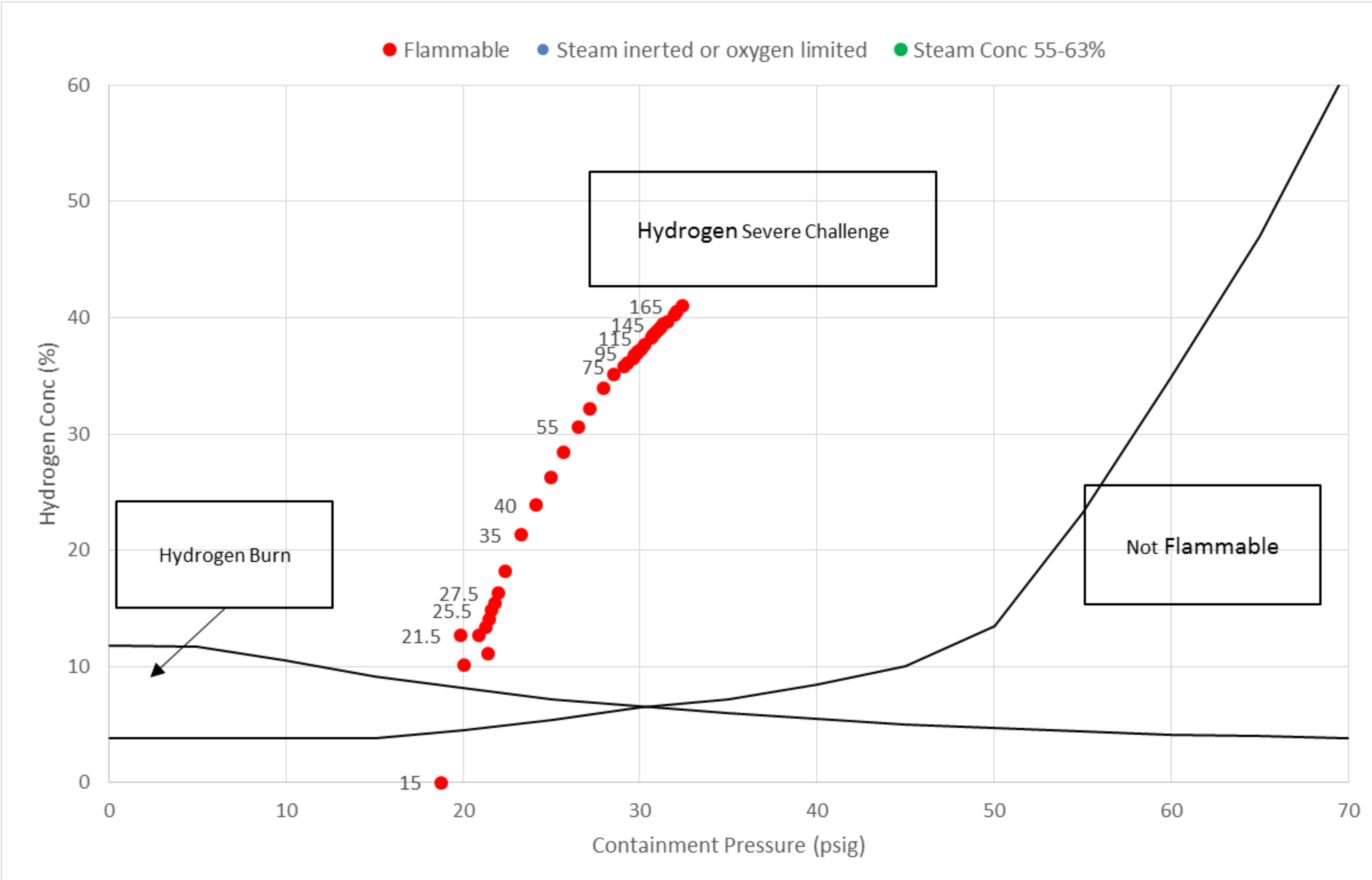


Figure 4-29: Containment Conditions in Case 6B Superimposed onto the SAMG Computational Aid

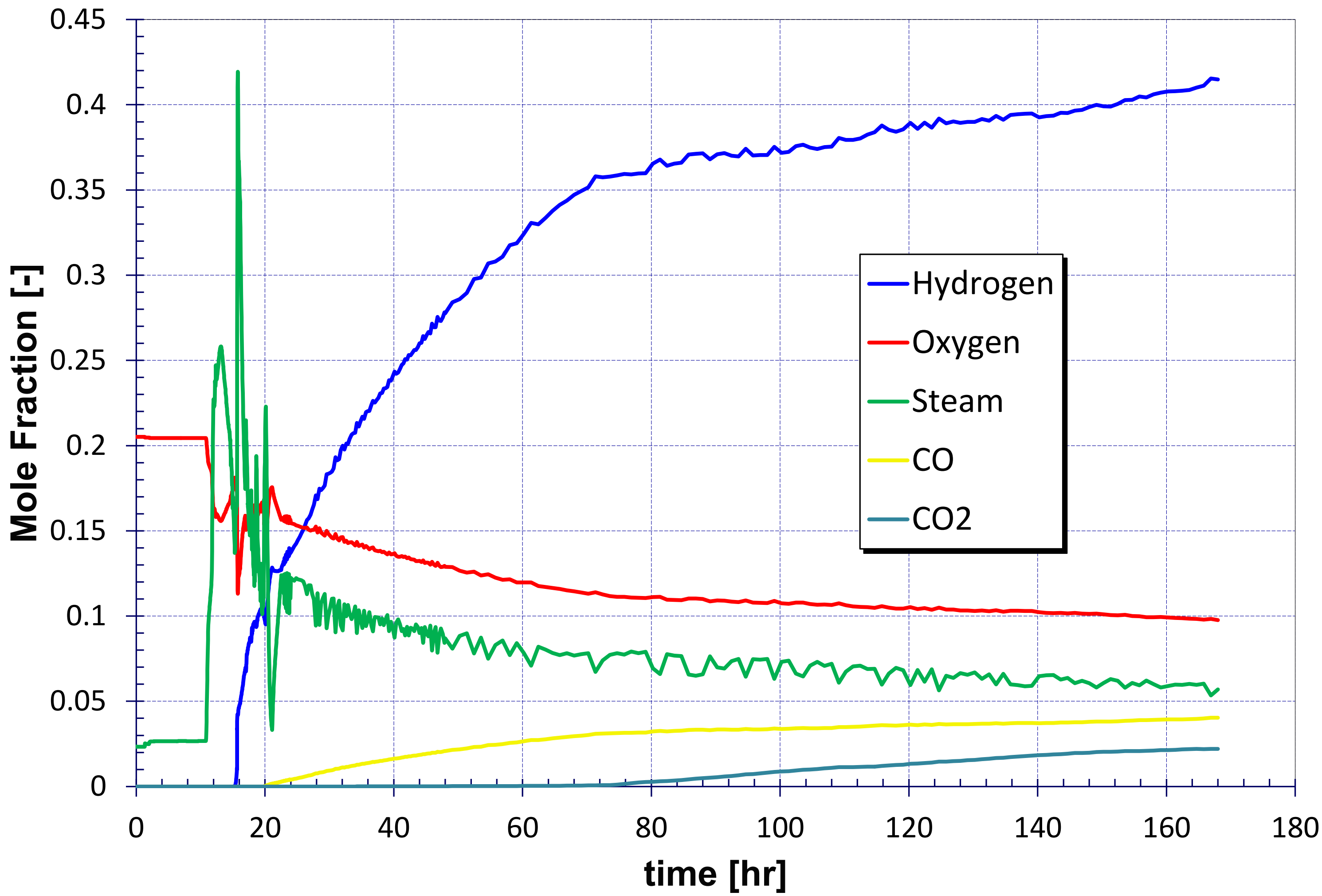


Figure 4-30: Concentrations within the containment dome for Case 6B

Table 4-21: Maximum containment loads by deflegration time

Phase	Time [hr]	MELCOR Max Pressure [bar]	Corresponding ERPRA-BURN Cumulative Probability
I	17.0	3.6	0.56
I	18.0	4.1	0.66
I	19.0	4.3	0.71
II	19.8	4.2	0.07
III	40.0	11.0	0.79
III	60.0	12.3	0.90
III	100.0	12.8	0.96

The maximum containment loads in this sensitivity lie well within the upper and lower limits predicted by ERPRA-BURN. The pressure spike experienced at vessel breach is relatively small, and this is consistent with the ERPRA-BURN results.

Qualitatively, the results of ERPRA-BURN for this case are supported by these results. Namely, in phase I and II, a hydrogen combustion large enough to overpressurize and fail containment is unlikely. For phase III, a hydrogen combustion is likely, and without prior burns, could pressurize containment on the upper end of what was predicted by ERPRA-BURN and could severely challenge containment.

For completeness, a similar analysis is conducted here for Cases 1A and 1A1 (see Section 1.1 of Appendix B). However, in this case, containment fan coolers are not running, and a high concentration of steam exists in containment due to hot leg nozzle creep rupture occurring prior to lower head failure. Sensitivity cases were performed to determine the impact of several input parameters. The following sensitivity cases were considered for the SBO scenario (Case 1A1):

- The ignition source probabilities of 0.99 for all phases (i.e., transforming the scenario into a transient with AC power but no containment heat removal)
- Changing the relative humidity in the containment for phase III from 0.3 (based on the MELCOR results) to 0.5
- Using a burn efficiency of 0.9 instead of 1.0 to account for the impact of heat losses to containment structures

Figure 4-31 shows that the largest containment loads are found to occur during the period of MCCI (i.e., phase III) for the base ERPRA-BURN 1A1 scenario. In this ERPRA-BURN scenario, containment loads prior to vessel breach and during vessel breach (i.e., phases I and II) are negligible. However, the loads during phase III can challenge the containment fragility curve and therefore, the containment integrity. ERPRA-BURN predicts a net containment failure probability of 0.21 for Case 1A1 in phase III. The sensitivity to the ignition source probability results in a slightly higher combined containment failure probability of 0.26, indicating that the dominant factor is high steam concentration rather than frequency of ignition source. Changing the relative humidity in phase III has a marginal impact on the containment failure probability for S1A1, whereas decreasing the burn efficiency to account for heat losses to structures reduces the combined containment failure probability slightly to 0.18. The minimum and maximum containment pressure resulting from these calculations is given in **Table 4-22**.

Table 4-22: Maximum and minimum pressures taken from Figure 4-31

	ERPRA-BURN	
	MIN [bar]	MAX [bar]
Phase I	1.1	5.5
Phase II	2.5	6.5
Phase III	4.3	14.0

According to **Figure 4-32** and **Figure 4-33**, concentrations in the containment dome are never conducive to a spontaneous combustion. The default parameters in MELCOR for spontaneous combustion are:

- less than 55% inertant concentration,
- greater than 10% hydrogen concentration, and
- greater than 5% oxygen concentration.

The relatively high steam concentration is a determining factor suggesting low likelihood of combustion. Similar results are obtained for Case 1A1, as shown in **Figure 4-34** and **Figure 4-35**. and **Table 4-23**.

Table 4-23: Maximum Containment Loads by Deflegration Time

Phase	Time [hr]	Max Pressure [bar]	Corresponding ERPRA-BURN Cumulative Probability
I	18.1	3.1	0.72
I	19.0	3.1	0.72
II	21.1	3.2	0.36
III	22.0	5.1	0.01
III	24.0	3.5	< 0.01
III	28.0	4.0	< 0.01
III	32.0	4.4	< 0.01

The maximum pressures seen in the containment dome are listed in **Table 4-23**. The phase I and II predicted maximum pressures show reasonable agreement with the ERPRA-BURN results. For phase III, the ERPRA-BURN results show containment pressure loads that can challenge containment integrity; however, the predicted concentrations for the Case 1A and Case 1A1 sensitivities suggest combustion is unlikely.

In total, these sensitivity studies demonstrate that:

- Situations with containment heat removal are susceptible to having combustible gas mixtures in phase III that are capable of failing containment, with oxygen starvation from prior burns being a key determinant of the likelihood
- Situations without containment heat removal (i.e., conditions in SBO scenarios) are susceptible to having combustible gas mixtures in phase III that are capable of failing containment, with steam inerting being a key determinant of the likelihood
- Ignition source presence and hot leg nozzle creep rupture have important influences on the above

It can be further inferred from these sensitivity analyses that the ERPRA-BURN results are appropriate and credible for modeling combustion events that can result in containment failure, but the conditional probabilities they generate are particularly sensitive to modeling uncertainty.

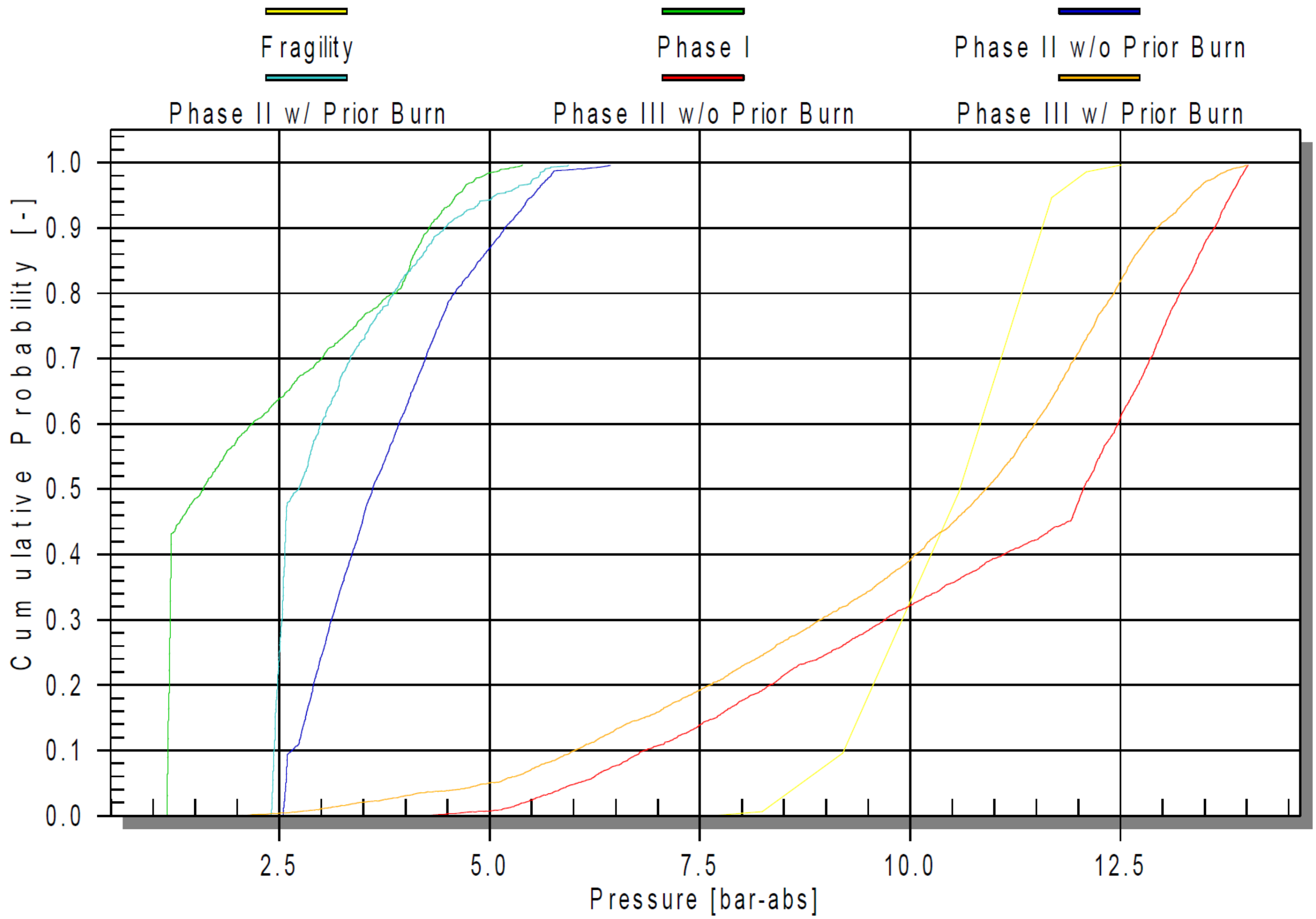


Figure 4-31: ERPRA-BURN peak global deflagration pressure loads for Case 1A1

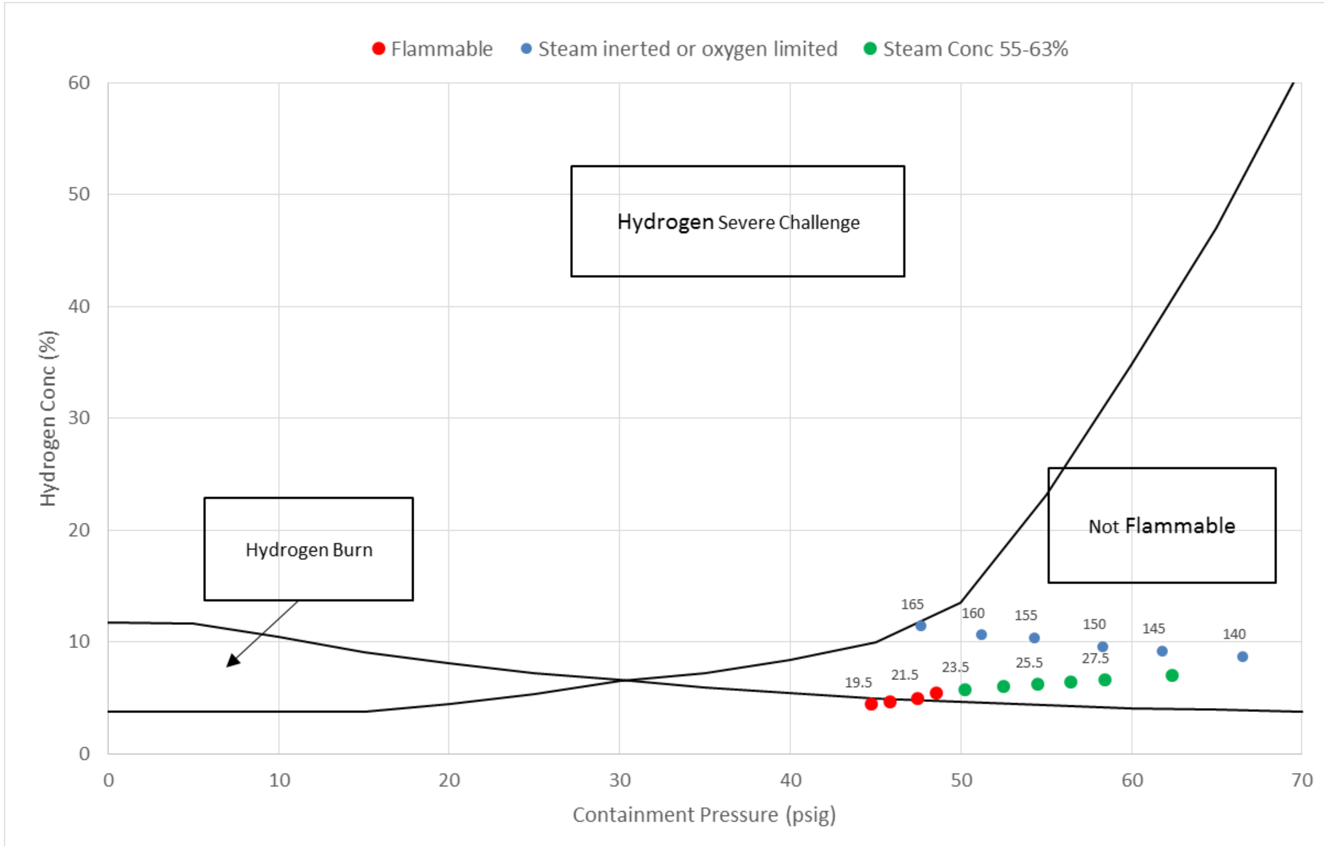


Figure 4-32: Containment Conditions in Case 1A Superimposed onto the SAMG Computational Aid

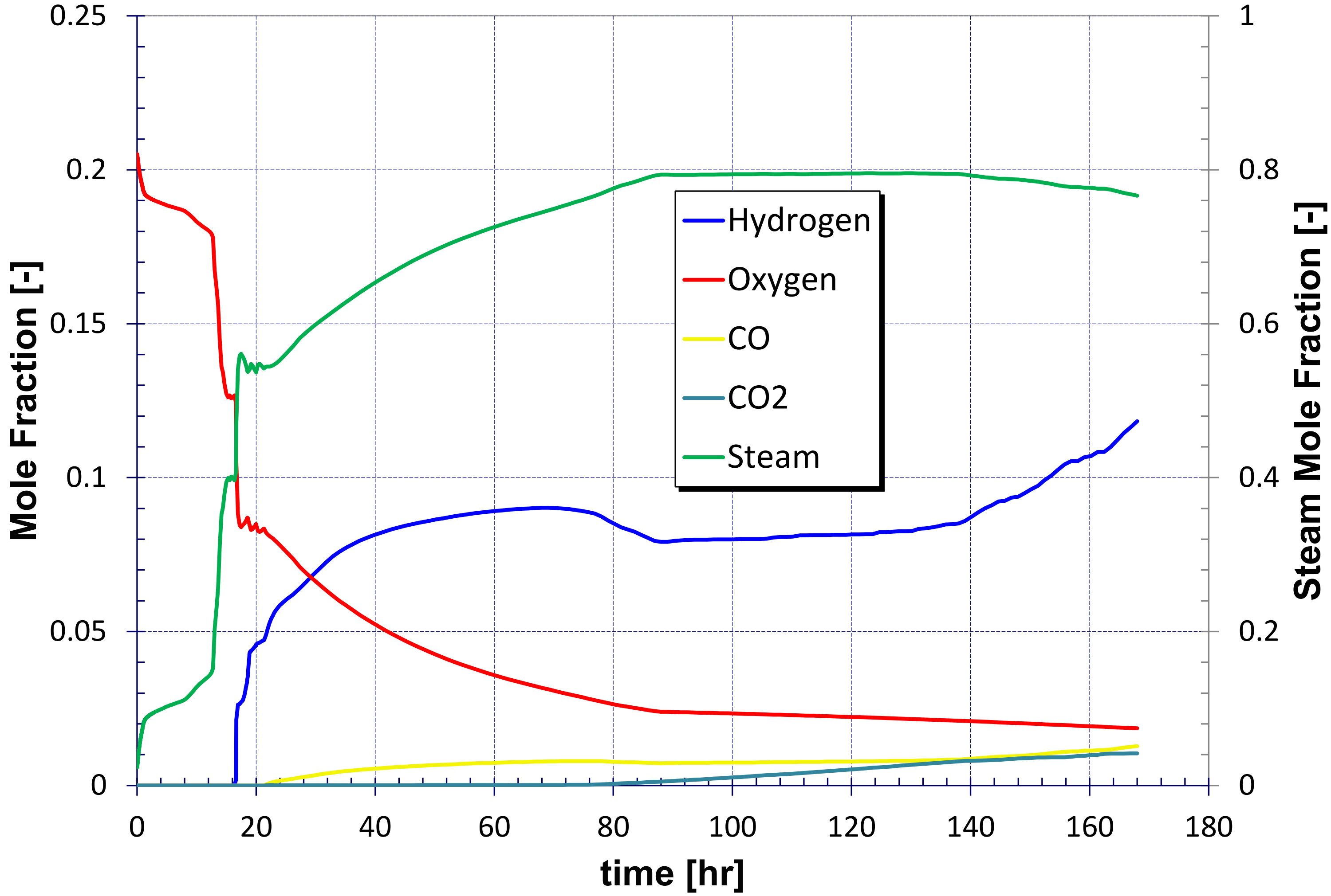


Figure 4-33: Concentrations within the Containment Dome for Case 1A

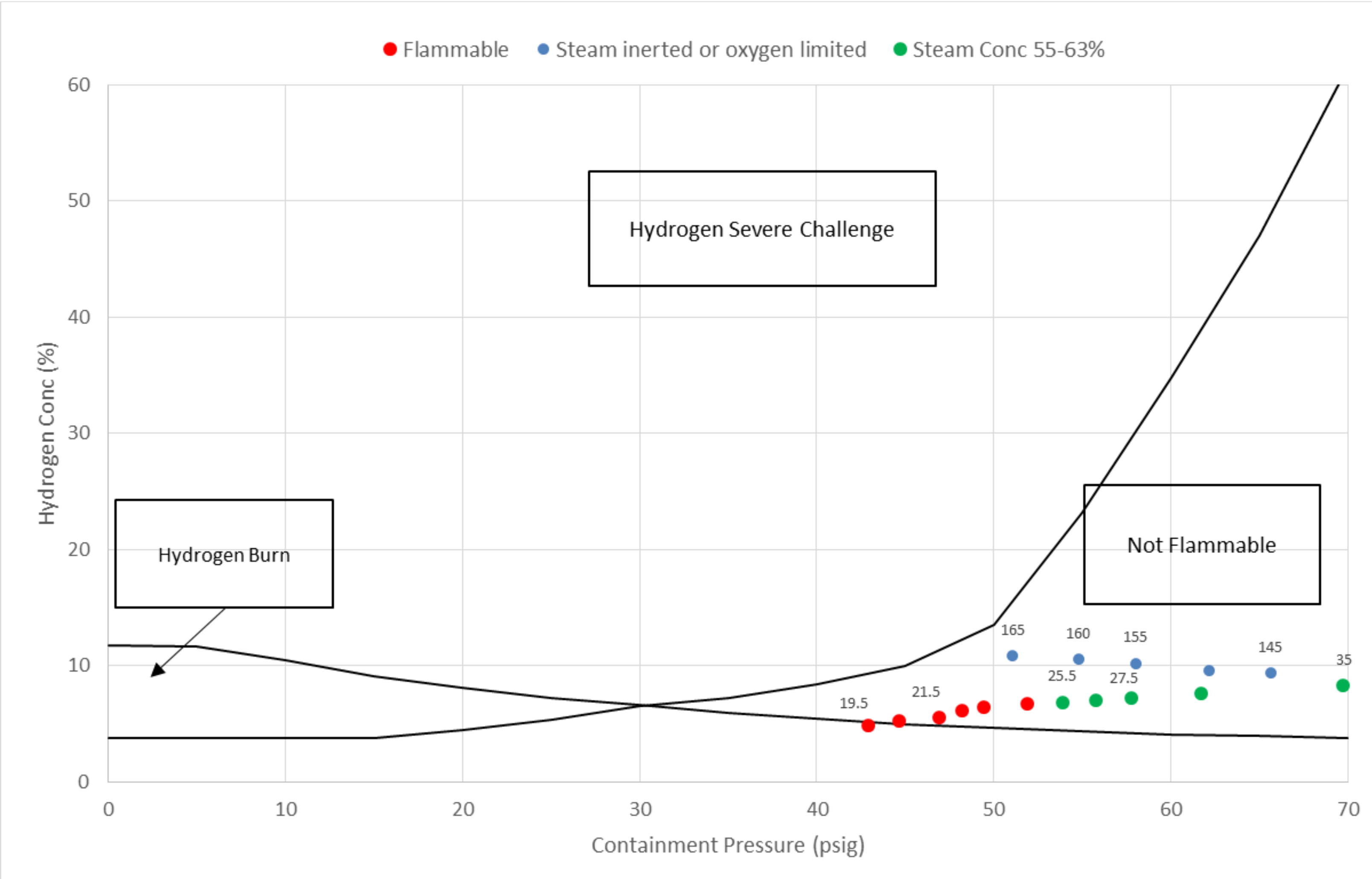


Figure 4-34: Containment Conditions for Case 1A1 Superimposed onto the SAMG Computational Aid

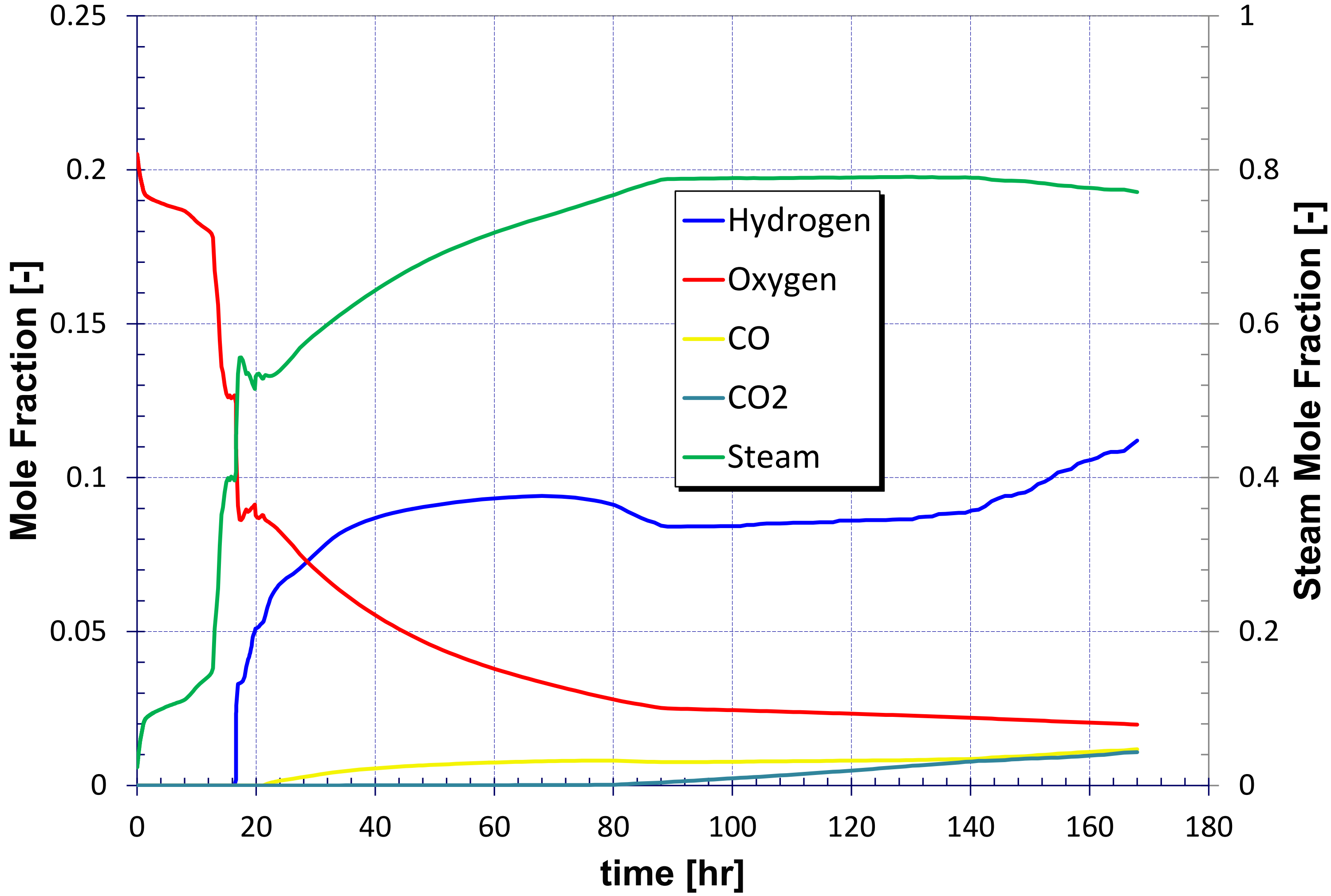


Figure 4-35: Concentrations within the Containment Dome for Case 1A1

4.8 Long-Term Containment Pressurization and Failure Modeling

4.8.1 Identified Uncertainties in Long-Term Containment Pressurization and Failure Modeling

Table 4-24: Uncertainties in Long-Term Containment Pressurization and Failure Modeling

Item	Description	Other comments
Primary containment structural vulnerabilities	Unique containment failure modes/effects can lead to different release characteristics. The intent of the identified uncertainty is that plant-specific analysis be performed and that different probable failure modes/effects be considered (e.g., penetration failure; temperature-induced seal degradation).	These modes were identified and characterized in this study, but the accident progression calculations were not repeated with alternate failure locations (except for the sensitivity studies herein).
"Normal" containment leakage	Normal (i.e., design-basis) containment leakage can vary between zero and 0.2 wt%/day (larger leakage would constitute an isolation failure). The maximum allowable design-basis leakage is used in the base MELCOR model (0.2 wt%/day), whereas actual values will vary (e.g., a 2010 integrated leak rate test on Unit 2 estimated a leakage rate of 0.07 wt%/day).	A lower leakage rate will generally act to reduce fission product leakage prior to containment failure. However, it will also serve to increase the rate of containment pressurization, and thereby potentially lead to an earlier over-pressurization failure (though competing phenomenological effects may affect the linearity of this relationship).
Containment failure location	The containment capacity analysis identified possible candidates for failure locations due to long-term pressurization. The containment wall-basemat junction and equipment hatch locations were considered equally likely. The containment wall-basemat junction was selected for modeling the containment release path.	Though the selection of containment failure location can influence the predicted radiological releases to the environment, many other aspects of the accident sequence modeling (e.g., availability of containment sprays and dynamic interactions with other connected volumes) can also influence the predicted releases. MU-8.1 considers alternative modeling aspects of the containment overpressure release path at the wall-basemat junction. As part of the L3PRA reactor-at-shutdown Level 2 PRA, alternate containment failure locations were modeled for both the wall-basemat junction and near the equipment hatch.

Table 4-24: Uncertainties in Long-Term Containment Pressurization and Failure Modeling

Item	Description	Other comments
Containment failure modes given quasi-static loads	Unique containment failure modes/effects can lead to different release characteristics. Note that the effect of different failure locations (e.g., equipment hatch versus wall-basemat junction) related to secondary building attenuation is contemplated in a source term uncertainty entry and also discussed in Section 20 of Appendix D.	The intent of the identified uncertainty is that a variety of approaches can be used and the outcomes are important. The draft Surry SOARCA UA (SNL, 2016a) identifies fragility curve, "design" leakage (before break) path length, and containment convection heat transfer multiplier as uncertain inputs.
Quasi-static failure threshold methods and correlation between failure pressure and leak rate	The state-of-practice is not mature for defining leak-before-break containment response for concrete containments.	This is discussed further in Section 6 of Appendix D. The reader is also referred to the analyses done in the draft Surry SOARCA UA (SNL, 2016a).
Plugging and decontamination in containment leakage paths by aerosols	This can lead to temporary decrease in containment leakage area and associated pressure rises, potentially leading to opening of leakages elsewhere, and/or decontamination in the flow paths.	Discussed in Appendix C of (EPRI, 2012a), which provides a simple equation that estimates the amount of aerosol material that must pass through a flow path before it is plugged ($m = K^*d^3$). A 2009 NEA State-of-the-art Report (SOAR) (NEA, 2009) states that experimental studies have demonstrated decontamination factors (DFs) in the range of 10 to 100, and provides the same correlation for plugging as in (EPRI, 2012a). It also states that "existing models are not mature enough and a sound, reliable and representative database against which to validate them is still missing," and references ongoing (at the time) work by Institut de radioprotection et de sûreté nucléaire (IRSN) and French Alternative Energies and Atomic Energy Commission.

4.8.2 Alternative Treatment(s) of Uncertainties in Long-Term Containment Pressurization and Failure Modeling

MU-8.1 A MELCOR simulation was performed (based upon Case 1B2 [see Section 1.2.2 of Appendix B]) where the containment overpressure failure mode/location was modeled to go through the tendon gallery and completely to the auxiliary building. This will show the effect of a different postulated failure on the environmental releases. This sensitivity differs from the base case in that none of the release path makes its way directly to the environment but blows down from the tendon gallery directly to the auxiliary building. The impact of the auxiliary building in minimizing the environmental release was demonstrated by a comparison of this sensitivity and base case source terms.

MU-8.2 A MELCOR simulation was performed (based upon Case 1B2) where the normal containment leakage was lowered to ~0.07 wt%/day to show the effect of having leakage that is notably below the maximum allowable design-basis leakage value, and in particular the effect on the time of containment long-term over-pressurization failure.

MU-8.3 A revisitation of the source term in MELCOR simulation 3A3 (see Section 3.3 of Appendix B) was performed in which a decontamination factor for aerosol plugging/turbulent deposition is applied to the fission products passing through the tendon gallery during containment failure.

4.8.3 Sensitivity Analysis for Containment Over-Pressurization without Tendon Gallery to Environment Pathway (MU-8.1)

Containment over-pressurization is modeled in the L3PRA project MELCOR deck with a pathway at the containment basemat junction opening to the tendon gallery when the pressure limit is reached. From there, two pathways open, one to the environment with 2/3 flow area and the other to the auxiliary building with the remaining 1/3 flow area. Contrary to expectation, reverse flow occurs in the path from the tendon gallery to the auxiliary building (discussed in Section 20 of Appendix D). In other words, at the time of containment over-pressurization, there is not an increase in fission products entering the auxiliary building via the tendon gallery, but rather all fission products entering the tendon gallery are either deposited there or go directly to the environment.

For this sensitivity, a MELCOR simulation based upon Case 1B2 was constructed in which the path from the tendon gallery to the environment was effectively removed, forcing the tendon gallery to pressurize and any airborne fission products entering the tendon gallery to make their way to the auxiliary building. The containment failure pressure set-point was reached at 54.9 hours and flow opens up to the tendon gallery, and from there, the auxiliary building (**Figure** 4-36).

At 66.9 hours in the sensitivity case, the auxiliary building reaches the assumed failure pressure of 1.1 bar-abs. At this point, a 1 m^2 path with no fan or filter opens from the auxiliary building to the environment due to the auxiliary building failure. A sizeable deflagration also occurs at this time and the resulting pressure surge causes a step increase in fission products to the environment, as seen in **Figure** 4-37. The two subsequent step increases at around 81 and 109 hours are also due to large deflagrations in the auxiliary building, as seen from the cumulative hydrogen burned in **Figure** 4-38.

Table 4-25 gives the fractional retentions in the auxiliary building and releases to the environment. With the removal of the direct pathway to the environment via the tendon gallery, there is a significant decrease in the environmental release fraction. However, care should be taken when directly comparing the retentions and releases in the sensitivity case to the base case. The rate of release of fission products to containment is not the same as in the base case due to threshold effects around the time of vessel breach. Small perturbations in the calculations lead to accumulators injecting sooner and the lower head failing later in this case. This has the effect of delaying the release of fission products from the RCS (**Figure 4-39**) and causing an increased retention there. Also, cesium is re-volatilized from the dried-out PRT at a faster rate (**Figure 4-40**) due to an increased PRT heat structure temperature in this case. Because of this, at the time of containment failure, fission products in containment differ between the base and sensitivity cases (**Figure 4-41**). Nevertheless, significant auxiliary building retentions are seen in the sensitivity case, but limited by subsequent building damage.

Table 4-25: Releases and Retentions for MU-8.1

Representative Element	Auxiliary Bldg		Environment	
	MU-8.1	**1B2**	**MU-8.1**	**1B2**
Xe	7.5E-2	6.7E-4	8.2E-1	8.9E-1
Cs	2.5E-2	9.6E-3	8.0E-4	4.3E-2
I	3.8E-2	1.1E-2	2.6E-3	8.2E-3
Te	1.0E-2	1.0E-2	2.7E-4	4.2E-3
Ba	3.3E-4	1.7E-4	1.1E-5	2.6E-4
Ru	1.8E-5	1.5E-5	1.5E-7	4.0E-7
Mo	5.3E-2	1.7E-2	1.1E-3	4.5E-2
Ce	7.8E-7	3.4E-7	2.0E-8	3.8E-7
La	4.8E-7	8.6E-8	1.7E-8	3.8E-7
UO2	6.3E-5	7.6E-6	2.4E-6	5.9E-5
Cd	3.9E-2	1.1E-2	1.5E-3	3.5E-2
Ag	3.7E-2	1.3E-2	9.4E-4	4.3E-2

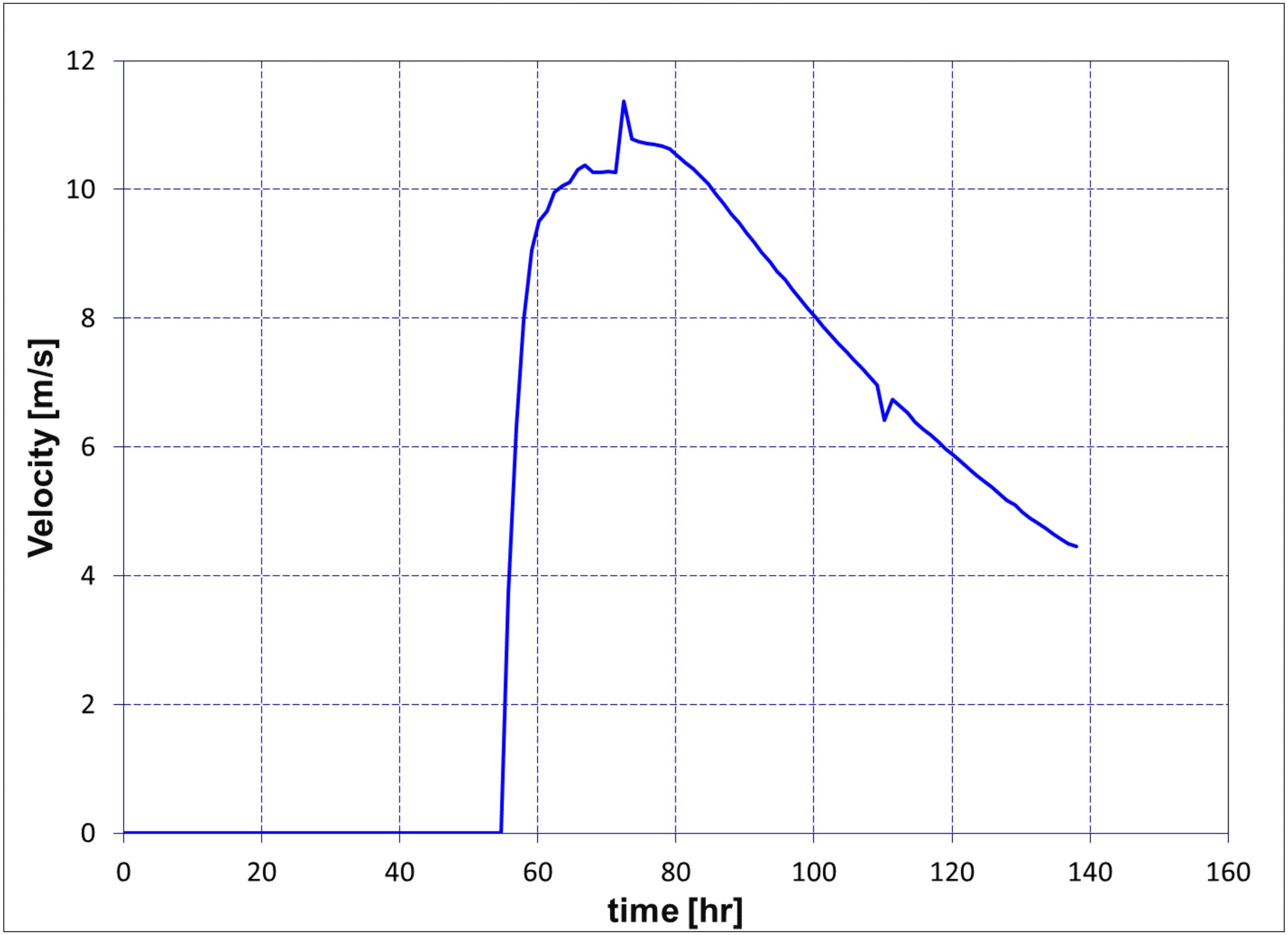


Figure 4-36: Flow Rate from the Tendon Gallery to the Auxiliary Building

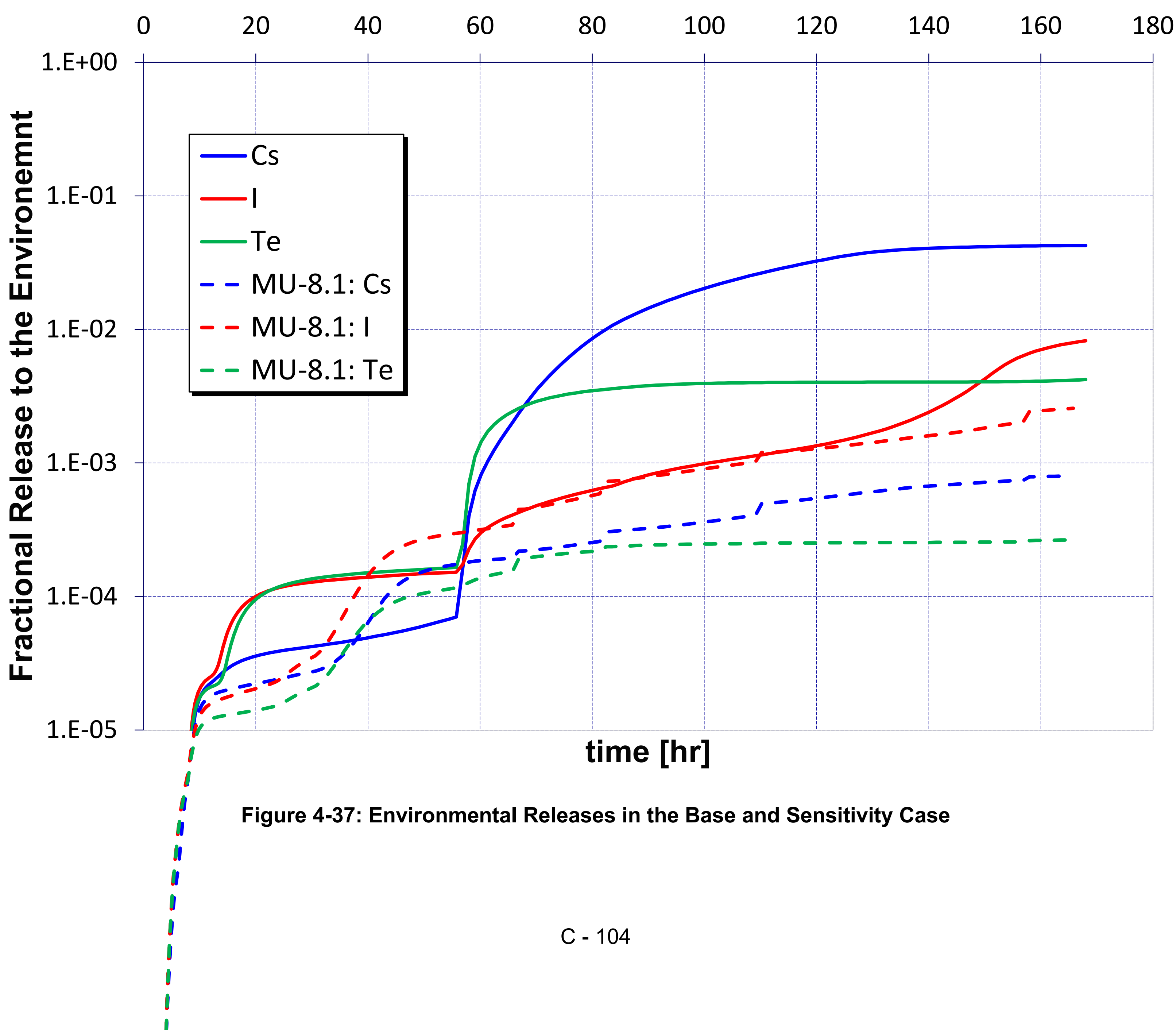


Figure 4-37: Environmental Releases in the Base and Sensitivity Case

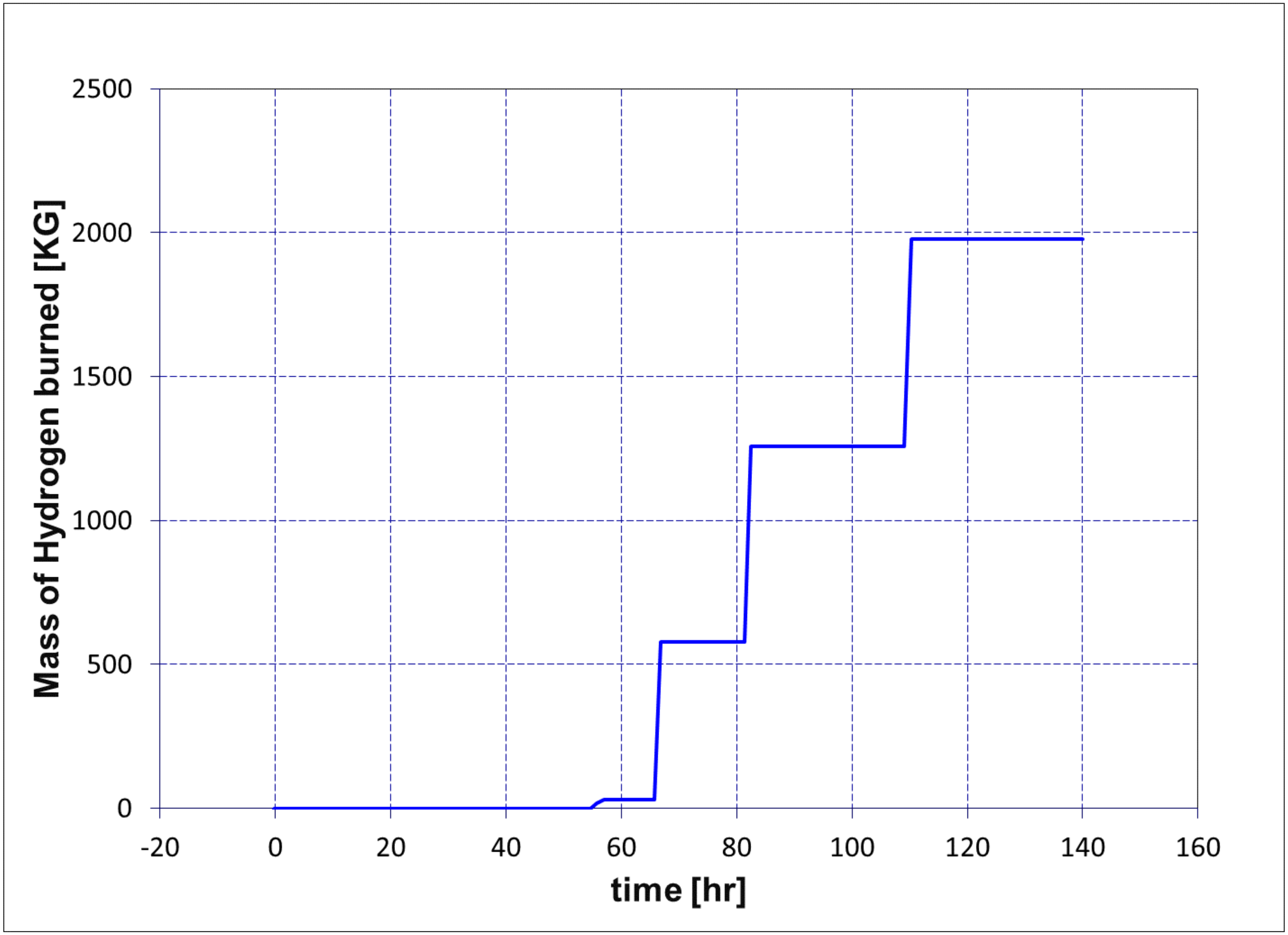


Figure 4-38: Mass of Hydrogen Burned in the Auxiliary Building for Sensitivity MU-8.1

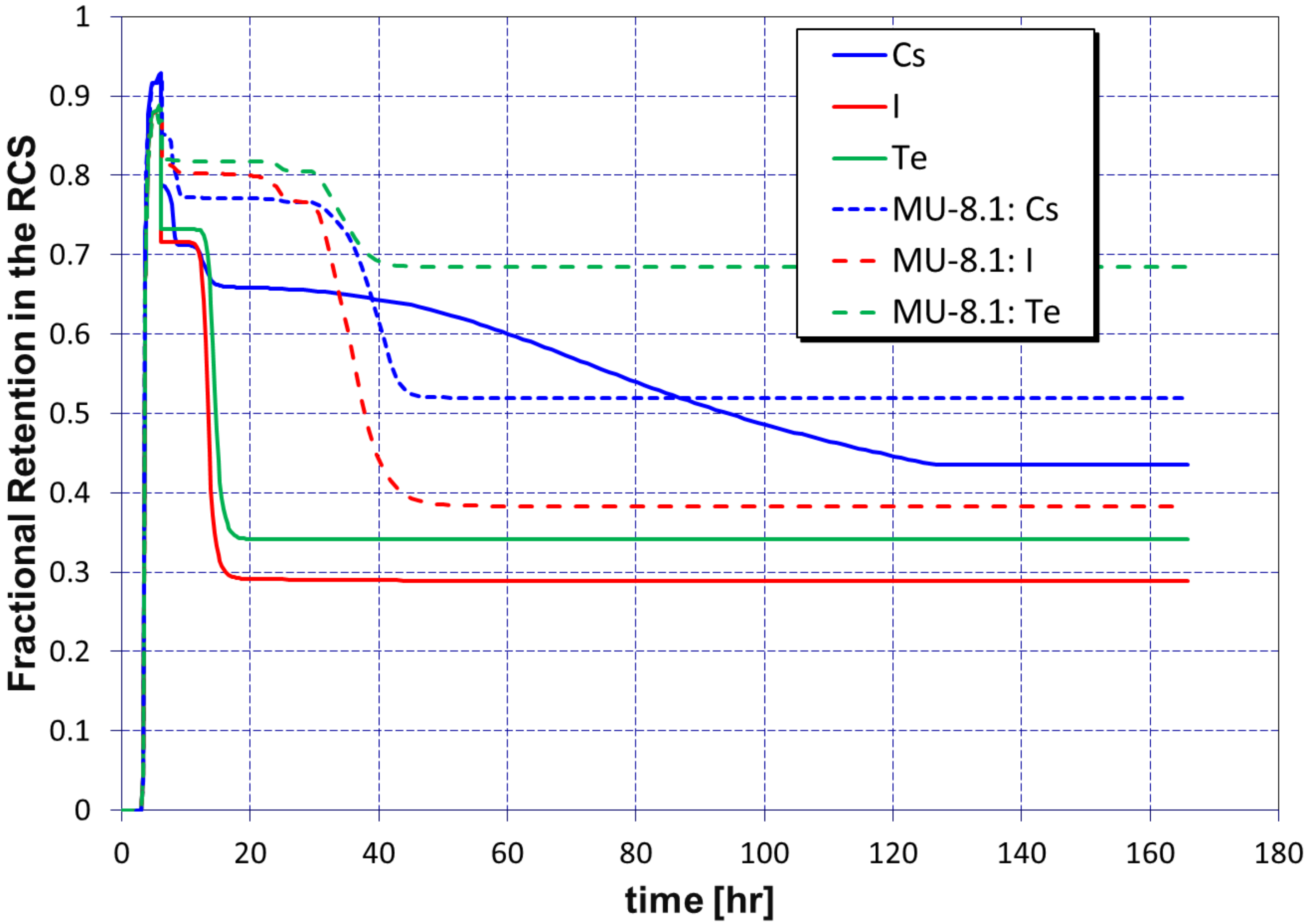


Figure 4-39: Retentions in the RCS for the Base and Sensitivity Case

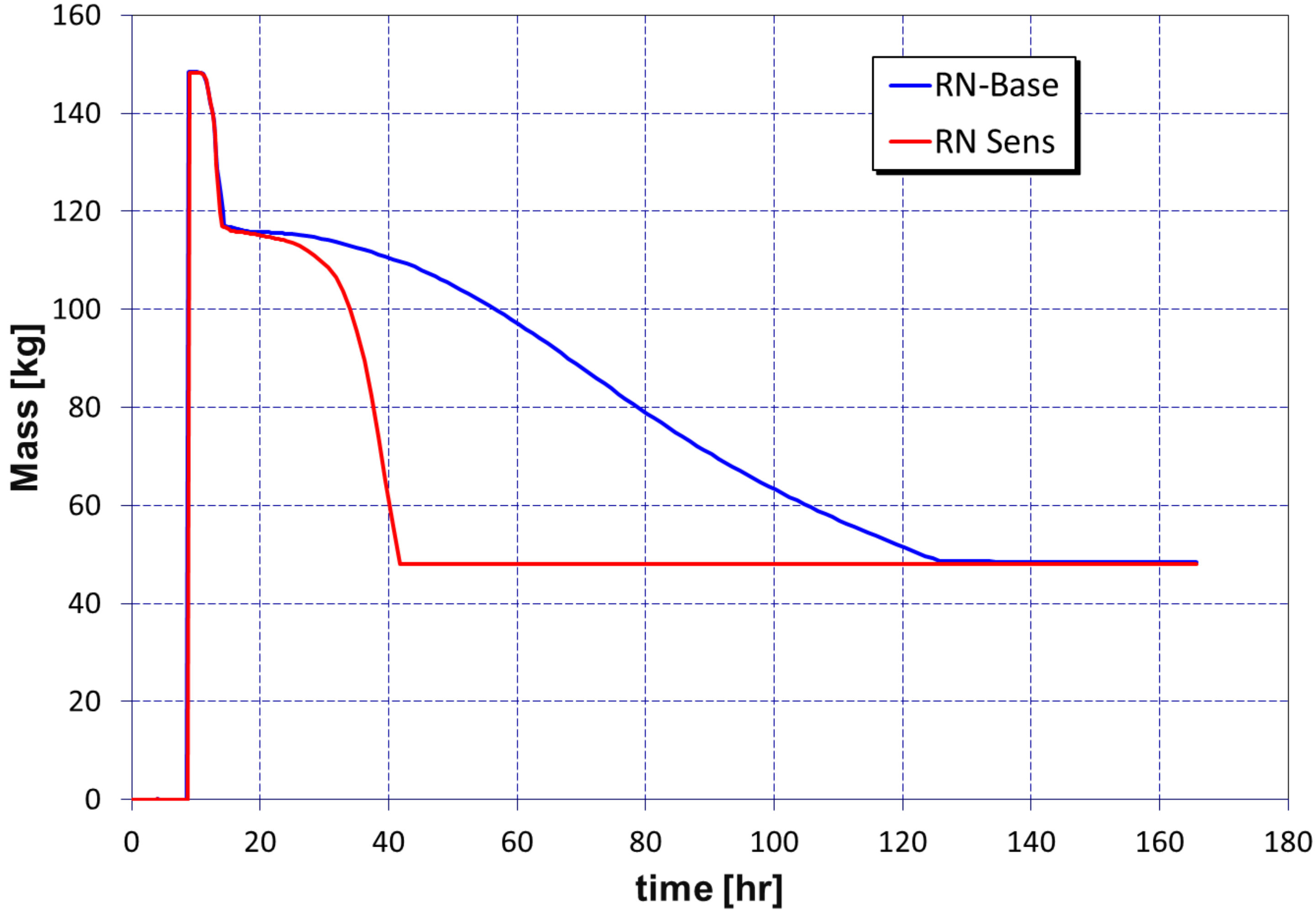


Figure 4-40: Radionuclides Deposited on the PRT Heat Structure for MU-8.1

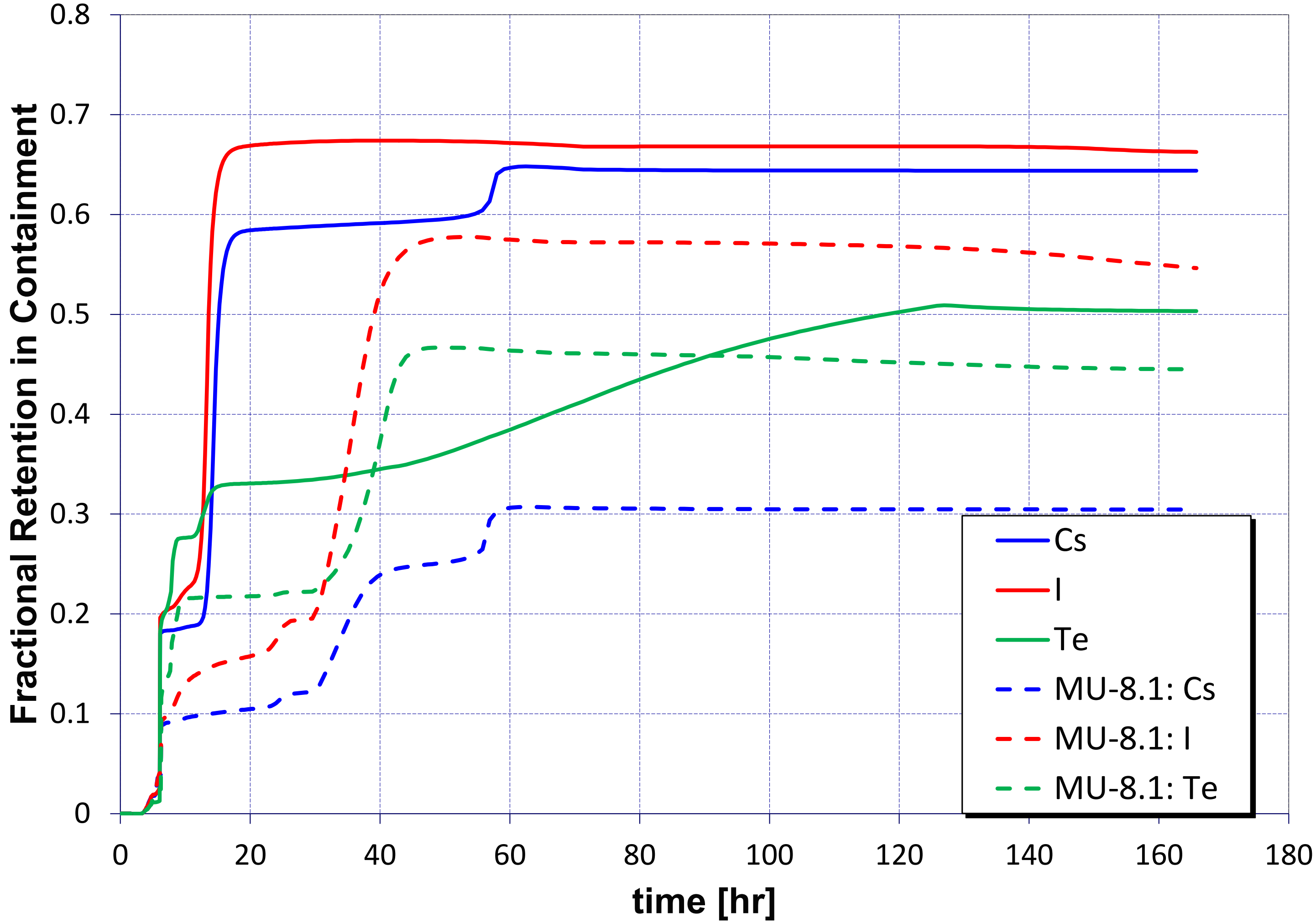


Figure 4-41: Fractional Retentions in the Containment for the Base and Sensitivity Case

4.8.4 Sensitivity Analysis on the Effect of Leak Rate on the Rate of Pressurization and the Timing of Over-Pressurization (MU-8.2)

In the L3PRA project MELCOR deck, there are five flowpaths that represent leakage out of the containment to either the environment or the auxiliary building. The flow area of these flowpaths was originally adjusted until a leak rate of about 0.2% air mass/day was seen. For this sensitivity, based upon Case 1B2, a reduced leak rate of 0.07%/day was used instead. As a coarse realization of this, the area of each of the five referenced flowpaths was adjusted by a fraction of 0.07/0.2=0.35.

The decreased flow rate of the containment leakage has little impact on the rate of containment pressurization (**Figure 4-42**). The time of failure for the MU-8.2 sensitivity is 55.8 hours versus 56.2 hours in the 1B2 Base Case. However, small changes in the containment atmosphere pressure (**Figure 4-43**) affect the pressure within the RCS around the time of fuel failure (**Figure 4-44**). This has a trickle-down effect on the timing of accumulator injection and vessel breach. In the sensitivity, accumulators inject sooner, and lower head failure occurs later (at 6.6 versus 6.1 hours). This has the effect of delaying the release of fission products from the RCS (**Figure 4-45**) and causing an increased retention there. The environmental release (**Figure 4-46**) in Case MU-8.2 is slightly greater than the base case. The exception is cesium which is re-volatilized from the dried-out PRT at a greater rate (**Figure 4-47**) due to an increased PRT heat structure temperature in the case of MU-8.1.

Because of the differences in RCS retentions from event timings, care should be taken when directly comparing the retentions and releases in the sensitivity to the base case given in **Error! Reference source not found.**.

This sensitivity demonstrates that the choice in leak rate has little effect on the rate of pressurization and the timing of over-pressurization. It also demonstrates the sensitivity of key event timings such as accumulator injection and vessel breach to changes in the RCS and containment pressures.

Table 4-26: Containment Retentions and Environmental Releases for MU-8.2

Representative Element	Containment		Environment	
	MU-8.2	1B2	MU-8.2	1B2
Xe	1.9E-1	1.1E-1	9.0E-1	8.9E-1
Cs	3.8E-1	5.0E-1	1.6E-2	4.3E-2
I	4.6E-1	6.6E-1	2.3E-2	8.2E-3
Te	3.0E-1	6.4E-1	5.1E-3	4.2E-3
Ba	4.6E-3	1.1E-2	2.5E-4	2.6E-4
Ru	2.7E-3	2.8E-3	2.6E-6	4.0E-7
Mo	8.0E-1	8.2E-1	3.8E-2	4.5E-2
Ce	1.8E-5	1.7E-5	4.4E-7	3.8E-7
La	5.2E-6	5.5E-6	4.0E-7	3.8E-7
UO2	5.6E-4	6.4E-4	5.6E-5	5.9E-5
Cd	4.6E-1	6.2E-1	3.2E-2	3.5E-2
Ag	5.9E-1	6.2E-1	3.2E-2	4.3E-2

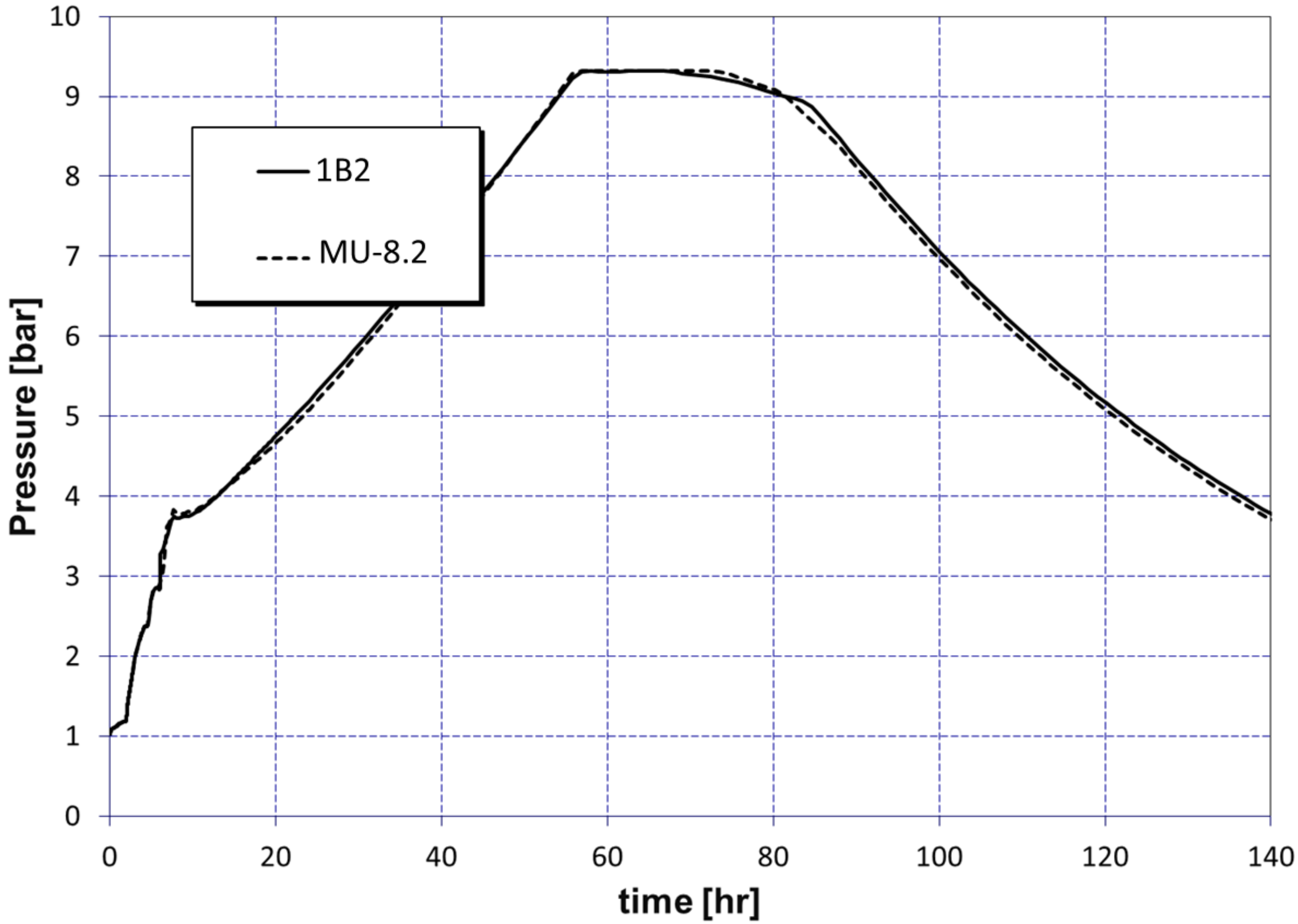


Figure 4-42: Containment Pressure in the Base and Sensitivity Case

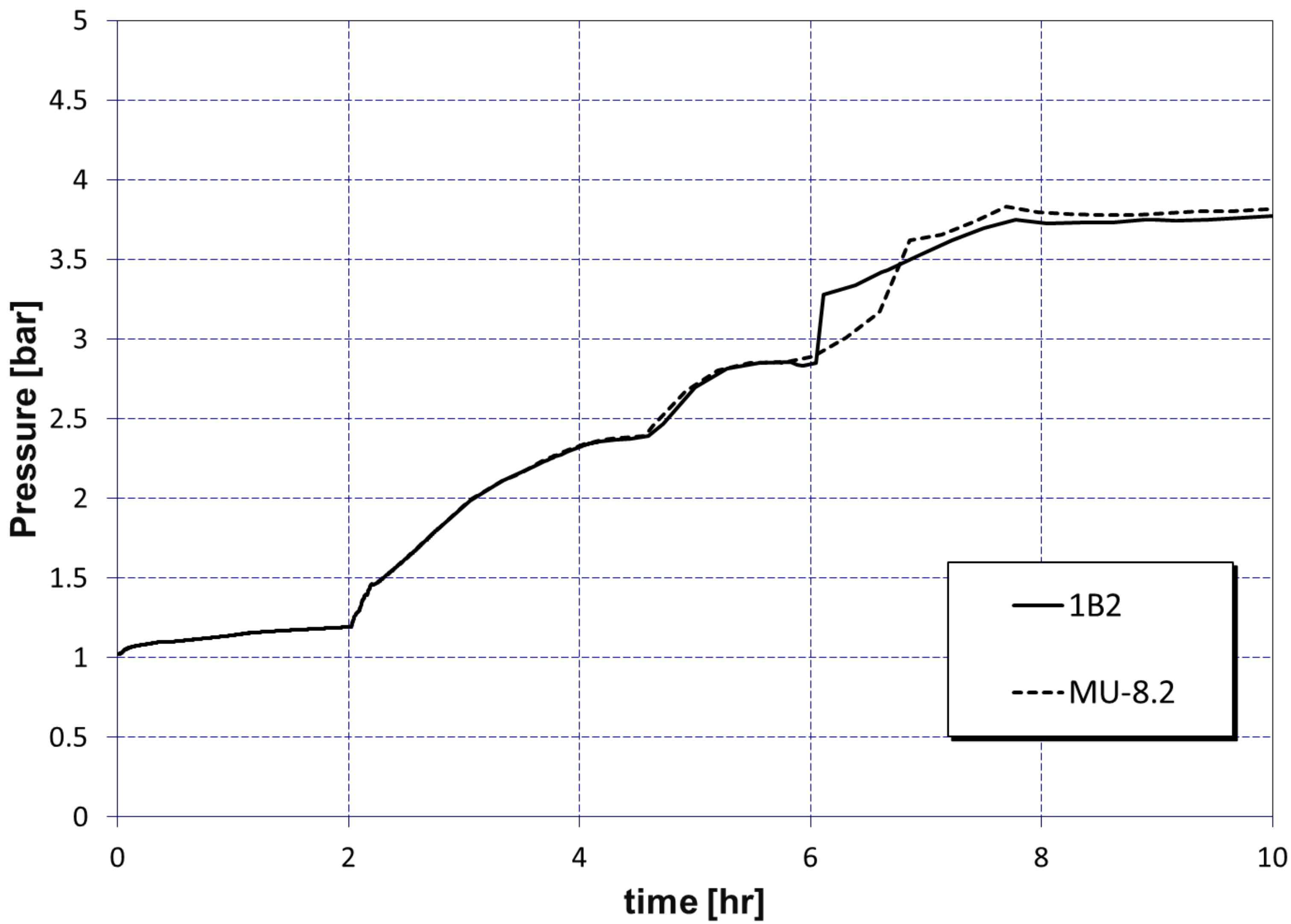


Figure 4-43: Containment Pressure in the Base and Sensitivity Case – Detailed

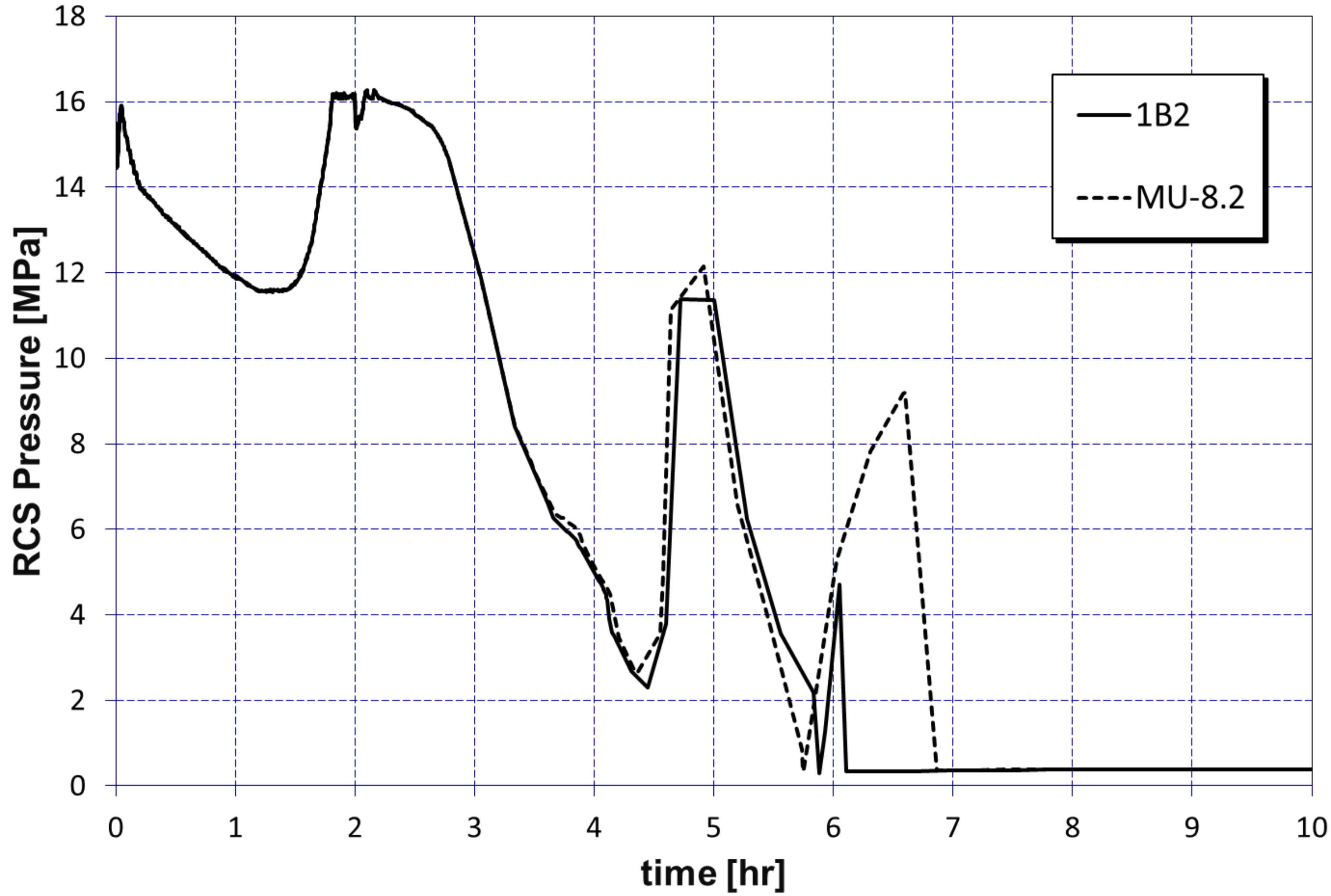


Figure 4-44: RCS Pressure in the Base and Sensitivity Case

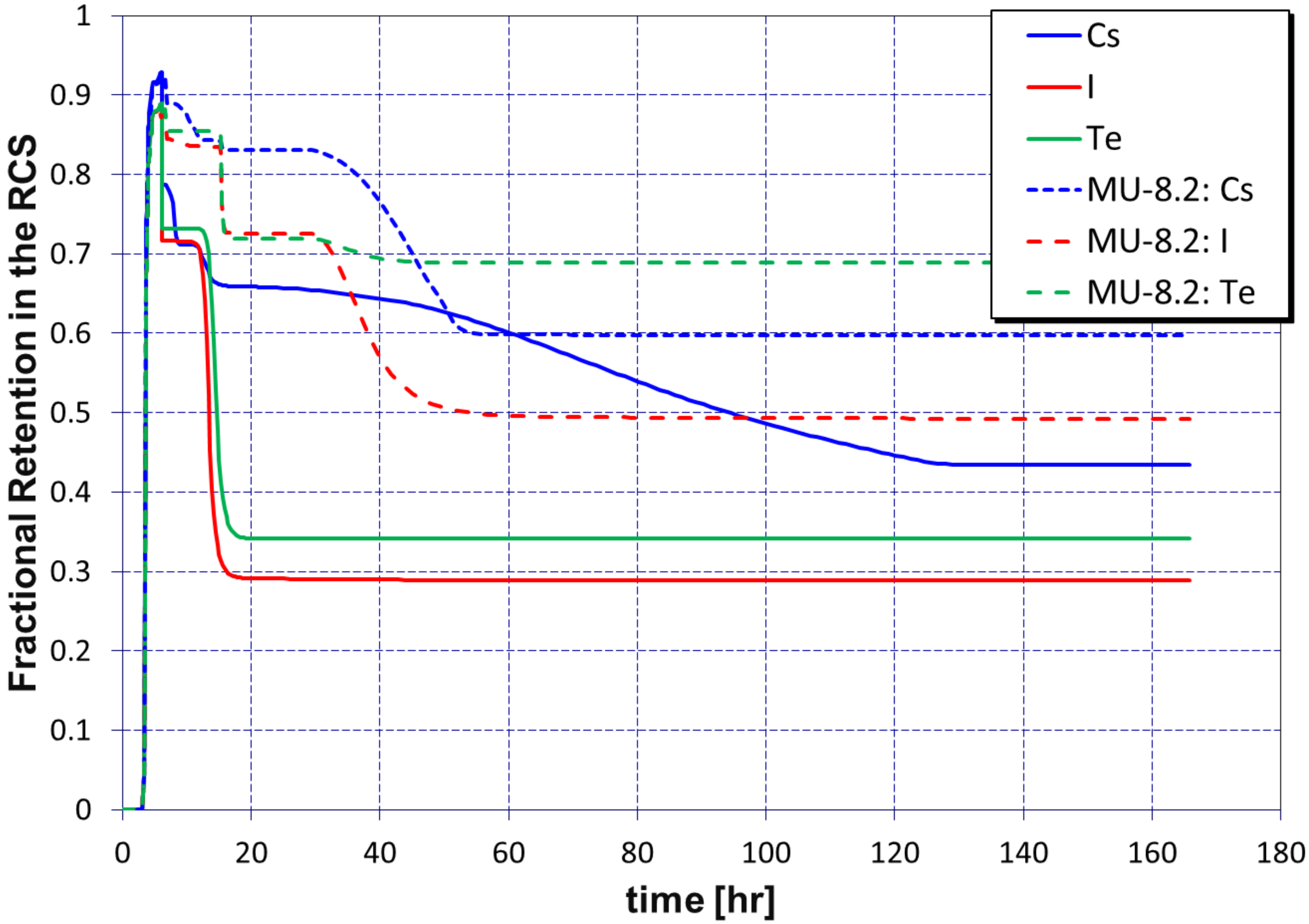


Figure 4-45: Retentions in the RCS for the Base and Sensitivity Case

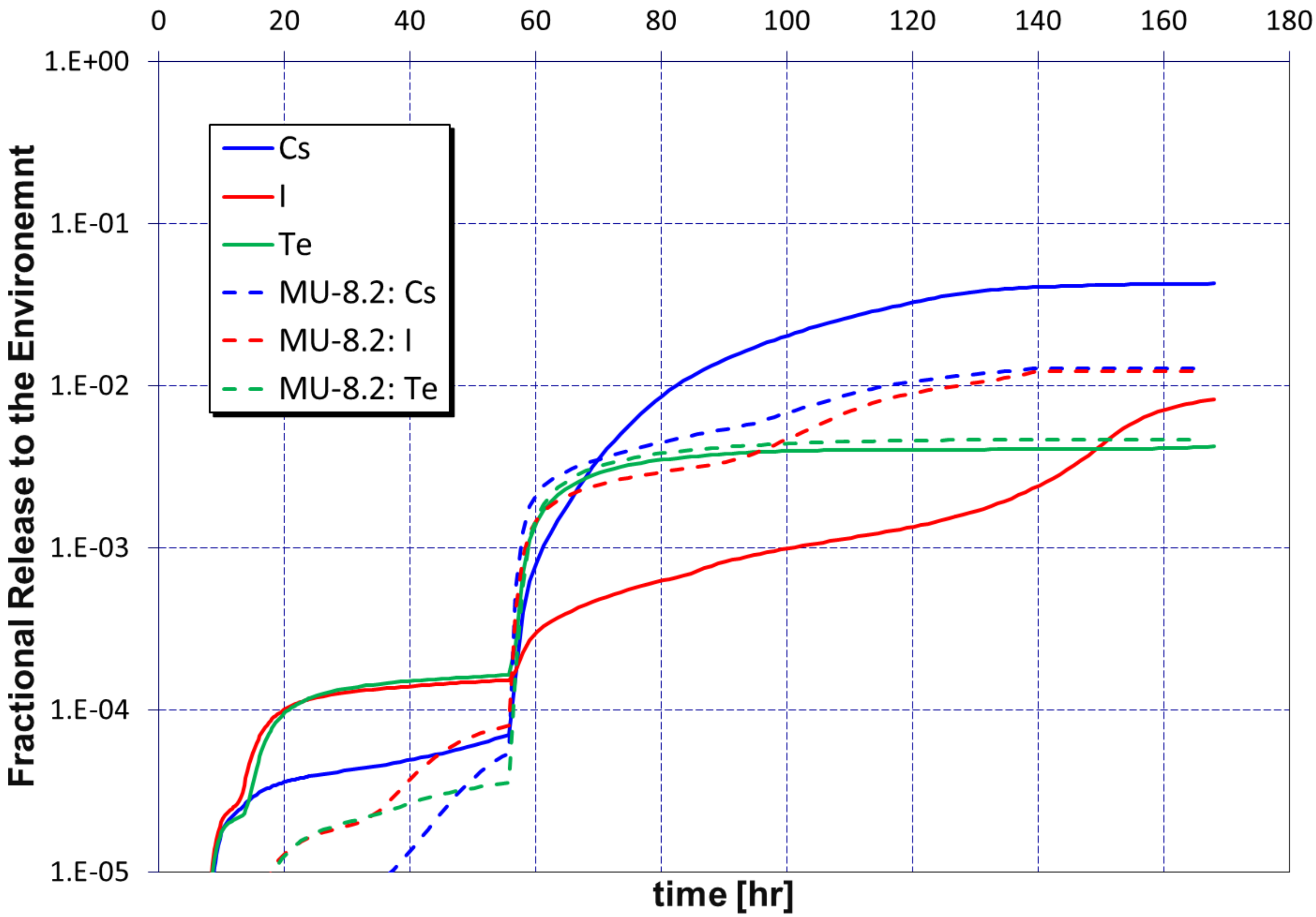


Figure 4-46: Environmental Release Fraction in the Base and Sensitivity Case

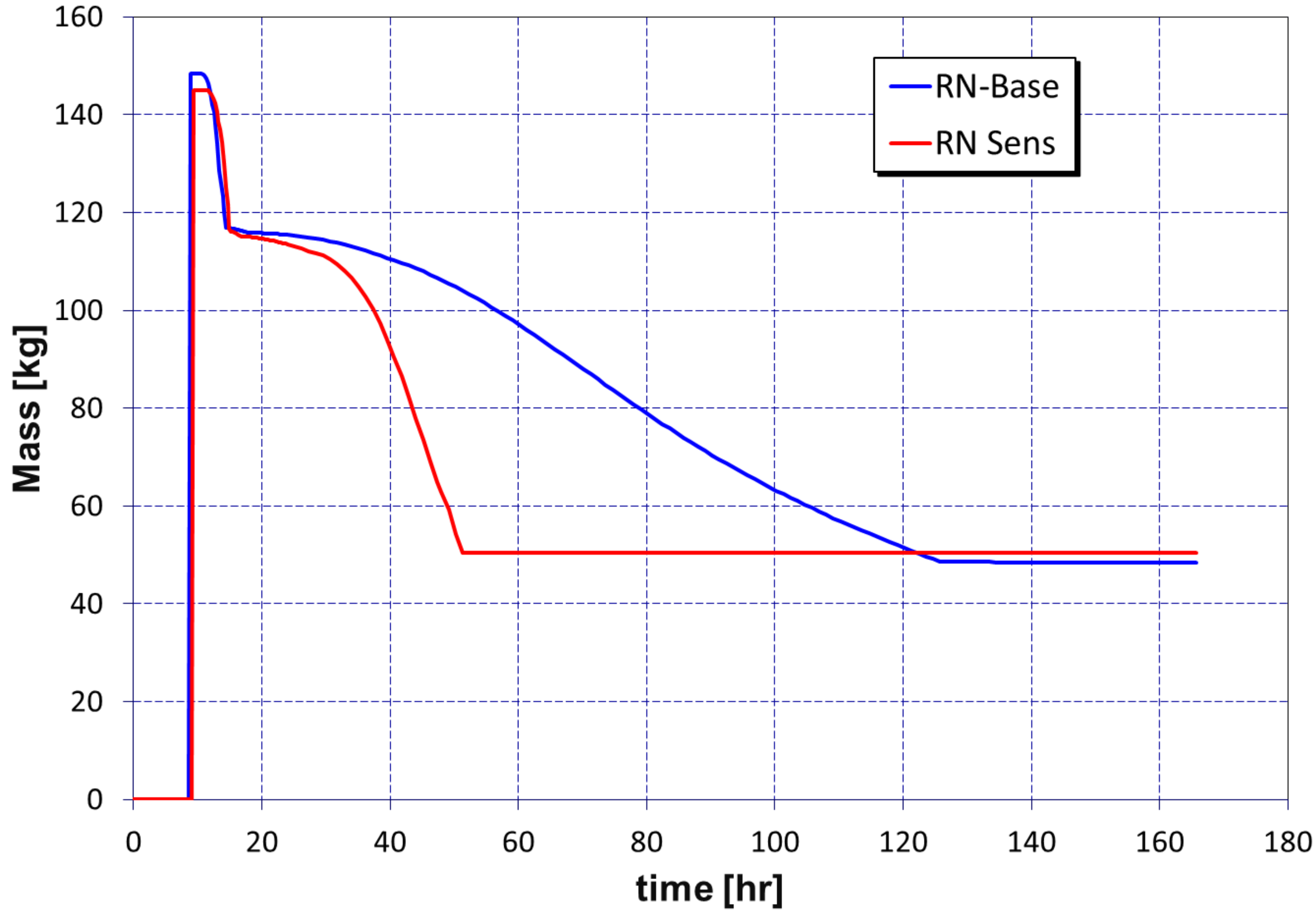


Figure 4-47: Radionuclides Deposited on the PRT Heat Structure for MU-8.2

4.8.5 Sensitivity Analysis on Effect of Decontamination Factor of Containment Over-Pressurization Leakage Path to the Tendon Gallery (MU-8.3)

This sensitivity re-considered Case 3A3. In that case, containment overpressure failure occurs at 71.3 hours, at which time flow path opens to the tendon gallery. Assumed failure of the tendon gallery doors from the depressurization then opens a path to the environment and channels fission products upward to the environment. While there is also an opening from the tendon gallery to the auxiliary building, there is only reverse flow of air occurring in this flow path (discussed in greater detail in Section 20 of Appendix D). As the fission products pass through the tendon gallery, the larger particles are removed and deposit on the floor. The other classes of fission products have DFs ranging from 1.07 to 1.33 (e.g., CsI and Mo have DFs of 1.07 and 1.33, respectively). Most of this deposition is due to gravitational settling of the fission products as they agglomerate and settle out in the large volume of the tendon gallery. Turbulent deposition in the containment crack, tendon gallery, and shaft is not modeled.

In reality, the velocity of the gas going into the tendon gallery at the time of containment over-pressurization is very high (see **Figure 4-48**). Turbulent deposition can be expected to occur in the crack through containment which is not currently being modeled in these MELCOR calculations and, thus, not part of the DFs previously discussed.

Consider, first, the possibility of the crack being plugged altogether. Experiments such as those performed by (Morewitz, 1979) and (Vaughan, 1978) have alluded to this possibility. However, these experiments showed plugging at differential pressures that were much lower than those seen in a containment over-pressurization scenario. They also did not consider the possibility of aerosol resuspension, which is likely given high flow velocity through the failure path (see **Figure** 4-48).

It is likely, however, that deposition due to turbulent flow through the leakage path would occur (see, for example, experiments at COLIMA [Parozzi, 2013]). After the first layer deposits, the subsequent layers of fission products are more likely to be resuspended and swept away by the high leak velocity. Simple hand calculations are used to approximate the fission products that may deposit in the failure pathway through the concrete using the results of Case 3A3 (namely the masses of fission products passing through the failure path). Values used in the calculation can be found in **Table 4-27**. Without detailed knowledge of the nature of the crack (surface area, tortuosity, etc.) and the amount of resuspension expected to occur, it is difficult to predict the DF.

Table 4-27: Constants and Calculated Values for Calculating Turbulent Deposition Velocity

Constant	Units	Value	Note:
Density of particle	kg/m^3	2650	SOARCA value[1]
Crack flow area	m^2	9.29E-4	Case 3A3 value
Crack length	m	3.2	(Candra, 2014a)
Crack thickness	mm	0.7	See text for derivation
Crack width	m	1.3	

[1]This is a more realistic density which is distinct from the default 1000 kg/m^3 density used in the MELCOR calculations.

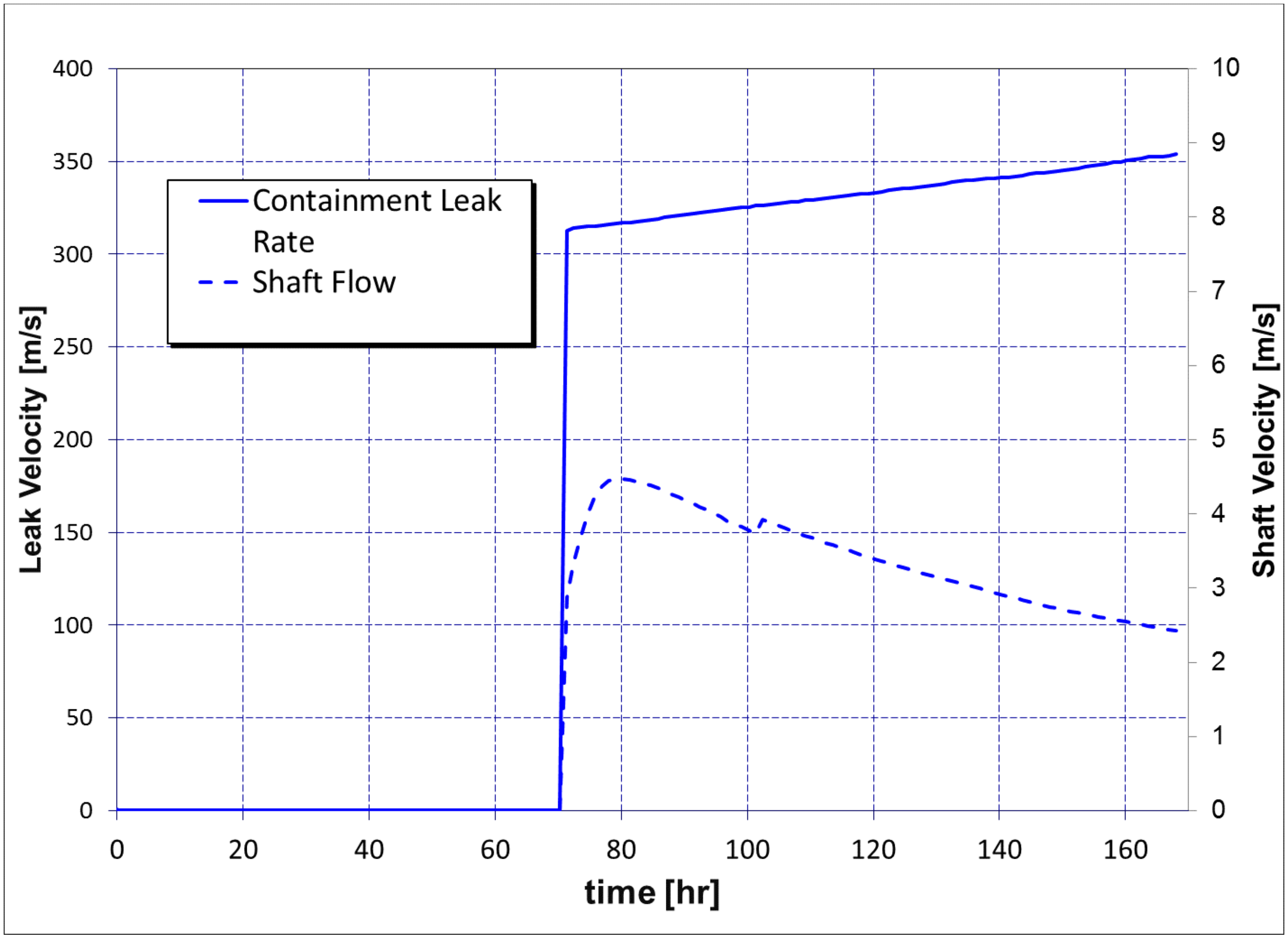


Figure 4-48: Flow Rate of Containment Leak Through Tendon Gallery to Environment and Through Shaft from Tendon Gallery to Auxiliary Building

The containment leakage pathway for MELCOR is derived from an independent finite element analysis performed for this project (discussed in Section 2.2.2 of the main body of this report). Here, an option for catastrophic failure of containment is described as a 1 ft^2 leakage path around the entire perimeter of containment leading to the tendon gallery. This implies a crack thickness of approximately 0.7 mm. In Case 3A3, a smaller flow area of 0.01 ft^2 is sufficient to arrest containment pressurization. If we assume the same thickness, the width of the smaller crack implied in the MELCOR calculations would be 1.3 m. The concrete is 3.2 m thick here, giving approximate dimensions for the crack as 0.7 mm x 1.3 m x 3.2 m. The planar surface area is then 8.32 m^2. Given the uncertainty in the nature of the crack (width, surface roughness, tortuosity) and the possibility of having a nonuniform deposition pattern on the surfaces, it is not unreasonable that the effective surface area could be an order of magnitude higher. A surface area of 83 m^2 is therefore assumed to account for these effects.

Given the mass $Q_{\ell,k}$ of each aerosol component type (k) and size section (ℓ), the number of particles, $n_{\ell,k}$, and their cross-sectional surface area, $SA_{\ell,k}$, is determined using the log mean diameter of the particles of that size (d_ℓ) via the equations:

$$n_{\ell,k} = 6\frac{Q_{\ell,k}}{\rho_p \pi d_\ell^3}$$

and

$$SA_{\ell,k} = \frac{3Q_{\ell,k}}{2\rho_p d_\ell}.$$

While turbulent deposition velocity is dependent on particle diameter, for convenience it is assumed all particles are equally likely to deposit. A total of 13 kg of fission products passes through the failure pathway from containment to the tendon gallery over the course of the scenario and could coat a surface area of approximately 4000 m^2. Scaling these calculations shows that 0.28 kg of material could deposit on the 83 m^2 crack surface, yielding a DF of 1.02. These calculations neglect deposition that may occur in the space behind the containment liner before getting to the concrete crack.

A similar calculation can be performed for the shaft from the tendon gallery to the environment. The overall surface area of the shaft is approximately 48 m^2, which would allow about 0.16 kg of material depositing giving a DF for the shaft of 1.01. In the large volume of the tendon gallery, the velocity is significantly less, and turbulent deposition would be small compared to the other modes of deposition.

This sensitivity demonstrates that the DF brought on by turbulent deposition in the containment over-pressurization leakage path to the tendon gallery is not significant. Deposition is therefore dominated by the gravitational settling in the tendon gallery that is already modeled in the calculations.

4.9 Ex-Vessel Coolability and MCCI Modeling

4.9.1 Identified Uncertainties for Ex-Vessel Coolability and MCCI Modeling

Table 4-28: Uncertainties for Ex-Vessel Coolability and MCCI Modeling

Item	Description	Other comments
Ex-vessel debris bed coolability	This can affect when releases trail off and whether the accident can be terminated prior to basemat melt-through.	Discussed in Section 21 of Appendix D. In (SNL, 2013) the ex-vessel uncertain input parameters included debris overflow head as a function of debris temperature for solid and liquid.
Impact of core debris / concrete interactions	This affects the response of containment (e.g., pressurization), generation of combustible gases, and fission product speciation; it also has the theoretical potential to degrade the structural supports of the RPV itself.	This includes a sub-item on ex-vessel Cs_2MoO_4 treatment. Appendix B also includes considerations related to core-concrete interaction erosion rate as it affects the timing of basemat melt-through relative to other containment failure modes; the issue of under-mining RPV support is discussed briefly in Section 11 of Appendix D. The draft Surry SOARCA UA (SNL, 2016a) conducted separate sensitivity analyses for concrete aggregate material, and the amount of rebar in the concrete.

4.9.2 Ex-Vessel Coolability and MCCI Sensitivities

No sensitivity studies were conducted for ex-vessel coolability and MCCI modeling.

4.10 ISLOCA Modeling

4.10.1 Identified Uncertainties in ISLOCA Modeling

Table 4-29: Uncertainties in ISLOCA Modeling

Item	Description	Other comments
Characterization of the ISLOCA break (size and location)	This can affect the accident progression for ISLOCA, based on significant uncertainties in the exact location and type of break (source term aspects are identified separately).	An expert elicitation on ISLOCA was conducted under the L3PRA project (NRC, 2019a), which provides some related information. Section 5 of Appendix B includes both 2-inch and 8-inch breaks.
Turbulent deposition in the connected piping	In the Surry SOARCA study (NRC, 2013), turbulent deposition in RHR piping was predicted to provide significant scrubbing, largely due to back-side flooding of the RHR pipe preventing re-vaporization.	See Section 5 of Appendix B of this report for discussion of deposition mechanisms in connected piping. In this study no attempt was made to model turbulent deposition in the RHR piping.

Table 4-29: Uncertainties in ISLOCA Modeling

Item	Description	Other comments
Piping penetration area filtration and exhaust system (PPAFES) filter efficiency, clogging, and heat load damage	Assumptions are made regarding how the PPAFES filters will respond to significant passage and deposition of radiological material beyond their design-basis.	This issue is discussed briefly in Section 5 of Appendix B.
Auxiliary building blowdown failure during RHR ISLOCAs	The SAPHIRE model presumes that all RHR ISLOCAs are large, and this in turn presumes that they will fail the auxiliary building during blowdown. As captured in the expert elicitation (NRC, 2019a), a portion of the RHR ISLOCAs will result in an effective flow area below the 2.5”-equivalent diameter break that is correlated to auxiliary blowdown failure in Section 5 of Appendix B.	This assumption will tend to over-estimate the contribution of ISLOCAs to consequences and risk.
*Note that uncertainties related with fission product retention in surrounding structures is captured in **Section 4.13**.*		

4.10.2 Alternative Treatment(s) of Uncertainties in ISLOCA Modeling

MU-10.1 A MELCOR simulation was performed based on Case 5B (an ISLOCA with a double-ended, 8-inch, submerged break in the RHR line), but with a slightly smaller 6-inch break. This provided an intermediate result between the 2-inch and 8-inch ISLOCAs of Cases 5 and 5B, respectively. The key event timings of this sensitivity case were compared with those of Case 5 and Case 5B to demonstrate the effect of break size on accident progression.

4.10.3 Sensitivity Analysis on ISLOCA Break Size (MU-10.1)

Cases 5 and 5B have significant differences in their releases to the environment with ISLOCA break sizes of 2 and 8 inches, respectively, due to the failure of the auxiliary building in the latter case. For this sensitivity, the break size was set to 6 inches to serve as an intermediate result between the two existing scenarios. All other parameters, including break elevation and RHR pump operation, are the same as in Case 5B.

Key event timings for the three cases are provided in **Table 4-30**. In Case MU-10.1 the blowdown in the auxiliary building is less (**Figure 4-49**) due to the smaller break size, but is still high enough to fail the auxiliary building around the same time as Case 5B. The unfiltered flow path that opens to the environment due to this failure causes an increased release of fission products to the environment as compared to Case 5. **Table 4-31** gives the fractional environmental release and the auxiliary building retention for radionuclides in the three cases. The large retentions seen in the auxiliary building in MU-10.1 in **Figure 4-50** are due to increased scrubbing of fission products in the pool overlying the RHR break. The smaller break size allows for greater breakup of the gas stream and results in fewer fission products making their way to the auxiliary building atmosphere (**Figure 4-51**), which also contributes to the lower fission product releases to the environment.

This sensitivity demonstrates that ISLOCA break size has a large impact on the amount of expected environmental release in two significant ways. First, the auxiliary building will quickly pressurize following the pipe break, and whether it fails, depends strongly on the break size. Second, the break size has an impact on the effectiveness of the overlying pool (if it exists) to scrub the fission products.

Table 4-30: Selected MELCOR Event Timings for Case 5, 5B and MU-10.1

Key Accident Parameter	Event Timing*		
	5	**MU-10.1**	**5B**
Auxiliary building failure	-	9 sec	7 sec
Start of accumulator injection	18 min	4 min	2 min
RWST at Lo-3	4.3	1.2	1.2
Start of core uncovery (level at TAF)	7.6	1.3	1.2
Onset of core oxidation	9.4	2.9	2.7
First gap release	9.4	2.9	2.7
Spatially maximized clad temperature exceeds 2200F	9.5	3.1	2.9
Average temperature of coolant at the core exit exceeds 1200F	9.5	3.1	2.8
Dryout of the lower plenum	12.2	5.7	5.5
Vessel Breach (through-wall yield)	13.4	5.7	6.2

* Timings are in hours unless otherwise stated.

Table 4-31: Fractional release and retentions of radionuclides for Cases 5, 5B and MU-10.1

Representative Element	Calculation End (72 hr)					
	Retention in Auxiliary Building			Release to Environment		
	5	**MU-10.1**	**5B**	**5**	**MU-10.1**	**5B**
Xe	2.6E-3	1.1E-1	1.2E-1	9.9E-1	8.9E-1	8.6E-1
Cs	7.5E-1	7.9E-1	7.3E-1	6.4E-4	6.4E-2	9.2E-2
I	8.8E-1	8.6E-1	7.9E-1	1.1E-3	8.2E-2	1.2E-1
Te	8.5E-1	8.6E-1	7.9E-1	7.6E-4	7.4E-2	1.1E-1
Ba	1.3E-2	1.4E-2	1.5E-2	9.4E-6	7.6E-4	1.0E-3
Ru	2.3E-2	3.1E-2	2.8E-2	1.0E-5	8.9E-4	1.3E-3
Mo	2.1E-1	2.2E-1	1.8E-1	1.8E-4	1.8E-2	2.3E-2
Ce	1.5E-5	2.0E-6	5.9E-6	1.5E-7	2.2E-7	4.8E-7
La	9.2E-7	7.3E-7	7.5E-7	4.5E-9	3.7E-8	4.4E-8
UO2	3.0E-3	3.9E-3	3.4E-3	1.3E-6	1.1E-4	1.6E-4
Cd	7.1E-1	7.6E-1	6.8E-1	4.3E-4	4.3E-2	7.1E-2
Ag	6.3E-1	6.5E-1	6.0E-1	3.6E-4	3.4E-2	5.1E-2

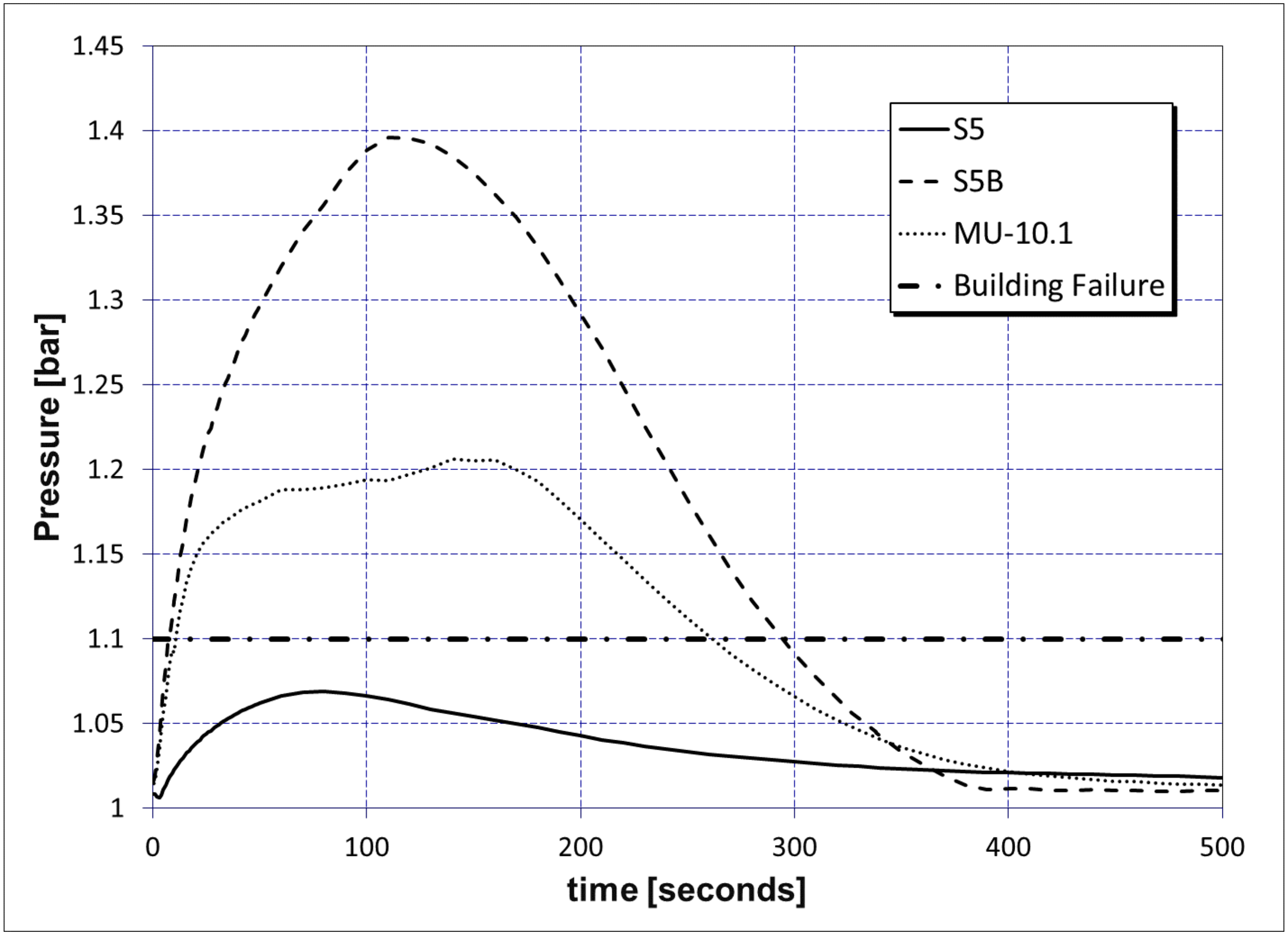


Figure 4-49: Pressure in the Auxiliary Building in the Cases 5, 5B, and MU-10.1

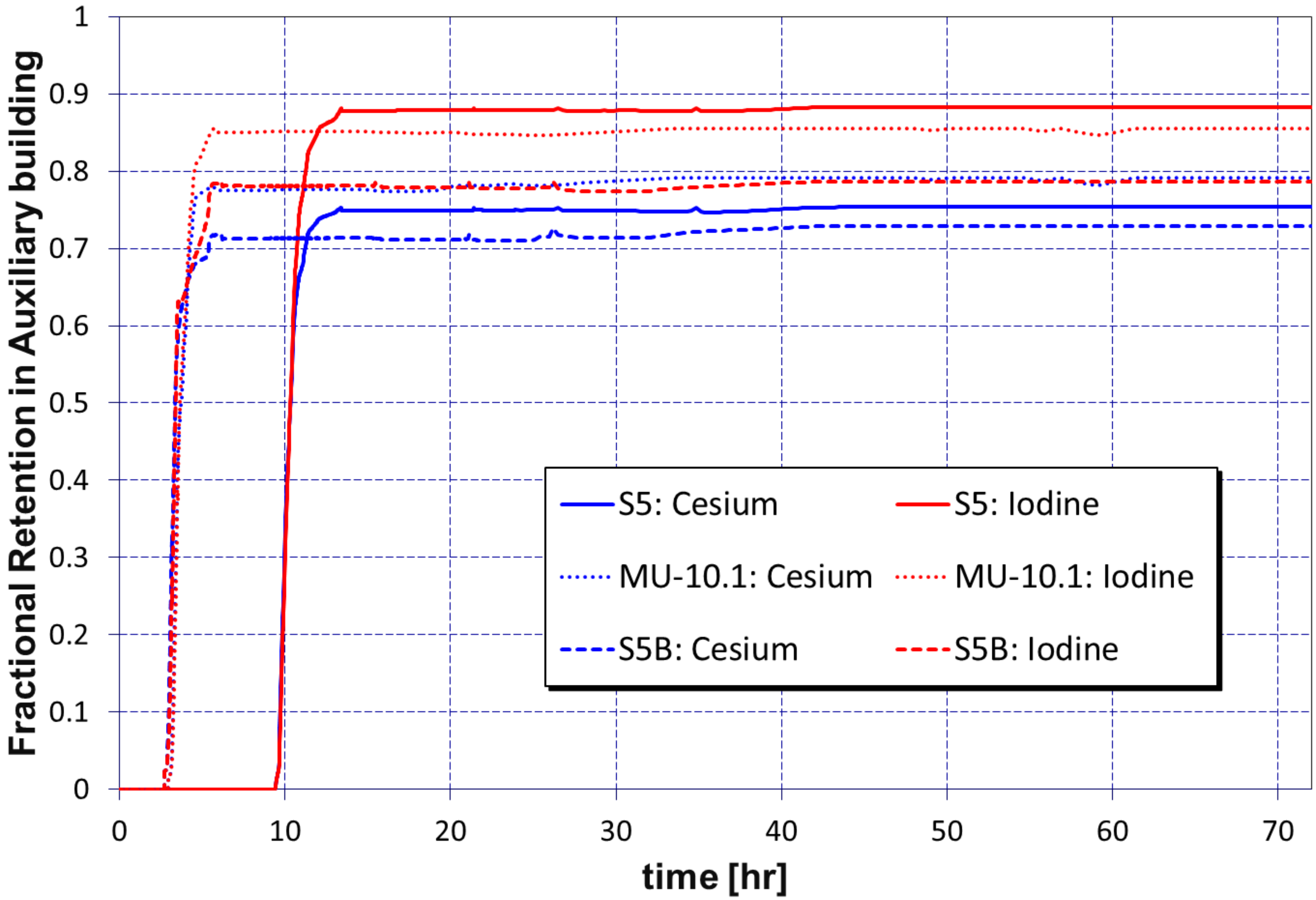


Figure 4-50: Retention of Cesium and Iodine in the Auxiliary Building for Cases 5, 5B and MU-10.1

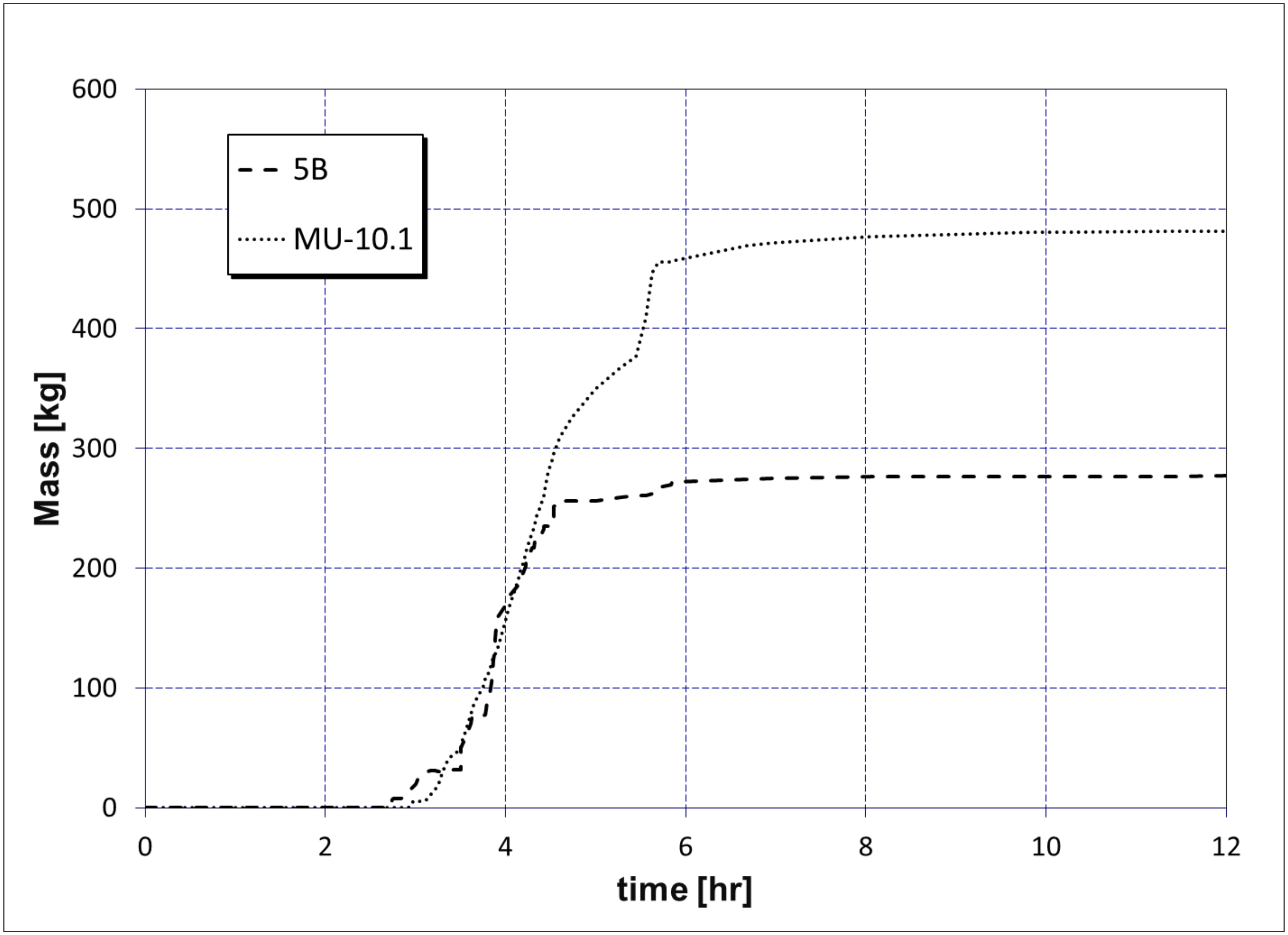


Figure 4-51: Aerosol Mass Retained in the Pool of Level C in the Auxiliary Building for Cases 5B and MU-10.1

4.11 SGTR and Induced SGTR Modeling

4.11.1 Identified Uncertainties in SGTR and Induced SGTR Modeling

Table 4-32: Uncertainties in SGTR and Induced SGTR Modeling

Item	Description	Other comments
Binning of C-SGTR into a single failure probability / break size	C-SGTR is treated as a threshold effect (i.e., discrete probability that a leak of a specified size will occur). It can alternatively be viewed as a continuum of leakage area versus probability resulting in smaller, higher-probability failures and larger, lower-probability failures.	The failure probability of a leak size approaching zero leakage area is conceptually one in a continuous representation. Meanwhile, the C-SGTR calculations performed for this project (NRC, 2018) estimated a failure probability of less than 1×10^{-2} for a leakage area the size of two double-ended tube breaks. Discretizing this spectrum into several bins (probabilities and accompanying source terms) may alter the estimation of risk impact. The draft Surry SOARCA UA (SNL, 2016a) included a separate mini-UA on SGTR and included the effects of multiple tubes rupturing along with other influential uncertain inputs for SGTR.
Time between C-SGTR and subsequent hot leg creep rupture	This delta-t is uncertain and has a fairly direct impact on predicted source term. Case 3A3 (see Section 3.3 of Appendix B) assumes a 50-minute delta-t, a value which is exaggerated based on nuances about the history of these calculations.	Case 3A3 is believed to provide a somewhat exaggerated response, while Case 3A2 (see Section 3.2 of Appendix B) provides the logical extreme of no subsequent hot leg failure. A sensitivity with a shorter delta-t was deemed to be reasonable. The results of the draft Surry SOARCA UA (SNL, 2016a) showed the calculated time difference between induced SGTR and hot leg rupture ranged from 0.12 minutes to 144 minutes, with a mean of 28.1 minutes, and the time of hot leg rupture had a large impact on total releases.
Break elevation in the SG tubes and secondary-side retention	Break elevation and assumptions about secondary-side retention affect the overall SGTR source term.	These issues are investigated in the draft Surry SOARCA UA, and are also discussed in Section 5.1 of Appendix B.
*Note that modeling assumptions related to the accident progression modeling up until the point of C-SGTR are captured in **Section 4.5**.*		

4.11.2 Alternative Treatment(s) of Uncertainties in SGTR and Induced SGTR Modeling

MU-11.1 A MELCOR simulation was performed based upon MELCOR calculation S3A3 with steam generator tube rupture occurring 15 minutes (compared to 50 minutes in the base case) prior to hot leg creep rupture at 10.9 hours. The impact of this timing assumption on the source term was compared to Cases 3A3 and 3A2 (see Sections 3.3 and 3.2 of Appendix B).

MU-11.2 A revisitation was performed of Case 3A2 in which the Powers[5] decontamination factor was applied in two ways to the calculated releases. These two approximations of the enhanced retention in the SG dryers and separators give an overly and underly conservative estimate of the environmental release.

[5] Researchers at Sandia National Laboratories have developed models to calculate the decontamination factors at the tube support plates and in the steam generator separators and dryers based on the experimental results (hereafter referred to as the Powers model). The same decontamination factor model was used for the draft Surry SOARCA UA (SNL, 2016a).

4.11.3 Sensitivity Analysis on the Effect of the Timing of SGTR in Relation to Hot Leg Nozzle Creep Rupture on Fission Product Retention and Environmental Release (MU-11.1)

SGTR is modeled in Case 3A3 as occurring 50 minutes prior to the predicted time of hot leg creep rupture in Case 3 (10.9 hours). The tube rupture occurs at the same time in Case 3A2; however, unlike Case 3A3, creep failure is disabled. In the L3PRA project MELCOR model, hot leg nozzle creep rupture is code-calculated, while SGTR is user-specified. This section discusses a sensitivity based upon Case 3A3 where SGTR is forced to occur 15 minutes prior to creep failure that still occurs at 10.9 hours. A summary of the key event timings for the three cases is given in **Table 4-33**.

Table 4-33: Key Event Timings (in hours) for Cases 3A3, 3A2, and MU-11.1

Key Accident Parameter	3A2	3A3	MU-11.1
SGTR	10.1	10.1	10.7
Clad temp >2200F	10.6	10.6	10.5
Creep failure of hot leg nozzles	N/A	10.9	10.9
Vessel Breach	13.2	14.9	15.0
Containment failure	87.8	70.5	67.6

When the tube rupture occurs, a driving head between the RCS and the environment causes a rapid burst of fission products to be released, Because of the later SGTR in MU-11.1, the amount of time wherein this driving head exists is lessened, and the fission product releases are greatly reduced from both other cases (**Figure 4-52**). Also, the RCS is at a higher pressure at the time of creep rupture because the tube rupture has only recently occurred, with pressurizer pressure around 13.8 MPa, versus 8.9 MPa in Case 3A3 (**Figure 4-53**). A rapid blowdown into containment occurs, and while containment over-pressurization occurs sooner in this sensitivity (**Figure 4-54**), there is also a greater retention of fission products in containment (**Figure 4-55**).

A breakdown of the containment and RCS retentions and environmental releases are given in **Table 4-34** for Cases 3A2, 3A3, and MU-11.1. The general result of the later SGTR in this sensitivity is less fission product retention in the SGs, greater fission product retention in containment, and a reduced release to the environment.

This sensitivity demonstrates that the timing of SGTR in relation to that of hot leg nozzle creep rupture has a large impact on where fission products are retained and the extent of the release to the environment.

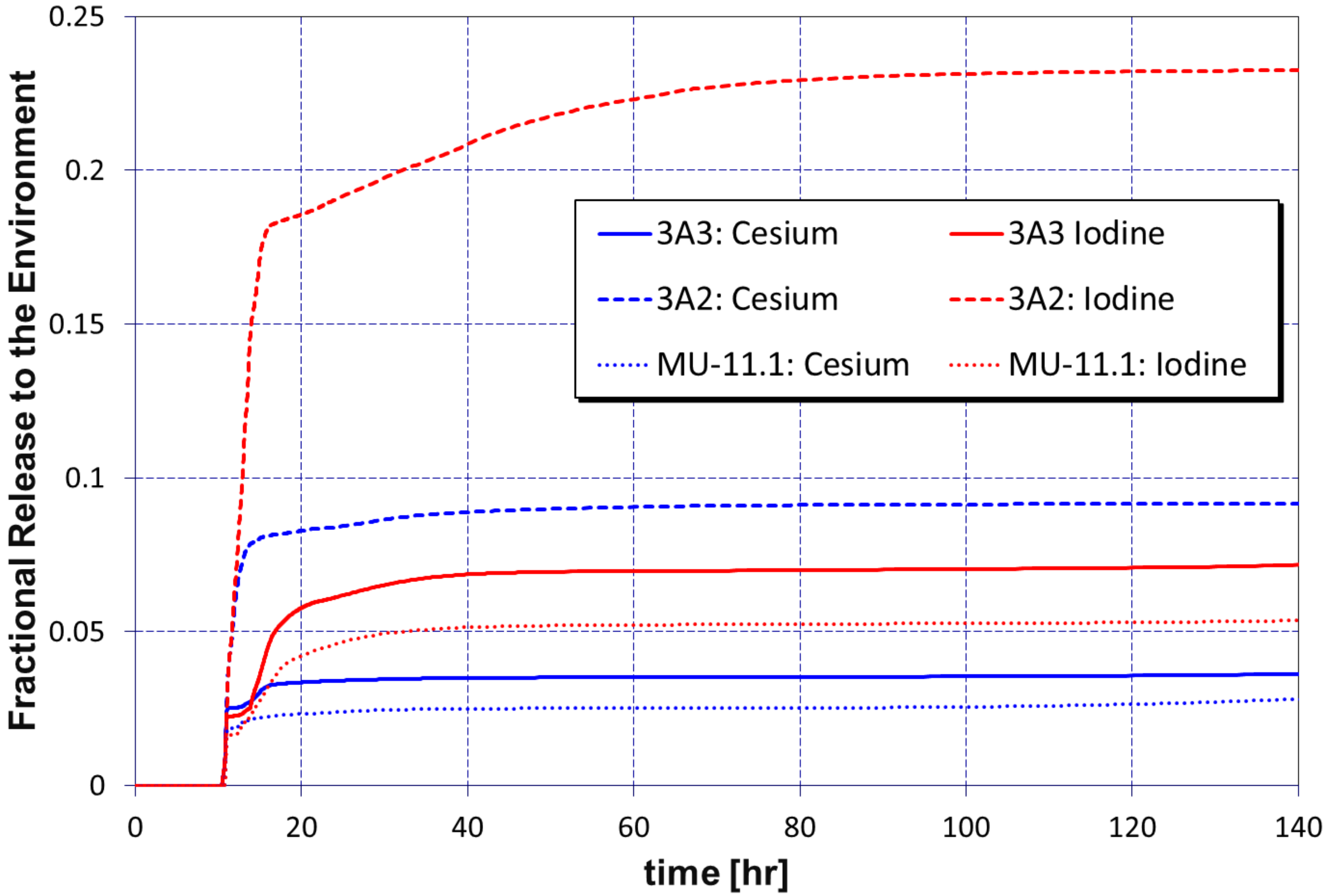


Figure 4-52: Fractional Release to the Environment for Cases 3A2, 3A3 and MU-11.1

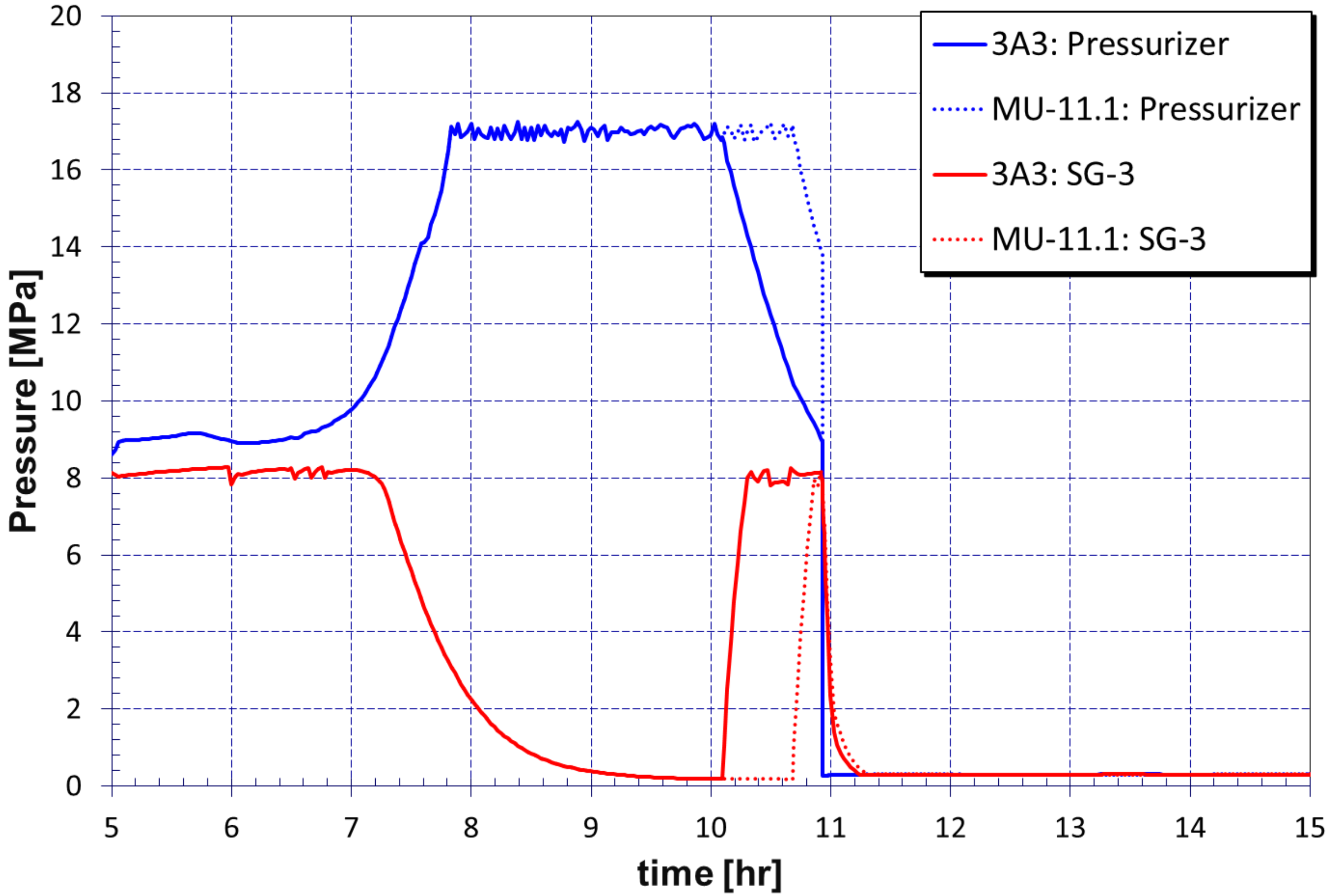


Figure 4-53: Pressure in the Pressurizer and SG3 (the ruptured SG) for Cases 3A3 and MU-11.1

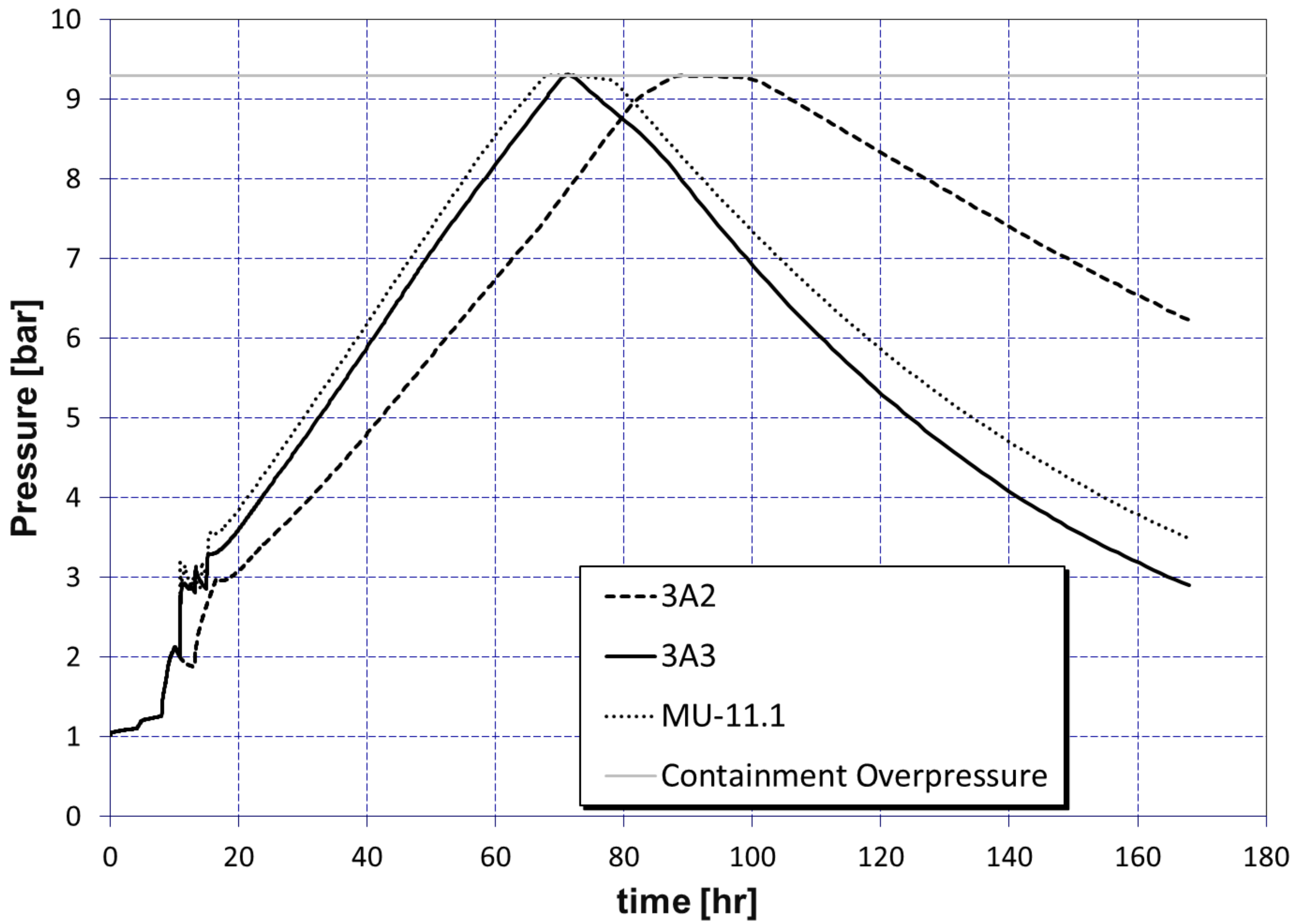


Figure 4-54: Containment Pressure in Cases 3A2, 3A3 and MU-11.1

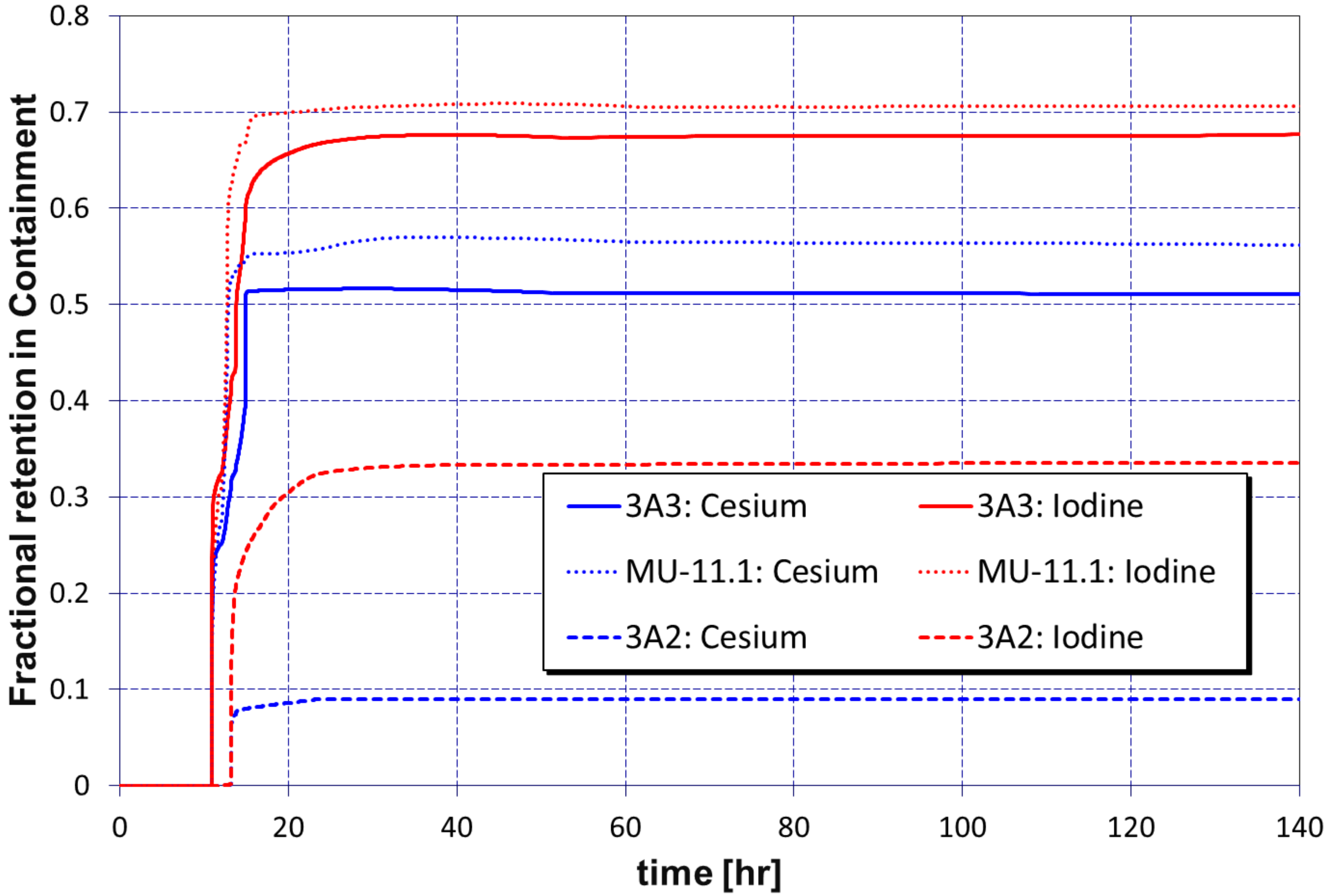


Figure 4-55: Fractional Retention in Containment for Cases 3A2, 3A3, and MU-11.1

Table 4-34: Fractional Retentions and Environmental Release for Cases 3A2, 3A3 and MU-11.1

Representative Element	SG			Containment			Environment		
	3A2	3A3	MU-11.1	3A2	3A3	MU-11.1	3A2	3A3	MU-11.1
Xe	6.9E-5	2.1E-4	2.6E-4	5.0E-2	1.3E-1	1.6E-1	9.5E-1	8.7E-1	8.4E-1
Cs	1.9E-1	6.9E-2	6.6E-2	8.9E-2	5.1E-1	5.6E-1	9.2E-2	3.8E-2	3.1E-2
I	2.0E-1	3.4E-2	3.4E-2	3.4E-1	6.9E-1	7.1E-1	2.3E-1	7.6E-2	5.7E-2
Te	1.3E-1	6.8E-2	6.1E-2	1.6E-1	5.6E-1	6.6E-1	1.9E-1	3.8E-2	2.2E-2
Ba	2.0E-3	2.4E-3	1.1E-3	1.4E-2	2.8E-2	1.6E-2	4.7E-3	2.8E-3	6.4E-4
Ru	2.1E-3	2.2E-4	3.3E-4	2.6E-4	5.5E-3	2.1E-2	6.9E-4	9.4E-5	1.8E-4
Mo	5.0E-2	2.5E-2	2.4E-2	7.5E-1	8.0E-1	8.2E-1	2.2E-2	3.5E-2	3.4E-2
Ce	1.1E-7	6.8E-6	2.7E-6	1.3E-3	1.9E-3	1.2E-3	2.4E-7	6.1E-6	3.8E-6
La	9.7E-8	1.8E-7	1.5E-7	3.4E-5	3.4E-5	3.4E-5	1.8E-7	4.6E-7	4.7E-7
UO2	2.9E-4	4.2E-5	5.6E-5	3.8E-4	1.1E-3	2.9E-3	1.3E-4	6.2E-5	7.4E-5
Cd	1.5E-1	3.6E-2	3.6E-2	3.1E-1	6.6E-1	7.0E-1	7.1E-2	4.6E-2	2.7E-2
Ag	1.3E-1	2.6E-2	2.6E-2	4.3E-1	6.8E-1	7.0E-1	5.1E-2	2.9E-2	2.1E-2

4.11.4 Fission Product Retention Modeling for SGTR MELCOR Simulations (MU-11.2)

The base case only considers deposition of fission products in the structures going from the SG to the environment and fallout of the aerosol. Two approaches for applying the DFs from the Power's model are suggested in Section 3.2 of Appendix B of this report.

Approach 1 is to take what the Powers model predicts coming out of the SG and assume that it makes its way directly to the environment. This is a generally pessimistic approach as it does not consider deposition in the steam line and the increased growth and fallout of the aerosols during transport.

Approach 2 is to apply the Powers model DF directly to the release calculated by MELCOR out the SG relief valve (or other secondary-side release pathway). This is perhaps an overly optimistic approach in that it may double-count some deposition effects by assuming the DFs can be applied on top of one another.

For this sensitivity, the DFs calculated by the Powers model are applied to the results of Case 3A2 in these two ways. **Table 4-35** gives the fission products going to the environment as calculated by these two approaches. The best estimate on the actual release lies between the two results, but likely closer to the values of the more optimistic Approach 2.

Table 4-35: Environmental Releases as Calculated by Two Applications of the Powers DFs

Representative Element	Approach 1	Approach 2	Base Case 3A2
Xe	9.5E-1	9.5E-1	9.5E-1
Cs	6.3E-2	2.0E-2	9.2E-2
I	3.5E-1	1.8E-1	2.3E-1
Te	1.5E-1	8.5E-2	1.9E-1
Ba	2.1E-3	1.5E-3	4.7E-3
Ru	2.5E-4	6.3E-5	6.9E-4

Table 4-35: Environmental Releases as Calculated by Two Applications of the Powers DFs

Representative Element	Approach 1	Approach 2	Base Case 3A2
Mo	9.2E-3	3.9E-3	2.2E-2
Ce	2.1E-7	1.7E-7	2.4E-7
La	1.5E-7	1.2E-7	1.8E-7
U	6.1E-5	3.0E-5	1.3E-4
Cd	2.9E-2	1.3E-2	7.1E-2
Ag	2.1E-2	6.7E-3	5.1E-2

4.12 Containment Isolation Failure Modeling

4.12.1 Identified Uncertainties in Containment Isolation Failure Modeling

Table 4-36: Uncertainties in Containment Isolation Failure Modeling

Item	Description	Other comments
Containment Isolation System (CIS) screening	Screening of CIS penetrations is based on fission product considerations and a binary (> or ≤ 2-inch) criteria for active isolation failures.	In reality, releases will follow leak size more linearly, and this approach provides poor resolution in some regards. MELCOR calculations were run to show that an effective 2-inch (5.8-cm) leakage size is a reasonable threshold for defining a containment isolation failure from a containment pressurization standpoint. See Section 6 of Appendix D for discussion of containment leakage sizes.
CIS pre-existing tear assumption	The adopted licensee CIS logic includes a probability of a pre-existing tear. No active CIS failures (which might be more likely to lead to larger failure sizes) contribute significantly to CIS frequency.	
Characterization of the containment isolation failure location and size	In Cases 7 and 7A (see Section 7 of Appendix B), the pre-existing containment isolation failure is assumed to be a 2-inch-equivalent diameter break leading into the environment – a smaller/larger leakage area, or a location leading directly to the auxiliary building, would change the source term.	The leakage size aspect is somewhat related to the first line item in this list, while the leakage location aspect is somewhat related to MU-8.1 (see Section 4.8.2). Note that failure directly to the auxiliary building would make that area less habitable and might decrease the likelihood of using the EDMG pump to spray water into containment through the containment spray air test lines (as part of an SCG-1 action).

4.12.2 Alternative Treatment(s) of Uncertainties in Containment Isolation Failure Modeling

MU-12.1 To address the effect of the specification of the isolation failure size, Case 7 (see Section 7 of Appendix B) was re-run using a 4-inch equivalent diameter leakage area (directly to the environment). Though not important for the MELCOR model input change, this leakage area can be notionally thought to correspond to an active isolation failure of the containment penetration for service air and the post-LOCA purge air supply.

MU-12.2 Regarding the effect of the dominant contributor to containment isolation failure (1-L2TEAR), this parameter is part of the propagation of parameter uncertainty. However, it can also be viewed in isolation of other uncertain parameters, because its frequency has a nearly linear effect on the CIF and CIF-SC release categories, and thus on LERF and LRF.

4.12.3 Sensitivity Analysis on Effect of the Size of the Containment Isolation Failure Path to the Environment (MU-12.1)

In this sensitivity based on Case 7, the containment isolation flow path's open fraction is increased to model a 4-inch rather than a 2-inch containment isolation failure path directly to the environment. The large leak in containment is present from the start of the simulations.

As the accident progresses, the rapid pressure increase seen in the base case is dampened in MU-12.1 (**Figure 4-56**), due to the larger leak size. The decrease in pressure around 75 hours in both cases corresponds to the time when there is no more water in the cavity. Even though there is a large driving pressure in the base case, the larger flow path in the sensitivity allows for a much greater flow rate out the failure pathway (**Figure 4-57** and **Figure 4-58**), particularly between 12 and 24 hours when the greatest releases occur. The environmental releases and containment retentions for both the sensitivity and base cases are shown in **Table 4-37**. In general, the radionuclide releases to the environment are roughly twice that of the base case (**Figure 4-59**). The rate of radial concrete ablation in the reactor cavity is much greater for this case (MU-12.1) (**Figure 4-60**). Sidewall melt-through occurs at 121 hours (versus 149 hours in the base case).

This sensitivity demonstrates that containment isolation failure size can have a large impact on the magnitude of the environmental release.

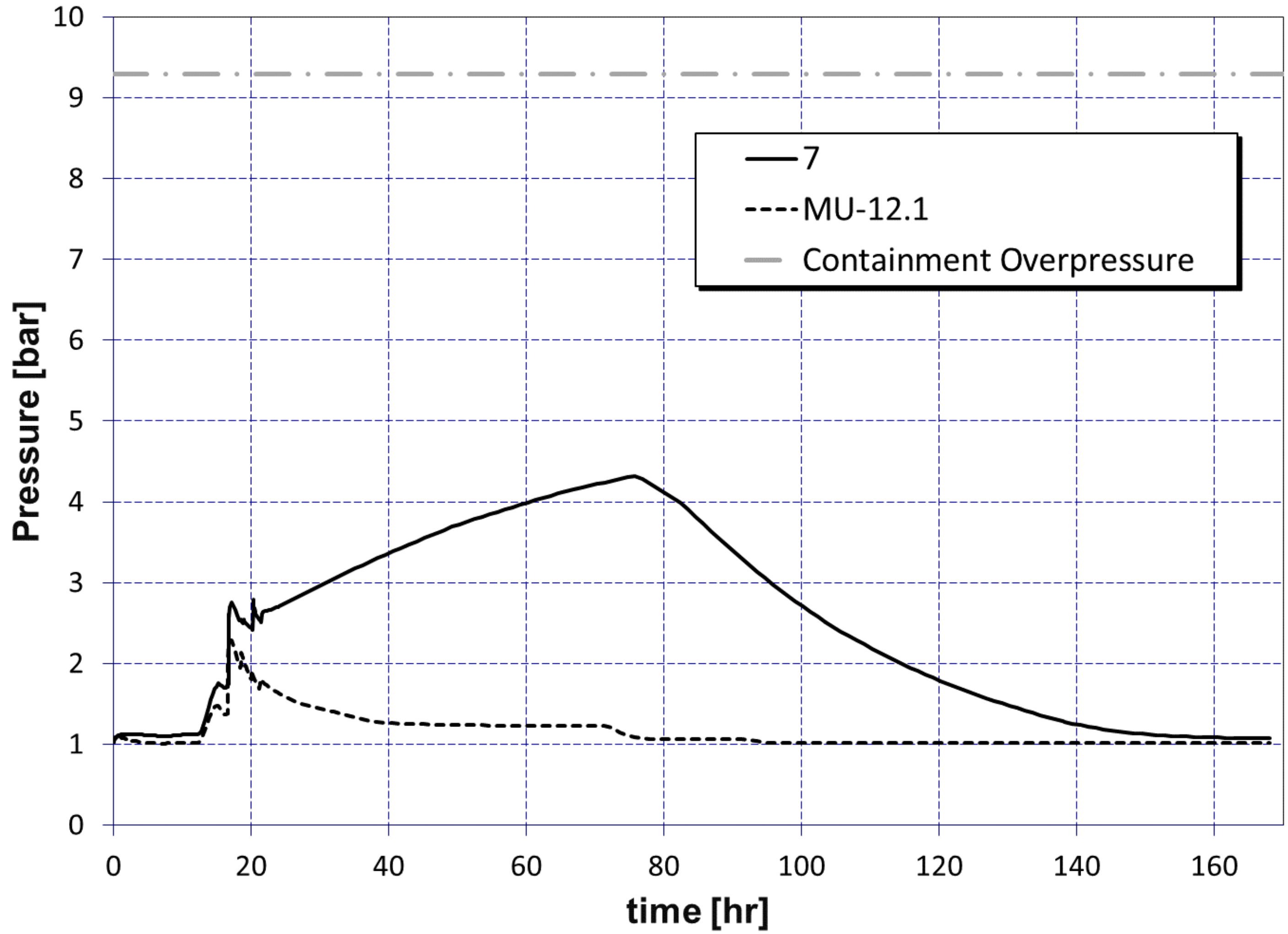


Figure 4-56: Containment Pressure in the Base and Sensitivity Cases

Table 4-37: Containment retentions and environmental releases for Cases 7 and MU-12.1

Representative Element	Containment		Environment	
	7	MU-12.1	7	MU-12.1
Xe	1.6E-02	2.6E-04	9.8E-01	1.0
Cs	5.8E-01	5.1E-01	6.6E-02	1.4E-01
I	6.9E-01	6.2E-01	7.9E-02	1.8E-01
Te	6.7E-01	5.9E-01	7.5E-02	1.8E-01
Ba	2.2E-02	2.5E-02	2.2E-03	6.5E-03
Ru	1.8E-02	1.3E-02	1.6E-03	3.4E-03
Mo	8.1E-01	6.9E-01	8.0E-02	2.1E-01
Ce	1.6E-03	1.4E-03	1.2E-04	3.1E-04
La	4.3E-05	4.1E-05	4.2E-06	1.0E-05
UO2	2.6E-03	2.1E-03	2.8E-04	5.3E-04
Cd	6.8E-01	6.4E-01	7.7E-02	1.8E-01
Ag	6.6E-01	6.2E-01	6.9E-02	1.8E-01

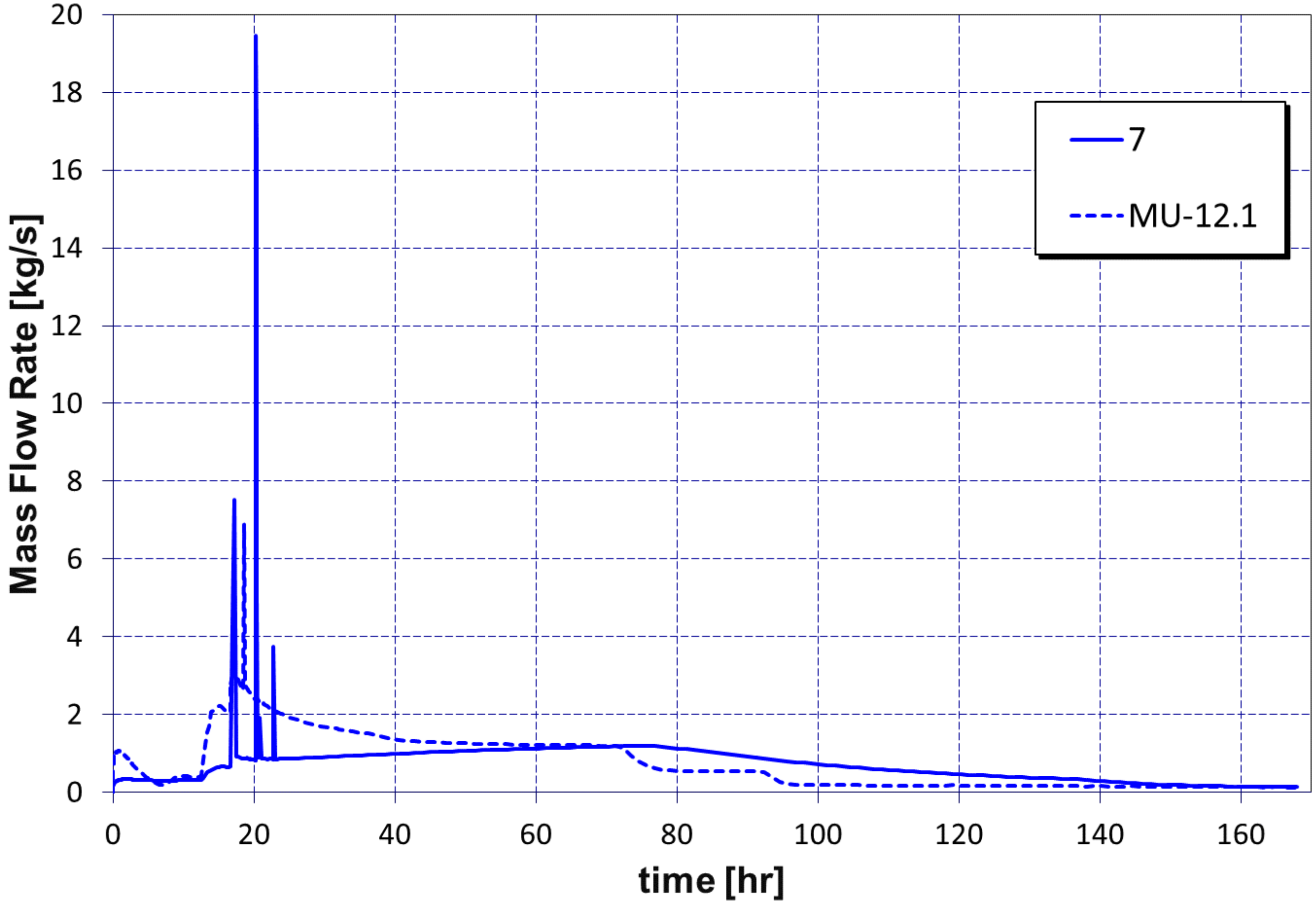


Figure 4-57: Total Mass Flow Rate Through the Containment Isolation Failure Path for Cases 7 and MU-12.1[6]

[6] The peaks in the mass flow rate correspond to instances where liquid instead of gas flows through the failure pathway.

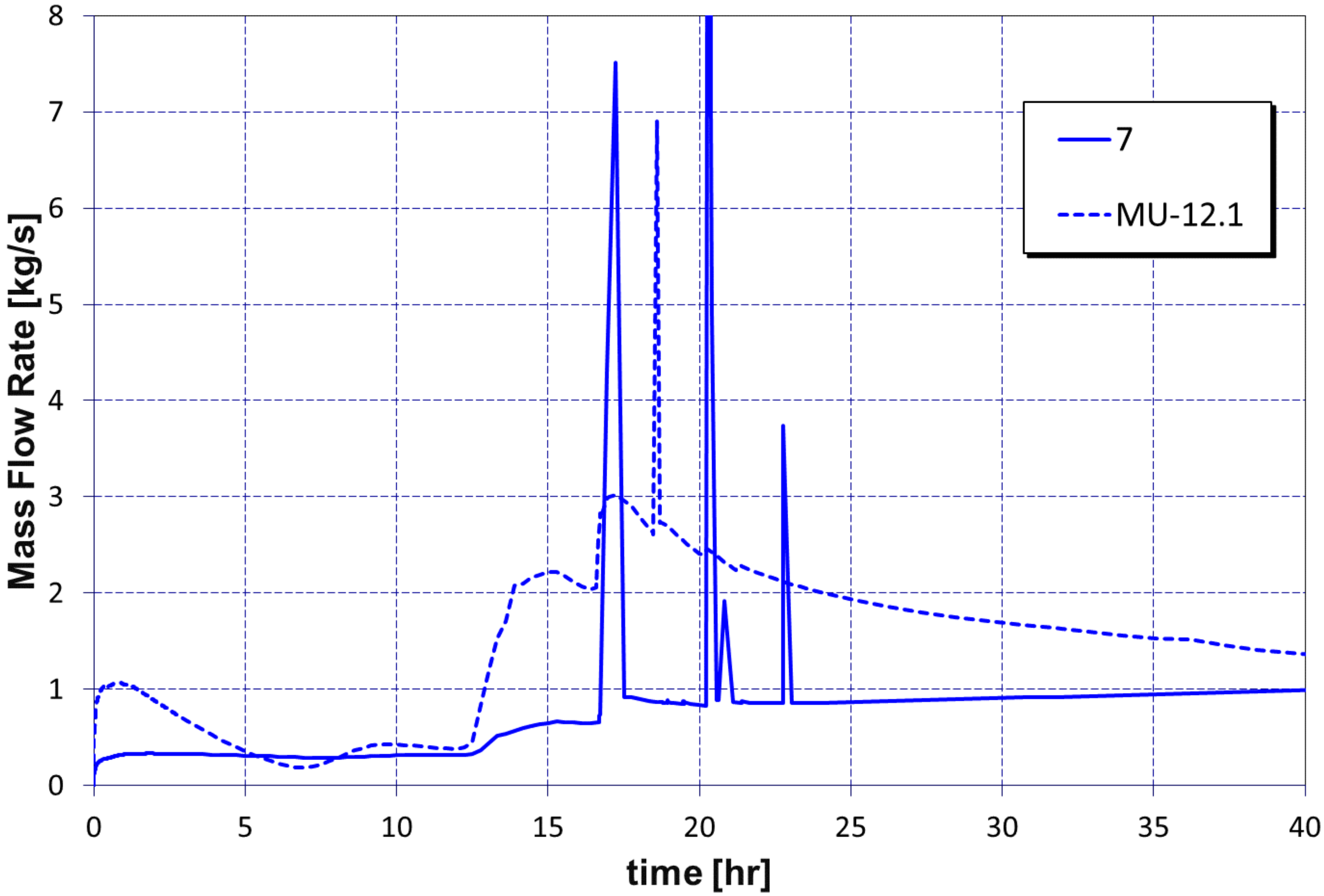


Figure 4-58: Total Mass Flow Rate Through the Containment Isolation Failure Path for Cases 7 and MU-12.1 – Detailed[6]

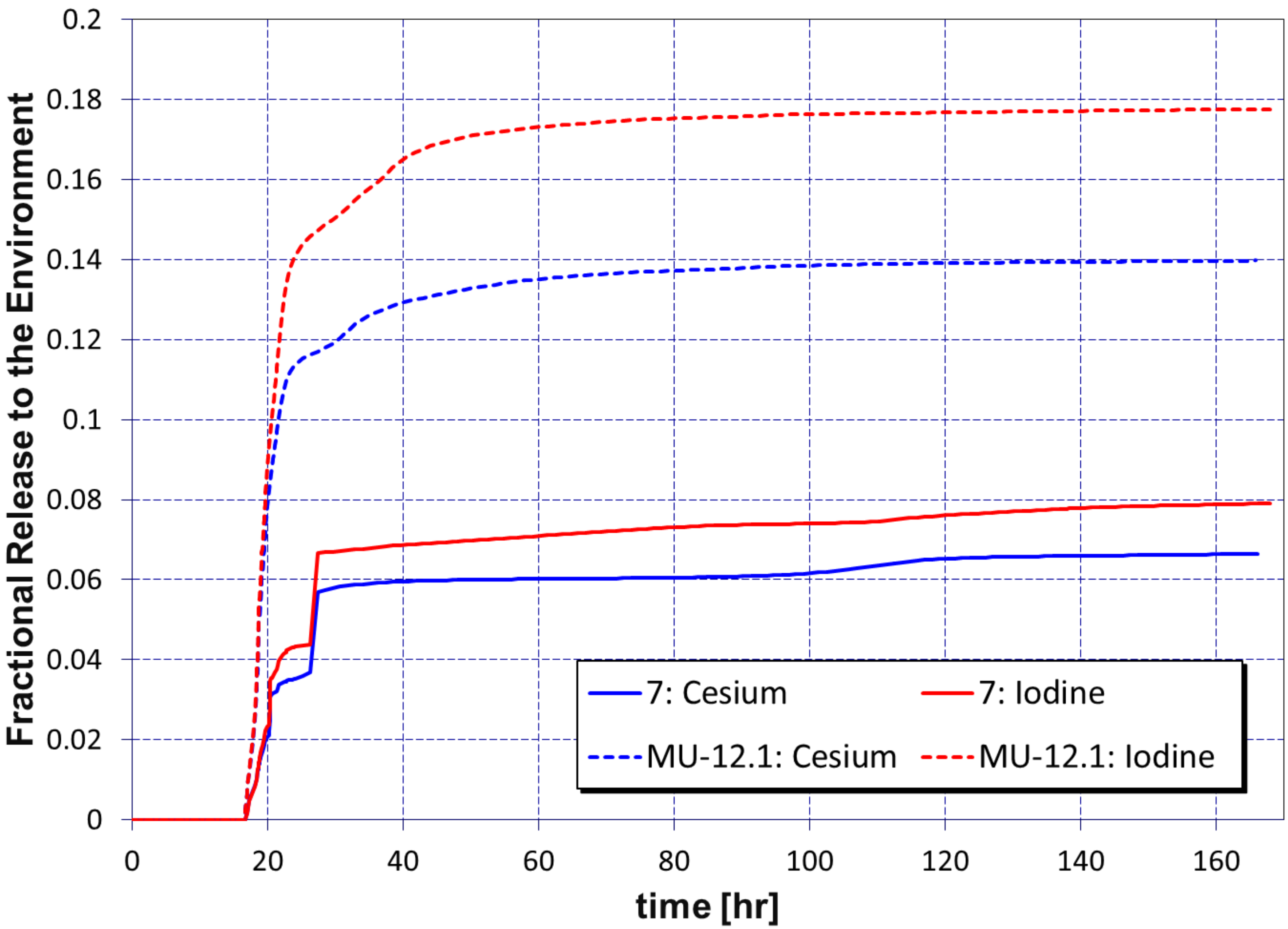


Figure 4-59: Fractional Releases to the Environment for Cesium and Iodine in Cases 7 and MU-12.1

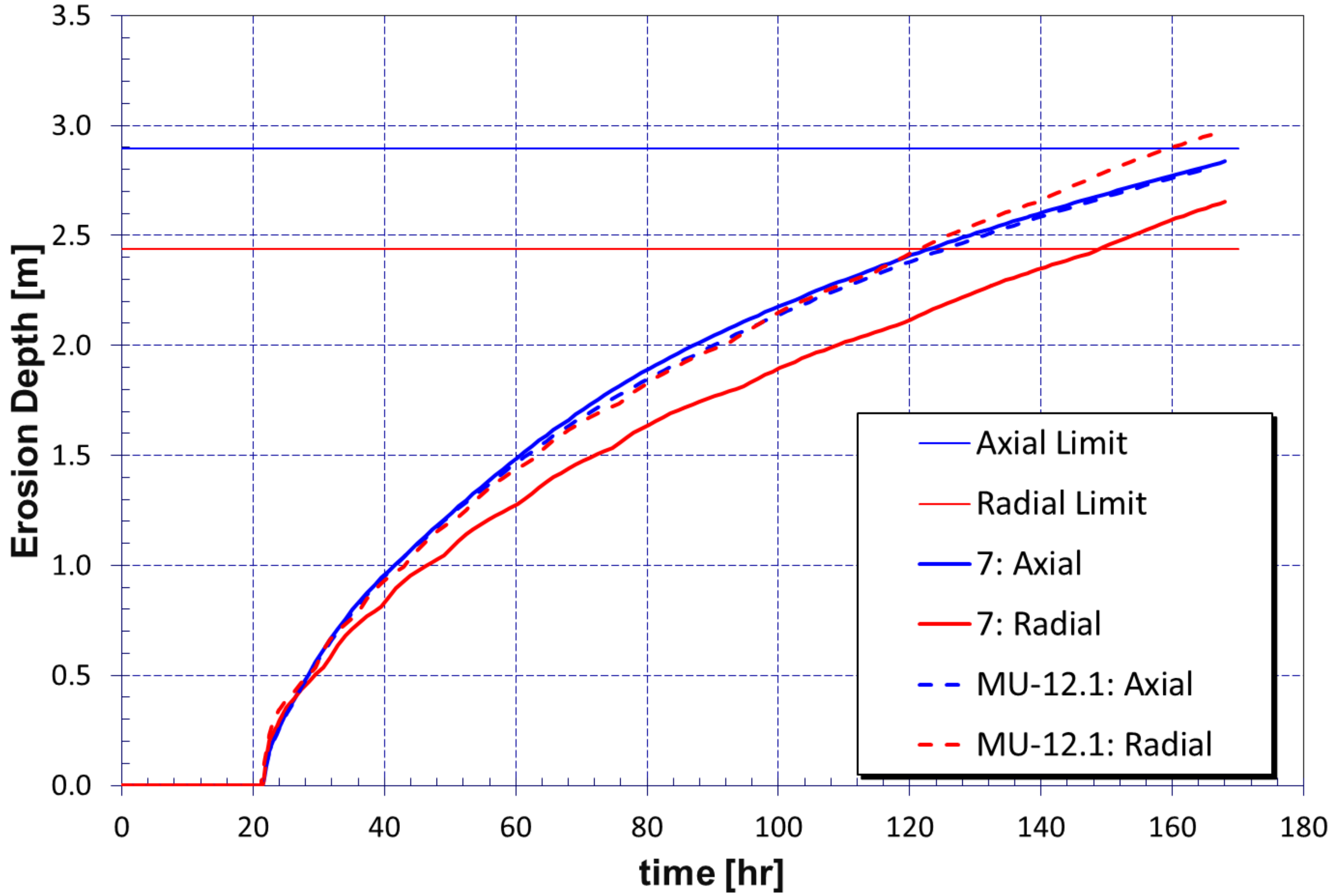


Figure 4-60: Axial and Radial Extent of Cavity Erosion in Cases 7 and MU-12.1

4.12.4 Sensitivity Analysis on Effect of Containment Isolation Failure Due to Pre-existing Maintenance Errors (MU-12.2)

Containment isolation failure (CIF) in the current model is dominated by the event 1-L2TEAR ("CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS"). Thus, the relative contribution of 1-REL-CIF and 1-REL-CIF-SC release categories to overall release frequency, LERF, and LRF, is heavily influenced by the failure probability for this event. To illustrate this effect, consider the following information:

	Contribution from 1-L2TEAR*	Contribution from all other containment isolation failures
1-REL-CIF	6.3E-8/yr	2.5E-9/yr
1-REL-CIF-SC	1.1E-11/yr	0

* CET sequence number 74 should have been assigned to the 1-REL-CIF-SC release category rather than the 1-REL-CIF release category, since in-vessel recovery occurs. The change would raise the 1-REL-CIF-SC frequency to $5x10^{-9}$/yr, while reducing the 1-REL-CIF frequency to ~$5.8x10^{-8}$/yr (the 1-CET-074 sequence frequency is ~$5x10^{-9}$/yr), discounting any minimization that would occur. This would not affect the release categories' percent contribution to overall release frequency (<0.1 percent and 0.1 percent, respectively).

From this, one can see that if 1-L2TEAR were one order of magnitude higher, the overall CIF frequency would increase to ~$6.3x10^{-7}$/yr. If 1-L2TEAR were one order of magnitude lower, the overall CIF frequency would decrease to ~$8.8x10^{-9}$/yr. The corresponding change to LERF and LRF, if the increased/decreased CIF frequency did not impact other release category frequencies contributing to LERF and LRF, would be small (as described below).

Since 1-REL-CIF does not meet the criteria for early fatalities LERF, there would be no change to this metric regardless of the change to 1-L2TEAR. With a one order of magnitude increase, early injuries LERF would double from 1 percent to 2 percent of overall release frequency. LRF would increase slightly from 14 percent to 15 percent for termination times of 36 hours and 60 hours, respectively, but would remain unchanged for a 7-day termination time. With a decrease by a factor of 10, there is no change to either the early injuries LERF or LRF values regardless of termination time.

This sensitivity demonstrates that for the current model results, the dominant contributor to containment isolation failure (1-L2TEAR) has a large effect on early injuries LERF, a small effect on LRF, and no effect on early fatalities LERF.

4.13 Release Pathway Modeling

4.13.1 Identified Uncertainties in Release Pathway Modeling

Table 4-38: Uncertainties in Release Pathway Modeling

Item	Description	Other comments
Elevation of a low-lying "normal containment leakage" flow path to auxiliary building	Can affect the extent to which fission products in the containment sump leak into the auxiliary building after the RWST has been injected; this is one of several normal leakage paths	This effect is most pronounced in cases with the RWST contents injected, and containment heat removal available, thus leading to situations with long periods of water slowly draining through this low-lying leakage path without other containment failure (see Cases 2R2, 6C, and 6D in Appendix B [sections 2.2, 6.4, and 6.5, respectively]).
Only airborne pathways are considered	Containment failure at a low elevation, containment bypass, containment isolation failure, containment basemat melt-through, and diversion of contaminated water all have the potential to lead to aqueous releases. These release pathways would lead to accident consequences (most likely to manifest themselves in onsite personnel exposures and accident cleanup costs) that are not accounted for here.	Contaminated water transiting to the tendon gallery and auxiliary building are captured within the MELCOR modeling domain, but no aqueous releases are passed to MACCS. This is also discussed in Section 1 of Appendix D.
Source term attenuation in structures outside the primary containment	Source term characterization for leakage pathways through adjoining structures, including: • Penetration leakage into the auxiliary building • Leakage through the tendon gallery (see Section 20 of Appendix D) • ISLOCA into the auxiliary building • Modeling of auxiliary building filtration, where applicable This affects the amount of radiological material retained by adjoining structures (versus how much is released to the environment).	For the L3PRA project, this is relevant to the equipment building, the main steam valve room, the tendon gallery, and the three tendon gallery access shafts. If penetration failure becomes more prominent, the control building and possibly the fuel handling building would also come into play. The ISLOCA issues are discussed in this document, in Section 2.1 of Appendix D, and in Section 5 of Appendix B.

Table 4-38: Uncertainties in Release Pathway Modeling

Item	Description	Other comments
In-vessel recovery (IVR) is binned with non-IVR in the release categorization for ECF, and IVR is not queried for ISLOCA and SGTR release categories	A potential conservatism in the modeling is that 1-REL-ECF releases are binned irrespective of in-vessel recovery, while containment bypass occurring prior to core damage does not query in-vessel recovery.	The impact of this simplification is expected to be small, in that 1-REL-ECF has a very small contribution to overall release frequency, while SGTR and ISLOCA sequences are still subject to large releases even if IVR occurs (depending on precisely when during the in-vessel melt progression it occurs).

4.13.2 Release Pathway Modeling Sensitivities

No sensitivity studies were conducted for release pathway modeling.

4.14 Other Fission Product and Emergency Preparedness-Related Modeling

4.14.1 Identified Uncertainties in Other Fission Product and Emergency Preparedness-Related Modeling

Table 4-39: Uncertainties in Other Fission Product and Emergency Preparedness-Related Modeling

Item	Description	Other comments
SCALE analysis uncertainties	This could have small effect systemically on the MELCOR analyses.	See SCALE sensitivity analyses in Section 2.3.1 of the main body of this report; although this uncertainty affects both decay heat and fission product inventory, it is grouped here.
SCALE analysis uncertainties specific to MOC/EOC assumption	This could have moderate effect systemically on the MELCOR analyses; BOC would have lower decay heat, whereas EOC would be similar to MOC; source terms would be expected to be lower for BOC and higher for EOC.	Current MELCOR model uses MOC; although this uncertainty affects both decay heat and fission product inventory, it is grouped here. The draft Surry SOARCA UA (SNL, 2016a) sampled time-at-cycle as BOC (7 days), MOC (200 days), and EOC (505 days) in the integrated UA. In addition, for each of these time-at-cycles, family of decay heat curves were sampled to represent deviations from the nominal decay heat curve.
Timely emergency action level (EAL) monitoring is assumed	Any impacts from delayed EAL escalation would not be captured; this includes delays with activating the TSC since this is assumed to happen at an ALERT.	The basis for this modeling assumption is described in Section 7 of Appendix B. Effects on consequence modeling fall within the scope of the Level 3 PRA (i.e., the consequence/risk analysis task).
Source term characteristics	This is a blanket uncertainty in (EPRI, 2012b) which covers release model, fission product transport and deposition, chemistry model, MCCI effects, uncertainties, revolatilization, etc.	These uncertainties are largely addressed by the numerous related items previously identified herein; note that the draft Surry SOARCA UA project identifies the chemical form of cesium as the key item in this category. Of the cesium remaining after it reacts with iodine to form CsI and after 4.65 percent of the remaining is introduced to the fuel gap as CsOH, the draft Surry SOARCA UA varied the fraction of cesium as cesium molybdate between 0 to 1, with mode of 0.8.

Table 4-39: Uncertainties in Other Fission Product and Emergency Preparedness-Related Modeling

Item	Description	Other comments
Assume all iodine combines with cesium to form CsI (i.e., initial gaseous Iodine fraction is 0)	The chemical form of the iodine released will impact the timing and extent of the environmental release.	The draft Surry SOARCA UA (SNL, 2016a) uses a surrogate release fraction based on the fission gas release to the gap. Experiments (Pontillon, 2005) estimated that it could vary from 0.2% to 7%. The draft Surry SOARCA UA varied the gaseous iodine percentage in the range of 0 to 3%.
Aerosol shape factor set to one (sphere)	The shape of a particle greatly affects its rate of condensation, agglomeration (growth), and deposition.	Little is known concerning the shape of particles during severe accidents, though they tend to be more spherical than chain-like (Kissane, 2008). The default MELCOR value is one. The draft Surry SOARCA UA (SNL, 2016a) varies the dynamic shape factor between 1 and 5.
Fire suppression system impacts on auxiliary building source term attenuation	Affects release characterization for ISLOCA and containment isolation failures. Actuation of the fire suppression system has the potential to promote fission product scrubbing, de-inert a steam-inerted area and cause combustion, and affect the performance of the charcoal filters (in the case of the filter fire suppression equipment).	Related to this, a combustion event may damage the fire suppression equipment, causing it to inadvertently actuate, or not actuate under valid conditions. Due to the large phenomenological uncertainties and the patchwork spatial arrangement of fire suppression components, the impact would be very difficult to characterize. The sustainability of a fire will be affected by high steam environments.

4.14.2 Alternative Treatment(s) of Uncertainties in Other Fission Product and Emergency Preparedness-Related Modeling

It was decided that no straight-forward sensitivities could be performed in the context of this report that would shed light on these issues. Rather, the reader is referred to the draft Surry SOARCA UA (SNL, 2016a). For instance, in that study it was found that for an unmitigated short-term SBO, regression analyses showed time-at-cycle, fraction of gaseous iodine, chemical form of cesium, and aerosol shape factor to be among the top six most influential varied input parameters for 48-hour iodine and cesium release magnitudes in non-SGTR realizations.

4.15 Accident Termination Modeling

4.15.1 Identified Uncertainties in Accident Termination Modeling

Table 4-40: Uncertainties in Accident Termination Modeling

Item	Description	Other comments
Modeling of offsite power recovery	Crediting offsite power recovery subsequent to core damage would have a yet-to-be-determined effect on the results. The subset of cases it applies to are discussed in Section 2.1.3 of the main body of this report. It has the positive effect of potentially returning mitigating systems to service, and the negative effect of increasing the likelihood of a severe combustion event due to equipment sparking or de-inerting of containment.	Additional considerations related to Level 1 PRA LOOP recovery data are also described in Section 2.4.1 of the main body of this report. Meanwhile, (Troll, 2015) provides additional information on applying power recovery in Level 2 PRA, and states that most Level 2 PRAs do not consider power recovery during the period of time between core damage and vessel failure (when it would be useful for potential in-vessel recovery). That paper provides an example where inclusion of power recovery makes a few percent difference in LERF; however, the example also assumes that re-alignment of systems and in-vessel degraded core recovery is assured and that containment challenge likelihood is not altered by the additional hydrogen production.
Impact from accident duration truncation of sequence runs	This affects the amount of radiological material released, as discussed further in Section 21 of Appendix D and (Helton, 2016).	This is related to the issue of treating offsite resources for accident mitigation. This project treats this issue fairly conservatively, adopting an across-the-board truncation time (e.g., 48 hours after accident initiation) would lead to smaller estimated releases.
*Related uncertainties regarding HRA modeling are captured in **Section 4.3**.*		

4.15.2 Alternative Treatment(s) of Uncertainties in Accident Termination Modeling

MU-15.1 A sensitivity analysis was performed in which the simulation truncation time was varied and its impact on a chosen definition of LERF and LRF was identified to explore the impact of simulation end-times.

4.15.3 Sensitivity Analysis on Impact of Simulation Truncation Time on the Risk Metrics LERF, LRF, and CCFP (MU-15.1)

Table 4-41 demonstrates the impact of simulation truncation time on the risk metrics LERF, LRF, and CCFP. The LERF metric is unaffected by the choice of end-time for the MELCOR simulation. The increases in LRF and CCFP are driven by the inclusion of the LCF and LCF-SC release categories (which account for nearly half the overall frequency). The risk surrogates for LCF and LCF-SC are provided by the MELCOR calculations S1B and S2R2, respectively, in which containment failure occurs at 47.9 hours and 120 hours, respectively.

For LRF, there is a measurable impact on the value from “60 hours after SAMG entry” to “7 days after SAMG entry” since both S1B (LCF) and S2R2 (LCF-SC) have not met the criteria for a “large” release by 60 hours (but do before 7 days). In the case of CCFP, this marked increase occurs earlier because the LCF release category experiences containment failure by “60 hours after SAMG entry,” whereas the LCF-SC release category does not experience containment failure until later.

Table 4-41: Risk Surrogate Results Given Varying Simulation End-Times

	Assumed Accident Termination Time[1,2]		
	36 hrs after SAMG entry	**60 hrs after SAMG entry**	**7 days after initiator**
LERF (early fatalities)	0.01	0.01	0.01
LERF (early injuries)	0.01	0.01	0.01
LRF	0.14	0.14	0.61
CCFP	0.18	0.60	0.65

[1] This refers to the time after SAMG entry (in the case of the first two categories), which can range quite a bit depending on the scenario. The third category is measured from the start of the accident, and always occurs well after the first two categories. In viewing these results, understand that limitations in the HRA and the phenomenological modeling make the longer-term results quite uncertain, and likely pessimistic.

[2] Values in the table are fractions of overall release frequency.

This analysis demonstrates the significance of selecting the simulation end-time, as well as the large impact of the LCF and LCF-SC categories on the LRF and CCFP metrics.

4.16 MELCOR Solution Robustness

4.16.1 Identified Uncertainties in MELCOR Solution Robustness

Table 4-42: Uncertainties in MELCOR Solution Robustness

Item	Description	Other comments
Miscellaneous threshold effects in plant response	The existence of thresholds in the plant response modeling (e.g., PRT rupture disk failure pressure, time-at-temperature structural failure of Ring 5 supporting structures) mean that trivial input changes sometimes result in notable output changes	These effects are numerous and unpredictable and are inevitable in this type of analysis. As an example of these effects, see Section 4.8.3 (MU-8.1).
Mass conservation errors introduced by interference between the flashing and hygroscopicity models in MELCOR	These models are known to have some issues with respect to conserving mass when activated simultaneously. Conversely, disabling either model removes some viable physics from the calculation. This is true for the version of the code used in the present analysis.	Activating the hygroscopic model tends to increase the aerodynamic mass median diameter in containment when the atmosphere is not saturated. Otherwise, activating this model has little effect on the calculation. Turning off the flashing model affects the partitioning of water between the pool and atmosphere in flow paths in which the flashing model had been active in previous revisions (i.e., pipe break, seal leakage, and relief valve flow paths). The hygroscopic/flow path flashing models were corrected in Revision 2.1.8611 of the MELCOR code.
Numerical variance caused by minor changes in user-defined simulation parameters	Small changes to input parameters (e.g., time step or flow-path shuffling) can cause perturbations in the calculation that lead to differences in otherwise identical calculations.	The “noise” in MELCOR calculations has long been acknowledged and the uncertainties are not additive in nature. Efforts to both characterize and reduce this inherent variance are currently being done by developers at Sandia National Laboratories (Humphries, 2016).

4.16.2 Alternative Treatment(s) of Uncertainties in MELCOR Solution Robustness

As is discussed above, some level of uncertainty is inherent to MELCOR itself (and any other complex code of this type). A source of noise can be traced to the matrix solver for the flow path calculation, which can then be amplified by various physics models during core degradation, as well as bifurcations in accident progression paths. In addition, code revisions and modeling improvements (e.g., reflood quench) often lead to variations in the output parameters (such as in-vessel hydrogen production). The nature of these uncertainties and their impact on calculations is not explored further here.

4.16.3 MELCOR Solution Robustness Sensitivities

No sensitivity studies were conducted specifically for the L3PRA project.

5. Summary and Conclusions

Parameter uncertainty distributions have been defined for all Level 2 PRA basic event parameters, and these parameter uncertainties have been simultaneously propagated through the model with the Level 1 PRA basic event parameters. Results are presented in **Section 3** for individual release category frequencies, total release frequency, LERF, and LRF.

In addition, dozens of model uncertainties have been identified and discussed. For a subset of these, sensitivity analyses have been performed. **Table 5-1** provides a summary of these sensitivity analyses. It is important to understand that these are summaries of individual sensitivity analyses for the reference plant/PRA and not broad statements about the sensitivity to given model uncertainties. For applicable MELCOR re-calculations, the cumulative release fraction of cesium and iodine to the environment (typically at 7 days after the initiating event) is given in the rightmost columns, along with that of the associated base case(s). The authors place less significance in changes to cumulative release fractions when they are less than roughly 1 percent of the initial radionuclide inventory. In this low-end range, changes can be thought of as a combination of actual influences versus indirect and unrelated effects (analogous to “noise” in signal processing). For applicable SAPHIRE calculations, a summary of the results is provided.

Figure 5-1 and **Figure 5-2** show the release fractions (blue bars) of iodine and cesium for each of the MELCOR sensitivity re-calculations. Also shown is the factor (orange dots) by which these values differ from the base calculation (sensitivity/base value). Factors higher than one denote instances where the sensitivity produces a larger release than its associated base case, while factors less than one denote the opposite. This compilation of sensitivity results gives some indications that the baseline MELCOR results used to develop the Level 2 PRA (and define the representative source terms) may exhibit a general tendency of under-predicting iodine releases and over-predicting cesium releases. However, given the limited number of results and the general expected correlation between iodine and cesium releases, this is not judged to be a robust conclusion.

Given the results of these sensitivities (as well as the insights gained from the myriad of past and ongoing severe accident studies), the central tendency of the cumulative MELCOR release fractions can reasonably be expected to vary within a factor of 3 (for those values greater than ~1 percent as discussed above). This is not to say that adjusting a single parameter would not alter the release fraction by more (or less), and this is, in fact, illustrated by the iodine releases in MU-5.1B. Rather, it recognizes that individual changes only affect some aspects of the results, and that multiple simultaneous changes have the potential to either exaggerate or diminish the overall changes to the results. A different means of expressing the same general point would be: given the same state-of-knowledge, accident simulation truncation time, and initial conditions, results obtained by a different user or obtained by applying other credible modeling assumptions, would not be expected to change by more than a factor of 3.

For relevant release categories, **Figure 5-3** through **Figure 5-16** plot the environmental release fractions for all source terms in that category (including sensitivities). The source term chosen as the representative source term is marked with an asterisk (e.g., 1B*). These figures show a mix of outcomes, including cases where:

- The sensitivity results are enveloped by the pre-existing cases (1-REL-V-F-SC, 1-REL-NOCF)
- The sensitivity results show larger releases than the highest of the pre-existing cases (1-REL-ECF for iodine, 1-REL-LCF for iodine, 1-REL-ICF-BURN for iodine, 1-REL-CIF)

- The sensitivity results show smaller releases than the lowest of the pre-existing cases (1-REL-ISGTR, 1-REL-LCF, 1-REL-ICF-BURN for cesium)

For the sensitivity cases run here, relative to the pre-existing cases used in developing the Level 2 PRA, iodine releases are more commonly outliers than are cesium releases.

Several modeling uncertainties are worth highlighting as being important, and these generally comport with uncertainties that have been found to be important in other contemporary severe accident studies:

- Significant changes in the Level 1 PRA failure-to-run modeling assumptions (e.g., extended battery life during station blackout)
- The timing of primary-side relief valve failure and realistic modeling of PRT behavior, in that they can be very important in terms of cumulative iodine release, if it is proximate to the time of containment failure
- Other modeling assumptions (e.g., accumulator modeling) that significantly affect the in-vessel melt progression
- Induced failure of containment heat removal systems, and the subsequent effect on preventing late containment failure
- The location and size of containment failure, in that failure to the auxiliary building rather than directly to the environment can reduce the release, while the size of the failure (particularly in the case of energetic failures and isolation failures) can also have a prominent affect
- The timing of SG tube and hot leg nozzle creep rupture for severe accident-induced SGTR, as well as uncertainty related to secondary-side retention (e.g., in the dryers and separators) of fission products in all SGTRs
- Uncertainties in ISLOCA modeling, in that several choices can impact the release including the initial break size, whether the break is covered, turbulent deposition in the piping, and downstream effects on auxiliary building status

Another facet of uncertainty not touched upon elsewhere in this document is that of user effect (i.e., the influence that an analyst has by virtue of the many assumptions that are made in the conduct of a given analysis), which may be as, if not more, important than model uncertainty. International Standard Problems (ISPs) (NEA, 2000) provide a good example of a sub-component of this uncertainty, in terms of showing variability amongst multiple analysts using similar tools to study what is ostensibly the same problem.

Table 5-1: Recap of Model Uncertainty Sensitivity Analyses				
#	**Sensitivity Description**	**Qualitative Outcome of the Sensitivity Analysis**	**Quantitative Outcome of the Sensitivity Analysis (as appropriate)**	
MU-1.1 A&B	SAPHIRE calcs	This sensitivity showed that the ambiguities in PDS logic that were flagged as indeterminate in the PDS formulation have virtually no impact on the final release category frequency results. This lack of sensitivity is influenced by the dominance of SBO and loss of NSCW in the Level 1 PRA results, along with the Level 2 HRA's lack of credit for post-core-damage actions during station blackouts.		
MU-1.2	Recalculation of Case 1A in which the safety-related battery life is extended from 4 to 13 hours	Earlier/later battery depletion time has a somewhat linear impact on the timing of containment failure and environmental releases.	Iodine Release Frac: 8.8E-4 (Case 1A: 3.7E-3)	Cesium Release Frac: 1.0E-3 (Case 1A: 4.2E-3)
MU-1.3	Hand calculations to show how the baseline release category profile might differ if the frequency associated with CET sequence #1 was routed through the CET	This sensitivity showed that alternate assumptions regarding the treatment of long-term blind feeding SBO scenarios could cause a large shift from 1-REL-NOCF to 1-REL-LCF. Nevertheless, this large shift is caused by effectively applying containment pressurization rates over a time period that is not indicative of the extended blind-feeding scenarios. Thus, the sensitivity results are not as defensible as the baseline results. Separately, a small increase in 1-REL-ISGTR was estimated.		
MU-2.1	Recalculation of Case 6 with containment fan coolers assumed to fail at the time of a sizeable hydrogen combustion at 15.7 hours	Containment fails due to lack of cooling resulting in a larger release. This demonstrates that susceptibility to combustion-induced damage of the containment fan coolers would tend to shift release category frequency from the BMT release category to the LCF release category (represented by Case 1B).	Iodine Release Frac: 3.2E-2 (Case 1B: 1.2E-2)	Cesium Release Frac: 9.7E-3 (Case 1B: 9.9E-3)
MU-2.2	Recalculation of Case 6R1 with reduced flow of RHR assuming clogging of alternative source of water	This demonstrates that the in-vessel recovery probability is not sensitive to the available injection flow rate, within the range of flow rates considered.	Iodine Release Frac: 1.6E-5 (Case 6R1: 1.4E-5)	Cesium Release Frac: 1.37E-5 (Case 6R1: 1.2E-5)
MU-3.1 A&B	Two SAPHIRE calculations with all Level 2 post-core-damage HEPs moved either up or down	These sensitivities demonstrated that significant systematic changes to the HEP values has a notable effect on the 1-REL-ICF-BURN, 1-REL-ICF-BURN-SC, 1-REL-LCF, 1-REL-LCF-SC, and 1-REL-NOCF release category frequencies.		
MU-3.2	Hand calculations to show how long-term recovery might affect LERF, LRF, and CCFP	This sensitivity demonstrate that reliability of recovery actions would need to be on the order of one decade for each of the three classes of recovery (controlling combustion, controlling long-term containment pressure, flooding the cavity and preventing basemat melt-through) to counter-act the effects of longer accident simulation times.		
MU-4.1	Recalculation of Case 2 with accumulator injection rate increased	There is a notable impact on event timing but little impact on environmental release.	Iodine Release Frac: 9.3E-4 (Case 2: 2.0E-3)	Cesium Release Frac: 9.8E-4 (Case 2: 4.2E-3)

MU-4.2	Recalculation of Case 1B2 with the PRT modeled as non-adiabatic	There is a notable impact on environmental releases for this scenario, but it is not expected to impact other calculations.	Iodine Release Frac: 1.8E-2 (Case 1B2: 8.2E-3)	Cesium Release Frac: 7.3E-3 (Case 1B2: 4.3E-2)
MU-4.3 A&B	Two recalculations of Case 1A2 with the break-out and eutectic temperatures increased or decreased	These two sensitivities show that the selected parameters have a measurable effect on in-vessel behavior, but the effect on cumulative environmental releases is modest.	Iodine Release Frac: 4.8E-2 and 4.0E-2 (Case 1A2: 4.3E-2)	Cesium Release Frac: 1.9E-2 and 2.8E-2 (Case 1A2: 3.2E-2)
MU-5.1 A&B	Two recalculations of Case 1A2 with the timing of SRV failure varied	The timing of valve failure during core damage can significantly alter the time and nature of release to the containment. If the timing corresponds to the time of containment failure, then the environmental release can also be affected.	Iodine Release Frac: 1.0E-1 and 1.9E-1 (Case 1A2: 4.3E-2)	Cesium Release Frac: 1.2E-2 and 2.9E-2 (Case 1A2: 3.2E-2)
MU-6.1	Recalculation of Case 2A with RPV failure mode changed from vessel melt-through to penetration failure	The timing of RPV failure is shifted slightly and fission product releases are not significantly affected.	Iodine Release Frac: 1.5E-1 (Case 2A: 1.5E-1)	Cesium Release Frac: 1.5E-1 (Case 2A: 1.6E-1)
MU-6.2 A&B	Two recalculations of Case 3A4 where varying hole sizes are assumed at the time of RPV failure	The posited low-probability events (and thus low-frequency PRA sequences) involving early energetic containment failure lead to significantly larger environmental releases than the situation where early energetic containment failure events do not occur. This would move release category frequency from the LCF release category (represented by Case 1B) to the ECF release category (represented by Case 2A).	Iodine Release Frac: 8.7E-2 and 2.7E-1 (Case 3A4: 1.5E-2) For comparison to the representative case for ECF, Case 2A: 1.5E-1	Cesium Release Frac: 3.7E-2 and 1.1E-1 (Case 3A4: 9.9E-3) For comparison to the representative case for ECF, Case 2A: 1.6E-1
MU-7.1	Multiple recalculations to Cases 6B and 1A to explore the maximum containment pressure due to hydrogen deflagrations	This demonstrates qualitatively that results of ERPRA-BURN for the Case 6B are supported. Namely, in phase I and II, a hydrogen combustion large enough to over-pressurize and fail containment is unlikely. For phase III, a hydrogen combustion is not only likely, but without prior burns, could pressurize containment on the upper end of what was predicted by ERPRA-BURN and could severely challenge containment.		
MU-8.1	A recalculation of Case 1B2 where the containment overpressure release path is directed entirely to the auxiliary building	With the removal of the direct pathway to the environment via the tendon gallery, there is a significant decrease in the environmental release fraction (though only part of this is directly attributable to the release pathway).	Iodine Release Frac: 2.6E-3 (Case 1B2: 8.2E-3)	Cesium Release Frac: 8.0E-4 (Case 1B2: 4.3E-2)
MU-8.2	A recalculation of Case 1B2 where the containment leakage is decreased by 35%	This demonstrates that the choice in leak rate has little effect on the rate of pressurization and	Iodine Release Frac: 2.3E-2 (Case 1B2: 8.2E-3)	Cesium Release Frac: 1.6E-2 (Case 1B2: 4.3E-2)

		the timing of over-pressurization, though it does have indirect effects on other results.		
MU-8.3	A discussion on the possible decontamination factors for aerosol deposition in containment failure pathways considering Case 3A3	This demonstrates qualitatively that turbulent deposition of fission products in the containment over-pressure failure path is small compared to other modes of deposition. The approximate DF is roughly calculated to be 1.02 in the crack pathway and 1.01 in the tendon gallery shaft.		
MU-10.1	A recalculation of Case 5B with the ISLOCA break size set to 6 inches (vs. 8 inches in Case 5B)	This demonstrates that ISLOCA break size has a large impact on the magnitude of estimated environmental releases.	Iodine Release Frac: 8.2E-2 (Case 5B: 1.2E-1)	Cesium Release Frac: 6.4E-2 (Case 5B: 9.2E-2)
MU-11.1	A recalculation of Case 3A3 with tube rupture occurring 15 minutes prior to hot leg nozzle creep rupture	This demonstrates that the timing of steam generator tube rupture in relation to that of hot leg nozzle creep rupture has a large impact on where fission products are retained and the extent of the release to the environment.	Iodine Release Frac: 5.7E-2 (Case 3A3: 7.6E-2 and Case 3A2: 2.3E-1)	Cesium Release Frac: 3.1E-2 (Case 3A3: 3.8E-2 and Case 3A2: 9.2E-2)
MU-11.2	Hand calculations in which the Powers model for aerosol deposition in the SG is applied in two ways	This demonstrates that in the case of a steam generator tube rupture, more deposition than is currently modeled in MELCOR could occur in the SG dryers and separators (for most chemical classes).	Iodine Release Frac: Between 1.8E-1 and 3.5E-1 (Case 3A2: 2.3E-1)	Cesium Release Frac: Between 2.0E-2 and 6.3E-2 (Case 3A2: 9.2E-2)
MU-12.1	A recalculation of Case 7 with a 4 inch rather than 2-inch containment isolation failure	This demonstrates that containment isolation failure size can have a large impact on the magnitude of the environmental release.	Iodine Release Frac: 1.8E-1 (Case 7: 7.9E-2)	Cesium Release Frac: 1.4E-1 (Case 7: 6.6E-2)
MU-12.2	Explores the effect of the dominant contributor to containment isolation failure (1-L2TEAR) by increasing and decreasing the 1-REL-CIF and 1-REL-CIF-SC release categories	Increasing the 1-L2TEAR probability by an order of magnitude increased LRF and LERF (early injuries) by at most 1 percent of overall release frequency. This is a doubling for early injuries LERF and a small change for LRF. Decreasing them had no measurable impact on either.		
MU-15.1	Explores the impact of simulation end-time on the LERF, LRF, and CCFP metrics	This demonstrates the significance of selecting the simulation end-time, as well as the large impact of the LCF and LCF-SC categories on the LRF and CCFP metrics.		

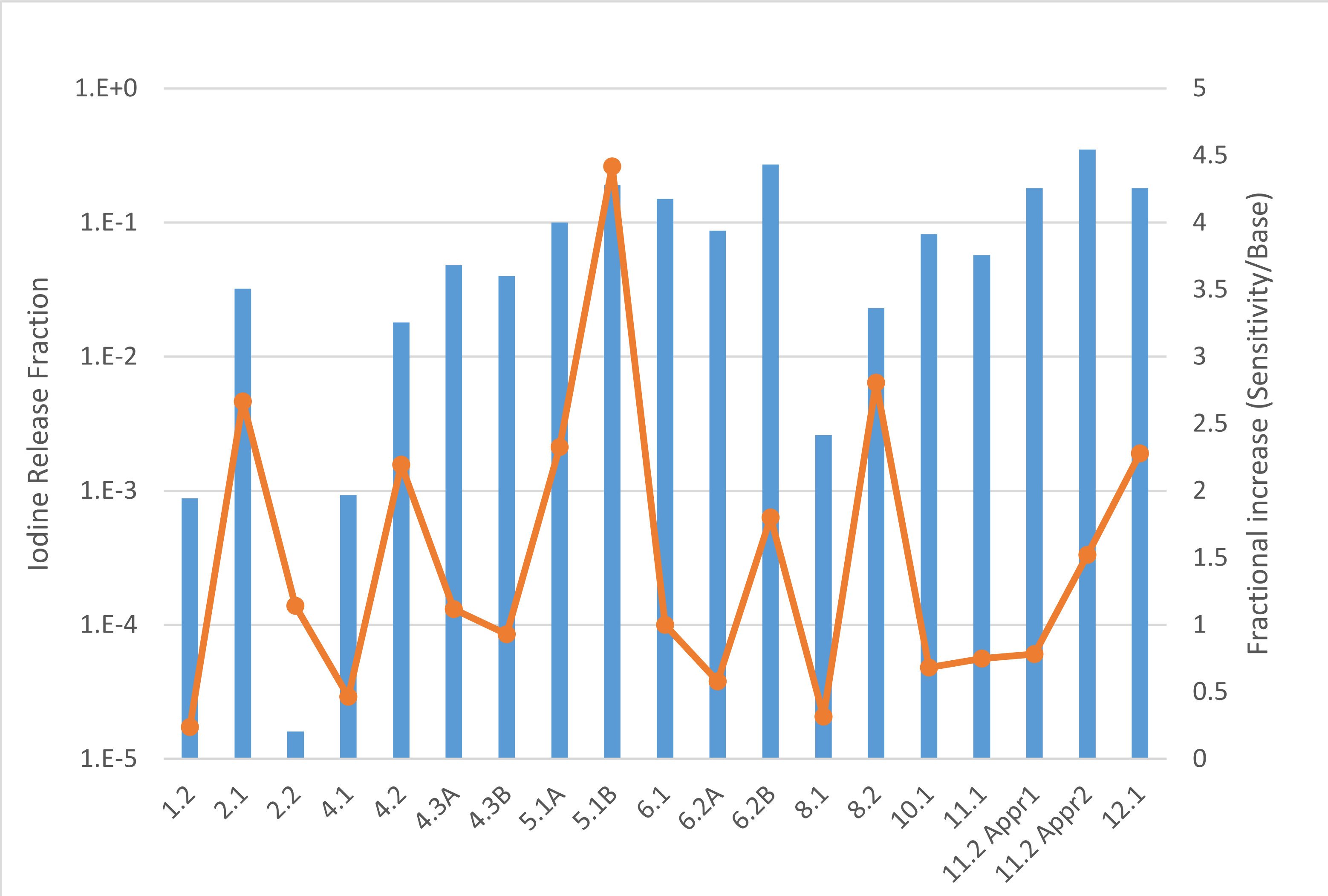


Figure 5-1: Summary of the Environmental Releases for All Sensitivity Cases - Iodine

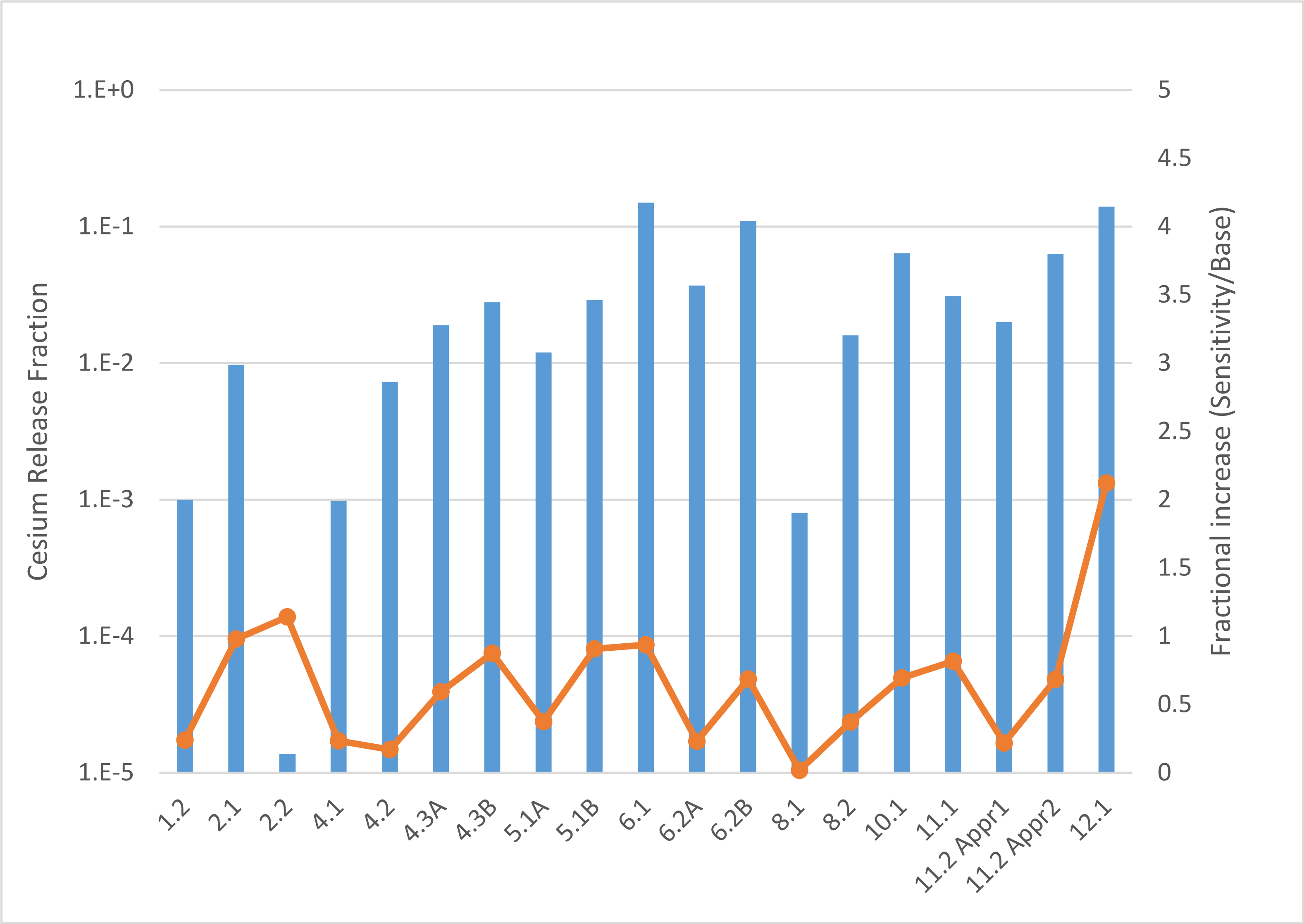


Figure 5-2: Summary of the Environmental Releases for All Sensitivity Cases - Cesium

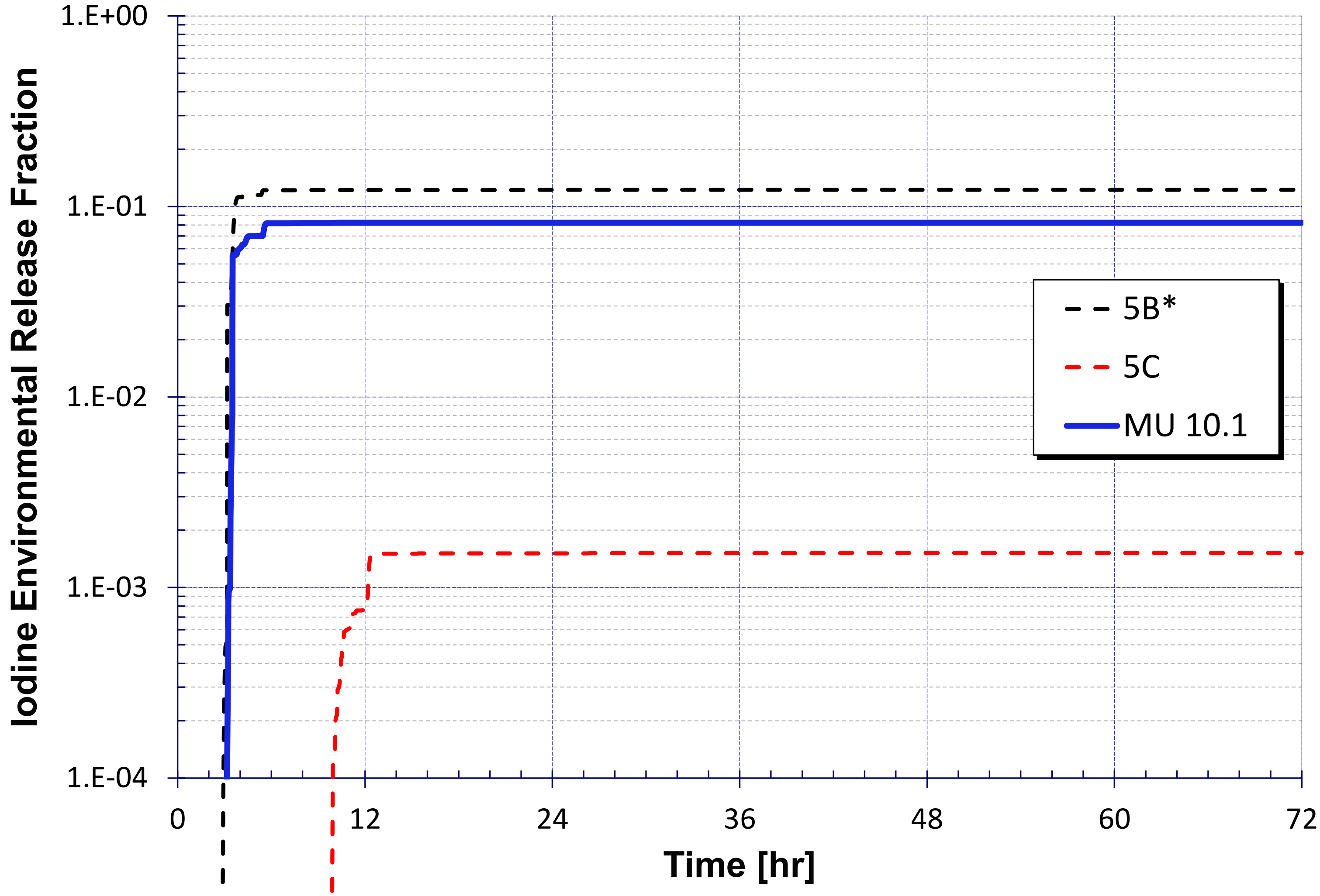


Figure 5-3: Fractional Release of Iodine to the Environment for 1-REL-V-F-SC Calculations

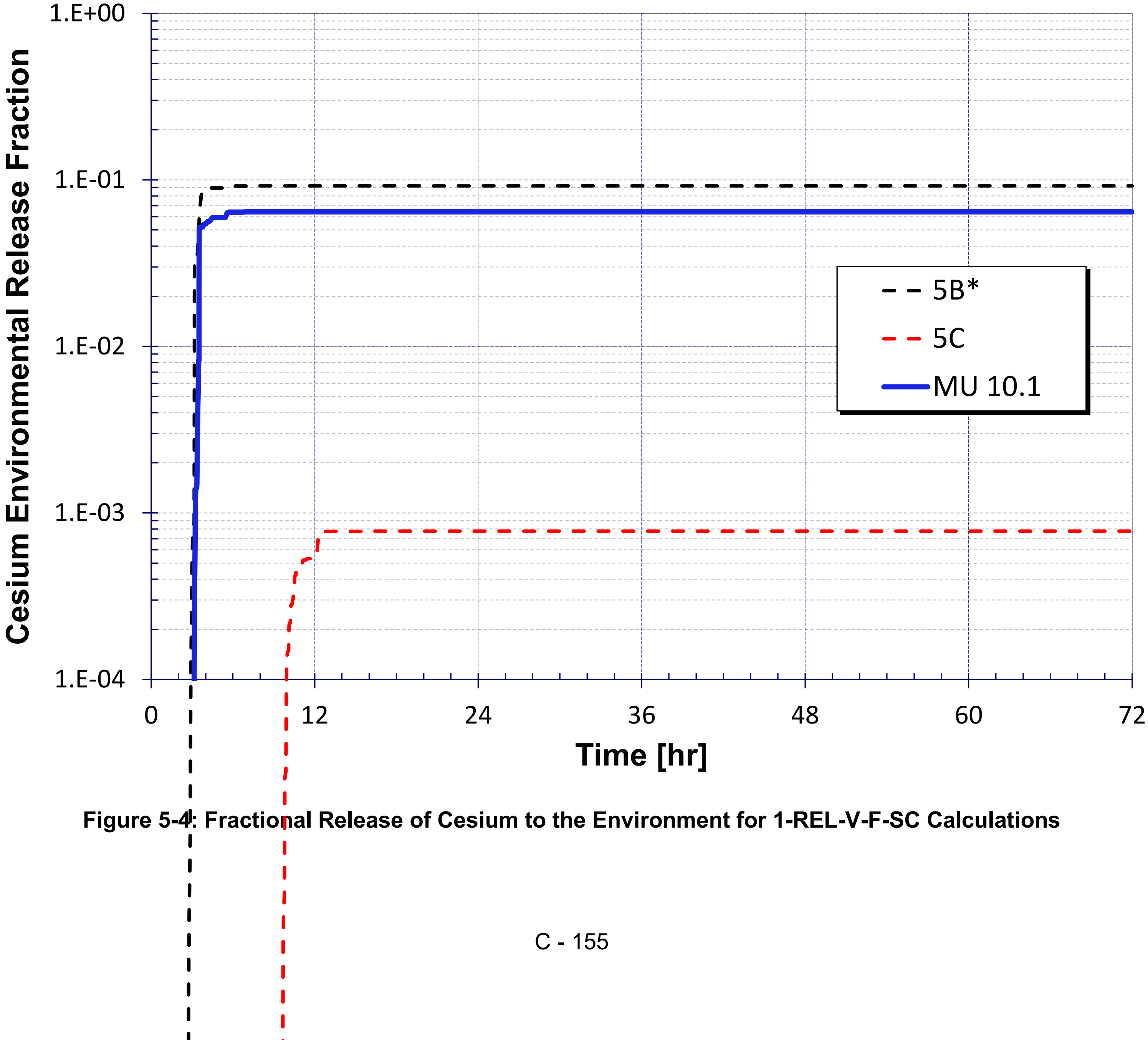


Figure 5-4: Fractional Release of Cesium to the Environment for 1-REL-V-F-SC Calculations

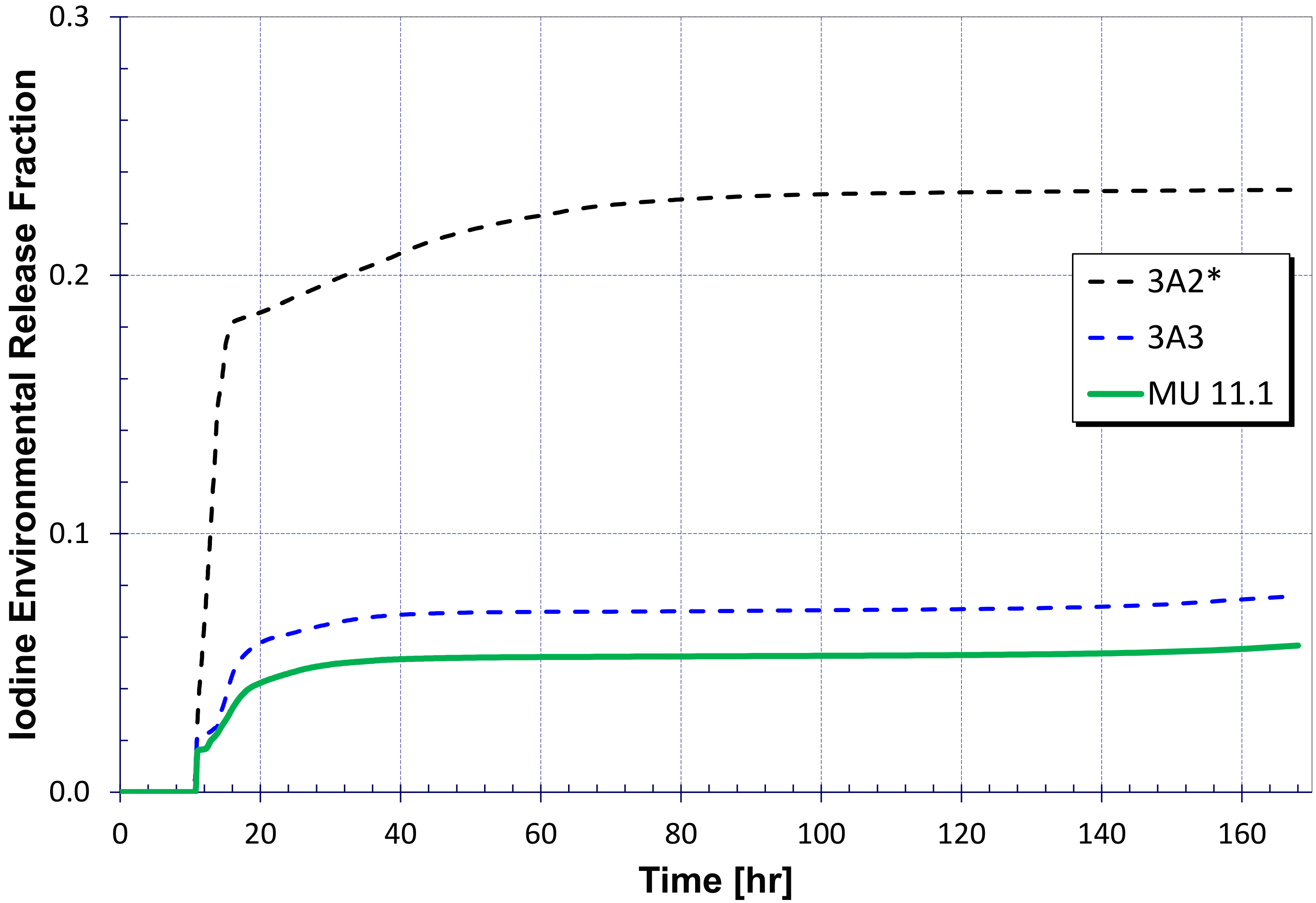


Figure 5-5: Fractional Release of Iodine to the Environment for 1-REL-ISGTR Calculations

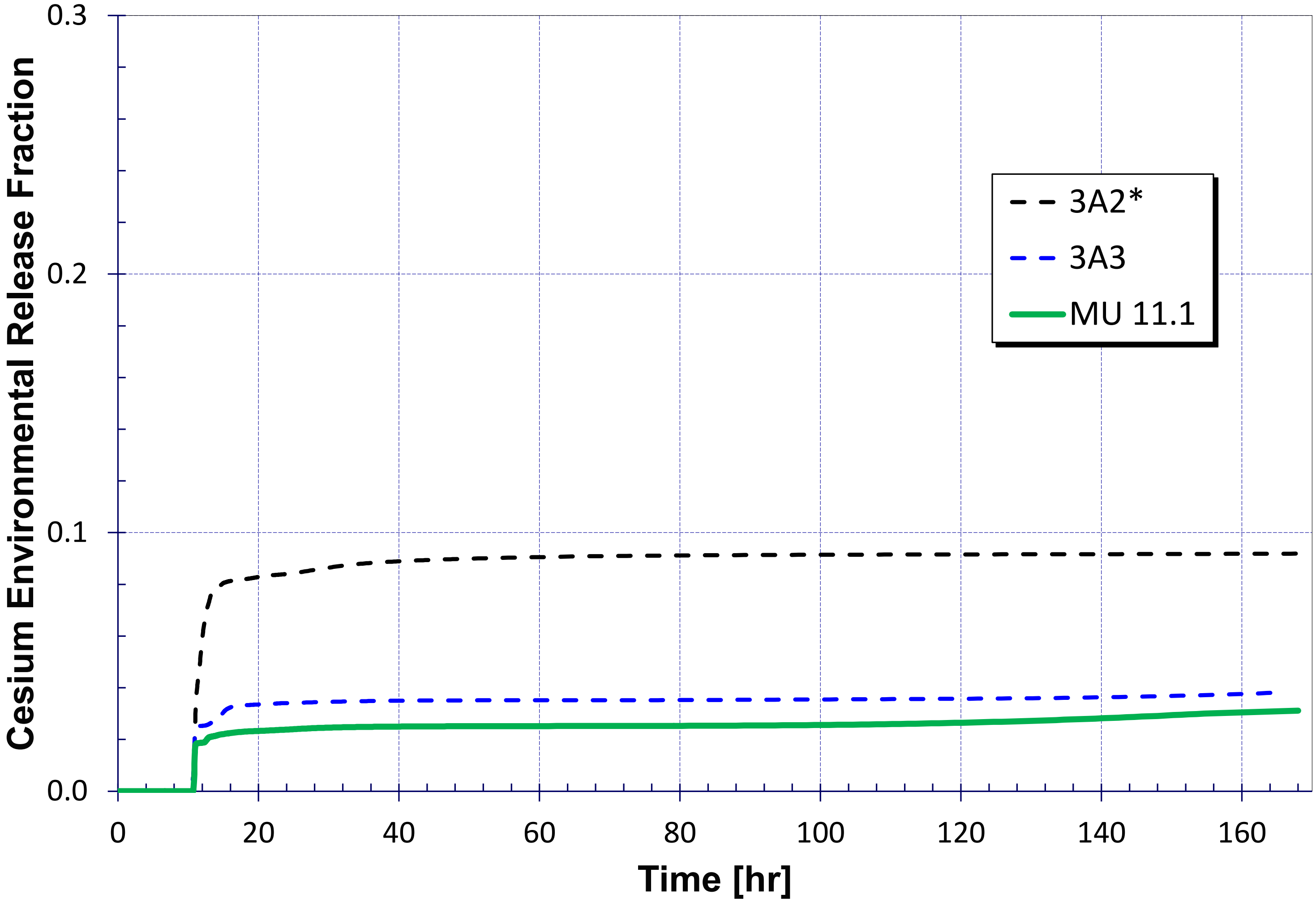


Figure 5-6: Fractional Release of Cesium to the Environment for 1-REL-ISGTR Calculations

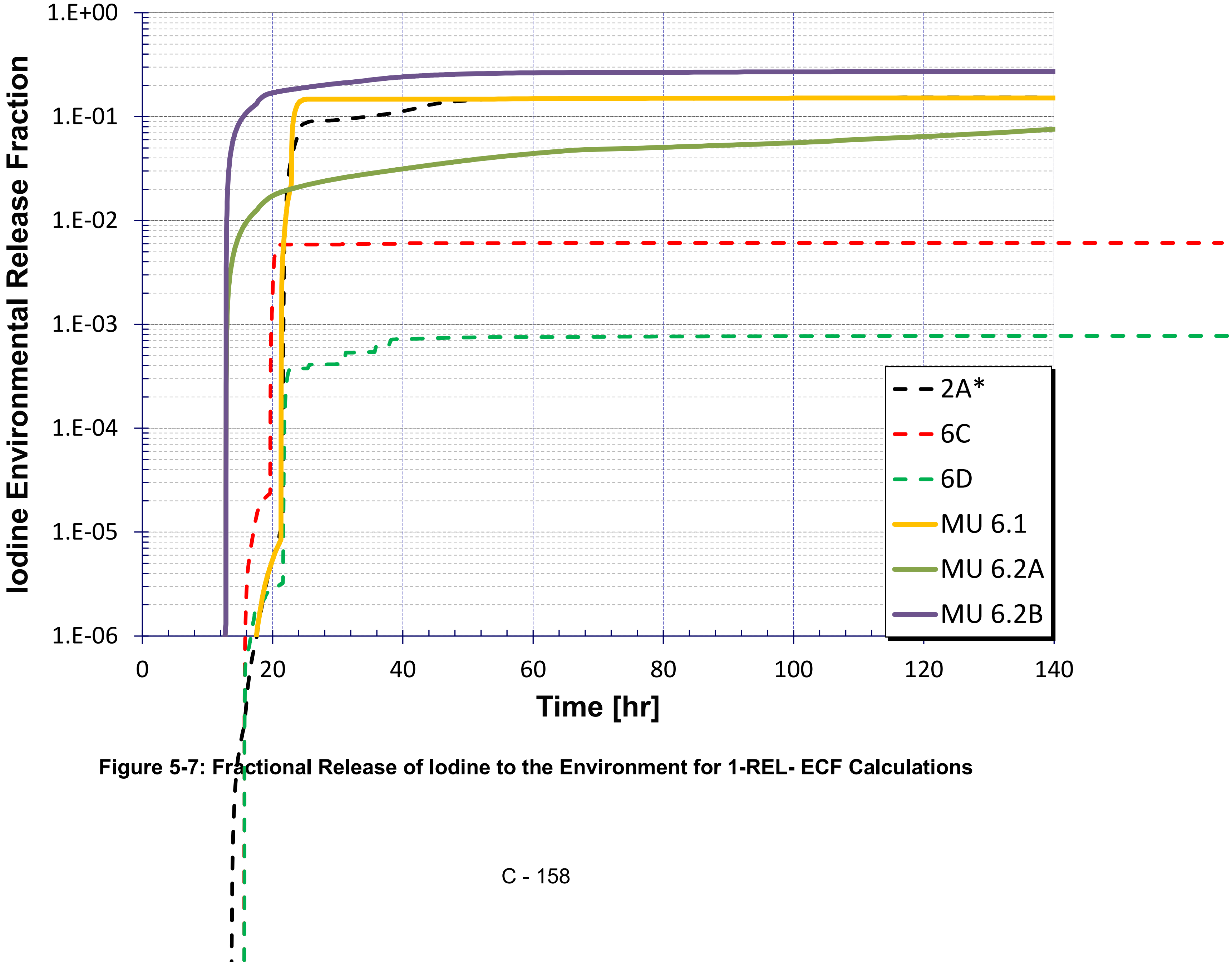


Figure 5-7: Fractional Release of Iodine to the Environment for 1-REL- ECF Calculations

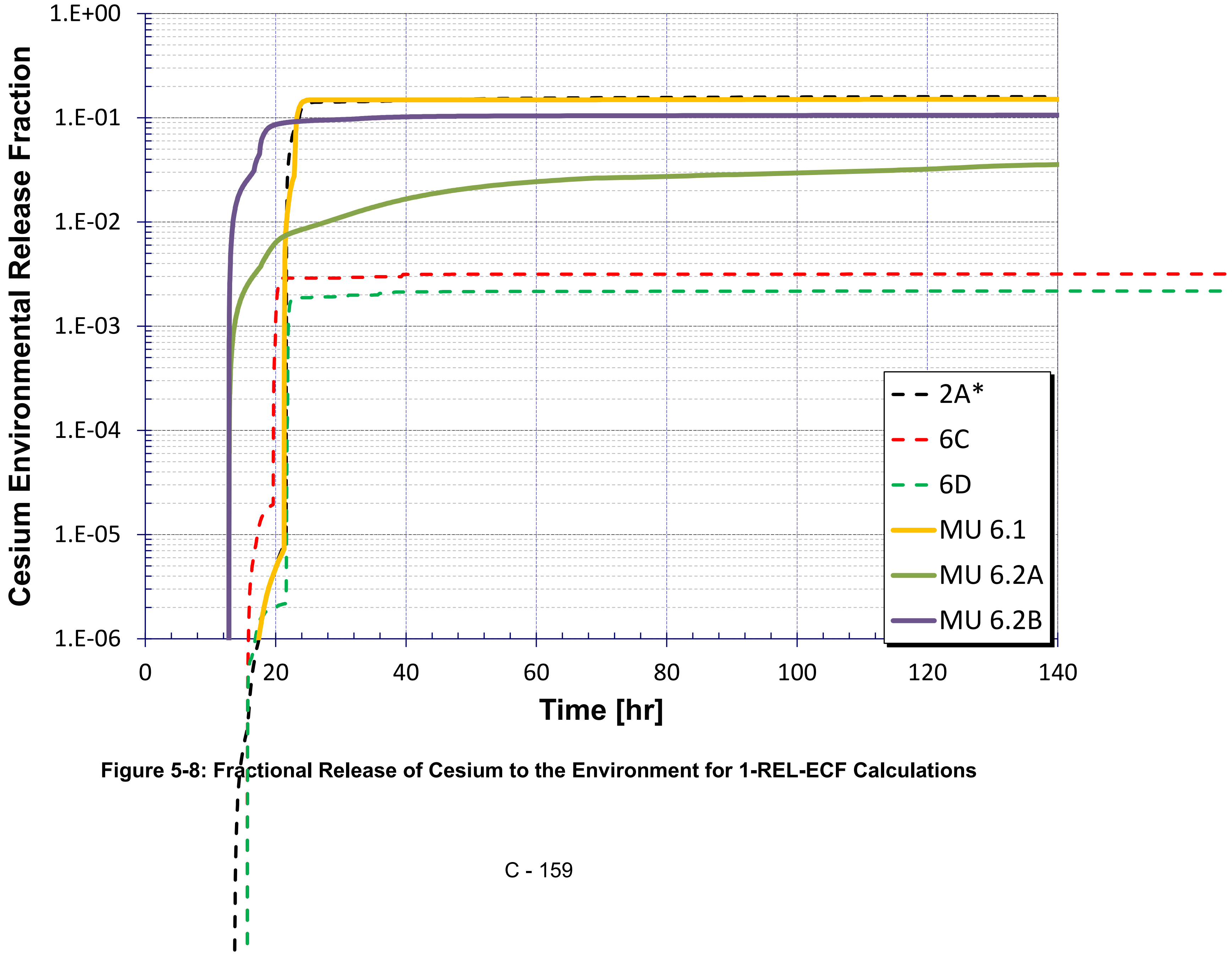


Figure 5-8: Fractional Release of Cesium to the Environment for 1-REL-ECF Calculations

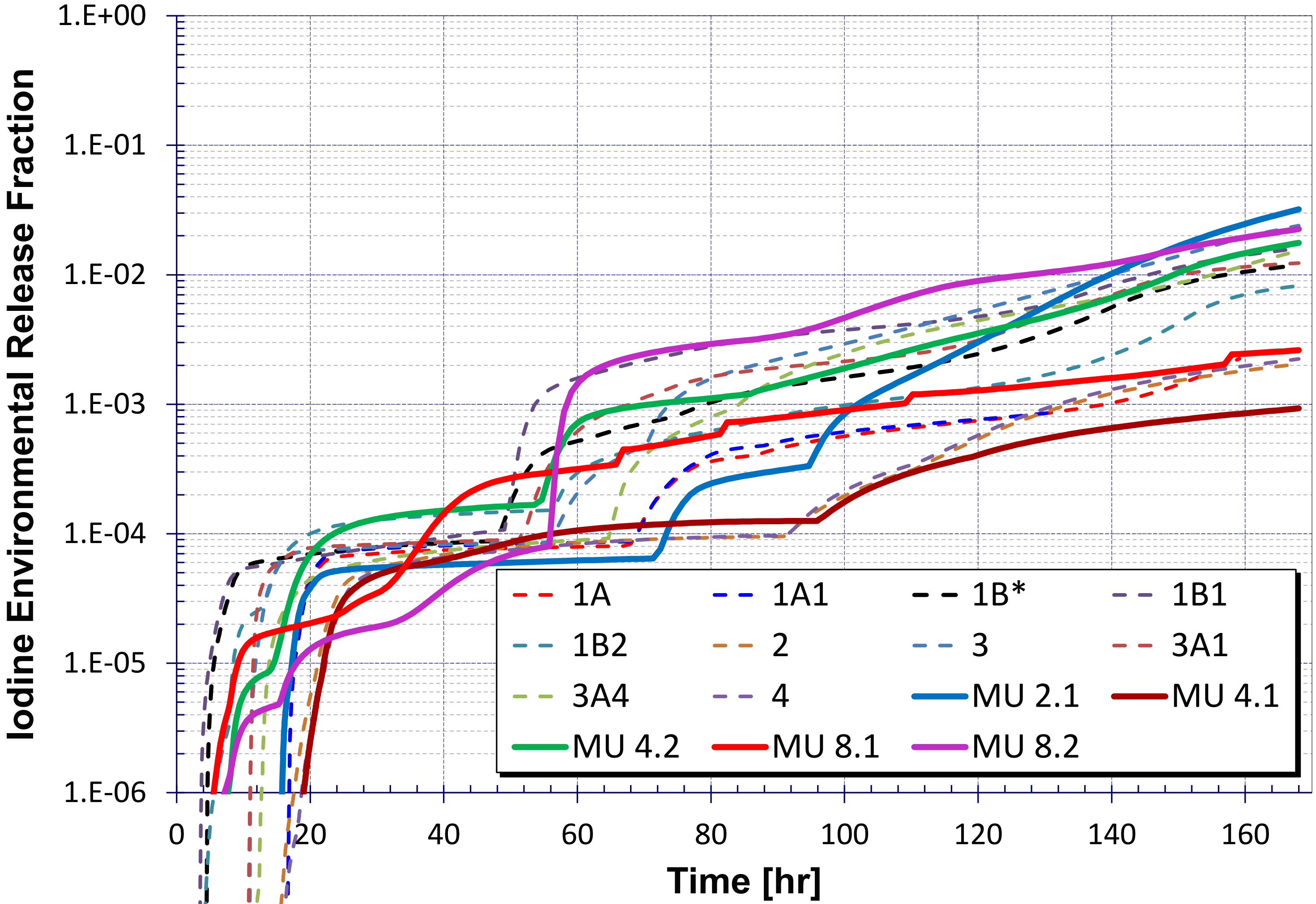


Figure 5-9: Fractional Release of Iodine to the Environment for 1-REL-LCF Calculations

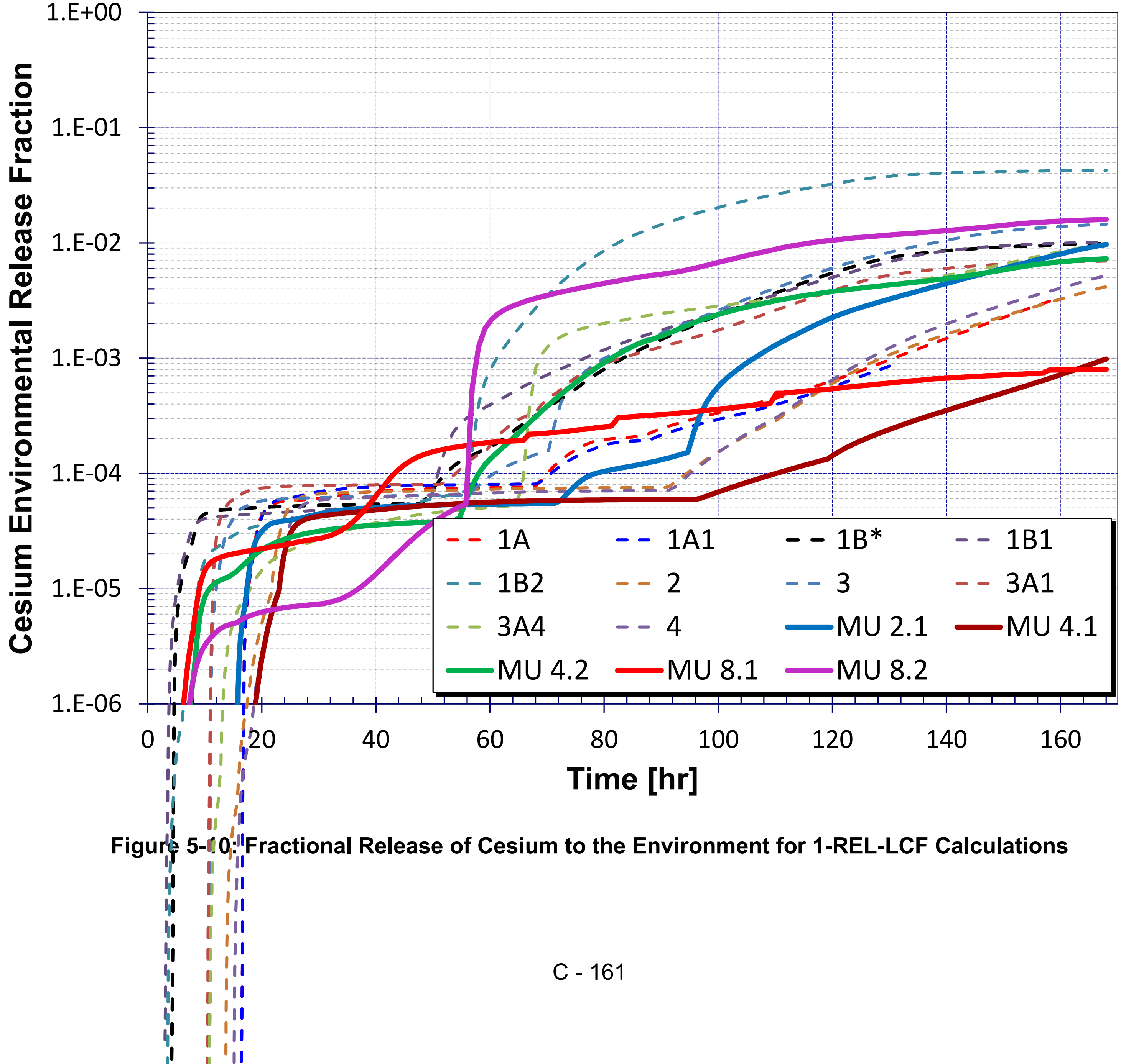


Figure 5-10: Fractional Release of Cesium to the Environment for 1-REL-LCF Calculations

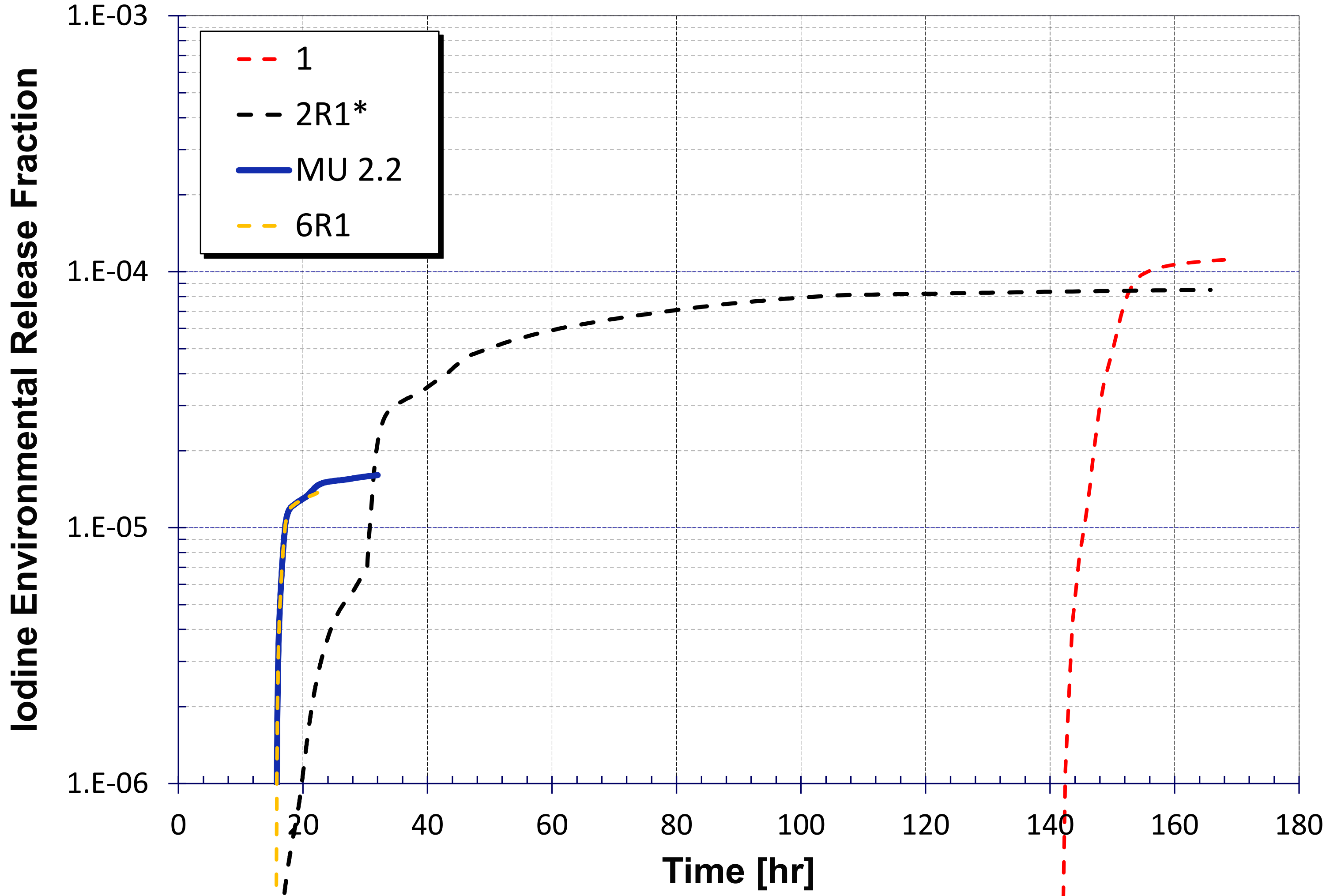


Figure 5-11: Fractional Release of Iodine to the Environment for 1-REL- NOCF Calculations

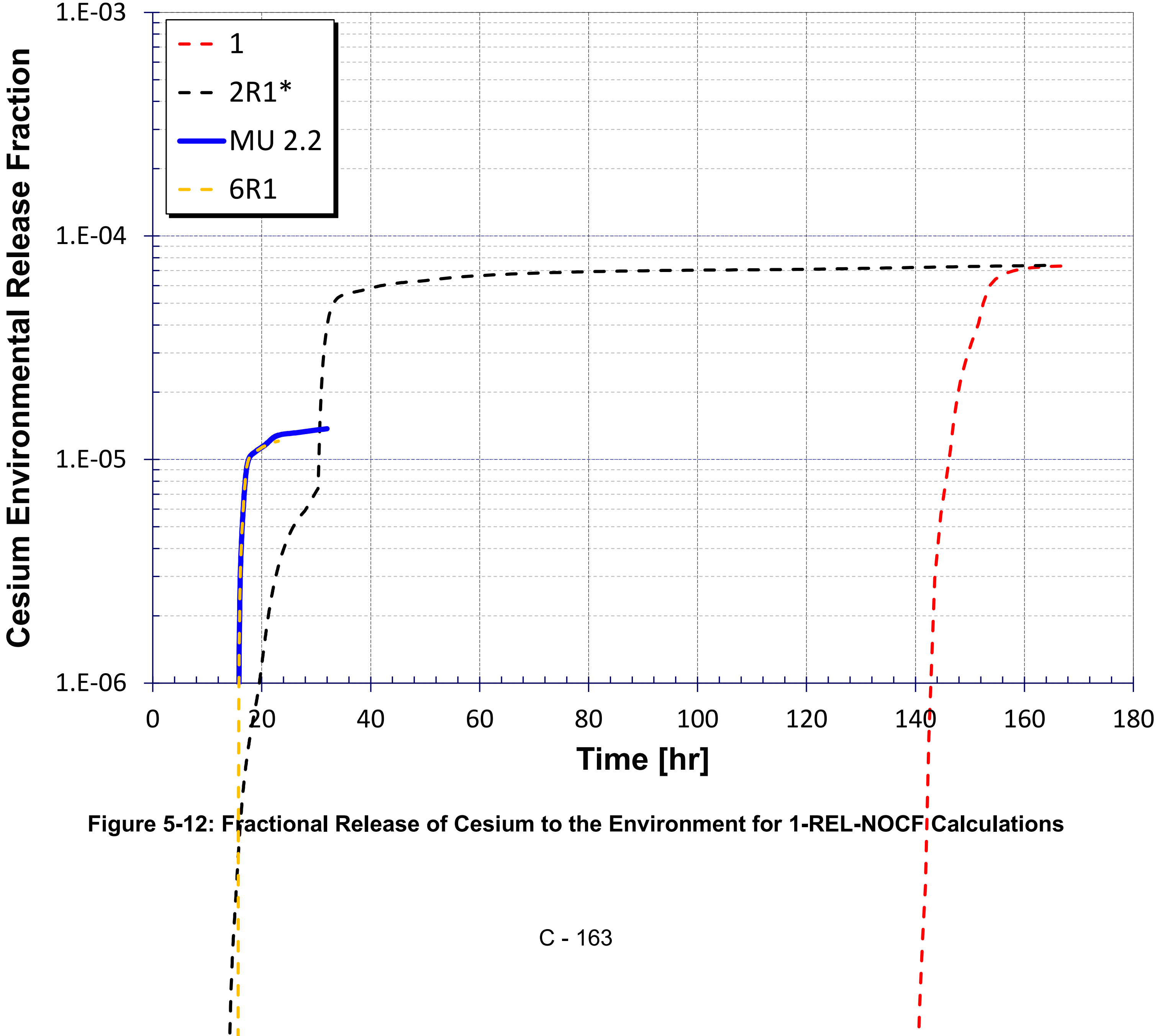


Figure 5-12: Fractional Release of Cesium to the Environment for 1-REL-NOCF Calculations

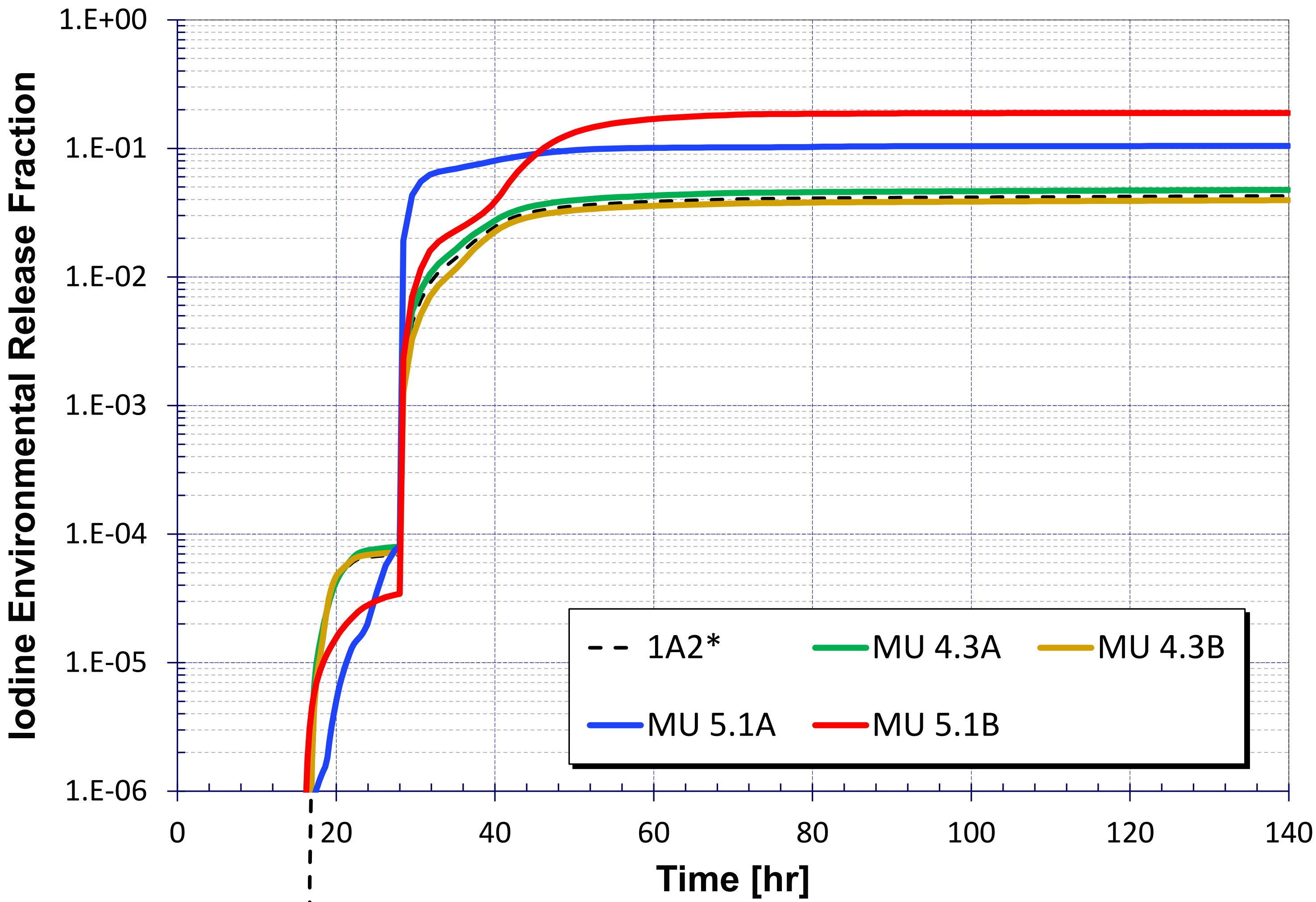


Figure 5-13: Fractional Release of Iodine to the Environment for 1-REL-ICF-BURN Calculations

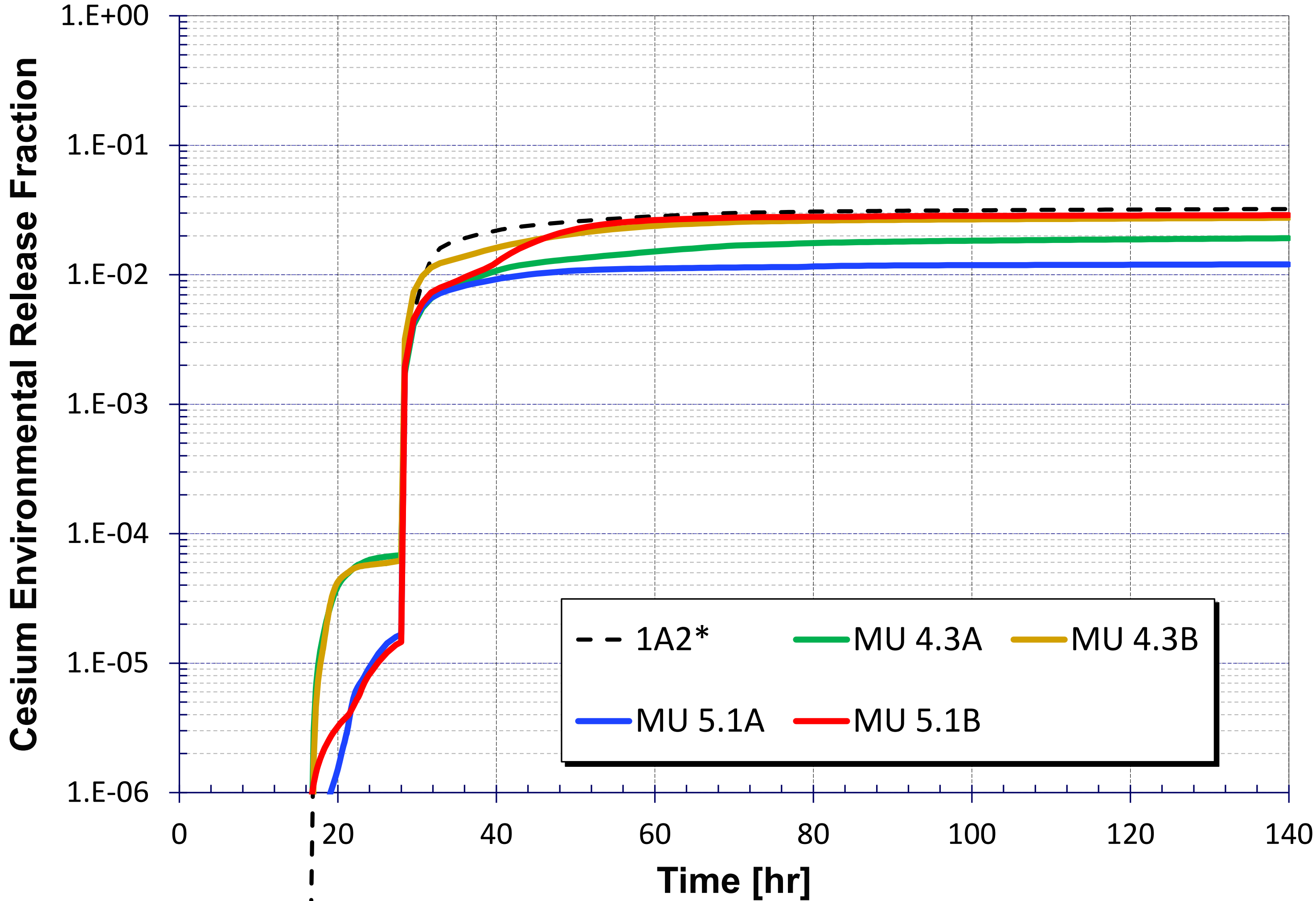


Figure 5-14: Fractional Release of Cesium to the Environment for 1-REL-ICF-BURN Calculations

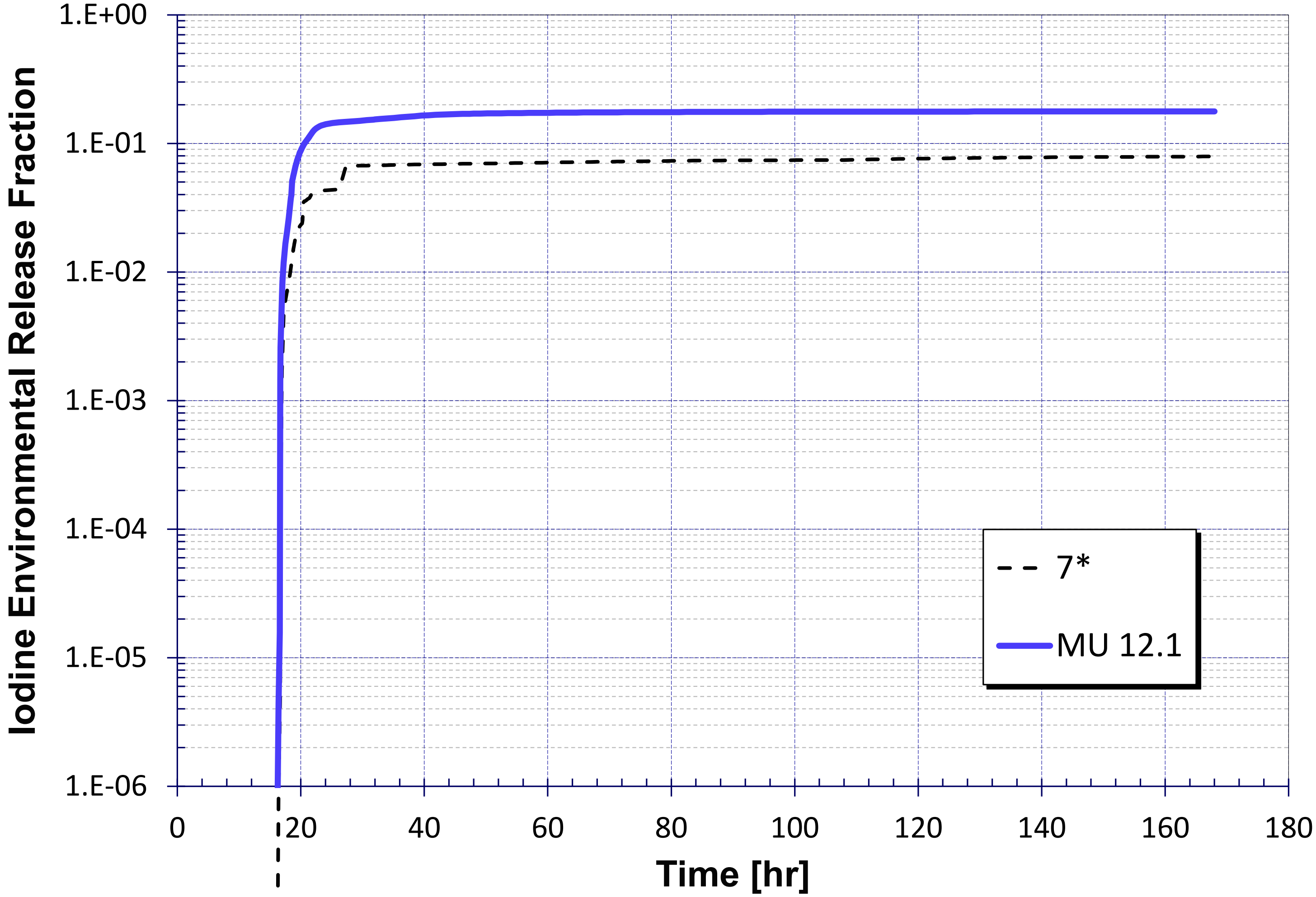


Figure 5-15: Fractional Release of Iodine to the Environment for 1-REL-CIF Calculations

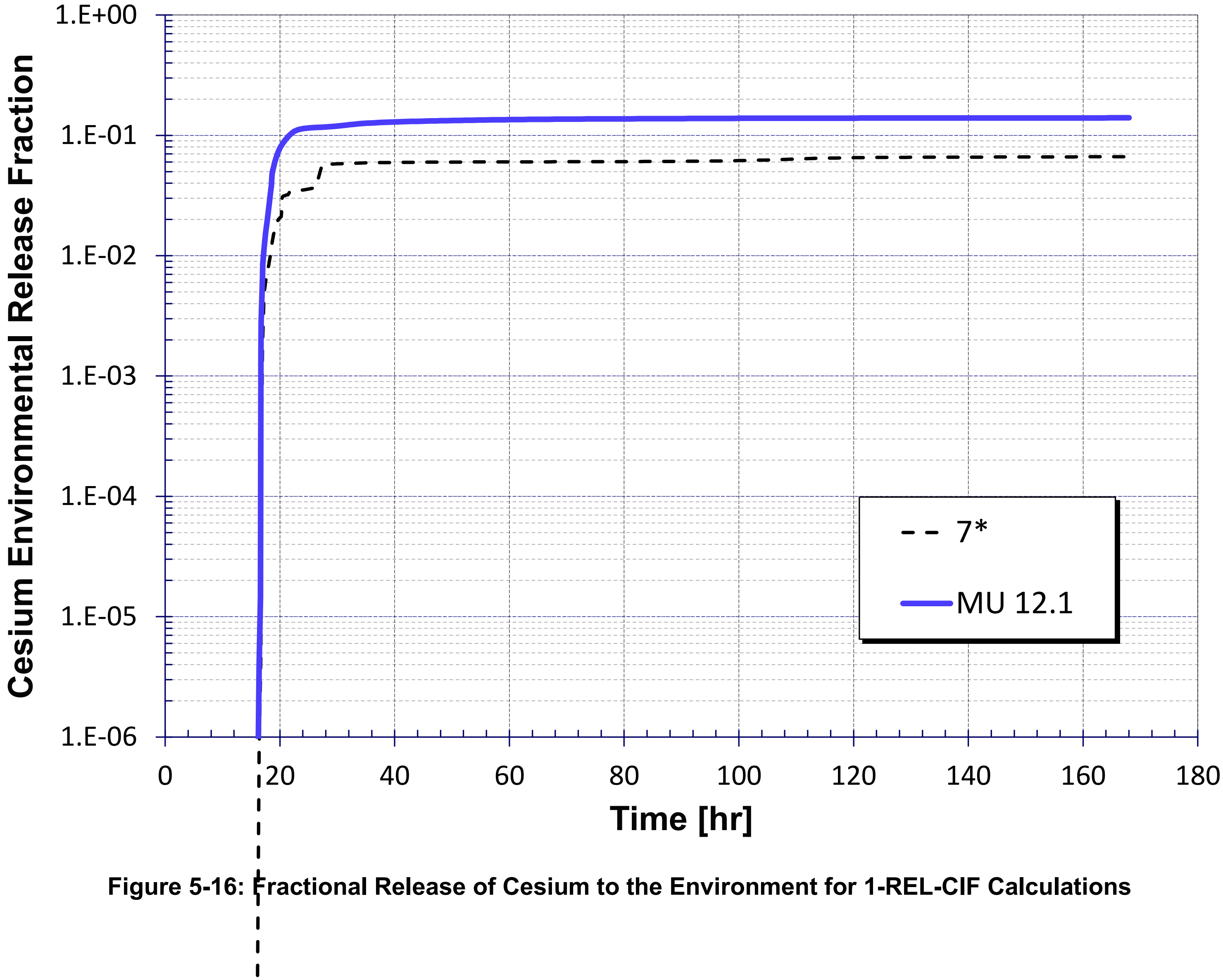


Figure 5-16: Fractional Release of Cesium to the Environment for 1-REL-CIF Calculations

6. Parameter Uncertainty Distribution Bases

The following tables provide the basis for parameter uncertainty distributions that do not use the standard approach of specifying a log-normal distribution by basing the magnitude of the error factor (EF) upon the mean.

Table 6-1: Parameter Uncertainty for 1-L2TEAR

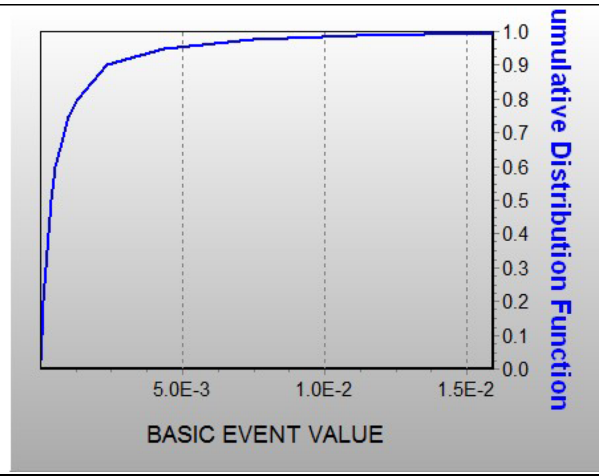
<table>
<tr>
<td>

Distribution:

Lognormal with mean of 1.1E-3 and EF = 10

See description below and CDF to the right.

</td>
<td>



</td>
</tr>
<tr>
<td colspan="2">

Basis: This basic event represents the failure of containment due to pre-existing maintenance errors, cracks, or tears which allow a direct pathway outside containment. The probability of such a pathway existing (and having gone unrecognized) depends in part on the interval between integrated leakage rate tests (ILRTs), i.e., a Type A containment leakage test. A failure probability of 1.1E-3 is used in the base model, which is taken from Revision 5 of WCAP-15691, “Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension,” dated March 2004 (Westinghouse, 2004). The variability in the expert elicitation data in Table F-6 of EPRI 1009325 (EPRI, 2007), selection of the appropriate leakage size to be associated with the basic event, and selection of an uncertainty distribution type was reviewed. This included a review of Table D-2 from EPRI 1009325 that provides a comparison of the pre-existing leakage probabilities developed by different methods. A lognormal distribution was selected for the uncertainty distribution because it captures both a clear central tendency (anchored around the point estimate), while also capturing a distribution that spans several orders of magnitude (given the large difference between expert estimates). An error factor of 10 was selected to effectively vary the distribution from zero to one, again in light of the span in expert elicitation results. A beta distribution was also explored (motivated partly by the relief valve approach used elsewhere in this report and Table D-2 of the aforementioned EPRI report), starting with a Jeffries Non-Informative Prior and selecting several tests that with zero observed failures would result in the desired mean. However, it was concluded that the resulting distribution did not offer any advantage over the aforementioned log-normal distribution.

</td>
</tr>
</table>

Table 6-2: Parameter Uncertainty for 1-L2-BE-ABFANS-IND-FAIL

<table>
<tr>
<td>Distribution:

See description below and Histogram to the right.

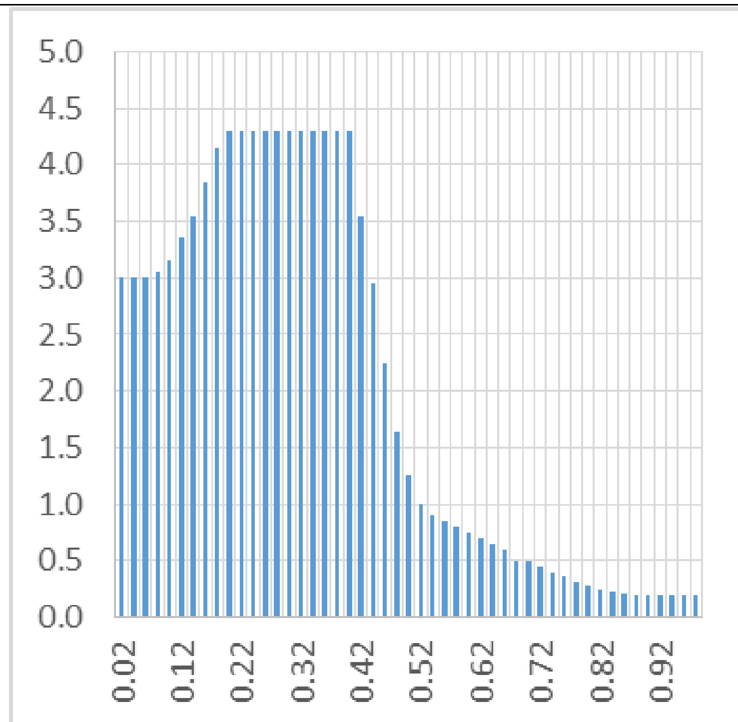
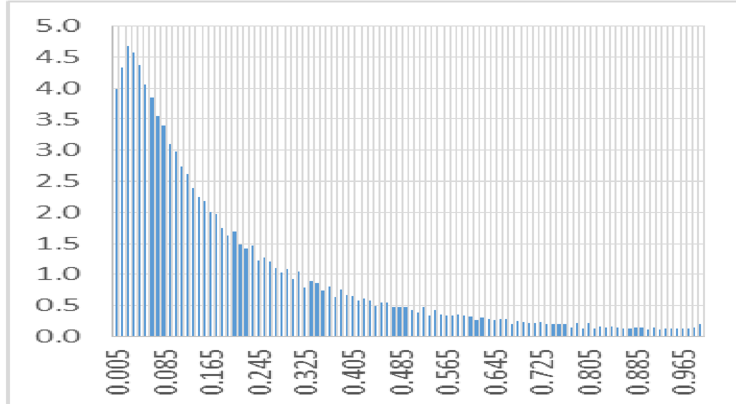
Encoded in the model as HIST-L2-ABFANSL.</td>
<td>
</td>
</tr>
<tr>
<td colspan="2">Basis: This basic event represents the likelihood that the PPAFES (post-accident) heating, ventilation, and air conditioning (HVAC) system will not fulfill its mission (i.e., will fail-to-start or fail-to-run). Failure modes include general mechanical unreliability and overloading of the filters with aerosols generated during core damage. The basic event development subjectively assigns a notional range of 0.1 to 0.5, with the point estimate being the middle of this range (0.3). Some scoping work had already been performed in as described in Section 5 of Appendix B to the main body of this report. It was decided that roughly 2/3 of the probability density would be spread relatively uniformly in the 0.1 to 0.5 range. The remaining probability density was split between 0-0.1 and 0.5-1.0. The use of a diffuse distribution that goes all the way to 1.0 (high probability of failure) was based on the belief that several possible and varying failure modes exist. The rationale for extending the distribution to zero (high probability of success) was based on the notion that the system only really needs to perform during a several-hour period when releases are at their peak, and that some very uncertain upstream effects (e.g., turbulent deposition in piping) could decrease filter loading. The remainder of the detailed histogram specification was guided by the desire to have a mean of 0.3.</td>
</tr>
</table>

Table 6-3: Parameter Uncertainty for 1-L2-BE-ARVSTUCK-SGTR

Distribution:

See description below and Histogram to the right.

Encoded in the model as HIST-L2-ARVSTUCK-SGTR.



Basis: This basic event represents the likelihood that a SG relief valve will be predominantly open during a SGTR initiating event where the ruptured SG is isolated prior to core damage. Failure modes include general mechanical unreliability and the various failure modes associated with repeated valve operation (modulation for ARVs and cycling for SRVs). The variability and uncertainty in the sequence characterization (most notably the number of valve cycles that will be experienced for an average simulation), along with the applicability of pre-core damage failure data to post-core-damage situations with higher (but not excessive) temperatures, is quite large. To develop the uncertainty distribution (as well as the mean value which is used as the baseline model's point estimate), a load and capacity distribution was developed. The load distribution (# of cycles) is based on review of available MELCOR/MAAP information, while the capacity distribution (likelihood of failure) is obtained from NUREG/CR-7037 (INL, 2010). Note that the failure data in question includes failures that involve leaking or weeping of valves, such that it over-estimates the likelihood of the valve failing largely or fully open. Two sets of failure data are processed. The first uses main steam ARV data, while the second uses main steam SRV data. The former is used in the model because the model does not have sufficient resolution to distinguish which valves are cycling and the ARV data results in a more inclusive (i.e., higher) failure probability.

Type of Valve	Distribution of Probability of Failure-to-Close on Initial Demand	Distribution of Probability of Failure-to-Close on Subsequent Demand	Source of Data from NUREG/CR-7037 (INL, 2010), Failure to Close, All Failures
MSS PORV/ARV	Beta(α = 1.5, β = 506.5)	Beta(α = 2.5, β = 227.5)	Table 18, MSS PORVs, Automatic demands
MSS safety vent valve (SVV) data	Beta(α = 15.5, β = 558.5)	Beta(α = 0.5, β = 196.5)	Table 20, MSS Code Safety Valves
Distribution of # of valve cycles: lognormal distribution with mean = 30, 5th percentile = 3, 95^{th} percentile = 100			

The load and capacity distributions were numerically convolved using a 100x100 inner/outer loop Excel-based stratified sampling approach (i.e., 10,000 data points), resulting in a distribution with the following attributes:

- Mean = 0.21
- Median = 0.14
- 5^{th} percentile = 0.01
- 95^{th} percentile = 0.71

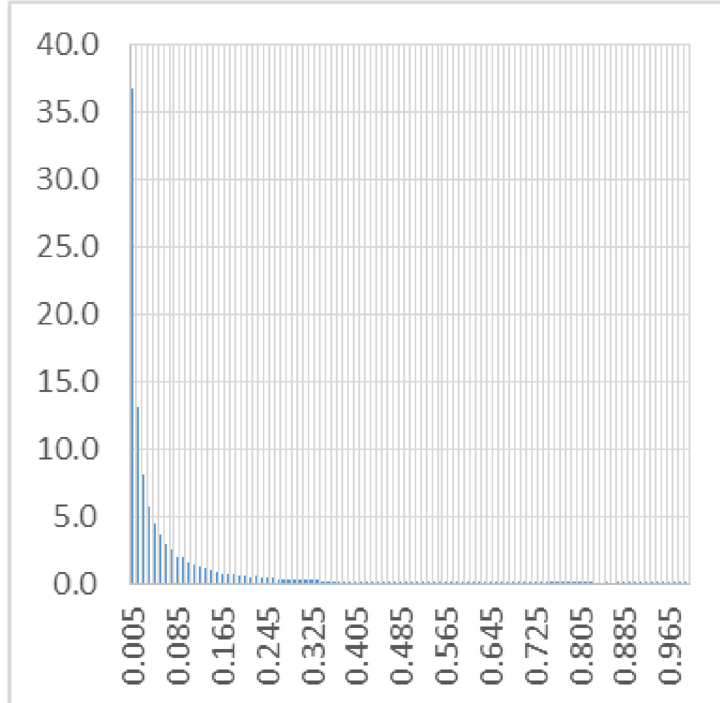
This data was then approximated and input to SAPHIRE using a 100-point "percentage"-type histogram.

Table 6-4: Parameter Uncertainty for 1-L2-BE-PZRVSTUCK-PORV

Distribution:

See description below and Histogram
to the right.

Encoded in the model as HIST-L2-PZRVSTUCK-PORV.



Basis: This basic event represents the likelihood of a pressurizer PORV failing due to cycling during the post-core-damage phase of high-pressure sequences. The effects of high temperatures on the valve performance are not explicitly considered, which is a limitation. At the same time, the authors note that high-temperature valve seizure is of significantly less concern for PWRs than it is for BWRs, owing to the much more tortuous path to the valves in question (owing to the surge line and pressurizer). The uncertainty treatment here is very similar to that described above in Table 6-3. In fact, the number of post-core-damage PORV lifts was estimated to be very similar to the number of total (pre- and post-core-damage lifts) for the secondary-side valves, and so the same load distribution is used (while the capacity distribution is different, and again taken from NUREG/CR-7037 (INL, 2010)). As before, the failure data in question includes failures that involve leaking or weeping of valves, such that it over-estimates the likelihood of failing the valve largely or fully open.

Type of Valve	Distribution of Probability of Failure-to-Close on Initial Demand	Distribution of Probability of Failure-to-Close on Subsequent Demand	Source of Data from NUREG/CR-7037(INL, 2010), Failure to Close, All Failures
Pressurizer PORV	Beta(α = 0.5, β = 100.5)	Beta(α = 0.5, β = 181.5)	Table 18, RCS PORVs, Automatic demands
Distribution of # of valve cycles: lognormal distribution with mean = 30, 5th percentile = 3, 95th percentile = 100			

Numerically convolving these distributions results in a distribution with the following attributes:

- Mean = 0.07
- Median = 0.02
- 5th percentile = 1E-4
- 95th percentile = 0.32

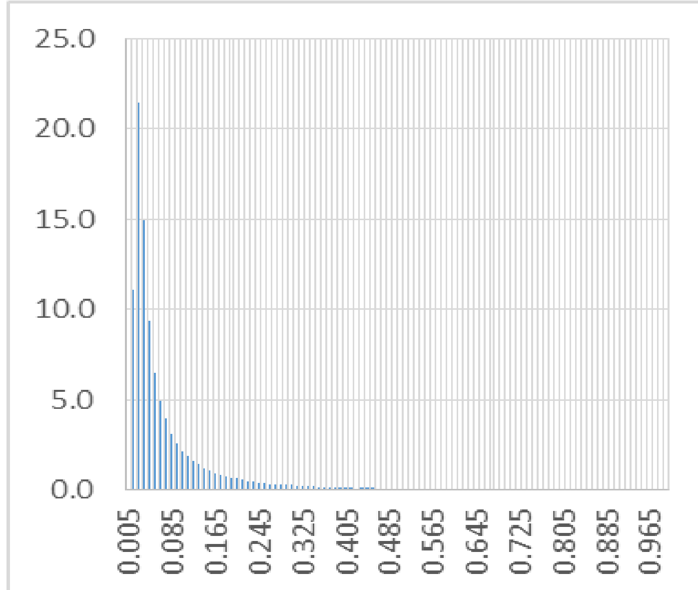
This data was then approximated and input to SAPHIRE using a 100-point "percentage"-type histogram.

Table 6-5: Parameter Uncertainty for 1-L2-BE-PZRVSTUCK-SRV

Distribution:

See description below and Histogram to the right.

Encoded in the model as HIST-L2-PZRVSTUCK-SRV.



Basis: This basic event is the companion to 1-L2-BE-PZRVSTUCK-PORV, but applicable to sequences where the SRV is the predominantly cycling valve during core damage. It is generated in the same way. Two sets of failure data are processed. The first uses main steam SRV data, as arguments have been made that this data is applicable to pressurizer SRVs. The second uses sparse pressurizer SRV operating experience data in concert with abundant test data (which has debatable applicability to accident situations). The former is used in the model.

Type of Valve	Distribution of Probability of Failure-to-Close on Initial Demand	Distribution of Probability of Failure-to-Close on Subsequent Demand	Source of Data from NUREG/CR-7037 (INL, 2010), Failure to Close, All Failures
Pressurizer SVV (based on MSS SVV data)	Beta(α = 15.5, β = 558.5)	Beta(α = 0.5, β = 196.5)	Table 20, MSS Code Safety Valves
Pressurizer SVV (based on sparse Pressurizer SVV data, and test data)	Beta(α = 2.5, β = 1807.5)	Beta[7](α =2.5, β = 1807.5)	Table 20, RCS Code Safety Valves and Table 22, RCS Code Safety Valves
Distribution of # of valve cycles: lognormal distribution with mean = 30, 5th percentile = 3, 95th percentile = 100			

Numerically convolving these distributions results in a distribution with the following attributes:

- Mean = 0.08
- Median = 0.04
- 5th percentile = 0.02
- 95th percentile = 0.31

This data was then approximated and input to SAPHIRE using a 100-point “percentage”-type histogram.

[7] Neither the operating experience data in Table 20 nor the test data in Table 22 are divided into initial and subsequent demands, so the same estimate is used for both.

Table 6-6: Parameter Uncertainty for 1-L2-BE-H2IGNSRC-E-NAC

<table>
<tr>
<td>Distribution:

See description below and Histogram to the right.

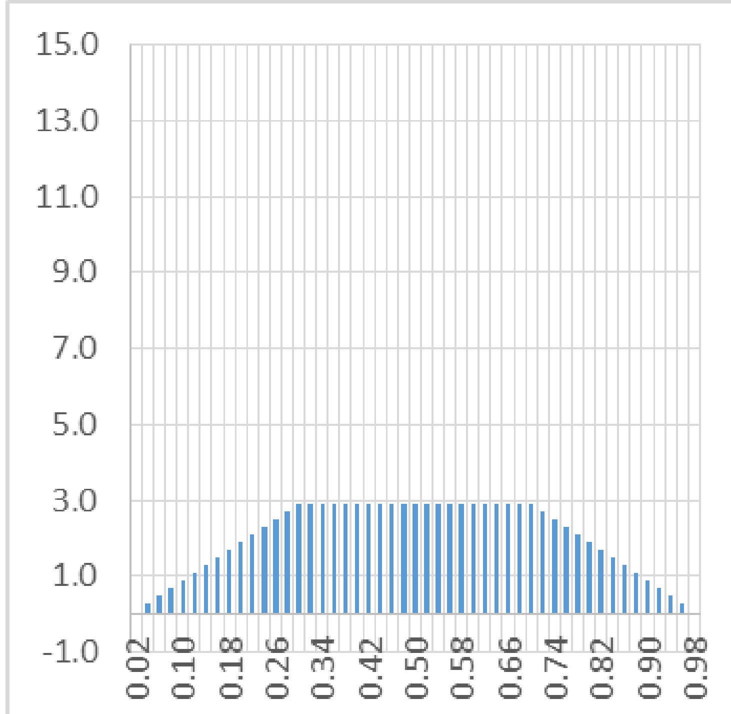
Encoded in the model as HIST-L2-H2IGNSRC-E-NAC.</td>
<td>
</td>
</tr>
<tr>
<td colspan="2">Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting), around the time of vessel breach, when all AC and DC power is unavailable. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. The distribution represents a uniform probability between 0.3 to 0.7, and a linearly decreasing probability between these values and the bounds of 0 and 1.</td>
</tr>
</table>

Table 6-7: Parameter Uncertainty for 1-L2-BE-H2IGNSRC-L-NAC

<table>
<tr>
<td>Distribution:

See description below and Histogram to the right.

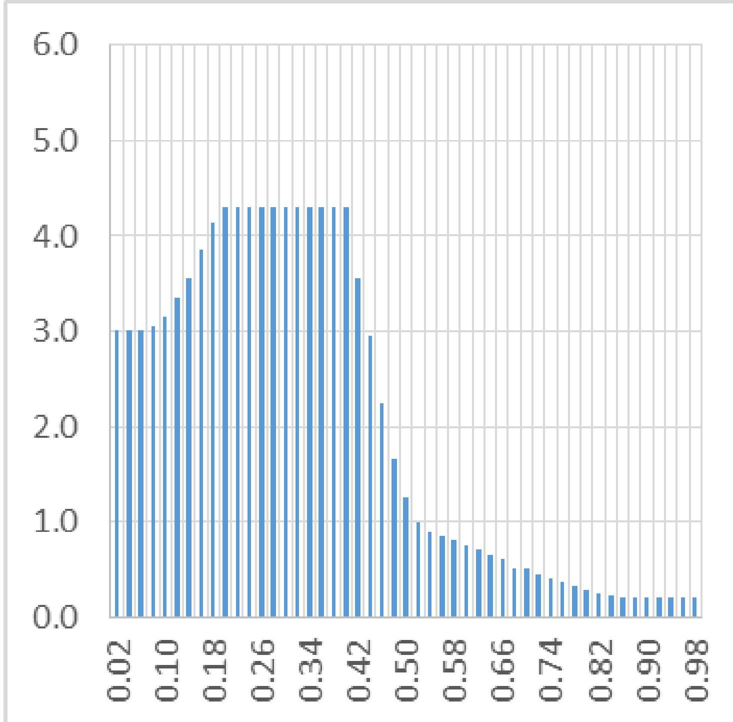
Encoded in the model as HIST-L2-H2IGNSRC-L-NAC.</td>
<td>
</td>
</tr>
<tr>
<td colspan="2">Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting), in the intermediate and late timeframe (i.e., well after the time of vessel breach but prior to containment quasi-static over-pressure failure), when all AC and DC power is unavailable. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which assumes the point estimate is the mean (as a simplifying assumption), assumes a uniform probability around that mean, and assumes the probability drops off quasi-exponentially to the bounds of the distribution (0, 1). The details of the shape of the distribution were chosen iteratively to enforce the mean value. This reflects a force-fitting that would not be needed if the uncertainty distribution had been defined prior to selecting the point estimate used in the modeling.</td>
</tr>
</table>

Table 6-8: Parameter Uncertainty for 1-L2-BE-H2IGNSRC-VE-NAC

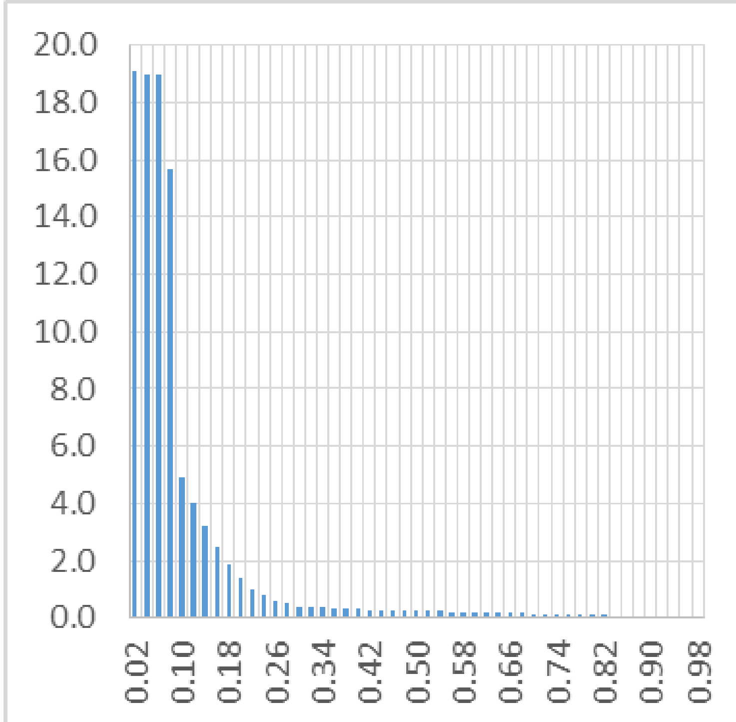
Distribution: See description below and Histogram to the right. Encoded in the model as HIST-L2-H2IGNSRC-VE-NAC.	
Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where containment combustible gases are collecting), in the very early timeframe (i.e., prior to vessel breach), when all ac and dc power is unavailable. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. This distribution assumes that the probability is relatively uniform around the assumed mean value, and then drops off quasi-exponentially, extending across much of the probability range (0,1). The details of the shape of the distribution were chosen iteratively to force the mean value. This reflects a force-fitting that would not be needed if the uncertainty distribution had been defined prior to selecting the point estimate used in the modeling.	

Table 6-9: Parameter Uncertainty for 1-L2-BE-ABFH2-FANS

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<td>Distribution:

See description below and Histogram to the right.

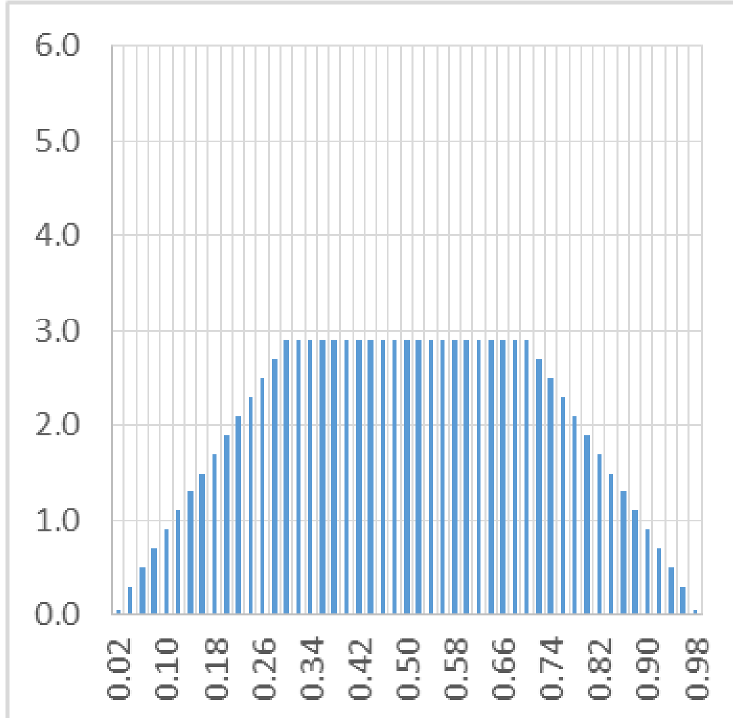
Encoded in the model as HIST-L2-ABFH2-FANS</td>
<td>
</td>
</tr>
<tr>
<td colspan="2">Basis: This basic event represents the probability of having a combustion event in the auxiliary building during an ISLOCA initiator, when the Piping Penetration Area Filtration Exhaust System is operating. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment and MELCOR scoping investigations. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. More specifically, the probability is assumed to be uniform around the assumed mean value, and to drop off linearly in both directions to the bounds of the probability range (0,1).</td>
</tr>
</table>

Table 6-10: Parameter Uncertainty for 1-L2-BE-ABFH2-NOFANS

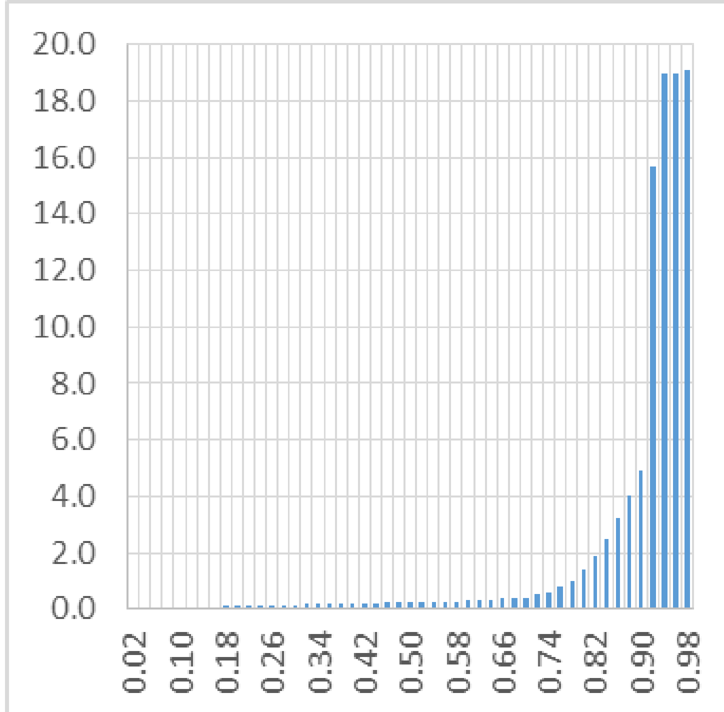
Distribution: See description below and Histogram to the right. Encoded in the model as HIST-L2-ABFH2-NOFANS	
Basis: This basic event represents the probability of having a combustion event in the auxiliary building during an ISLOCA initiator, when the Piping Penetration Area Filtration Exhaust System is not operating. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment and MELCOR scoping investigations. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. This distribution assumes that the probability is relatively uniform around the assumed mean value, and then drops off quasi-exponentially, extending across much of the probability range (0,1). The details of the shape of the distribution were chosen iteratively to force the mean value. This reflects a force-fitting that would not be needed if the uncertainty distribution had been defined prior to selecting the point estimate used in the modeling.	

Table 6-11: Parameter Uncertainty for 1-L2-BE-ISLOCASUBM-LRG

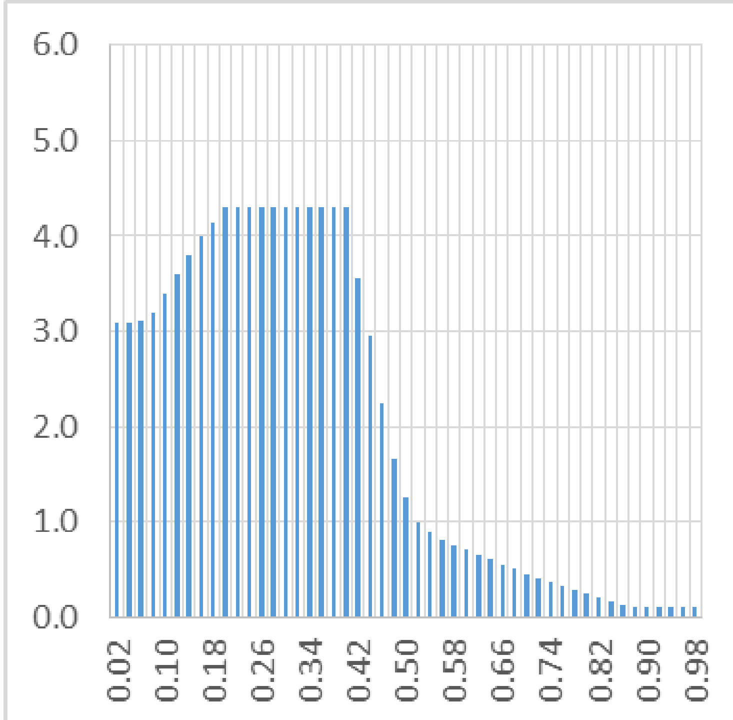
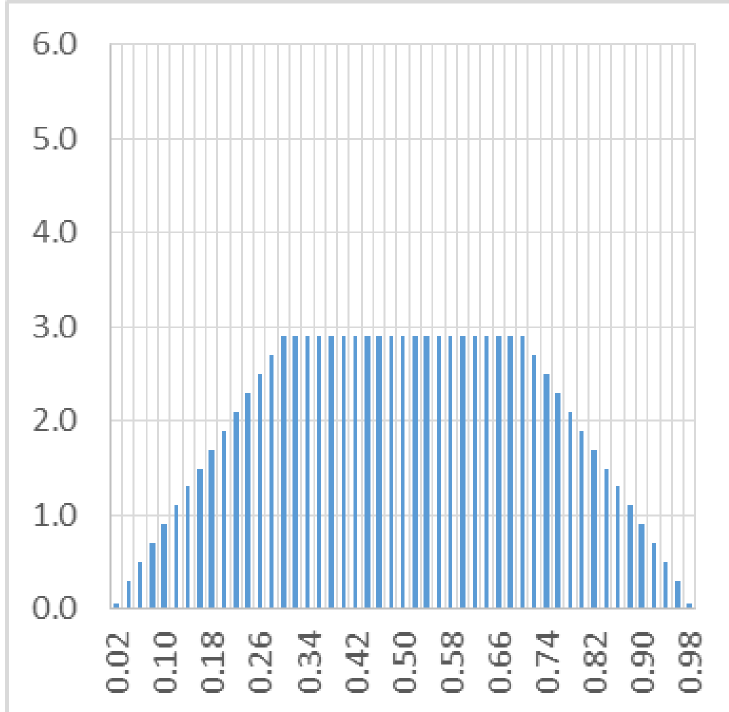
Distribution: See description below and Histogram to the right. Encoded in the model as HIST-L2-ISLOCASUBM-LRG	
Basis: This basic event represents the probability of the ISLOCA break being predominantly submerged in the auxiliary building for a large-break ISLOCA. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment, but also leveraging some results from the Level 1 ISLOCA expert elicitation. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. This distribution assumes that the probability is uniform around the assumed mean value, and then drops off quasi-exponentially in both directions, extending across all of the probability range (0,1). The details of the shape of the distribution were chosen iteratively to force the mean value. This reflects a force-fitting that would not be needed if the uncertainty distribution had been defined prior to selecting the point estimate used in the modeling.	

Table 6-12: Parameter Uncertainty for 1-L2-BE-ISLOCASUBM-SM

Distribution: See description below and Histogram to the right. Encoded in the model as HIST-L2-ISLOCASUBM-SM	
Basis: This basic event represents the probability of the ISLOCA break being predominantly submerged in the auxiliary building for a small-break ISLOCA. The probability of this occurrence was subjectively assigned during the baseline model development based primarily on practitioner judgment. The team felt that the uncertainty associated with this event is greater than with many of the other events in the Level 2 model, owing to the more subjective nature of its assignment. As such, the use of the default scheme (log-normal distribution with an error factor based on the point estimate magnitude) does not adequately capture the degree of uncertainty. For this reason, a histogram was constructed which reflects a more diffuse distribution, but which maintains the previously-developed point estimate as the mean of the distribution. More specifically, the probability is assumed to be uniform around the assumed mean value, and to drop off linearly in both directions to the bounds of the probability range (0,1).	

7. Model Uncertainty Supplements

This section provides supplemental information on the reasoning behind the modeling choices in select sensitivity cases. The sensitivity cases are described in detail in **Section 4**.

7.1 Supplemental Information for Section 4.4

7.1.1 Supplemental Information on Sensitivity Analysis of Timing of Initial and Final Accumulator Injection (MU-4.1)

For those cases in which there is a significant delay (more than an hour) between the time of initial and final accumulator injection, **Table 7-1** gives the minimum RCS pressure (taken from the pressurizer) prior to hot leg creep rupture (HLCR) and the accumulator inventory at the associated time.

Table 7-1: RCS Pressure and Accumulator Inventory Just Prior to Hot Leg Creep Rupture

Case	Minimum pressure prior to hot leg creep rupture (psig)	Accumulator inventory (per accumulator) prior to hot leg creep rupture (m^3)[1]
1	322	17.77
1A	364	19.40
1A1	364	19.40
1A2	364	19.40
2	223	11.85
2a	223	11.85
2R1	223	11.85
2R2	226	11.85
4	222	11.91
5	216	12.54
5A	342	18.73
5C	342	18.73
7	377	19.93
8B	405	19.23

[1] The initial inventory of each accumulator is ~25.5 m^3

The hold-up of accumulator injection is caused by the significant depressurization of the tank as the nitrogen gas expands. The relationship between the pressure in the accumulator and the delivered volume of water used in the MELCOR analysis is shown in **Figure 7-1** (ideal gas expansion: $\frac{P}{P_0} = \left(\frac{V}{V_0}\right)^{\gamma}$). Note that the full volume cannot inject until the RCS pressure is less than 125psig.

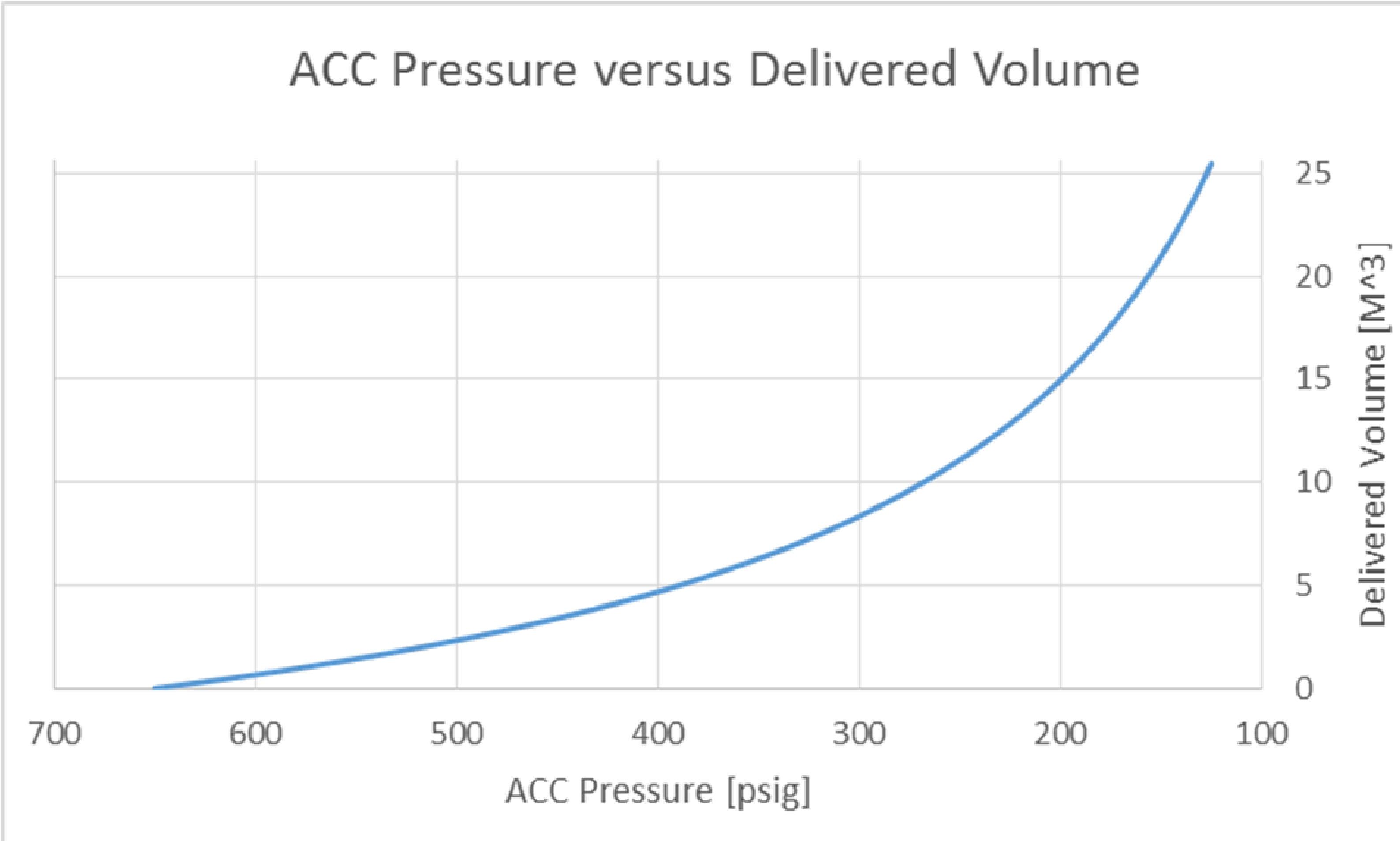


Figure 7-1: Accumulator Delivered Volume as a Function of Pressure for Isentropic Process

The sensitivity calculation chosen in this case was to alter the value of gamma (the specific heat capacity ratio) from 1.4 (assumes an isentropic process) to 1 (assumes an isothermal process, shown in **Figure 7-2**) so that the full inventory could inject closer to 200 psig.

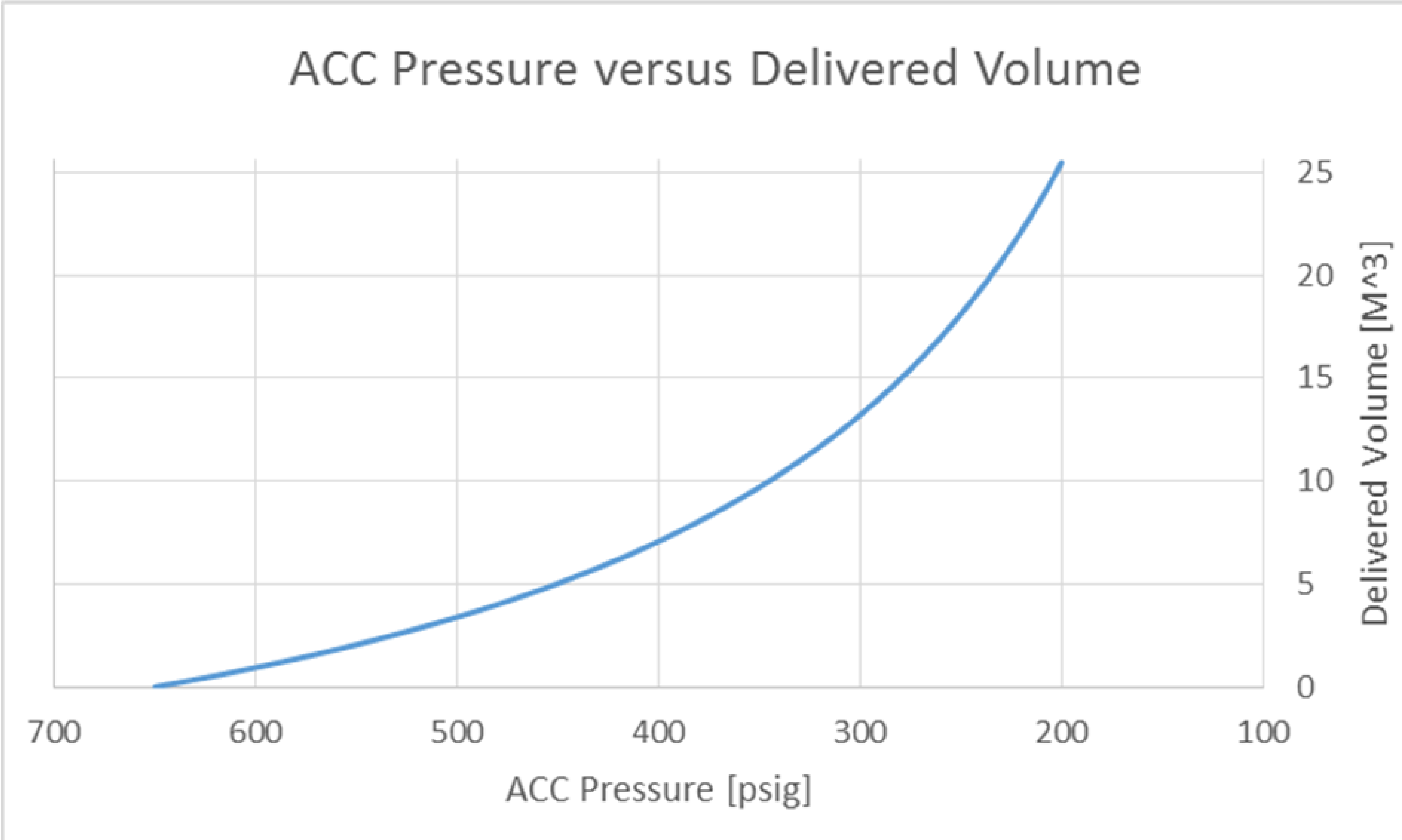


Figure 7-2: Accumulator Delivered Volume as a Function of Pressure for Isothermal Process

Case 2 was selected as a good candidate for this sensitivity study since there is a significant time delay between the start and end of accumulator injection with pressure hanging around 220 psig. **Figure 7-3** below gives the RCS pressure and accumulator delivered volume as a function of time for Case 2.

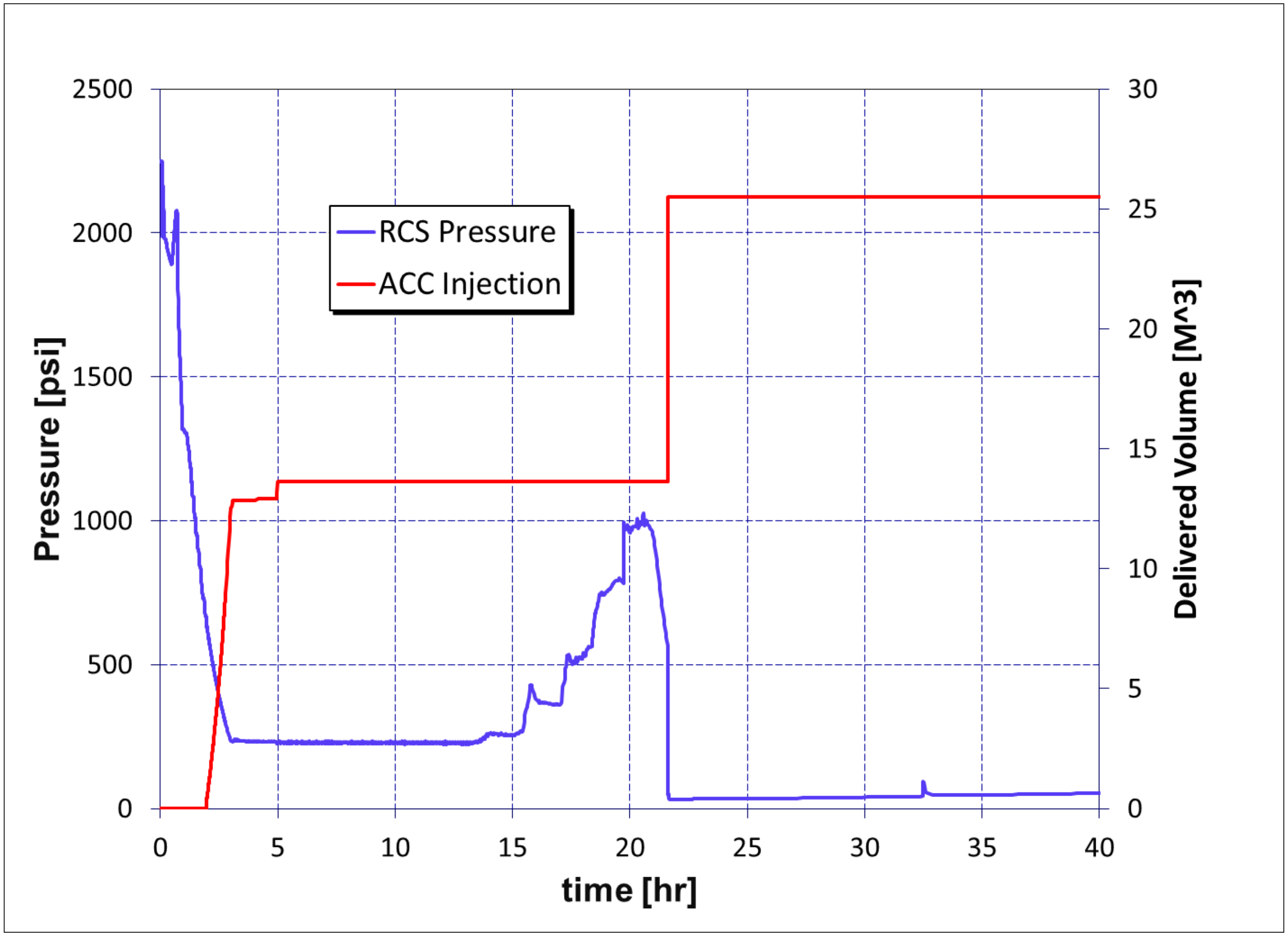


Figure 7-3: Net Accumulator Injection and RCS Pressure as a Function of Time for Case 2

7.1.2 Supplemental Information on Sensitivity Analysis of Adiabatic Heatup of PRT (MU-4.2)

Cases 1B2 and 3A4 briefly described in Sections 1.2 and 3.4 of Appendix B are the only simulations in which the PRT dries out. With the PRT being modeled as adiabatic (insulated), the concern is that it is not modeling the convective cooling that would take place to the atmosphere of containment leading to a greater release to the environment. Note that this is not a major concern in Case 3A4 since the dryout occurs later in the simulation (86 hours) and little re-volatilization of the deposited radionuclides occurs. The following attempts to outline the issue using the results of 1B2 (see **Figure 7-4** below).

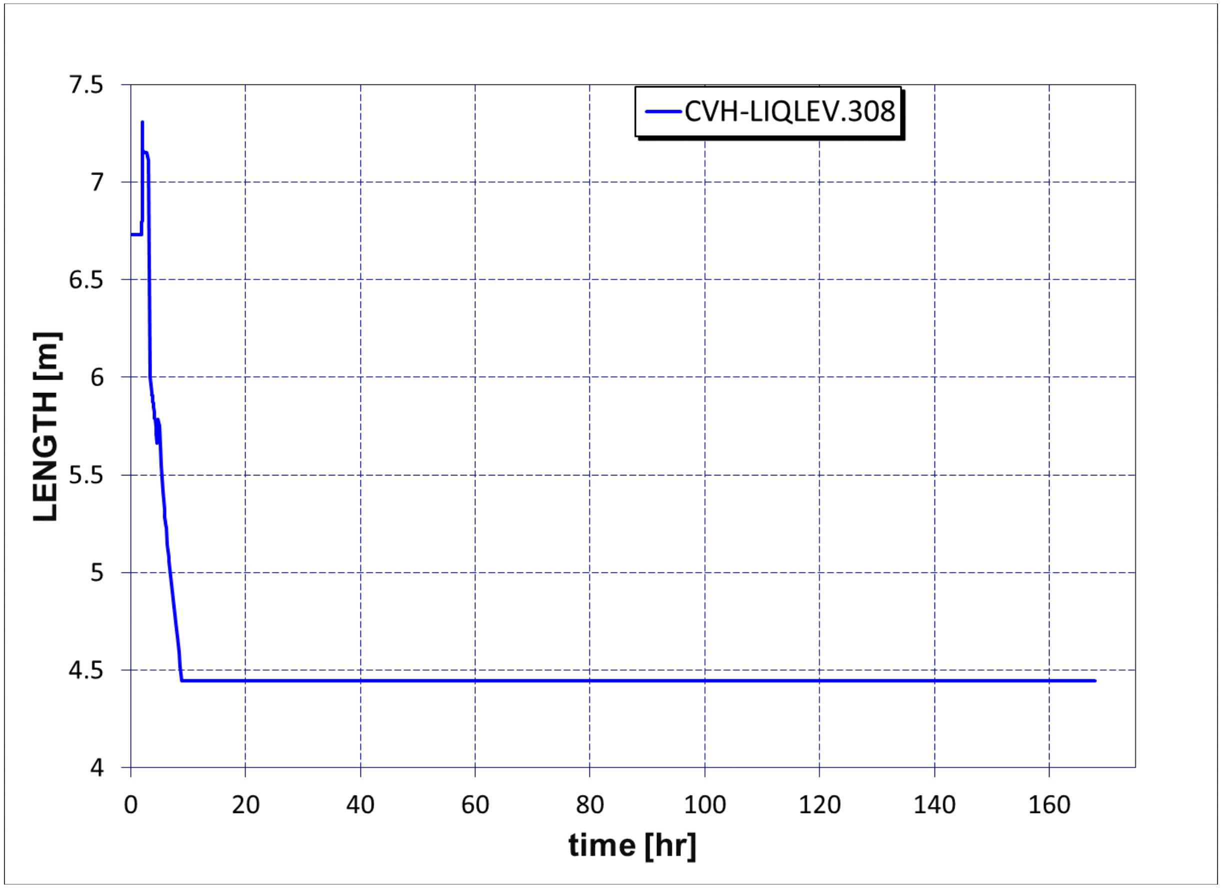


Figure 7-4: Water Level in the PRT for Case 1B2

When the PRT dries out around 9 hours, the suspended radionuclides deposit on the PRT heat structure causing it to heat up quickly (see **Figure 7-4** and **Figure 7-5**). Since the PRT is modeled as adiabatic, the heat does not dissipate through to the containment atmosphere and remains at an elevated temperature, only decreasing as the radionuclides volatilize. Note that MELCOR does not use the (anomalously) high temperature seen in **Figure 7-5** to model the melting of this heat structure; it is only used to model the re-vaporization of deposited fission products on the associated heat structure.

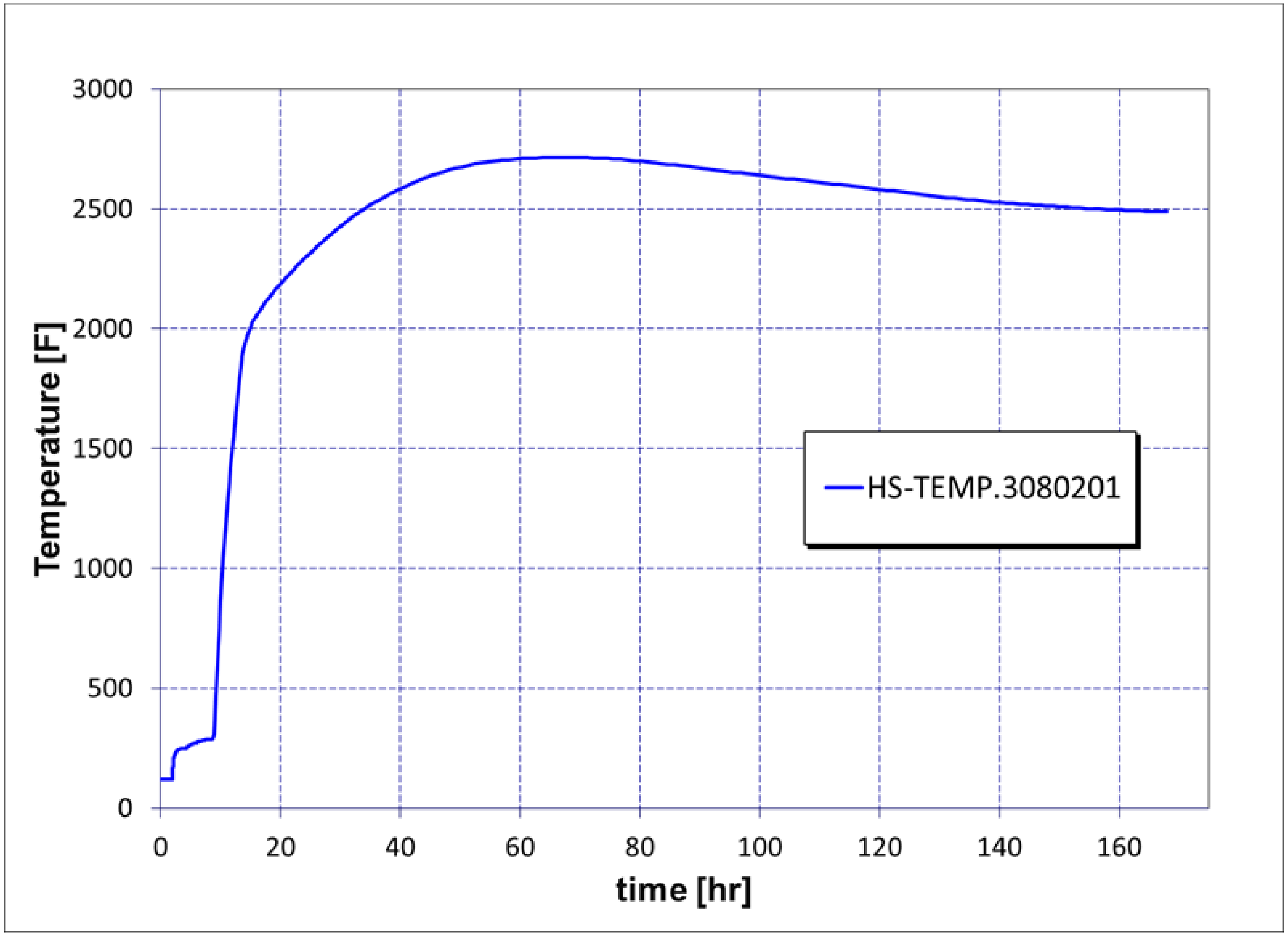


Figure 7-5: Temperature of the PRT Heat Structure

In **Figure 7-6** and **Figure 7-7**, note that the decreasing retention of cesium in the RCS (which includes the PRT) mirrors the decreasing radionuclide mass deposited on the PRT heat structures. This is due to the volatilization of the cesium with the PRT at a sustained high temperature.

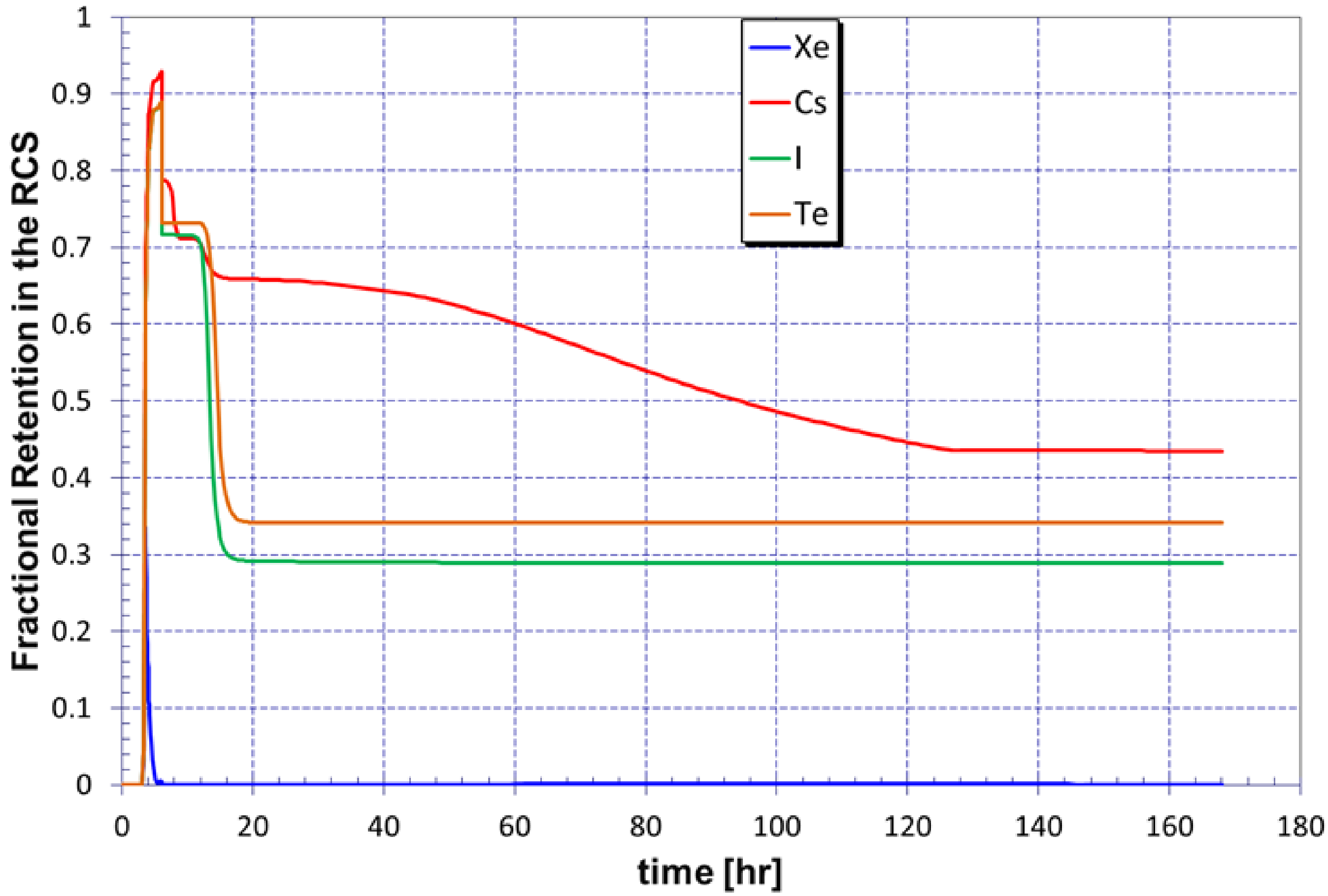


Figure 7-6: Retention of volatile fission products in the RCS

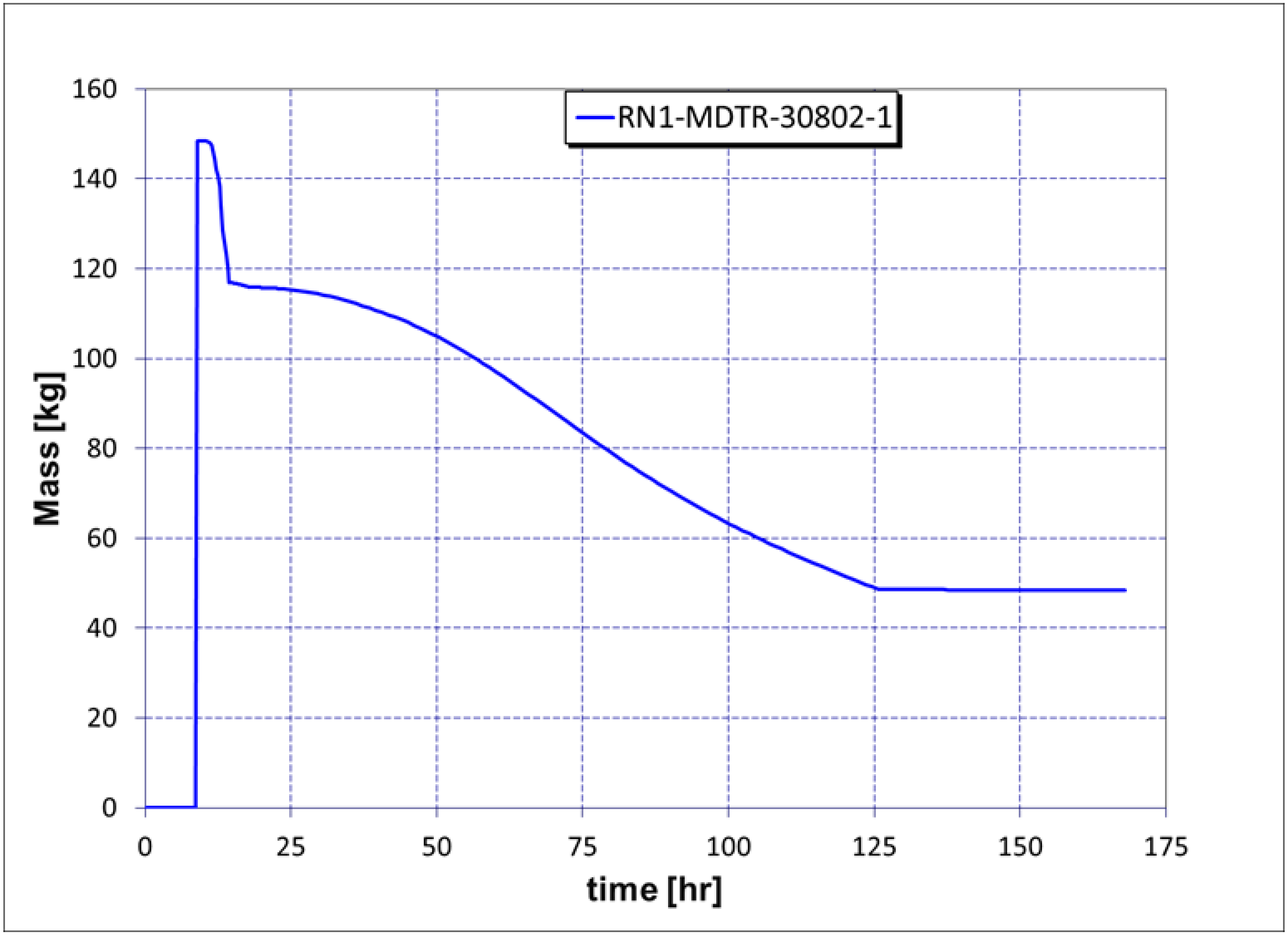


Figure 7-7: Fission Products Deposited on the PRT Heat Structure

The cesium being released from the PRT becomes airborne, making its way to containment and subsequently (when containment failure occurs at ~57 hours) to the environment. This leads to a greater release of cesium to the environment as compared to Case 1B in which the PRT does not dry out. See **Figure 7-8** and **Figure 7-9** below.

A sensitivity study was performed for this case (see Section 4.4.4 on MU-4.2) in which the PRT was modified to be non-adiabatic.

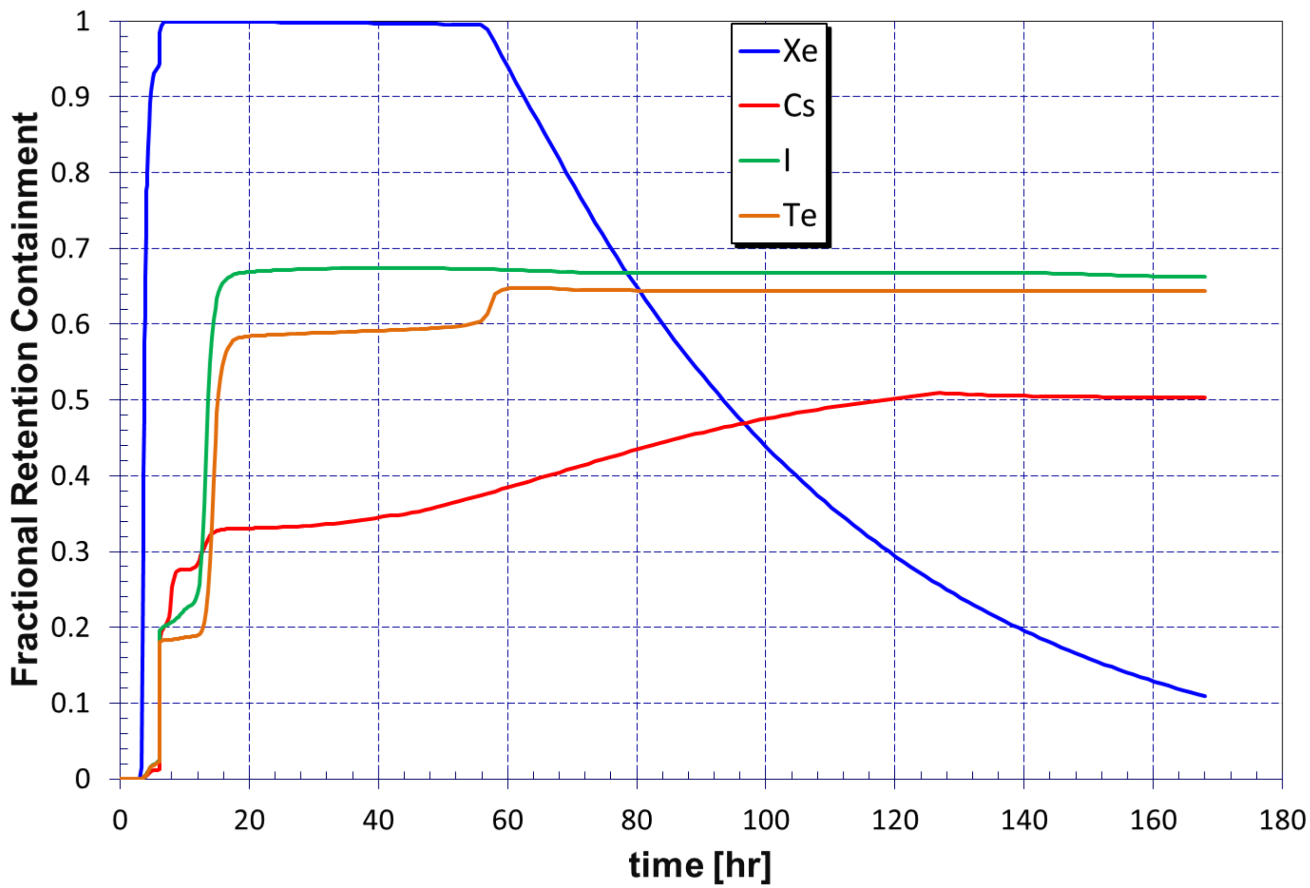


Figure 7-8: Fraction Retention of Volatile Fission Products in the Containment

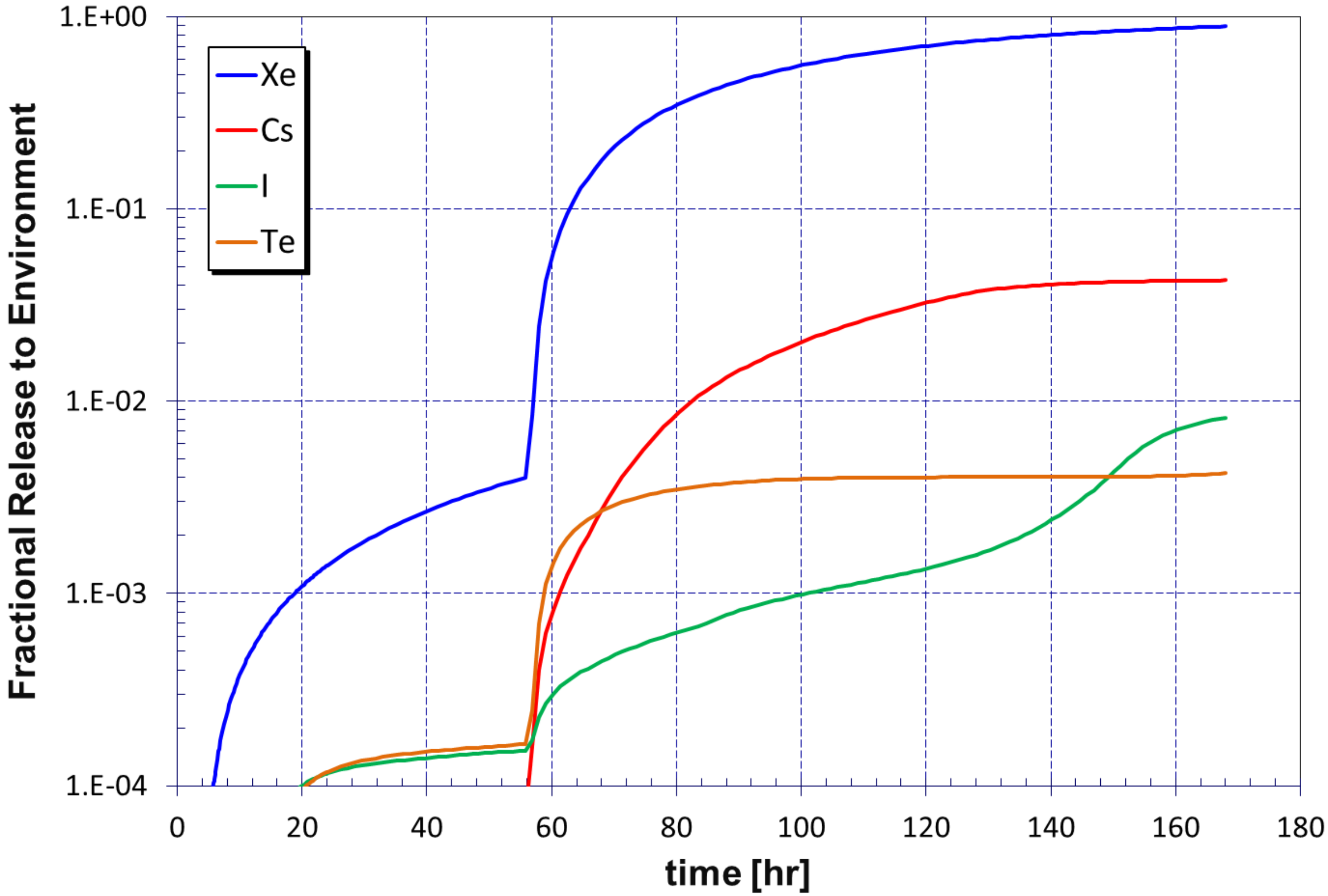


Figure 7-9: Fractional Release of Volatile Fission Products to the Environment

7.2 Supplemental Information for Section 4.5

7.2.1 Supplemental Information on Sensitivity Analysis on Effect of a Stuck Open Pressurizer Safety Relief Valve (MU-5.1)

In Case 1A2 briefly discussed in Section 1.1 of Appendix B (an SBO with early containment failure due to a global hydrogen deflagration), the primary SRVs cycle 118 times during the period of 12.6 and 16.6 hours. The PORVs do not cycle since they are unavailable after 4 hours due to end of battery life. **Figure 7-10** shows the primary-side pressure holding around 17 MPa as the SRVs are cycled. Hot leg nozzle creep rupture occurs at 16.6 hours causing the pressure to quickly drop off.

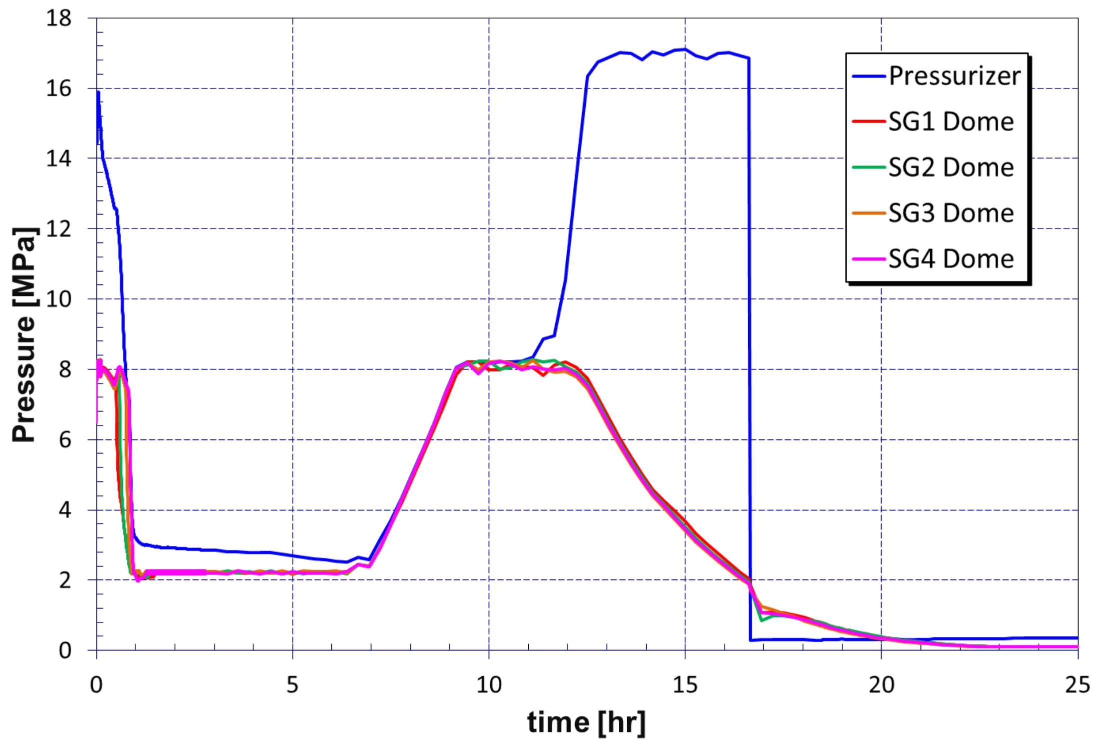


Figure 7-10: Pressure in the RCS and Steam Generators

When the SRV initially opens, it passes liquid water, transitioning to steam at 13 hours, as seen in **Figure 7-11**.

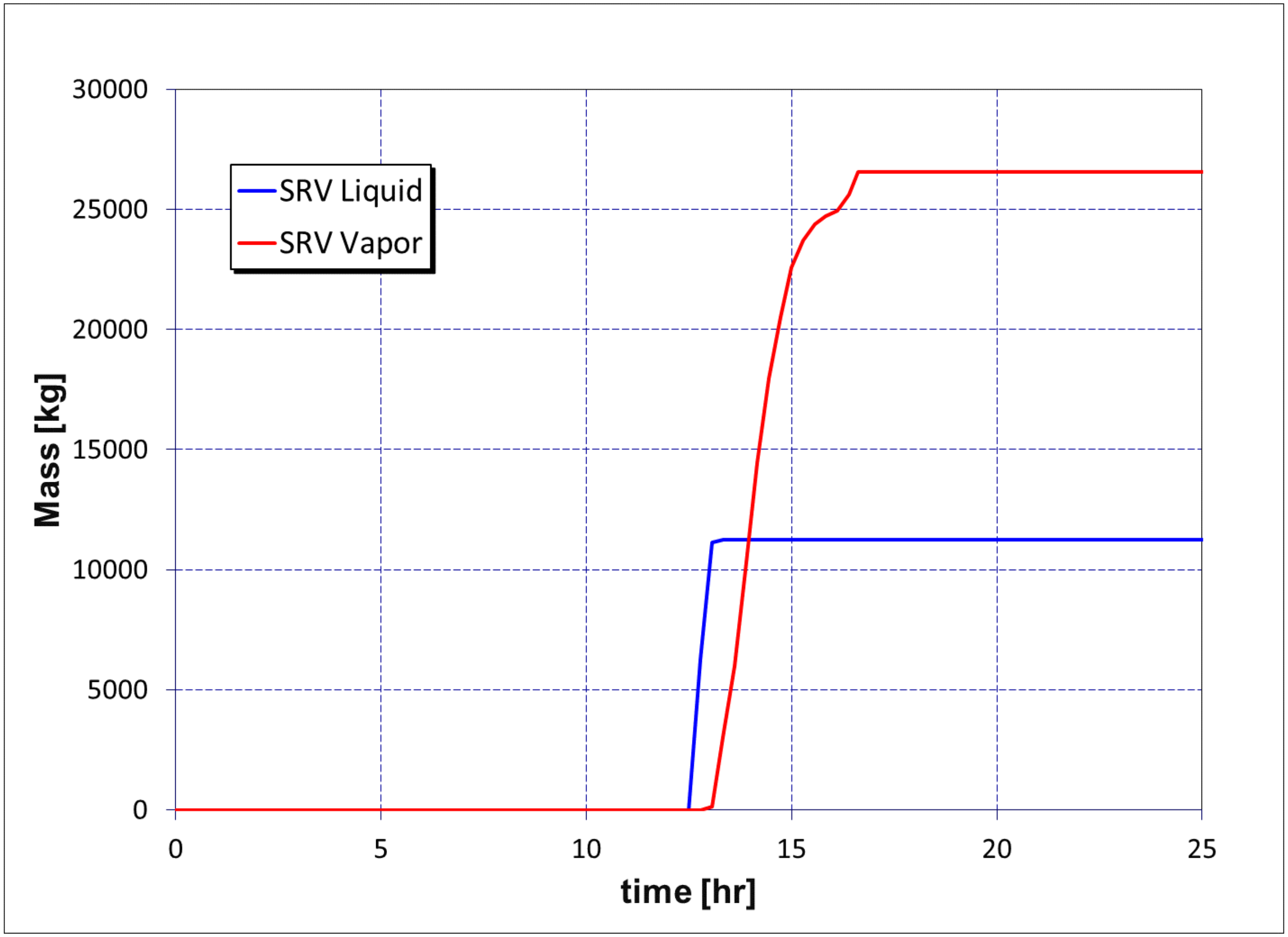


Figure 7-11: Mass of Water Liquid and Vapor Passing Through the SRV

Figure 7-12 provides the temperature within the pressurizer which rises to and levels off around 250F. The peak in vapor temperature coincides with peak nodal clad temperature exceeding 2200F (1204 C).

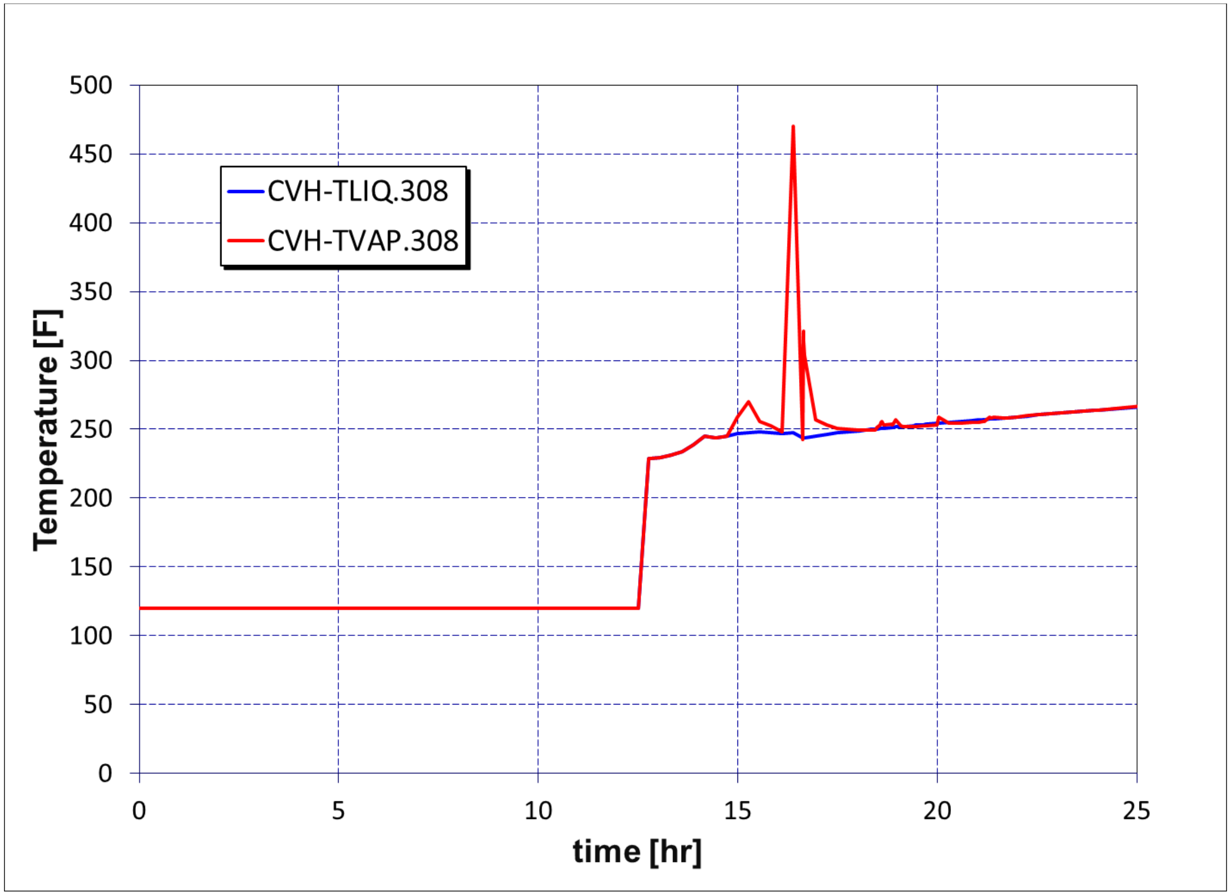


Figure 7-12: Vapor and Liquid Temperatures Within the Pressurizer

Given these conditions, it is not unreasonable to assume that the SRV may become stuck open over the interval of 12.6 and 16.6 hours. For the first sensitivity (see Section 4.5.3 on MU-5.1A), the SRV is assumed to fail open on its first cycle (around 12.6 hours). The second sensitivity (see Section 4.5.3 on MU-5.1B) assumes failure of the valve at 14.5 hours, in order to create three distinct realizations (failure upon first lift, failure roughly halfway between re-pressurization and hot leg creep rupture [HLCR)] and moot because of HLCR).

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Appendix D: Modeling and Other Issues Affecting the Level 2 Reactor, At-Power, Internal Event and Flood PRA for the Level 3 PRA Project

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1. Use of MELCOR in predicting Aqueous Releases

Due to the current state-of-practice and past assessments that showed that aqueous releases pose less overall public health and environmental risk than airborne releases, the L3PRA Project Level 2 probabilistic risk assessment (PRA) does not consider aqueous releases. To address this issue, an evaluation was done to determine whether MELCOR analysis results could be used as input to separately study accident aqueous releases.

MELCOR tracks the transport of radiological material within the containment (and within other structures that are prescribed in the input model), whether that material is in a gaseous or aqueous phase. In MELCOR terminology, each control volume within the defined system has multiple phases that can occupy the control volume (atmosphere, pool, fog, etc.), and these thermodynamic constituents are tracked in the control volume hydrodynamic (CVH) package. In concert, the RN (radionuclide) package tracks the radiological material that is in these constituents, by chemical class. Each chemical class is initially populated by the core material, and as core degradation phenomena occur (clad rupture, fuel melting) this material migrates out of the core. While in the gas phase, MELCOR will model most of the physics that this radiological material may experience, including gravitational settling, agglomeration, plate-out, thermophoresis, etc. However, once the material goes in to an aqueous solution, the material is assumed to be homogenously mixed in the control volume's pool, and very little physics or chemistry is modeled (because MELCOR is generally not used to estimate aqueous environmental releases). Two notable exceptions are that limited chemistry modeling can be employed to account for the effects of sump pH on iodine retention, and separately, the decay heat given off by the radiological material is added to the pool's enthalpy. Consistent with past studies (e.g., Chang, 2012a) the iodine chemistry models were not active for the L3PRA Project calculations. Also note that if the pool (aqueous phase) in a particular control volume dries out (boils off), the radiological material is deposited on "floors" (i.e. horizontally oriented heat structures) if any are present. Otherwise, the radiological material is re-introduced in to the control volume atmosphere phase. If the pool drains in to, is pumped in to, or is hydro-dynamically blown in to, a different control volume, then the radiological material is carried along homogenously (i.e., settling and entrainment effects are not captured).

The MELCOR input model for the L3PRA Project uses roughly a dozen control volumes to nodalize the containment. Roughly another dozen control volumes represent the tendon gallery, surrounding structures, and the environment. Any radiological material that leaves containment in aqueous solution would end up in either a general control volume representing the environment, the tendon gallery, and/or one or more levels of the auxiliary building model. The MELCOR model also has a flow path from the reactor cavity representing basemat melt-through, however that flow path is clamped shut by default (basemat melt-though is manually assigned based on the ablation depth exceeding the concrete depth). The time-dependent mass of radiological material in each of these control volumes, by chemical class, is attainable via normal parameter outputs and/or control functions.

Reference:

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2. Treatment of Bypass and Unisolated Containment Events with Low Frequency

There are four classes of bypass and unisolated containment events typically captured by Level 2 PRAs. These are:

- Interfacing system LOCAs
- Any core damage sequence that is coincident with containment isolation failure
- Steam generator tube ruptures
- Steam generator tube ruptures induced by severe accident conditions

In the case of the internal events and floods L3PRA Project Level 1 PRA model (R02), in conjunction with the bridge tree (containment systems event tree), the first three of these classes have very low frequency relative to other potential plant damage states. The contributing initiators are highlighted in **Table 2-1** below.

Table 2-1: Breakdown of Bypass and Unisolated Containment Frequency Contributions

	Initiating event frequency (/yr)	Conditional core damage probability	Total Level 1 frequency (/yr)	Containment Isolation Failure probability
Interfacing system LOCAs initiators	--	--	3.4E-7	--
Sequences that include containment isolation failure	--	--	5.5E-5[(1)]	1.2E-3[(2)]
Steam generator tube rupture initiators	1.38E-3	1.1E-4	1.5E-7	--
Most of the results are based on the R02 model version and SAPHIRE code version 8.1.4.6 solved at a truncation of 1E-11; however, for convenience, some values were calculated using version 8.1.4.8 of the SAPHIRE code, which produces the same results.				

2.1. Interfacing System LOCAs

When solved using a truncation of 1E-11/yr, there are 601 plant damage state (PDS) cut sets associated with an interfacing system loss of coolant initiator (ISLOCA). Of these 601 cut sets, 148 include an initiator related to a low-pressure safety injection (LPSI) ISLOCA and represent 90.13% of the ISLOCA contribution to the total PDS frequency. These cut sets include one of the following initiating events: 1-IE-ISL-RHR-HLS, an ISLOCA via the hot leg suction lines; 1-IE-ISL-RHR-CLI-A or 1-IE-ISL-RHR-CLI-B, an ISLOCA via the cold leg injection lines. The remaining 453 of the 601 cut sets include the initiator 1-IE-ISL-RCP-S1LO as the initiating event, which represents an ISLOCA via the reactor coolant pump seal leak-off return line. The top six of the 601 cut sets consist of common cause failures (CCFs) of hot leg suction isolation valves and represent 58% of the ISLOCA contribution to the total PDS frequency. The top 22 of the 601 ISLOCA cut sets represent 87.36% of the ISLOCA contribution to the total PDS frequency.

As part of the Level 3 PRA Project, issues were identified pertaining to the modeling (and quantifying) of reactor coolant system and emergency core cooling system ISLOCA sequences. Some very limited data was identified that implied the potential for CCF of isolation valves (i.e.,

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1 This value represents the sum of the frequencies of the minimal set of Level 1 CD cut sets that pass through the 1-CIS failure branch in the 1-PDS-Q event tree.

2 The value represents the solution to the 1-CIS fault tree at a truncation of 1E-11.

large internal leakage) that could result in an ISLOCA. Due to the large uncertainty related to this data, and the risk-significance associated with ISLOCAs, an expert elicitation was performed to address these issues. A brief overview of the ISLOCA expert elicitation is provided in Section 3.5.1 of the Nuclear Regulatory Commission's (NRC's) report for the Level 1, at-power, reactor PRA for internal events (NRC, 2022).

Based on the general ISLOCA insights from NRC-sponsored research in the late 1980's and early 1990's, the results of the Reference Plant PRA model ISLOCA analysis, and the ISLOCA expert elicitation, a three-valve failure screening criterion was applied to determine which ISLOCA pathways to include in the L3PRA Project Level 1, at-power, reactor PRA model for internal events. This criterion simply states that if three or more valves are needed to fail to lead to an ISLOCA, then the applicable ISLOCA pathways are screened out from further consideration. The application of this criterion results in the inclusion of four ISLOCA pathways in the L3PRA Project Level 1 model, as noted below:

- Residual heat removal (RHR) system via hot leg suction lines from the reactor coolant system (RCS)
- RHR system via the cold leg injection lines to the RCS
- Auxiliary component cooling water system via the reactor coolant pump (RCP) thermal barrier heat exchangers
- RCP stage 1 seal leak-off

2.2. Containment Isolation Failures

The containment isolation system functionality during an accident sequence is modeled in the containment isolation system fault tree 1-CIS. Much of the functionality and structure of the Reference Plant's containment systems models are adopted in the SAPHIRE containment systems models without change. Several containment penetrations are excluded from the 1-CIS fault tree based on whether the penetration:

- is less than two inches in diameter
- is an administratively-controlled mechanical and fluid system containment penetration
- isolated a system that is closed to both the RCS and containment atmosphere
- isolated a system that is closed to the environment outside of containment
- connects containment through the containment sump suction lines to the refueling water storage tank (RWST)

At a truncation of 1E-11/yr, the solution to the 1-CIS fault tree consists of 163 cut sets, which sum to a total probability of 1.2E-3. The top four cut sets each consist of a single basic event that collectively constitutes 99.74% of the total 1-CIS solution probability, as shown in the following table:

Table 2-2: Containment Isolation Cutset Contributions

Cut Set	% Contribution	Probability
1-L2TEAR	92.67	1.11E-3
1-CIS-AOV-OO-2626_27B-CC	2.44	2.93E-5
1-CIS-AOV-OO-HV28_29B-CC	2.44	2.93E-5
1-CIS-AOV-OO-HV780781-CC	2.44	2.93E-5

The pre-existing containment leak probability 1-L2TEAR is based on a leakage rate of 100 L_a or greater, as taken from (Westinghouse, 2004). A 2-inch diameter hole-size is used to determine the representative containment leakage rate for cut sets that included this basic event; however, the 1-L2TEAR probability covers a break range of 1.2-inch equivalent and greater. Section 6 of this report addresses the derivation of this leakage size. The next highest cut set in the 1-CIS solution is an independent electrical component failure and operator failure to isolate. The remaining lower order cut sets are characterized by multiple independent component failures.

CCF modeling in the containment isolation system is included for three penetrations that are isolated by the following valve pairs in the following lines:

- Normal containment sump discharge, 3-inch, air-operated gate valves
- Normal containment purge supply and equalizing, 14-inch, air-operated butterfly valves
- Normal containment purge supply and equalizing, 14-inch, air-operated butterfly valves

Two of the containment penetrations each have a pair of 24-inch containment purge and exhaust lines and a pair of 14-inch containment purge and exhaust lines, for a total of four lines through each penetration. Although the 24-inch containment purge and exhaust lines are isolated by valve pairs and would require CCF modeling, the 24-inch containment purge and exhaust lines are normally isolated (closed) under administrative controls during plant operation and were therefore screened out by the second screening criterion. CCF of the isolation valve pairs on the 14-inch lines is modeled because they serve the containment mini-purge and exhaust systems that are allowed to be open intermittently during plant operations.

For the containment penetrations associated with the containment spray system, there is no accounting for any component failure due to common cause since no two valves that would fall in the same common cause group act as redundant barriers against containment leakage. It is assumed that the same basis could be applied to exclude CCF modeling of the valves for a 4-inch penetration that is isolated with an outboard air-operated valve and inboard check valve.

The following list of PDS end states represents those PDS end states that have a containment isolation failure (CIF) in the associated sequence logic and have contributing cut sets:

- PDS-02-6
- PDS-03-5
- PDS-05-6
- PDS-11-5
- PDS-11-6
- PDS-12-5
- PDS-12-6
- PDS-19-6
- PDS-23-5
- PDS-25-6
- PDS-29-5
- PDS-29-6
- PDS-35-5
- PDS-35-6
- PDS-43-5
- PDS-49-6
- PDS-63-5

These 17 PDS end-states contain a total of 565 cut sets when quantified using a truncation of 1E-11/yr. Of the 565 CIF cut sets, 493 cut sets include 1-L2TEAR and the remaining 72 cut sets are lower order 1-CIS cut sets.

Strictly for an historical perspective, it is noted that the failure probability estimates stemming from this evaluation are significantly lower than those from (Pelta,1985), which estimated the probability of a small leak that violates Technical Specifications to be 0.3 and the probability of a large leak to be 0.001 to 0.01.

2.3. Steam Generator Tube Ruptures

The final two classes mentioned above (spontaneous and consequential steam generator tube ruptures), do not warrant any additional special attention here. Spontaneous tube ruptures are modeled in the Level 1 PRA and have a low core damage frequency as captured in the earlier table. Despite this low frequency, they are carried forward and evaluated in the Level 2 PRA. Consequential steam generator tube ruptures are germane to the Level 2 (as opposed to Level 1) PRA, and thus their contribution to the overall risk is represented by a separate release category (1-REL-ISGTR). The consequential steam generator tube ruptures are discussed in the Level 2 PRA main report, and in **Section 5** of this appendix.

2.4. Pre-initiating Event Human Failure Events

To account for the modeling of pre-initiating event human failure events in the NRC's Level 1 at-power reactor PRA model, an evaluation was performed that surveyed the Idaho National Laboratory data (INL, 2019) used to calculate the failure probabilities for a sample of component types. The conclusion from this evaluation is that the pre-initiating event human failure events data is well-represented in the component failure data and that explicit modeling of the contribution of pre-initiating event human failure events is unnecessary and redundant. As such, the changes that were incorporated into the NRC's containment systems models do not involve the explicit modeling of pre-initiating event human failure events.

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3. Cesium Molybdate Treatment (Ex-vessel)

Releases from VANESA (i.e., the MELCOR subroutine that handles fission product release from the cavity) go into the cesium (Cs) and Molybdenum (Mo) classes, with no effort to recombine them as Cs_2MoO_4. Thus, VANESA essentially decomposes Cs_2MoO_4 into Cs and Mo.

The change was made to the 1.8.6 and 2.1 codes in early 2010. The problem was that MELCOR was not conserving the mass of Cs_2MoO_4 when it was mapped to VANESA class 19 (Cs). Thus, it was necessary to change the default treatment so that Cs_2MoO_4 is decomposed to Cs and Mo in the cavity. Training materials were updated to indicate that the old default mapping should not be used. For the L3PRA Project, MELCOR model revision 3 (and beyond) adopt the correct mapping.

The issue of properly treating the chemical form of Cs and Mo pushes the state-of-knowledge, but it's treatment here is viewed as reasonable based on the following perspective. The deposition patterns in the upper plenum of the Phebus test section suggest that cesium is in the chemical form of cesium molybdate (JRC, 2015). It is believed that Cs and Mo diffuse out of the fuel as atoms and later combine to form Cs_2MoO_4. This combination would be affected by local oxidation/reduction conditions. MELCOR does not explicitly account for all these processes, and the user must prepopulate the Cs/Mo class in an attempt to simulate observed cesium behavior in the RCS and in containment. This treatment then breaks down because VANESA does not include a Cs_2MoO_4 class, and so MELCOR must decompose Cs_2MoO_4 into Cs and Mo when the corium enters the reactor cavity to prevent large mass conservation errors. Whether Cs and Mo would combine in the cavity to form Cs_2MoO_4 and whether Cs and Mo would combine following release from the cavity to the containment atmosphere are not well understood. Within the timeframe of the L3PRA Project, the project team was not aware of any experiments that could be used to determine (with confidence) the chemical form of Cs and Mo following ex-vessel release.

References:

JRC, 2015 European Commission, Joint Research Centre, *Circuit and Containment Aspects of Phébus Experiments FPT0 and FPT1*, JRC Science and Policy Report, Institute for Energy and Transportation, ISBN 978-92-79-47900-7, 2015.

4. Comparison of MELCOR Results to Other Relevant Analyses

To provide perspective on how comparable the MELCOR results are to analyses for similar plants, a series of comparisons was performed that target scenarios analyzed for the L3PRA Project that have relevant counterparts (see below). The information discussed below provides confidence that the MELCOR results are reasonable for the purposes for which they are being used.

Table 4-1 compares, at a very high-level, the source terms computed for the L3PRA Project against those generated for the Reference Plant individual plant examination (IPE) submittal. In some cases, releases for station blackout in the L3PRA Project values are much higher than the IPE, owing somewhat to the longer accident termination times used (to encompass late containment failure), as well as the occurrence of an assumed hydrogen combustion event in one case that fails containment. It is also seen that the “non-volatile” release is much higher, due to changes in the modeling of molybdenum volatility during sustained molten core - concrete interaction (MCCI). Containment isolation failure estimates are comparable (other than the late molybdenum releases due to the longer accident termination time). The steam generator tube rupture (SGTR) ranges are comparable between the two studies, including the L3PRA Project’s treatment of induced SGTR. The ISLOCA release magnitudes are much lower in the L3PRA Project, owing to the treatment of passive and active retention mechanisms in the auxiliary building. For the final category, “All Others,” two sets of values for the current study are given. The first set is the releases at the end of the germane calculations (after ultimate containment failure) and they are significantly larger than their IPE analogues. The bracketed values are for the same calculations, but taken just prior to containment failure, and these are equivalent to the IPE results. This shows the effect of continuing the calculations beyond a shorter accident termination time, which has the benefit of capturing additional phenomenological events and the detriment of producing increasingly speculative results (both in terms of phenomenological modeling and accident management actions). This latter issue is discussed in more detail in **Section 21** of this report.

Table 4-2 provides a high-level comparison between the L3PRA Project results and those of the State-of-the-Art Reactor Consequence Analysis (SOARCA) Surry analysis (Chang, 2012a). Where notable differences in results are evident, the reason for the differences is explained. The differences are typically a combination of: (i) differences in plant design, (ii) differences in phenomenological and system modeling assumptions, and (iii) fundamental differences in the scope of the studies and the assessment technologies employed (PRA versus consequence analysis).

Table 4-1: Comparison of Source Terms with the Reference Plant IPE

	Reference Plant IPE	L3PRA Project
Blackout (with successful containment isolation)		
Onset of core melt (IPE) or time of severe accident mitigation guideline (SAMG) entrance (NRC L3 PRA), in hours	2.5 – 23	3 - 139
Cumulative noble gas release (%)	< 1%	Up to 99%
Cumulative volatile* fission product release (%)	< 1%	Up to 4%
Cumulative non-volatile** fission product release (%)	< 1%	Up to 14%
Containment isolation failure		
Onset of core melt (IPE) or time of SAMG entrance (NRC L3 PRA), in hours	9	16
Cumulative noble gas release (%)	92%	Up to 98%
Cumulative volatile* fission product release (%)	3%	Up to 4%
Cumulative non-volatile** fission product release (%)	< 1%	Up to 8%
SGTR (and for the current study, ISGTR)		
Onset of core melt (IPE) or time of SAMG entrance (NRC L3 PRA), in hours	23	49 to 95 (and 10)
Cumulative noble gas release (%)	96%	Up to 92% (and 95%)
Cumulative volatile* fission product release (%)	41%	Up to 34% (and 23%)
Cumulative non-volatile** fission product release (%)	4%	6% (and 4%)
Interfacing Systems LOCA		
Onset of core melt (IPE) or time of SAMG entrance (NRC L3 PRA), in hours	9	3 - 10
Cumulative noble gas release (%)	100%	Up to 99%
Cumulative volatile* fission product release (%)	90%	Up to 14%
Cumulative non-volatile** fission product release (%)	26%	Up to 3%
All Others		
Onset of core melt (IPE) or time of SAMG entrance (NRC L3 PRA), in hours	0 to 20	10 to 15
Cumulative noble gas release (%)	< 1%	Up to 99% [<1%]***
Cumulative volatile* fission product release (%)	< 1%	Up to 16% [< 1%]***
Cumulative non-volatile** fission product release (%)	< 1%	Up to 21% [< 1%]***

* In the L3PRA Project, this refers to the higher of cumulative I and Cs release

** In the L3PRA Project, this refers to the higher of cumulative Ba, Ru and Mo release; these classes become volatile during sustained MCCI due to depletion of other metals

*** The values in brackets are maximums prior to containment failure

Table 4-2: Comparison of L3PRA Project Internal Events, Internal Floods, Level 2 PRA Results to the SOARCA Surry Study

Containment Failure Mode	NUREG/CR-7110 Volume 2 (NRC, 2013b)	L3PRA Project Level 2 PRA	Comments
Interfacing System Loss-of-Coolant Accident	Significant release** at 13-14 hours 2-3% Cs release at 48 hours	Significant release at 3-8 hours <1 to 13% Cs release at 48 hours Release frequency is <1% of the total release frequency	SOARCA credits operator actions (even for the unmitigated case) that delay the onset of core damage. Without these actions accomplished, MELCOR predicts the onset of core damage much earlier. SOARCA analysis includes significant retention for all realizations due to turbulent deposition in piping and flooding on the backside of the piping (which suppressed re-volatilization). These mechanisms are either not credited (turbulent deposition) or do not apply to all L3PRA Project sequences (backside flooding).
Steam Generator Tube Rupture – Spontaneous	Radionuclide releases are not provided due to the low likelihood that operators would not prevent core damage	>50 hours before a significant release <1 to 25% Cs release Release frequency is <1% of the total release frequency	
Steam Generator Tube Rupture - Induced	Significant release at 6-12 hours < 1% Cs release at 4 days	Significant release at 11 hours Up to 9% Cs release Release frequency is <1% of the total release frequency	Timings are generally comparable between the two studies. The primary difference between the two studies is the timing of hot leg failure following SGTR, with the time period in the L3PRA Project being much longer (thus resulting in more containment bypass releases).

Table 4-2: Comparison of L3PRA Project Internal Events, Internal Floods, Level 2 PRA Results to the SOARCA Surry Study

Containment Failure Mode	NUREG/CR-7110 Volume 2 (NRC, 2013b)	L3PRA Project Level 2 PRA	Comments
Containment Isolation Failure	Not analyzed	Significant releases starting at ~18 hours Up to 3% Cs release Release frequency is <1% of the total release frequency	-
Early Energetic Induced Failures	Not analyzed	Significant release at 20+ hours <1% to 16% Cs Release frequency is <1% of the total release frequency	These L3PRA Project release categories have to do with phenomena such as hydrogen detonation, direct containment heating, vessel rocketing, and steam explosions that were excluded from SOARCA on the basis of low probability.
Late over-pressure failure	Significant release at 30-74 hours < 1% Cs release at 4 days	Earliest over-pressure failure is at 48 hours <1% to 4% Cs release Release frequency is roughly half of the total release frequency	Results are generally comparable. For the largest release in this category (4%) in the L3PRA Project, the release is actually smaller (~2%) at the comparable time (4 days).
Basemat Melt-Through	Not analyzed	No significant airborne release Minimal Cs releases Release frequency is ~1% of the total release frequency	Since Surry has a smaller free volume (sub-atmospheric design), long-term over-pressure failure is more prominent relative to basemat melt-through. Additionally, SOARCA generally assumes that containment flooding occurrs at 48 hours and is successful in terminating MCCI.

Table 4-2: Comparison of L3PRA Project Internal Events, Internal Floods, Level 2 PRA Results to the SOARCA Surry Study

Containment Failure Mode	NUREG/CR-7110 Volume 2 (NRC, 2013b)	L3PRA Project Level 2 PRA	Comments
Intact (or no core damage)	Multiple scenarios led to no core damage or otherwise intact containment with minimal or no radiological releases	No significant airborne release Minimal Cs releases Release frequency is roughly one-third of the total release frequency	

4.1. Comparison of station blackout pre-core damage results to various sources

A comparison was performed of Level 1 PRA station blackout success criteria and sequence timing information generated using the L3PRA Project MELCOR model to a handful of different sources (Westinghouse topical reports, Westinghouse Emergency Operating Procedure Bases, Reference Plant MAAP calculations). From those analyses the following are observed:

- Core uncovery timings estimated using the L3PRA Project MELCOR model are generally comparable to the SPAR and Reference Plant-generated MAAP4 uncovery timings. For 182 gpm/RCP and 480 gpm/RCP leak sizes, the L3PRA Project analysis predicts core uncovery somewhat earlier than the other two sources.
- The predicted time to core damage occurs later for the L3PRA Project MELCOR calculations for leaks driven by loss of feedwater (the 21 and 76 gpm/RCP cases), which is expected for that set of calculations because the analogous MAAP4 calculations only credit feedwater to 2 SGs. For the larger leaks, the results are comparable.
- Specific to 21 gpm/RCP simulations, the L3PRA Project MELCOR model predicts earlier steam generator (SG) dryout for unfed SGs (relative to the Reference Plant MAAP4 results) and consistent SG dryout for fed SGs. RCS pressure between the two sets of results is very comparable, and the timing of core uncovery shows excellent agreement. Once the core becomes completely uncovered, the MELCOR simulations predict significantly slower boiling dry of the region below the core, which delays the time of core damage notably.

One of the overall conclusions of the comparative analysis is that the MELCOR results are reasonable when compared to other relevant information sources.

4.2. Comparison of station blackout results to the draft Surry SOARCA Uncertainty Analysis study

The draft Surry SOARCA Uncertainty Analysis study documented in (Ross, 2016a) focuses extensive resources on understanding the (primarily epistemic) severe accident modeling uncertainty of a short-term station blackout (i.e., immediate loss of alternating current [ac] power, direct current [dc] power, and turbine-driven auxiliary feedwater) scenario, including the potential for an induced consequential steam generator tube rupture. **Table 4-3** presents these results for the scenarios where a consequential steam generator tube rupture (C-SGTR) does not occur, and shows:

- Very comparable timings, and delta-timings, for events (e.g., start of core uncovery, dryout of lower plenum water) that are not directly affected by the differences in boundary and initial conditions;
- Very comparable amounts of in-vessel hydrogen production, particularly given the very broad spread of results within the draft Surry uncertainty analysis (UA);
- Large differences in the ex-vessel hydrogen and carbon monoxide production, with the L3PRA Project results being higher – the reason for this was not investigated further. Note that both analyses treat the concrete as being basaltic.
- Very comparable environmental releases of volatile chemical classes at 48 hours, and somewhat higher environmental releases at later times for the L3PRA Project (accident termination timing effect).

Table 4-3: Comparison of Present 1B/1B2 Results to 2016 Draft Surry UA

Value/Timing	**L3PRA R02_L2 Case 1B** (ST-SBO)	**L3PRA R02_L2 Case 1B2** (ST-SBO with stuck power-operated relief valve [PORV])	**2016 Draft Surry SOARCA UA Base Case**	**2016 Draft Surry SOARCA UA Range for non-SGTR Cases**
Start of core uncovery (level at top of active fuel [TAF])	2.4	2.4	2.7	
Pressurizer valve sticks open (hours)	-	2.0	3.0	
Complete core uncovery (level at bottom of active fuel)	3.2	3.1	3.1	
First gap release (hours)	3.6	3.2	3.4	
Spatially maximized clad temperature exceeds 1478K (2200 °F) (hours)	3.9	3.3	3.8	
Creep failure of the hot leg nozzles (hours)	4.5	-	4.5	
Dryout of lower plenum water (hours)	6.4	6.1	5.6	
Vessel breach (hours)	7.7	6.1	7.7	
Containment overpressure failure (hours)	48	56	41.1, >48[1]	
Basemat melt-through (hours)	106	99	>48	
Number of pressurizer PORV cycles	243	Sticks on 63rd lift (1st water lift)	n/a	
Number of pressurizer safety relief valve (SRV) cycles	16	0	Sticks on 45th lift	
Number of SG SRV cycles	59	59	Sticks on 45th lift	
In-vessel hydrogen production from Zircaloy and steel oxidation (kg)	583	548	515	160 to 680
Pressurizer pressure just before vessel breach (MPa)	0.32	4.7	0.24	

Table 4-3: Comparison of Present 1B/1B2 Results to 2016 Draft Surry UA

Value/Timing	**L3PRA R02_L2 Case 1B** (ST-SBO)	**L3PRA R02_L2 Case 1B2** (ST-SBO with stuck power-operated relief valve [PORV])	**2016 Draft Surry SOARCA UA Base Case**	**2016 Draft Surry SOARCA UA Range for non-SGTR Cases**
Ex-vessel hydrogen production (kg)	3000 (48h) 5100 (168h)	3000 (48h) 5100 (168h)	1250	
Ex-vessel carbon monoxide production (kg)	4500 (48h) 8100 (168h)	4500 (48h) 8100 (168h)	2100	
Cumulative Cs release to the environment	<0.001 (48h) 0.01 (168h)	<0.001 (48h) 0.04 (168h)	<0.001	~0 to 0.002
Cumulative I2 release to the environment	<0.001 (48h) 0.01 (168h)	<0.001 (48h) 0.08 (168h)	<0.001	~0 to 0.003
Cumulative Mo release to the environment	<0.001 (48h) 0.04 (168h)	<0.001 (48h) 0.05 (168h)		
End of calculation (hours)	168	168	48	

[1] These are the times of liner yield and rebar yield, respectively.

Table 4-4 presents some additional comparisons for the environmental release for C-SGTR and shows that the L3PRA Project results (which assume a single tube failure) are comparable to the single-tube draft Surry UA results. The Surry UA sensitivity also includes multi-tube failures, which demonstrate a very wide range of results.

Table 4-4: Comparison of L3PRA Project 3A3 Results to 2016 Draft Surry UA

Value/Timing	**L3PRA R02_L2 Case 3A3** (non-station blackout [SBO] with no emergency core cooling system [ECCS])	**2016 Draft Surry UA (Single-Tube) Range**	**2016 Draft Surry UA Sensitivity (Multi-Tube) Range**
Time between first gap release and C-SGTR (hours)	-0.2 (specified as a boundary condition)	Not reported	Not reported
Time between C-SGTR and hot leg creep rupture (hours)	0.8 hours (specified as a boundary condition)	Not reported	Not reported
Cumulative Cs release to the environment	0.04	0.007 to 0.05	0.02 to 0.31
Cumulative I2 release to the environment	0.08	0.01 to 0.13	0.07 to 0.6
End of calculation (hours)	168	48	48

4.3. Comparison of Loss-of-Nuclear Service Water (Case 2) and Electrical Distribution Failures (Case 4) to Similar Reference Plant MAAP Results

For loss-of-nuclear service water (Case 2) and electrical distribution failures (Case 4) leading to situations where both trains of ECCS and containment sprays/coolers are disabled coincident with elevated RCP seal leakage, the Reference Plant had SBO MAAP cases that approximated these conditions with several key deviations. The three major deviations are the timing of equipment loss, the nature of the operator-enacted depressurization, and the availability of cold-leg accumulators.

In the Reference Plant cases, all systems powered by AC-power are lost at time zero, whereas in the L3PRA calculations, systems are lost at different times. Also, in the Reference Plant calculations the operator-induced depressurization rate is at the maximum rate (as specified in the ECA-0.0 procedure used for those analyses) versus being at 100°F/hour in the L3PRA calculations (consistent with procedures since it is a non-SBO situation). Finally, and most importantly, when paired with the fact that the system has been depressurized by operator action, the Reference Plant calculation discredits cold leg accumulators, while the MELCOR analysis includes them. (They are not queried in the reference plant PRA or the L3PRA project; however, their independent failure probability would greatly skew the associated sequence frequency and Level 2 PRA impacts.) **Table 4-5** captures the differences in boundary conditions and results between the various cases.

Table 4-5: Comparison of Cases 2 and 4 to Similar Reference Plant MAAP Results

Condition	NRC MELCOR Case 2	NRC MELCOR Case 4	Reference Plant MAAP sbocase7	Comments
Auxiliary feedwater (AFW) flow characteristics	Ample AFW to 4/4 SGs		570 gpm to 2/4 SGs	Substantively equivalent
Duration of feedwater	Beyond the time of core damage			
Seal leak rate / time	182 gpm/RCP starting at 43 minutes	182 gpm/RCP starting at 2.2 hours	182 gpm/RCP starting at 13 minutes	Failures occur at different times based on cause
Accumulators	4/4 available		0/4 available	See discussion above
PRZ PORVs	Available throughout	1 fails upon battery depletion	Available until battery depletion	Substantively equivalent for this scenario
Time of battery depletion	4 hours after loss of ac	1 train lost 4 hours after loss of ac bus	4 hours after loss of ac	Substantively equivalent for this scenario
Depress & cooldown started at / rate	30 minutes after seal failure (~1.2 hr) / 100°F per hr.	2.7 hours / 100°F per hr.	1 hour / maximum	Different timings based on cause
Reactor, main feedwater, & RCP Trip	5 minutes (manual)	2 hours	0 minutes (loss of all ac power)	Failures occur at different times based on cause

Table 4-5: Comparison of Cases 2 and 4 to Similar Reference Plant MAAP Results

Condition	NRC MELCOR Case 2	NRC MELCOR Case 4	Reference Plant MAAP sbocase7	Comments
Core level at TAF	5.8 hours*	5.6 hours*	4.2 hours*	TAF onset somewhat delayed in MELCOR calculations by slower depressurization rate and heatup greatly delayed by accumulator injection with reduced RCS leakage rate due to depressurization
Core-exit thermocouple > 1200F	12.8 hours*	12.7 hours*	5.2 hours*	
Core damage	14.6 hours*	13.2 hours*	5.5 hours*	

* These times have been adjusted to reflect the timing of this event relative to the loss of seal integrity, to make this a more meaningful comparison

References:

Chang, 2012a — Chang, R., et al., *State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*, NUREG-1935, November 2012. [ML12332A057]

Ross, 2016a — Ross, K., et al., *State-of-the-Art Reactor Consequence Analysis Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station*, Draft Report, January 2016. [ML15224A001]

5. Consequential Steam Generator Tube Rupture

Modeling of consequential steam generator tube rupture (C-SGTR) differs between the NRC L3PRA Project PRA and the current large early release frequency (LERF) best practices modeling used in the NRC's generic C-SGTR regulatory research of this issue. For the purpose of this discussion, the two are referred to as the "NRC L3PRA Project PRA" work and the "NRC generic C-SGTR" work.

Deterministic analysis using the L3PRA Project reactor MELCOR model broadly agrees with the findings of the current NRC generic C-SGTR work documented in (NRC, 2017a), though the NRC L3PRA Project PRA work does not rigorously model tube degradation or tube bundle temperature variations. Based on this global similarity and considering the modeling rigor, the NRC L3PRA Project PRA leverages the other NRC generic C-SGTR work (most notably the "C-SGTR Calculator" described in (NRC, 2017a)) in order to estimate the conditional failure probabilities. The inputs used for this process, and the results, are also described below. Following this, four areas where the two modeling approaches deviate or appear to deviate are identified. These deterministic modeling differences do not result in a significant change in the expected importance of these types of events (i.e., very low C-SGTR frequency).

5.1. Failure probability estimation

Two specific calculations were used as the inputs for use of the C-SGTR calculator in the NRC L3PRA Project R01_L2 (2014) PRA. These involve high-dry sequences with and without a depressurized secondary side (in the second case, SG number 2 depressurizes prior to core damage). From these calculations, the inner surface temperatures for the hot leg, surge line, SG tubes in up-flow, and SG tubes in down-flow were extracted, along with the pressures in the hot leg (centerline of the volume closest to the vessel for 1 loop) and the lower portion of the secondary side of the boiler region. In the case with the depressurized SG, primary-side temperatures were taken from the loop with the depressurized SGs (which are higher than those in the other loops).

From these simulations, the C-SGTR calculator was applied to develop a failure probability representing the likelihood that high-dry-low conditions would lead to a consequential steam generator tube rupture. The conditional probability of C-SGTR for the L3PRA Project for the high-dry-low simulation was estimated to be: P(CSGTR) = 0.024 for a total leak area of 4.3 cm^2 (0.66 sq. inches) – which is approximately equivalent to a double-ended guillotine break of a single tube used in the L3PRA Project (3.7 cm^2).

The uncertainties associated with this estimate support the assumption that the probability for the high-dry sequence would be even lower. The above value (rounded to 2%) is used in the L3PRA Project PRA (in the 1-L2-DET-CONTVE event tree) as a single-point estimate for the probability of C-SGTR for high-dry-low conditions. The L3PRA Project PRA model assumes that dry SGs are unable to maintain pressure, and all high-dry sequences are treated as high-dry-low.

With the development of the R02_L2 (2017) model, new MELCOR calculations were run, but unlike the R01_L2 MELCOR analyses, hot leg creep rupture was generally not suppressed. The R02_L2 MELCOR analyses always predict hot leg creep rupture for high-pressure sequences. To provide information for the C-SGTR and high-pressure melt ejection investigations, an additional calculation was run, deliberately suppressing hot leg creep rupture. This case was compared to the MELCOR results used in the C-SGTR calculator for the R01_L2 model, and this comparison is shown in **Figure 5-1** through **Figure 5-4** below.

The MELCOR results are very comparable for the two calculations for the hottest loop. The SG upflow and downflow temperatures are extremely close, the surge line temperatures and hot leg temperatures appear to be a little more challenging in the new results, and the RCS pressure (which affects all the components) is similar but more rounded. Qualitatively, this suggests that if the C-SGTR probability were to shift owing to these new MELCOR results, it would shift downward. Since (again qualitatively), the uncertainty in the estimated probability is driven as much by assumptions in the statistics of its estimation and the flaw distribution assumptions, the team concluded that it is appropriate to retain the previously-estimated value of 2%.

It is possible to develop a spectrum of probabilities versus break size, and to develop such a spectrum on a scenario-by-scenario basis. However, doing so is beyond the current scope of the L3PRA Project given that C-SGTR is not expected to be a significant contributor to the overall risk results. For these reasons, only a single double-ended tube break is considered, similar to the way that the SGTR initiating event is treated in the Level 1 PRA.

Finally, it should be noted that the deterministic analysis done for C-SGTR for the L3PRA Project involves a delayed loss of AFW (at 3 hours). Intuitively, one might expect a more challenging situation if an immediate loss of AFW is assumed (and thus a potentially higher conditional probability). However, the current understanding is that the immediate loss of AFW is more likely to lead to a less challenging (lower conditional probability) result. Calculations done for (NRC, 2017a) indicate that during the creep phase of the accident, earlier AFW failure leads to (i) higher hot leg temperatures, (ii) similar steam generator tube temperatures, and (iii) for station blackout accidents, slightly lower differential pressures owing to PORVs rather than SRVs being the relief mode during the earlier accident progression. These conditions all lead to a less challenging effect on the steam generator tubes.

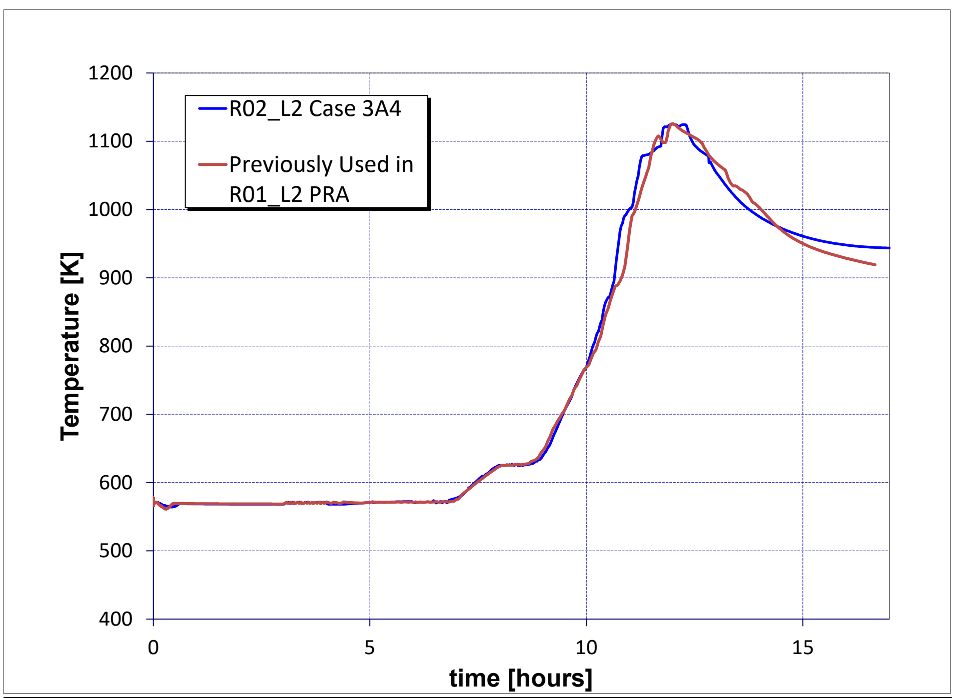

Figure 5-1: SG Up-Flow Temp Comparison for C-SGTR

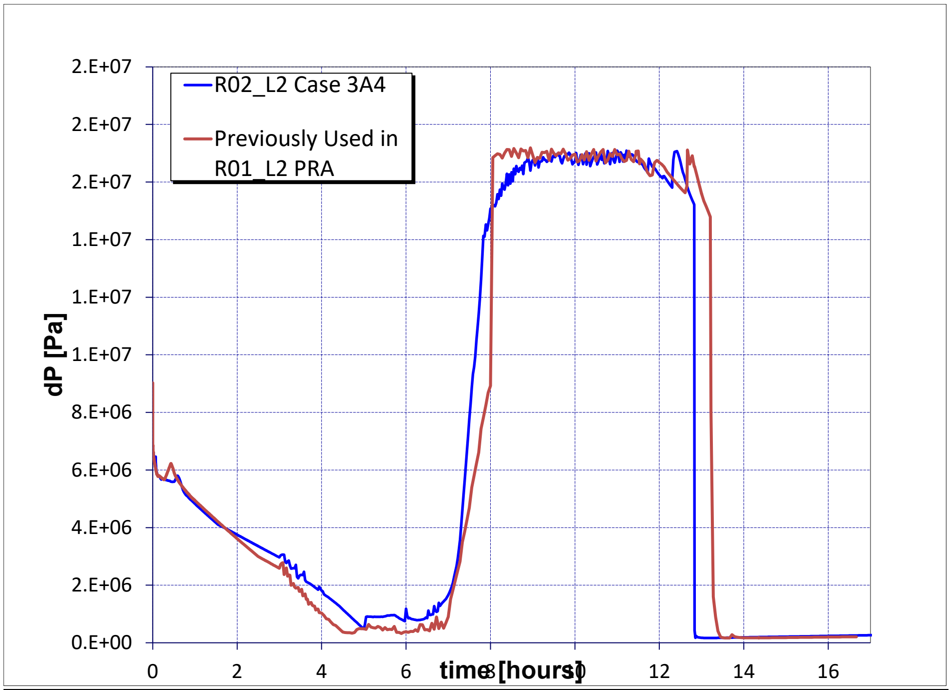

Figure 5-2: Delta-P Comparison for C-SGTR

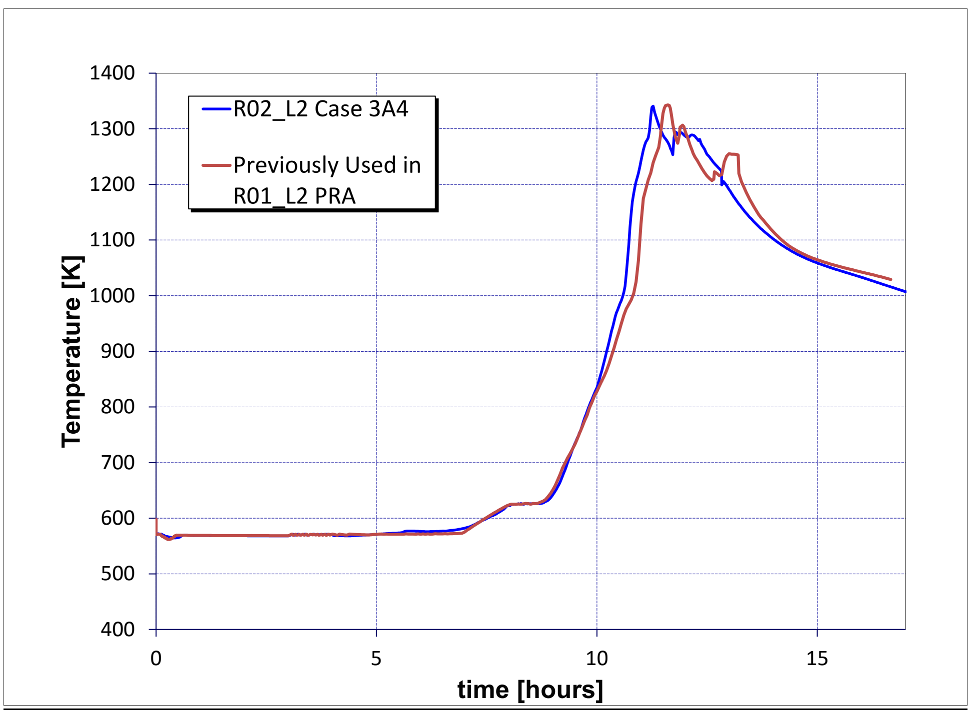

Figure 5-3: Hot Leg Temp Comparison for C-SGTR

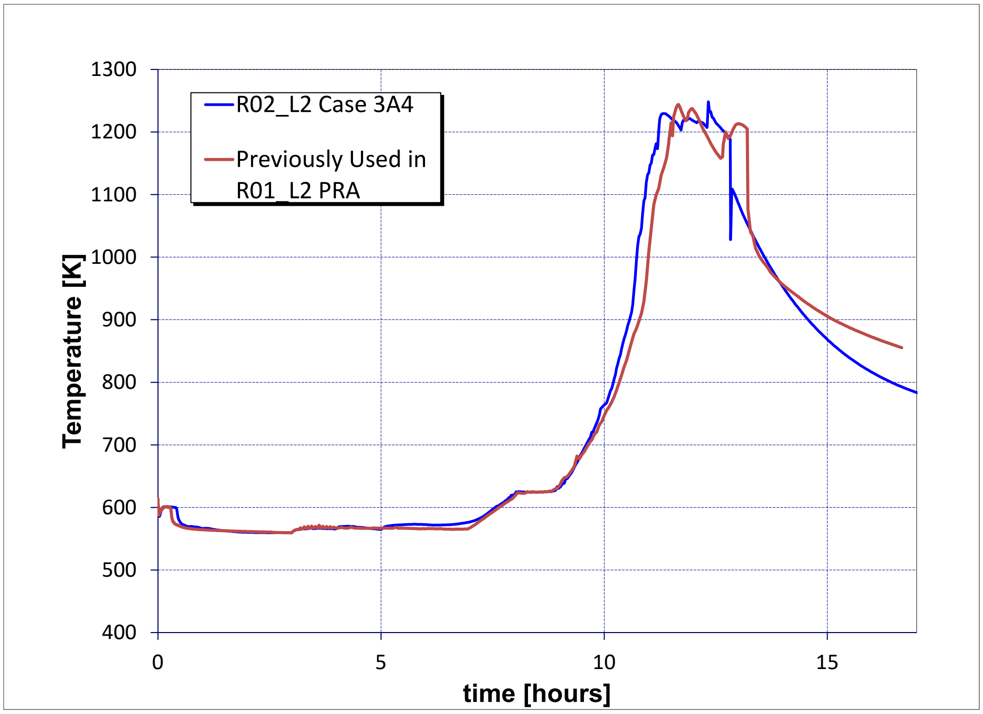

Figure 5-4: Surge Line Temp Comparison for C-SGTR

5.2. Loop Seal Clearing

The NRC generic C-SGTR work, mainly referring here to the SCDAP/RELAP5 Zion work [namely (Fletcher, 2010)], predicts that clearing of the loop seals results in a very high likelihood of consequential tube rupture, due to higher temperatures resulting from a lack of the benefits of heat transfer from counter-current hot leg flow and reduced steam generator inlet plenum mixing. However, the NRC generic C-SGTR work indicates that only very large seal leakage rates (namely the 480 gpm/RCP case from the WOG 2000 model) will clear, and then only if the leakage rate develops later in the accident (i.e., not 13 minutes after loss of seal cooling). This is based on detailed analysis of the thermal-hydraulics for a range of conditions, but as described in (NRC, 2017a), such modeling has inherently high uncertainty. Specifically, modeling choices associated with the thermal-hydraulic and system design boundary conditions can have a noticeable effect on the loop seal clearing behavior. Partly in acknowledgement of this, the PRA model described in (NRC, 2017a) conservatively (for the purposes of calculating LERF) assumes that all 480 gpm/RCP situations clear the loop seal, and that the 300 gpm/RCP situations (which only arise in the WOG2000 model in cases involving failed early primary-side depressurization) do not clear the loop seal and lead to early SGT failure.

The NRC L3PRA Project MELCOR calculations do not predict complete loop seal clearing (i.e., re-establishment of full-loop natural circulation in any loop) in any of the cases analyzed, which only investigate early (13 minute) RCP seal failure. For 182 gpm/RCP cases, these calculations show brief flow through the loop seal just prior to 2 hours (before core damage), but no coherent full loop circulation until after hot leg nozzle failure. This appears to be because there is still a significant amount of water in the loop seal through the time of hot leg nozzle creep failure, and the vessel lower plenum level does not begin to drop until after hot leg nozzle creep failure. The L3PRA Project calculations are in agreement with the analyses described in (Fletcher, 2010) and (NRC, 2017a). Only the interpretation of those results differs, with the L3PRA Project using a more realistic representation (since its goal was not to produce a conservative estimate of

LERF), wherein the 480 gpm/RCP case is not associated with loop-seal clearing. (In fact, 480 gpm/RCP cases are treated as intermediate pressure during core damage in 1-L2-DET-PRESVE, based on other project-specific work.) Nevertheless, a large difference is not seen in the C-SGTR likelihood results because these higher RCP leak rates have very low associated probabilities.

5.3. In-core Instrument Tube Failure During Core Degradation

The issue here is whether the way in which the instrument tubes degrade during core damage is such that small pathways are temporarily opened between the core region and the seal table room (in containment). If such pathways open, they would provide a temporary means of relieving primary-side pressure, transporting fission products to containment, and transporting hydrogen and steam to containment.

The NRC generic C-SGTR work concludes that it is unlikely that instrument tube failure will significantly impact the effect of C-SGTR on LERF (NRC, 2017a). This conclusion was drawn based primarily on work performed for the Zion plant in (Krall, 2009). This issue was not investigated in the SOARCA Surry study (NRC, 2013b), though the equivalent boiling water reactor (BWR) issue (failure of the in-core instrumentation associated with the TIP system) was investigated, which in the BWR case has the potential to lead to primary containment bypass without any subsequent induced failures. (NRC, 2013b) concludes that in-core instrument tube failure could be a mechanism for transporting hydrogen and fission products to the reactor building, but in-core instrumentation failure is unlikely to significantly impact the overall accident progression. The NRC L3PRA Project MELCOR (and associated probabilistic) model generally support the conclusion that in-core instrument tube failure is unlikely to significantly impact LERF. However, a more pronounced impact may be seen between the relative likelihood of other RCS failure locations (namely hot leg nozzle versus lower head failure), and observable changes in release magnitude can occur depending on whether in-core instrument tube failure is or is not treated. Note that in one case the NRC L3PRA Project PRA MELCOR model attempts to mechanistically model instrument tube failure and re-closure at each axial and radial node in the model based on local conditions, recognizing the significant uncertainty that exists in such modeling. A probabilistic model for the likelihood of instrument tube failure occurring and significantly altering accident progression was not pursued, based on the results of MELCOR calculations showing modest sensitivity to this phenomenon.

As described above, the C-SGTR and L3PRA projects view the effect of in-core instrument tube failure differently overall, but agree that it is relatively insignificant, if one is focused solely on LERF.

5.4. Steam generator leakage

The NRC generic C-SGTR work assumes that high/dry sequences result in low steam generator pressure on the basis that the steam generators are not be capable of retaining pressure once they boil dry. The basis for this assumption is discussed further in Appendix A of (LaChance, 2008a). As a result, all high/dry sequences become high/dry/low sequences.

The NRC L3PRA Project makes the same assumption in the probabilistic model, as well as in the relevant deterministic analyses. The only point of distinction is that for deterministic cases not focused on the likelihood of RCS piping failure, the SG leakage area is assumed to be 0.1 in^2 rather than 0.5 in^2 to minimize the benevolent effect that the larger area had on decay heat removal when feedwater is available.[3] The deterministic treatment is a simplifying assumption, in lieu of developing a model that (i) addresses the difference in leakage between situations where the MSIVs are open/closed, and (ii) builds in enhanced leakage based on secondary-side valve operations. In the case of both projects, the modeling is an acknowledgement that:

- The secondary-side leaks. With the turbine tripped, some combination of the steam dumps, ARVs, and SRVs leak, and this leakage likely becomes worse the more the various valves modulate. The MSIVs closing may or may not inhibit leakage downstream of the MSIVs.
- Assuming zero SG leakage will arguably over-estimate the time to SG dryout and CST depletion.
- As SG leakage increases, decay heat removal from the leakage increases, when the SGs are steaming. At 0.5 in^2 per SG (and prior to SG dryout), this approximates the situation of an operator deliberately dumping steam as part of a cooldown.
- Leaks greater than roughly 0.5 in^2 for dry SGs have little additional effect on the probability of C-SGTR, since the SGs will be fully depressurized.
- There is no basis to believe or refute that the SGs will act in a correlated fashion when exposed to the same conditions.

5.5. Material creep rupture properties

The NRC generic C-SGTR work (in this case the probabilistic calculator) (NRC, 2017a) is the most recent and comprehensive attempt to assign material properties to thermally treated alloy 600 and 690. The data is explicitly given in the corresponding material input files to the calculator and discussed in the calculator basis document. TT600 and TT690 appear to hold well against crack generation during normal operation, but once challenged by pressure and temperature transients, they are predicted to fail rapidly. The vintage of the underlying data set is such that it does not take in to account recent operating experience such as the steam generator degradation that occurred at San Onofre Nuclear Generating Station.

The NRC L3PRA Project PRA work does not address this issue deterministically but does directly utilize the NRC generic C-SGTR work in estimating failure probabilities.

References:

Fletcher, 2010	Fletcher, D., et al., *SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR*, NUREG/CR-6995, March 2010. [ML101130544]
Krall, 2009	Krall, A., et al., *Analysis of the Impact of Instrumentation Tube Failure on Natural Circulation During High-Pressure Severe Accidents,* ERI/NRC-09-206, December 2009. [ML100130402]
LaChance, 2008a	LaChance, J., et al., *Severe Accident Initiated Steam Generator Tube Ruptures Leading to Containment Bypass – Integrated Risk Assessment, Draft Final Letter Report*, January 2008. [ML080500084]

[3] It is estimated that in cases with feedwater available, the 0.5 in^2 leakage per SG could remove as much as 40 MW of decay heat; in the absence of firm evidence that this is a reasonable effect, a smaller leakage area is generally assumed.

NRC, 2013a Bixler, N., et al., *State-of-the-Art Reactor Consequence Analyses Project – Volume 1: Peach Bottom Integrated Analysis*, NUREG/CR-7110, Volume 1, Revision 1, May 2013. [ML13150A053]

NRC, 2013b US NRC, *State-of-the-Art Reactor Consequence Analyses Project – Volume 2: Surry Integrated Analysis*, NUREG/CR-7110, Volume 2, Revision 1, August 2013. [ML13240A242]

NRC, 2017a Sancaktar, S. et al., *Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes*, NUREG-2195 [ML18122A012]

6. Containment Leakage, Effective Sizes Under Normal and Accident Conditions

Various containment leakage areas are assumed in the L3PRA Project MELCOR model. These leakage areas are summarized in **Table 6-1** below, and the ensuing paragraphs explain their origin.

Table 6-1: Containment Leakage Size Summary

	Leakage area (m^2)	Approximate corresponding effective diameter (in.)
Normal leakage[a]	$1.88 \cdot 10^{-5}$	0.2
Isolation failure logic model - pre-existing tears or maintenance errors	$7.1 \cdot 10^{-4}$ and greater	1.2 and greater
Isolation failure logic model - active component failures	Greater than $2.0 \cdot 10^{-3}$	Greater than 2
Smaller vent pathway in SAMGs (post-loss of coolant accident [LOCA] purge)	$8.0 \cdot 10^{-3}$	4
Over-pressure failure due to gradual over-pressure – internal events and floods MELCOR analysis	Varies as a function of pressure (see text)	Varies as a function of pressure (see text)
Over-pressure failure due to energetic phenomena – internal events and floods MELCOR analysis	$3.3 \cdot 10^{-2}$	8
Larger vent pathway in SAMGs (containment mini-purge)	0.1	14

[a] "Normal leakage" is the sum of all unintended leakage paths caused by things such as pinholes in the containment liner, penetration gaps like weld discontinuities, and micro-orifices in organic seals/gaskets.

The design-basis leakage rate for Appendix J testing and Technical Specifications of 0.2 wt%/day is used in the MELCOR model. This leakage rate is distributed over multiple flow paths in the MELCOR model, and the effective leakage area is calibrated using a stand-alone sourced-air calculation. In the L3PRA Project MELCOR model, the leakage area is $1.88 \cdot 10^{-5}$ m^2.

The containment isolation logic model screens out pipes of diameter 2 inches or smaller. The quantification of this model is dominated by a single event associated with pre-existing tears or maintenance errors (see **Section 2**). This value, in turn, defines a large leak as one greater than $100L_a$ based on (Westinghouse, 2005). The translation of $100L_a$ to a leak size is described in the following paragraph.

L_a is the containment leakage rate at the maximum calculated containment pressure resulting from the limiting design-basis accident (per the Technical Specification Bases and Title 10 to the Code of Federal Regulations [10 CFR] Part 50 Appendix J). For the L3PRA Project, L_a = 0.2% containment atmosphere mass per day at a pressure of 37 psig. To calculate the mass of the containment atmosphere at 37 psig, the partial pressures of air and steam in the atmosphere must be known, as well as the temperature of the containment atmosphere at which L_a is evaluated. Based on Reference Plant information for large break loss-of-coolant-accidents (LBLOCAs) for the L3PRA Project, the containment atmosphere is initially at 120°F, 17.7 psia, and 20% relatively humidity. The saturation pressure of water at 120°F is 1.7 psia, so the initial partial pressure of water in containment is 0.34 psia. Thus, the initial partial pressure of air is 17.36 psia. Considering the peak containment pressure and temperature for several LBLOCA scenarios, the hot leg break scenario results in the highest peak containment pressure and temperature, which are 36.5 psig and 250°F, respectively. Assuming the atmosphere is saturated, the partial pressure of water is 29.8 psia. Using the ideal gas law, the partial pressure of air is 21.3 psia. Adding the air partial pressure to the steam partial pressure, the total containment pressure is 51.1 psia (36.4 psig). Iterating to get a peak pressure of 37 psig = 51.7 psia yields a temperature of 394.9 K (~251°F).

To calculate the leakage area that results in a leakage rate of 100 L_a, a simple MELCOR input deck was used that includes two time-independent control volumes that represent the containment atmosphere and the environment. The containment control volume has the following characteristics: temperature = 394.9 K, pressure = 3.565 bar (37 psig), humidity = 1.0, and volume = 77,900 m^3. Note that the total mass in this control volume is 192,500 kg, so 100 L_a = 0.445 kg/s. The flow area of a flow path between the two control volumes can be adjusted using control functions to achieve the desired leakage flow rate, resulting in a flow area of 7.07×10^{-4} m^2.

Regarding containment over-pressure failure, the failure area based on best estimate over-pressure failure analysis is initially small but quickly grows to an area where further pressurization does not occur. The flow area is zero below 120 psig, 0.1 ft^2 at 124 psig, 0.3 ft^2 at 127 psig, and 1 ft^2 at and above 130 psig.

Finally, the two vent pathways specified in SAMG computational aid (CA)-4 are the post-LOCA purge exhaust line (4-inch schedule 40 pipe) and the containment mini-purge exhaust line (14-inch schedule 30 std. pipe).

Westinghouse, 2004 Westinghouse Owner's Group, *Joint Applications Report for Containment Integrated Leak Rate Test Interval Extension*, WCAP-15691, Revision 5, March 2004. [ML041190628]

7. Emergency Action Level Monitoring Assumptions

Assumptions must be made with respect to the monitoring of emergency action levels (EALs), which in turn have a potential effect on the timing of the EAL declarations. Specifically, the formal emergency operating procedure (EOP) cue that prompts monitoring of the EALs is a step found throughout the EOPs that states some variant of, "Initiate emergency classification determination and initial actions." The placement of this step in the EOPs used for the L3PRA Project was evaluated. From this evaluation, it was determined that in most cases, the step to initiate emergency classification and initial actions would occur very early in the procedure path (e.g., first 10-20 minutes). However, in a few cases this step would not be invoked as part of the initial EOP pathway, but rather would be invoked upon reaching ORANGE or RED path conditions on the Critical Safety Function Status Trees (i.e., once plant conditions had begun to significantly degrade). This would include situations such as: (i) a reactor trip with AC power, but with no safety injection, leading to a relatively routine response; and (ii) a reactor trip with AC power, with safety injection and decreasing pressure, without other signs of a LOCA (containment radiation, pressure, and sump levels normal), leading to a relatively routine post-LOCA cooldown and depressurization.

For cases where the explicit callout for this step would be delayed, it is still assumed that EAL monitoring starts promptly, based on commitments in the Emergency Plan and observations of emergency drills. Procedural guidance directs that the Emergency Director (ED) assess, classify and declare an emergency condition within 15 minutes after the availability of a plant indication or receipt of a report of an off-normal condition by plant operators up to and including the declaration of an emergency.[4] This is intended to ensure compliance with Appendix E to 10 CFR Part 50, Section IV.C.2., that requires licensees to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded. Licensee staff performance of timely and accurate emergency declaration is captured in the Reactor Oversight Process emergency plan (EP) performance indicator. Further this capability is observed by inspectors during evaluated exercises. The staff has the expectation, supported by many years of exercise and drill observation, as well as performance indicator data, that licensees can and will assess and declare emergencies within approximately 15 minutes of indication.

[4] The Emergency Plan requires there be an ED whenever an emergency is declared and this individual be trained to the ED level. The ED may not delegate classification of emergencies.

8. Understanding Equipment Survivability in the Screening of Human Reliability Assessments

To orient the analyst in the types of spatial equipment survivability issues that could arise in modeling operator actions post core-damage, the physical plant layout and sample environmental loads were investigated, along with comparing the resolution of the design-basis EQ envelope relative to the environmental conditions predictable using the MELCOR model. That work is described here.

The list below identifies the environmental conditions of potential concern, and provides a shorthand designator for each:

T_s	High temperature due to quasi-static events
T_d	High temperature due to dynamic events (e.g., combustion)
P_s	High pressure due to quasi-static events
P_d	High pressure due to dynamic events (e.g., combustion)
F	Flooding (i.e., submergence of equipment)
H	High humidity
R_c	High radiation due to contamination (direct contact with material)
R_s	High radiation due to shine from radiological material

Equipment survivability effects due to combustion were considered separately. Equipment damage due to aerosol deposition was not considered.

Figure 8-1 and **Table 8-1** through **Table 8-3** present:

- a schematic of the plant depicting the various regions of interest;
- a generic look at the potential aspects of concern during a severe accident for various locations in the plant, during various phases of the accident; in each case, this is intended to denote situations where temperature, pressure, humidity, or radiation may be more challenging than the design-basis environmental qualifications;
- a comparison of the EQ spatial nodalization to the nodalization of the reactor MELCOR model, and
- survivability observations referenced to their SAMG usage.

As mentioned previously in the main report: (i) manufacturing information necessary to develop specific failure characteristics of most of this equipment is not available, (ii) in many cases the equipment resides in an area where precise environmental loads have not been established in the MELCOR simulations, and (iii) resource limitations affected the extent to which more precise characterizations could be developed.

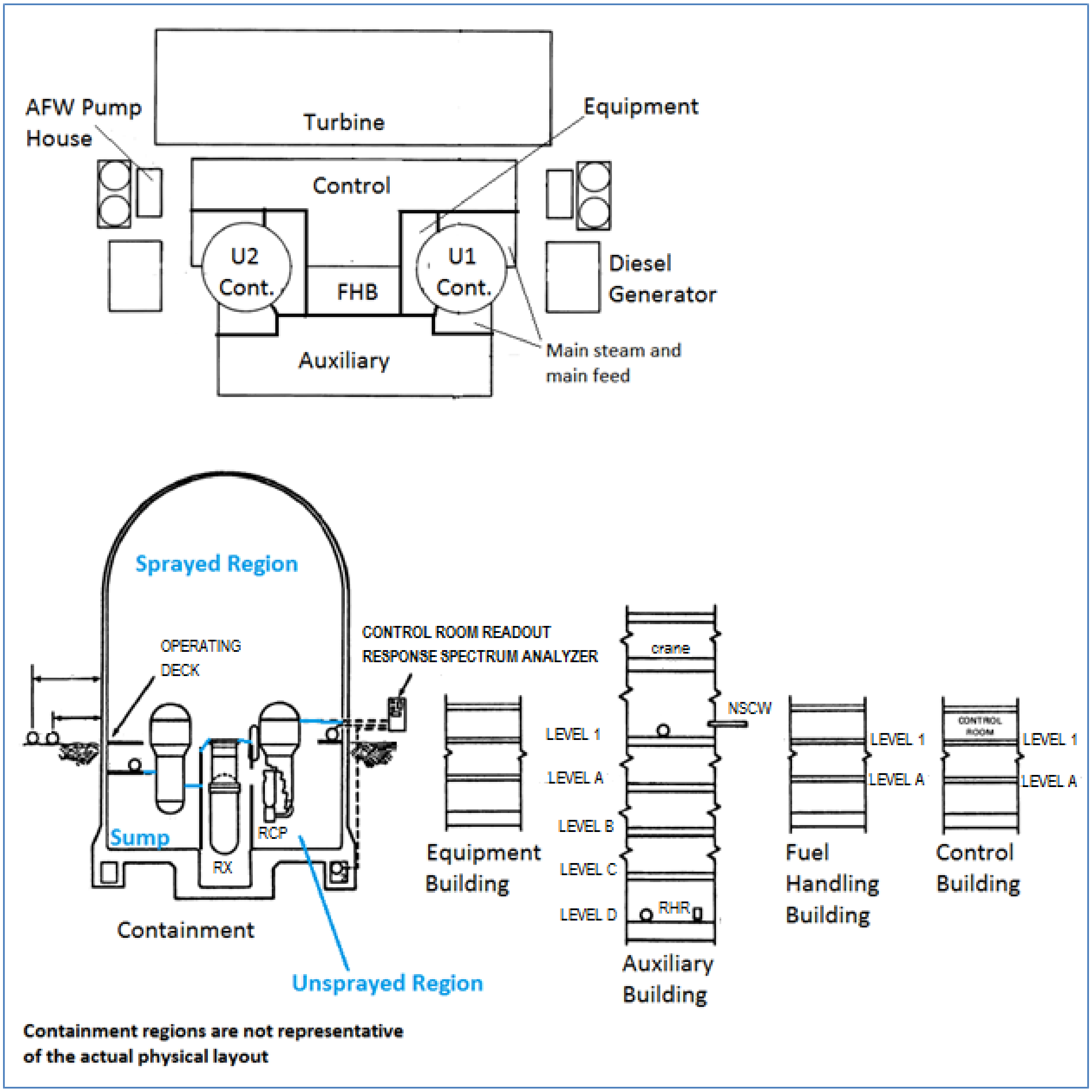

Figure 8-1: Overview of Plant Layout

Table 8-1: "Top Down" Snapshot of Potential Survivability Concerns for Reactor Severe Accidents for the L3PRA Project

Plant area	Potential conditions of concern: With core In-vessel	At vessel breach	Initial ex-vessel	Longer term (12-48 hrs after vessel rupture)	Instrumentation used in SAMG diagnostic trees and computational aids - Relevant mitigation equipment	Comments
Reactor pressure vessel	T_s, R_c, R_s	T_d, P_d [up to 150 bar drop] (if 1st RCS failure), R_c	R_c	-	Core exit thermocouples - None identified	
Reactor coolant system and connected within-containment piping	T_s, R_c, R_s	T_d, P_d [up to 150 bar drop] (if 1st RCS failure), R_c	R_c	-	RCS pressure Cold leg thermocouples PORV valve status - Reactor coolant pumps and seal cooling/injection lines Pressurizer sprays ECCS injection/recirculation lines	
Steam generator secondary side	T_s, R_s For tube rupture events: T_s, R_c, R_s	-	-	-	SG narrow range level - None identified	
Containment – Upper*	T_s [120C], R_c (if RCS leak through RCPs or cycling PORV)	T_d [300C], P_d [2 bar rise]	T_s [175C] T_d, P_d (if combustion or ex-vessel quenching occurs)	T_s [200C] P_s [150 psig] T_d, P_d (if combustion or ex-vessel quenching occurs)	Containment hydrogen sampling (train B) Cabling - core exit thermocouples Containment pressure (ext. range) Containment pressure Containment temperature Valve status for some vent paths - Containment fan coolers Containment spray nozzles Containment penetrations Cont. mini purge exhaust** Cont. post LOCA purge exhaust**	For long-term pressure response, containment failure is predicted to occur at about 120-140 psig

Table 8-1: “Top Down” Snapshot of Potential Survivability Concerns for Reactor Severe Accidents for the L3PRA Project

	Potential conditions of concern				Instrumentation used in SAMG diagnostic trees and computational aids - Relevant mitigation equipment	
Plant area	**With core In-vessel**	**At vessel breach**	**Initial ex-vessel**	**Longer term (12-48 hrs after vessel rupture)**		**Comments**
Containment – Lower	T_s [120C], R_c (if RCS leak through RCPs or cycling PORV)	T_d [700C], P_d [2 bar rise], R_c	T_s [175C] T_d, P_d (if H2 combustion or ex-vessel quenching occurs) R_c, R_s	T_s [200C] P_s [150 psig] T_d, P_d (if H2 combustion or ex-vessel quenching occurs) R_c, R_s	Containment hydrogen monitor (train A) Cabling – cold leg thermocouples (assumed) Cabling - SG narrow range level (assumed) Cabling – RCS pressure (assumed) Cabling – PORV status (assumed) Containment water level - RCP seal cooling/injection lines ECCS injection/recirculation lines	For long-term pressure response, containment failure is predicted to occur at about 120-140 psig
Containment – Cavity	T_s [150C] T_d, P_d (if hot leg nozzle fails) R_s	T_d [1100C], P_d [2 bar rise], R_c, R_s	T_s [300C] T_d, P_d (if combustion or ex-vessel quenching occurs) R_c, R_s	T_s [275C], P_s [150 psig] T_d, P_d (if H2 combustion or ex-vessel quenching occurs) R_c, R_s	None identified - None identified	
Containment – Seal Table Room	T_s, P_d, R_c (if leakage through incore instrument tubes occurs)	T_d [1100C], P_d [2 bar rise], R_c, R_s	T_s [300C] T_d, P_d (if combustion or ex-vessel quenching occurs) R_c, R_s	T_s [275C] , P_s [150 psig] T_d, P_d (if H2 combustion or ex-vessel quenching occurs) R_c, R_s	None identified - None identified	This assumes that the seal table itself does not block the cavity environmental hazards from affecting the seal table room.
Aux Bldg – Adjacent to containment (a.k.a., Equipment Bldg.)	For ISLOCA or containment isolation failure, near the failure point: T_s, F, R_c, R_s T_d, P_d, (if combustion occurs)			R_c (due to continued normal containment leakage)	ECCS pump flow rates - ECCS pumps ECCS injection / recirculation lines Containment spray pump A Connection to containment sprays (extensive damage mitigation guideline [EDMG])	

Table 8-1: "Top Down" Snapshot of Potential Survivability Concerns for Reactor Severe Accidents for the L3PRA Project

Plant area	Potential conditions of concern: With core In-vessel	At vessel breach	Initial ex-vessel	Longer term (12-48 hrs after vessel rupture)	Instrumentation used in SAMG diagnostic trees and computational aids - Relevant mitigation equipment	Comments
Aux Bldg – Elsewhere*	For ISLOCA or containment isolation failure, near the failure point: T_s, F, R_c, R_s T_d, P_d, (if combustion occurs)			R_c (due to continued normal containment leakage)	Cabling for RWST level (assumed) - Some duplicate EDMG equipment	
Fuel Handling Bldg – refuel floor	-	For containment isolation failure, near the failure point: T_s, R_c, R_s T_d, P_d, (if combustion occurs)		R_c (due to continued containment leakage)	None identified - Fuel transfer tube Some duplicate EDMG equipment	Assumes containment failure locations are basemat junction or equipment hatch
Fuel Handling Bldg – below refuel floor	For ISLOCA or containment isolation failure, near the failure point: T_s, F, R_c, R_s T_d, P_d, (if combustion occurs)			R_c (due to continued containment leakage)	None identified - Containment spray pump B Connected to containment sprays (EDMG)	
Control Building	For ISLOCA or containment isolation failure, near the failure point: T_s, F, R_c, R_s T_d, P_d, (if combustion occurs)			R_c (due to continued containment leakage)	All instrumentation (virtually all cabling goes to Control Building) - Motor control centers Switchgears Main control room Battery room Technical Support Center	Assumes containment failure locations are basemat junction or equipment hatch
Main Steam Safety Valve Room	For SG tube rupture events: T_s, F, R_c, R_s T_d, P_d, (if combustion occurs)			R_c (due to continued containment leakage)	None identified - Manual operation of SG PORVs SG PORVs / SRVs Connect. to AFW lines (EDMG)	
Turbine building	-	-	-	R_c (due to continued containment leakage)	None identified - Some duplicate EDMG equipment	

Table 8-1: "Top Down" Snapshot of Potential Survivability Concerns for Reactor Severe Accidents for the L3PRA Project

Plant area	Potential conditions of concern: With core In-vessel	At vessel breach	Initial ex-vessel	Longer term (12-48 hrs after vessel rupture)	Instrumentation used in SAMG diagnostic trees and computational aids - Relevant mitigation equipment	Comments
Maintenance building, production warehouse, north fire pump house, etc.	-	-	-	R_c (due to continued containment leakage)	None identified - Operations Support Center Demineralized water storage tank South fire water tank EDMG pump/trailer Other EDMG equipment	Offset from power block

Notes on the above table:

1. Containment temperature instrumentation, containment pressure (extended range and normal) instrumentation, containment mini purge exhaust and post-LOCA purge exhaust equipment (piping/ductwork/valves) are all assumed to be in upper (versus lower) containment, but this has not been confirmed
2. Peak values are intended to be illustrative; dynamic pressures and temperatures at the time of vessel breach (due to hydrogen combustion) are not bounding for these calculations. (They are relatively mild compared to an adiabatic isochoric complete combustion (AICC) situation.)
3. Effects on cabling between the instrument location and the cable spreading room are not accounted for except where cabling routes are readily inferable; focus here is on the instrumentation needed to assess the SAMG's Diagnostic Flow Chart, Severe Challenge Status Tree and Computational Aids, as opposed to the broader set of instrumentation that might be called upon when implementing SAMG strategies

* Shine or contamination radiation are certainly possible in these areas for specific scenarios, but here they are generally less likely than the other hazards

** These are the vent paths specified in SAMG Computational Aid #4 (CA-4)

Table 8-2: Relationship between the EQ Nodalization and the MELCOR Model Nodalization

<table>
<tr><th rowspan="2">Region</th><th colspan="2">EQ Temperature (F)</th><th colspan="2">EQ Pressure (psig)</th><th colspan="3">Dose</th><th rowspan="2">Relative Humidity</th></tr>
<tr><th>dynamic</th><th>static</th><th>dynamic</th><th>static</th><th>gamma</th><th>beta</th><th>integrated</th></tr>
<tr><td>Containment - sprayed</td><td colspan="4" rowspan="3">This was the only area where the MELCOR nodalization exceeds that of the EQ information.</td><td colspan="3" rowspan="11">The MELCOR model is not set up to provide this type of information.</td><td rowspan="11">The EQ envelope is almost always when steam would be expected. The Level 2 PRA only focuses on areas where steam occurs in a beyond-design basis accident, where it would not be expected to occur at all in a design-basis accident (DBA)</td></tr>
<tr><td>Containment - unsprayed</td></tr>
<tr><td>Containment - sump[1]</td></tr>
<tr><td>Equipment bldg - Level 1</td><td colspan="4" rowspan="5">The MELCOR model treats these on an elevation-by-elevation basis (though inter-elevation communication is not accurate due to not representing compartmentalization), but only as a single control volume that generally represents the combined equipment building and auxiliary building areas. Thus, conditions affected by compartmentalization, or that would be attributable to only one building, must be inferred by the scenario characteristics (e.g., ISLOCA in to the RHR pump room) rather than MELCOR output.</td></tr>
<tr><td>Auxiliary bldg - Level A-D</td></tr>
<tr><td>Auxiliary bldg - Level 1-3</td></tr>
<tr><td>Control bldg - Level A-C</td></tr>
<tr><td>Control bldg - Level 1-5</td></tr>
<tr><td>Fuel handling bldg - Level A-C</td><td colspan="4" rowspan="2">The reactor MELCOR model does not model this region, other than the fact that a combustion in the auxiliary building might open pathways to this air space.</td></tr>
<tr><td>Fuel handling bldg - Level 1-2</td></tr>
<tr><td>Other:
Main steam and feed areas
AFW pumphouse, et al.
Diesel Generator and related nuclear service cooling water (NSCW) and related
Steam tunnels
RWST et al.
Turbine building</td><td colspan="4">The MELCOR model does not consider these areas. They are handled based on scenario characteristics, and for the internal events/floods PRA, severe conditions are not expected.</td></tr>
<tr><td colspan="9">[1] Per SAMG Computational Aid #5 (CA-5), this is ~ 4 feet BELOW the spillover point (i.e., this is the DBA limit of 1 RWST, not the severe accident containment flooding level).</td></tr>
</table>

Table 8-3: Survivability Considerations for SAMG-Referenced Instruments

Strategy / CA	Parameter / value	SAMG/EDMG setpoint	Instrument range	Survivability observations
Entrance	Core-exit thermocouple (CETC) readout	1200F	200 to 2300F	There are 50 CETCs, and all operable CETCs factor into the control room readout, with only a small fraction of those required to be operable by Tech Specs (2 per train per quadrant; 16 total). Based on the Diagnostic Flow Chart (DFC), permanent errors occur starting around 2500F due to formation of new permanent junctions. (NRC, 2013a) and (NRC, 2013b) provide an agency perspective that the CETCs will provide adequate indication for SAMG entrance and subsidiary guideline assessment.
Severe Challenge Guideline (SCG)-1	Site doses	> 1R TEDE or 5R CDE Thyroid	n/a	Based on dose projections from the Technical Support Center (TSC) / Emergency Operations Facility (EOF), or field monitoring teams.
SCG-2	Containment pressure (extended range)	>102 psig	-5 to 160 psig	
SCG-3	Containment hydrogen	>6% and severe challenge per Computational Aid (CA)-3	0 to 10% partial pressure	The hydrogen sample lines will likely be plugged after vessel failure, according to the DFC.
SCG-4	Containment pressure (extended range)	< -5 psig	-5 to 160 psig	
Severe Accident Guideline (SAG)-1	SG narrow range level	<38% in any SG	0 to 100% of span	The DFC acknowledges various sources of bias in SG water level (e.g., high containment pressure), but asserts that these are “expected to have minimal impact on the SAMG span of interest.”
SAG-2	RCS pressure	>150psig	0 to 3,000 psi	

Table 8-3: Survivability Considerations for SAMG-Referenced Instruments

Strategy / CA	Parameter / value	SAMG/EDMG setpoint	Instrument range	Survivability observations
SAG-3	Core temperature	> 711F	200 to 2300F	The DFC lists several possible sources for this, with the CETCs being first. Assumptions about survivability are inter-related with assumptions about operator response and expected MELCOR-predicted accident conditions. (NRC, 2013a) and (NRC, 2013b) provide an agency perspective that the CETCs will provide adequate indication for SAMG entrance and subsidiary guideline assessment.
SAG-4	Containment water level	<23"	0 to 48" (narrow range)	
SAG-5	Site doses	> 1R TEDE or 5R CDE Thyroid	n/a	Based on dose projections from TSC/EOF, or field monitoring teams.
SAG-6	Containment pressure	>3.8 psig	0 to 75 psig	
SAG-7	Containment hydrogen	>4%	0 to 10% partial pressure	Based on the DFC, regular hydrogen monitoring is not accurate after vessel failure and will not measure carbon monoxide.
SAG-8	Containment water volume	< 1.3 million gallons	0 to 120in. (wide range)	Based on CA-5, top of range occurs at ~950,000 gallons. The CA instructs that water accumulation above this point must be manually accounted for based on injection estimates
CA-1	1. RCS pressure 2. # of Pressurizer PORVs opened 3. ECCS pump flow rates	n/a	1. 0 to 3,000 psi 2. binary indication 3. varies by system	
CA-2	1. Time since shutdown 2. Injection flow rate	n/a	1. n/a 2. varies by system	

Table 8-3: Survivability Considerations for SAMG-Referenced Instruments

Strategy / CA	Parameter / value	SAMG/EDMG setpoint	Instrument range	Survivability observations
CA-3	1. Containment pressure 2. Containment temperature 3. Containment hydrogen concentration	n/a	1. 0 to 75 psig 2. 0 to 300F 3. 0-10%	Containment temperature instrument range is limited, particularly if the core is ex-vessel, per the DFC. Based on the DFC, regular hydrogen monitoring is not accurate after vessel failure, and will not measure carbon monoxide.
CA-4	1. Containment pressure 2. Vent flow rate	n/a	1. 0 to 75 psig 2. varies by system	
CA-5	Containment water level	n/a	0 to 120in. (wide range)	See SAG-8
CA-6	1. Containment pressure 2. RWST water level	n/a	1. 0 to 75 psig 2. 0-100%	
CA-7	1. Containment pressure 2. Containment hydrogen concentration	n/a	1. 0 to 75 psig 2. 0-10%	See CA-3

Reference:

NRC, 2013a U.S. Nuclear Regulatory Commission, *Denial of Petition for Rulemaking PRM-50-105 Requesting Amendments Regarding In-Core Thermocouples at Different Elevations and Radial Positions Throughout the Reactor Core*, SECY-13-0063, June 2013.

NRC, 2013b U.S. Nuclear Regulatory Commission, *Staff Requirements – SECY-13-0063 - Denial of Petition for Rulemaking PRM-50-105 Requesting Amendments Regarding In-Core Thermocouples at Different Elevations and Radial Positions Throughout the Reactor Core*, SRM-SECY-13-0063, August 2013.

9. Habitability in Accident Management

Adverse environmental conditions were considered when addressing access for local actions as part of the SAMGs or EDMGs. When the HRA tasks associated with the reactor Level 2 PRA were performed, qualitative determinations were made with respect to accessibility/habitability for taking local actions, usually using scenario-level information (e.g., accident type) along with considerations about the spread of contamination/energy and the physical layout of the plant. Generally speaking, the following decomposition was used when considering adverse environmental conditions associated with radiation (shine and/or contamination):

- Areas potentially directly impacted by the event (e.g., residual heat removal heat [RHR] exchanger room during an RHR system ISLOCA) may have lethal or near-lethal doses and are assumed to NOT be habitable.[5]
- Areas nearby those areas that are directly impacted (e.g., a containment spray pump connection on the same elevation as a containment isolation failure) will have a broad range of possible dose fields and are assumed to NOT be habitable.
- Areas distant from areas directly impacted (e.g., an area on a different floor of the auxiliary building from a known containment leak path) will have a broad, but lower, range of possible dose fields, and are assumed to be habitable, though the time that an action is assumed to take may have been lengthened to account for additional Health Physics surveys or the need to take other additional worker protection measures (e.g., donning anti-contamination suits).
- Areas not proximate to directly-affected areas are assumed to be habitable.
- The Control Room and Technical Support Center are assumed to remain habitable during the phases of the accident where SAMG and EDMG-directed actions are modeled. This assumption was re-visited in those cases where the actions were significant contributors to risk reduction for bypass/unisolated containment sequences or sequences where the ventilation systems would be failed and, thus, for which the control room envelope habitability design would be decidedly different than the accident conditions.

Note that no SAMG or EDMG actions prompt the accident management team to enter containment (due in part to the very harsh environment that would be present), so no assumptions are necessary in that regard. The assumptions above consider the limited information available from the deterministic modeling with respect to radiation field conditions during a reactor accident (e.g., MELCOR did not directly compute radiation fields relevant for human dosimetry), as well as the limited spatial resolution of the structures surrounding containment.

The above treatment is intended to be conservative by only indirectly crediting fission product scrubbing for habitability determinations and only crediting fission product scrubbing in cases where such credit is inherent in physical building shielding and well-understood within-containment deposition processes.

Loss of ac and/or dc power can present additional non-radiological challenges during accident management. The two primary considerations are:

- Loss of emergency lighting, which is estimated to occur in about 90 minutes after loss of all AC power based on Reference Plant EOPs, if power for emergency lighting cannot be switched to the other unit.

[5] In this case, the room would also be judged inaccessible due to severe internal flooding.

- Loss of room cooling, although to be a major concern, it would need to be a failure where the partial loss of ac and/or dc power fails the heating, ventilation and air conditioning (HVAC) system and results in a core-melt accident but does not cause failure of the equipment (pumps, motor control centers) that contribute to the heat load.

9.1. Human Reliability Analysis Considerations

In the Level 2 HRA, adverse environmental conditions are also qualitatively handled, generally by ensuring that sufficient time is available to overcome these challenges in cases where ac or partial dc loss is present or limiting credit for actions when all ac and dc power is unavailable. A particular case of interest is heatup of the turbine-drive auxiliary feedwater (TDAFW) pump room during station blackout, since this represents a situation where extended operation of equipment occurs during a total loss of HVAC. Reference Plant specific calculations indicate that the room temperature of the TDAFW pump room 24 hours after the loss of room cooling would be significantly less than the 160°F calculated for the motor-drive auxiliary feedwater (MDAFW) pump rooms.

The rationale for not giving more or less credit for the ability to access these areas is made with the following set of related information in mind:

- Issues that have not received explicit consideration such as:
  - the effects of aftershocks on human performance (for seismic initiators),
  - heat-related effects (degradation of cognitive function, heat stroke, etc.),
  - humidity (e.g., steam burns, visual field),
  - hazards created by the initiating event or secondary effects of combustion events (e.g., displaced-by-flooding manhole covers in the areas between buildings as a fall hazard during the nighttime portions of the Fukushima recovery)
- Plant-specific emergency procedures, namely those that included statements about:
  - General access control, exposure limits, exposure control, etc.
  - It being permissible to perform radiation surveys in an area in parallel with commencement of work, if the work needs to be performed immediately
  - It not being permissible for any personnel to enter an area if dose rates were beyond the range of the instruments being used
  - The Emergency Director having sole authority to allow radiation exposures in excess of 10 CFR 20 limits
  - Exposures greater than 25 rem, for lifesaving missions, can only be performed by volunteers who have been made fully aware of the risks
  - If the task cannot be completed within the allowable time, or it was estimated that the permissible dose will be exceeded, personnel shall leave the radiation area
  - Use of potassium iodide by emergency workers as a thyroid blocking agent
- Plant-specific habitability system design, including the following considerations:
  - The control room post-accident habitability system includes 2 redundant, physically-separated trains with high-efficiency particulate air filters and charcoal absorbers, with each unit being associated with a different safety train. The air distribution aspects of the system are common between the two otherwise redundant trains.

- The systems are designed to address the most limiting design-basis accident.[6] Control room shielding and air handling requirements are dictated by a design-basis LOCA, to limit whole-body doses to below 5 rem from contributing modes of exposure for the duration of the accident.
- A control room isolation (CRI) signal causes the activation of the emergency air filtration systems for the control room (CR) and technical support center (TSC), if either a safety injection (SI) signal or a control room intake high radiation signal occurs.
- Insights from **Figure 9-1**, below that include:
  - the two control room air intakes are atop the equipment/control building (“1” in **Figure 9-1**),
  - the two plant vents for the containment / fuel handling / auxiliary building are atop the containment (“2” in **Figure 9-1**)
  - the radiological release points for the steam jet air ejectors and steam packing exhauster are atop the turbine building (“3” in **Figure 9-1**)

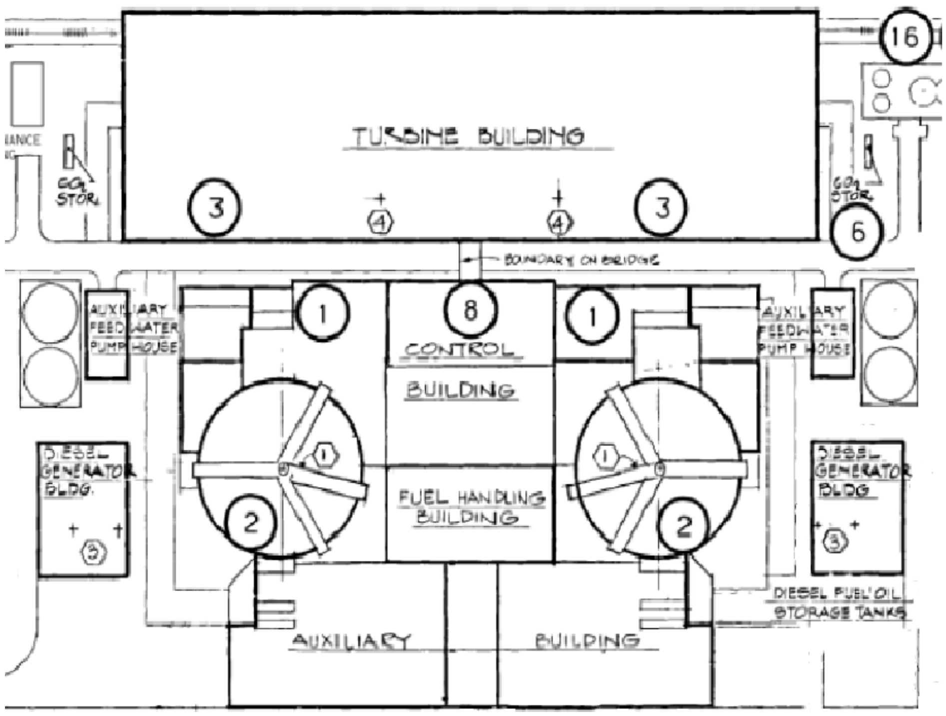

Figure 9-1: Ventilation and Release Pathways

- Reference Plant specific room heatup calculations supporting the licensee’s Level 1 PRA.
- Dose information from the March 2011 Fukushima Daiichi accident:
  - Between 9 to 12 days after the start of the accident, dose rates in the vicinity of the reactor building exteriors ranged from < 1mSv/hr to 130 mSv/hr (< 0.1 rem/hr to 13 rem/hr) (TEPCO, 2014a).

6 Note that General Design Criteria 19 covers control room habitability under “accident conditions” (i.e., without specifying the bounds of the accident domain); however, the design review is performed against DBA radiological sources (or the Alternative Source Terms for plants which have adopted it), as described in NUREG-0800, Section 6.4, Sub-section III, Item 5 (NRC, 2007a).

- During the two years following the accident, dose rates within the Unit 1 Reactor Building ranged from < 1 mSv/hr to 4,700 mSv/hr (< 0.1 rem/hr to 470 rem/hr). Note that this full range of values was seen on the same level of the building (TEPCO, 2014a).
- It has been estimated that approximately 4% of workers received total effective doses in excess of 5 rem (50 mSv), with less than 1% receiving doses greater than 10 rem (100 mSv), and less than 0.05% receiving doses greater than 20 rem (200 mSv) (WHO, 2013).

- Input from fire protection personnel about the dry environment conditions that typical nuclear power plant fire-fighting protective personnel equipment is designed for, and the limitations of this equipment in a hot, wet environment
- (NEI, 2002) that includes various Control Room Habitability Program issues, including enhanced unfiltered in-leakage observed during testing at some plants
- (EPRI, 2012), including statements such as:
    - "In addition to hydrogen release, extensive core damage will lead to the release of appreciable amounts of fission products from the fuel. In the event of enhanced leakage out of the containment into a reactor/auxiliary building, the presence of fission products in the discharge will significantly impair the capability of operational staff to gain access to either a large part of specific regions of a reactor/auxiliary building."
    - "If venting or purging is initiated after RCS damage condition BD, fission products deposited within the vent line or standby gas treatment system might reduce or prevent plant accessibility due to high radiation levels."
    - "With the plant design basis, the building ventilation and filtration systems would be sufficient, even under severe accident conditions, to reduce the levels sufficient for maintenance. If the ventilation system is not operational, as was the case for the Fukushima plants, then the radiation levels may be sufficient to prevent access to the building until the ventilation flow is restarted and the building atmosphere is recycled and filtered."
- Plant-specific considerations related to physical layout, habitability, and accident progression, as described in (Helton, 2014a)

References:

EPRI, 2012	Electric Power Research Institute, *Severe Accident Management Guidance Technical Basis Report*, EPRI TR-1025295, October 2012.
Helton, 2014a	Helton, D., et al., *Focus Areas for a Level 2 PSA That Supports a Site NPP Risk Analysis*, ESREL 2014 Conference, Wroclaw, Poland, September 14-18, 2014. [ML14230A077]
NEI, 2002	Nuclear Energy Institute, *Control Room Habitability Guidance*, NEI 99-03 Revision 1, Draft, dated January 2003 and transmitted to NRC in November 2002. [ML023330041]
NRC, 2007a	US NRC, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*, NUREG-0800, Revision 3, March 2007.
TEPCO, 2014a	Information derived from dose maps downloaded from the archival section of the public TEPCO website on April 23, 2014 (http://www.tepco.co.jp/en/nu/fukushima-np/f1/surveymap/index-e.html).

10. Nuclear Service Cooling Water (NSCW) Timing Assumptions

Assumptions are made in the L3PRA Project Level 2 PRA regarding downstream actions/failures caused by partial or total loss of NSCW. The L3PRA Project Level 1 PRA deals with most aspects of loss of NSCW; however, there are a handful of degraded condition situations where the Level 1 PRA does not need to provide detailed timing information to estimate core damage frequency. The assumptions unique to the Level 2 PRA in this regard are provided below, followed by the limited information from which these assumptions were developed.

10.1. Summary of NSCW Assumptions

- For failure of all six NSCW pumps:
  - Reactor trip and RCP trip are assumed to occur at 3 minutes via manual operator action based on abnormal operating procedures (AOPs).
  - ECCS, containment sprays, and containment fan coolers become unavailable prior to their actuation signals.
  - RCP seal failure, if it occurs, starts at 43 minutes following total loss of NSCW.
- For failure of 2 NSCW pumps in both trains,[7] with establishment of single-pump operation[8] in at least one train:
  - Manual reactor trip, operators tripping the RCPs, and isolating letdown are assumed to occur at 3 minutes[9] (manual operator action based on AOPs).
  - One train each of ECCS, containment sprays, and containment fan coolers are available, if they are first demanded after single-pump operations have been established (which is assumed to be 30 minutes after the loss of NSCW) and are assumed to be available for 3 hours thereafter prior to being secured due to degrading conditions. If they are demanded prior to 30 minutes, they are assumed to actuate and almost immediately (e.g., within 1 minute) fail non-recoverably.
  - RCP seal failure starts at 3 hours plus 13 minutes, if single pump operation is unsuccessful or ECCS is demanded after single-pump operation is set up.
- For failure of the cooling tower to start sprays or fans upon high temperature:
  - Reactor trip, operators tripping the RCPs, and isolating letdown depend on other characteristics of the accident or at the time associated with functional loss of NSCW described in the next bullet
  - It is assumed that all systems are available for surrogate timeframes associated with the application of a 'lifetime correlation' on a scenario-specific basis for the times-at-plant-condition articulated in **Table 10-1** below.
  - RCP seal failure, if it occurs, starts 13 minutes after the functional loss of NSCW

7 The L3PRA Project Level 1 PRA model includes combinations for 4, 5, and 6 pump CCFs for initiating event (i.e., IE LONSCW) and subsequent failures for other transients (e.g., TRANS).

8 Single pump operation is only credited in IE LONSCW scenarios for the 4- and 5-pump CCF.

9 Note that this quicker time does not apply for the loss of NSCW that occurs after different initiating events.

10.2. Background for NCSW Assumptions

- Typically, training material for component cooling water (CCW), NCSW, and ECCS does not provide timing information relative to how long supported components can operate without NSCW.
- Without NSCW almost all supported equipment will fail almost immediately.
- Assumed success criteria for NSCW operation:
    - If SI actuates – two-of-three pumps and three-of-four fans on one train
    - Otherwise – two-of-three pumps and one-of-four fans on one train
- Per manufacturer testing, RHR pumps remain operable under worst conditions without CCW cooling.
- With no NSCW flow, ECCS pumps will be unavailable.
- The description of the loss of NSCW event tree in the Reference Plant Level 1 PRA shows that:
    - The event tree does not consider temporary running or recovery of ECCS pumps.
    - The scenario was constructed under the assumption that an SI signal would not be present at the time of reactor trip, such that ECCS is not demanded until after single-pump operation could be established (if it was not a complete loss of NSCW event).
    - AFW is found to be unaffected by loss of NSCW; no direct cooling is provided to TDAFW or MDAFW pumps or rooms, and dc power would be available for at least 24 hours based on room heatup calculations.
    - The primary concern is loss of RCP seal integrity (due to loss of NSCW support for auxiliary component cooling water [ACCW], which in turn provides heat removal for seal injection (charging pumps) and the thermal barrier heat exchanger). The seal failure would be delayed due to ACCW heat capacity.
    - The time assumed to be available to implement single NSCW pump operation (1 hour) is conservative because the time between the reactor trip and total loss of RCP seal cooling occurrence could be several hours or more.
- The discussion of HRA assumptions in the Reference Plant Level 1 PRA shows that:
    - Operators have 35 minutes to establish one pump NSCW operation to avoid RCP seal failure based on 30 minutes (ACC with NSCW heatup time) plus 13 minutes (from WOG 2000 model) minus 8 minutes associated with "recovery time." The estimated time for operators to complete this action is 25 minutes.
    - It would take 9-12 hours for the cooling tower basin temperature to reach 95°F, if the cooling tower does not swap to spray mode at 75°F (assuming no loss of offsite power (LOOP), no SI, and no additional loads following reactor trip)
        - The starting point for the calculation (time zero) is reactor trip and basin temperature equal to 75°F.
- The EOPs do not appear to explicitly deal with a total loss of NSCW.
- Very simple heat balance calculations were performed to estimate the time it would take to heat up the cooling tower basin volumes to a point where CCW and ACCW function would become degraded (see **Table 10-1**). These timings should be viewed as surrogates. They

help to define specific conditions for the deterministic analyses and demonstrate relative effects between various situations of interest; however, they are uncertain given the assumptions made to produce them. Assumptions used in these calculations include:

- Normal operation – 43 MW (2 x 73.5x10^6 Btu/hr)
- Post-accident with SI or LOOP, without cooldown – 193 MW (2 x 330x10^6 Btu/hr)
- Post-accident without SI and LOOP, without cooldown – 19.7 MW (2 x 33.7x10^6 Btu/hr)[10]
- Post-accident with SI and LOOP, with plant cooldown – assumed to be 3.9 times the normal power heat load for two-train cooldown and 4.7 times more for one-train cooldown[11]
- Post-accident without SI or LOOP, with plant cooldown – assumed to be 1.7 times the normal power heat load for two-train cooldown and 2.6 times more for one-train cooldown (same basis as above)
- Cooling tower basin volume: 3.6 million gallons per train
- Starting temperature = 24C (75°F); ending temperature = 35C (95°F)[12]
- Evaporative heat losses were neglected, on the basis that they were predicted to be very small.

Table 10-1: NSCW Basin Heatup Times

		1-train operation	2-train operation
Normal operation		4 hours	8 hours
Post-accident without plant cooldown	LOOP, and SI actuates	50 minutes	100 minutes
	No LOOP / No SI	9 hours	18 hours
Post-accident with plant cooldown[13]	LOOP, and SI actuates	40 minutes	2 hours
	No LOOP / No SI	90 minutes	5 hours

Example calculation:

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10 This sums the non-ECCS / non-EDG loads using the SFP heat load as the only CCW heat exchanger load. Since the CCUs will reduce containment temperature significantly during this time, the containment cooling unit heat load was simplistically divided by 2. The resulting estimate still includes several simplifications, most notably that the values used are peak (time-independent) loads, and so a ratio is then applied of 330/441; this heat load is used for both the one and two train cases

11 Based on simple ratio of the values from Reference Plant information.

12 75°F is the point at which sprays initially actuate, while 95°F is the NSCW design temperature (for providing the necessary cooling to CCW and ACCW). The starting temperature could be higher (the system is routinely in spray mode), while the ending temperature could also be higher (even the design-basis ultimate heat sink analysis predicts temperatures above 95°F. The 20-degree dT is reasonable, and the slight shift in water heat capacity and density if higher temperatures were used for both values should be minimal relative to the many other simplifications/uncertainties in the calculations.

13 These are results associated with bringing the plant to cold shutdown, based on peak heat loads. They use time-independent heat loads to represent lengthy processes that involve the transition of decay heat removal from AFW to RHR.

$$dt \sim \frac{\rho \times V \times c_p \times dT}{q'} = \frac{\frac{996kg}{m^3} \times 13{,}600m^3 \times \frac{4.179kj}{kgK} \times 11K}{43{,}000\ kw} = 14{,}480s = 4\ hours$$

11. Reactor Cavity Rebar and Under-fill

The mass and composition of rebar in the reactor cavity can affect ex-vessel fission product releases. Further, the reactor cavity structure and backfill can affect basemat melt-through with the potential for the basemat melt-through to lead to failure of the vessel support structure.

11.1. Cavity Rebar

MELCOR calculations were run to determine the effects of the mass and composition of rebar in the reactor cavity on ex-vessel fission product releases in support of the L3PRA Project. The calculations simulate a high-pressure SBO with no turbine-driven auxiliary feedwater and with nominal containment leakage. The following three cases were analyzed: a case with no rebar in the concrete, a case with 100% iron rebar, and a case with rebar composed of SA-508 grade 5 steel. All three cases assume the concrete is the default CORCON BASALT concrete in MELCOR.

Using estimates of the cavity floor reinforcement, the default densities of carbon steel (7752.9 kg/m^3) and CORCON BASALT concrete (2340 kg/m^3) in MELCOR, the mass fractions of steel and concrete are approximately 0.05 and 0.95, respectively. This does not include the filler slab on the cavity floor or the steel liner; however, the filler slab and the liner account for a very small fraction of the volume of the cavity, so not including the filler slab and the liner is reasonable.

MELCOR requires the mass fractions of each compound in the concrete (including the rebar). Without knowing the exact composition of steel, one can assume that the rebar composition is 100% iron or pick a steel composition out of the ASTM standards. For the third case, steel with a composition corresponding to SA-508 grade 5 has been chosen because the carbon steel properties in MELCOR are based on SA-508 and because the maximum mass fractions of other elements (particularly chromium and nickel) for grade 5 is high compared to other grades. For elements that are not allowable, concrete components in the MELCOR cavity package (e.g. vanadium, molybdenum, copper), it is assumed that those elements are iron instead. Note that the actual rebar conforms to ASTM standard A615; however, the only requirement for the chemical composition of this steel is that phosphorous must be less than 0.06%. Thus, there is uncertainty in the actual rebar composition.

Calculation results show that there are relatively minor differences between the 100% iron and SA-508 grade 5 rebar cases. The largest difference in the environmental release fraction is 10% for molybdenum class. Other classes differ by about 5% or less, which is relatively small considering the large uncertainties in the fission product release fractions. There are also small differences in the timings of fission product releases from the cavity, but overall ex-vessel release fractions are almost identical for the two cases.

However, there are some significant differences between the no rebar case and the rebar cases (see **Figure 11-1** through **Figure 11-4** below). (The following discussion refers to the case with 100% iron rebar. Note that the results are similar between the case with 100% iron rebar and the SA-508 Grade 5 rebar.) While the ex-vessel release fraction at 72 hours is basically the same regardless of whether there is rebar in the cavity, the timing of ex-vessel releases changes because of the mass of metal in the cavity. Adding rebar delays the depletion of metal in the molten pool, which in turn delays release of molybdenum. For the case without rebar, metal is depleted around 27 hours, and molybdenum is fully released shortly thereafter as the molybdenum oxidizes. (Note that MELCOR assumes that iron oxidizes before molybdenum.) When rebar is present, the mass of metal drops to nearly zero at about 35 hours. However, because metal is continuously released from the cavity when rebar is present, the release of molybdenum following the layer transition is more gradual for the cases with rebar. At the same time, the aerosol mass in the containment atmosphere decreases with time, so that the aerosol concentration is 5 g/m^3 immediately before increased molybdenum releases for the no rebar case and 4 g/m^3 immediately before molybdenum releases for the rebar cases. The spike in molybdenum releases following metal depletion in the no rebar case results in an aerosol concentration of 11 g/m^3. In comparison, the peak aerosol concentration following metal depletion is less than 7 g/m^3 for the rebar cases.

Adding rebar to the cavity has an impact on the long-term containment pressure. Pressure is greater in the case with no rebar for several reasons. First, the increased mass of radionuclides in the sump (due to increased agglomeration and deposition of aerosols) in the case with no rebar leads to greater steam generation. Second, the fractional mass of water trapped in the concrete is reduced by approximately 5% when cavity rebar is present. Together, this leads to greater containment pressurization for the no rebar case.

Because the presence of rebar has a significant impact on containment aerosol concentration as a function of time, the presence of metal in the cavity also impacts agglomeration and deposition rates and affects the mass available for release to the environment. In the case of molybdenum, environmental releases for the rebar cases are almost four times greater than the releases when there is no rebar. Cesium and iodine releases increase by approximately 35%. The response might be different for cases with enhanced leakage or containment isolation failure because there would be less time for material to agglomerate before it is released, so there may be smaller differences in the releases. Either way, adding rebar to the cavity actually increases the environmental release fraction by reducing agglomeration and deposition in containment.

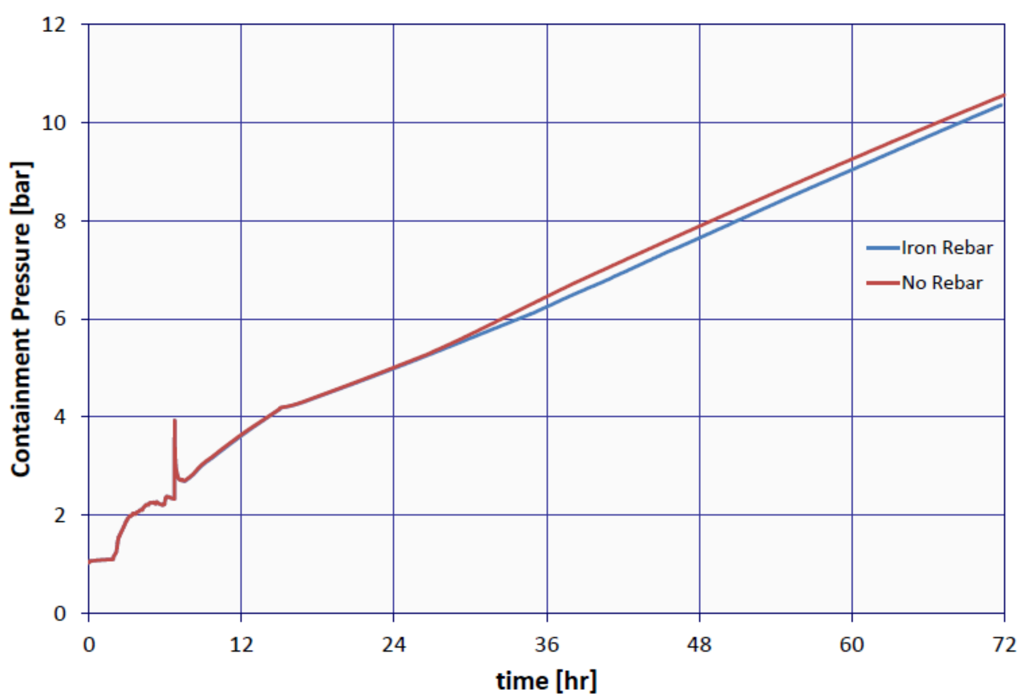


Figure 11-1: Effects of Rebar on Containment Pressure

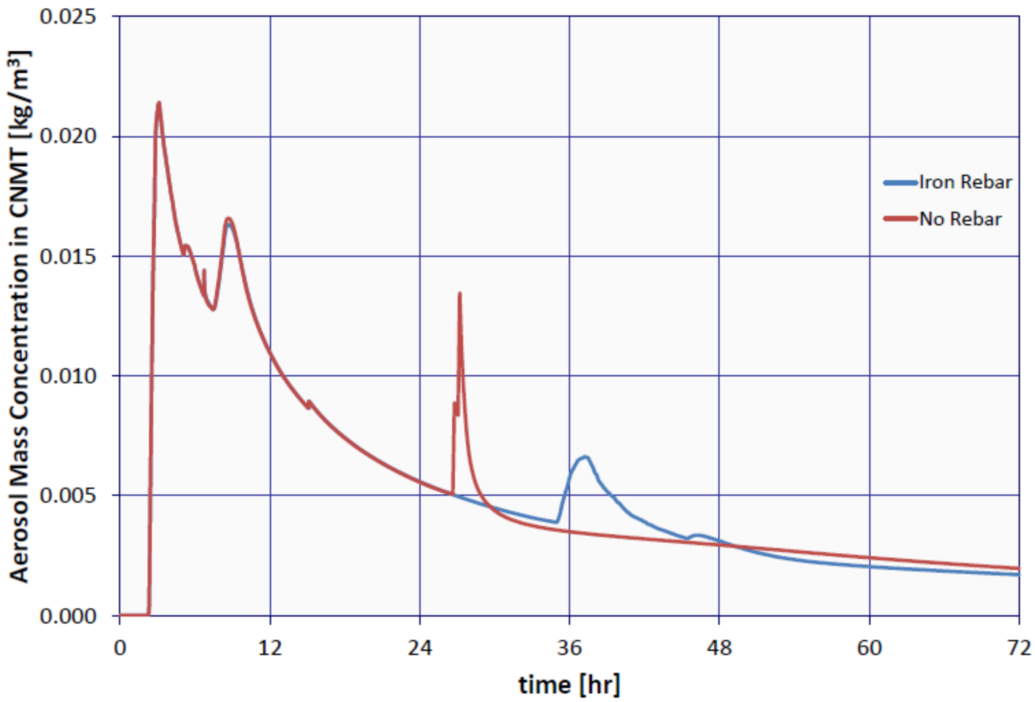


Figure 11-2: Effects of Rebar on Containment Aerosol Mass Concentration

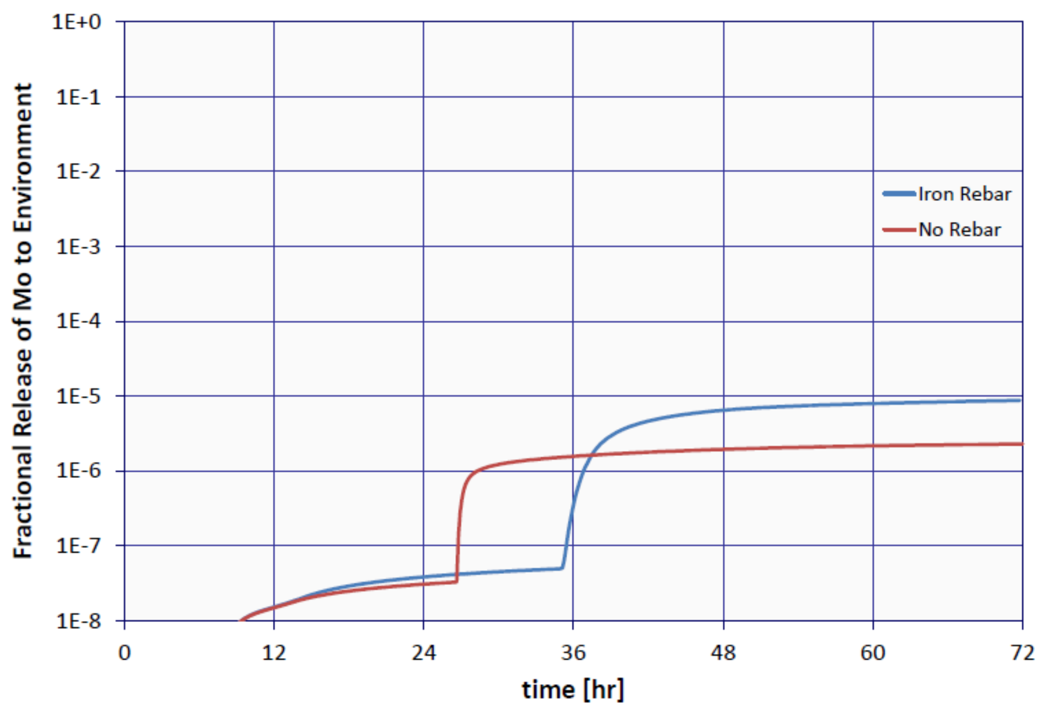


Figure 11-3: Effects of Rebar on Environmental Mo Release

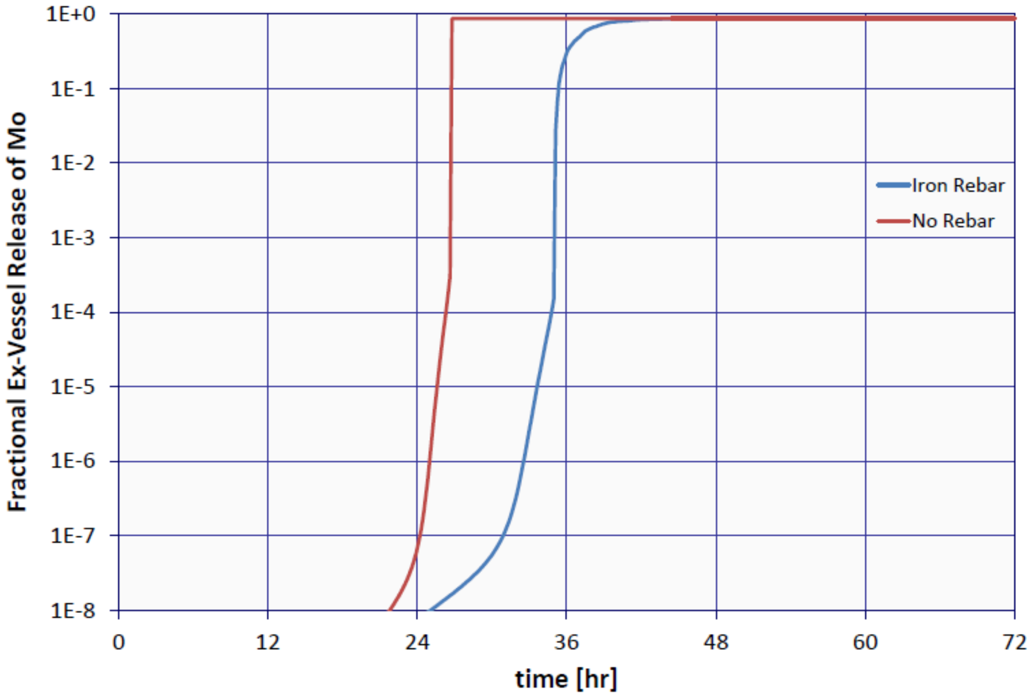


Figure 11-4: Effects of Rebar on Ex-Vessel Mo Release

11.2. Cavity Backfill

Considering basemat thickness, filler slab thickness, and concrete mud layer thickness, basemat melt-through would occur after roughly 2.9 m of axial erosion or 2.4 m of radial erosion (from MCCI). There are significant uncertainties associated with calculating cavity erosion. Even so, the time to erode this depth of material is on the order of days, and thus, these uncertainties should be contrasted with the significant uncertainty in modeling onsite accident management (including potential support from offsite resources) during that time. Finally, there are significant uncertainties with respect to when pressure-retention and fission product retention capability would be lost during the erosion process. On one hand, the embedded liner is penetrated well before 2.9 meters of axial erosion has taken place. On the other hand, the earth-backing and significant sub-surface depth of the cavity may greatly slow the depressurization process.

11.3. Vessel Support

Another issue considered was the potential weakening and failure of reactor pressure vessel supports due to substantial radial erosion in the cavity. If RPV supports were to fail, the resulting vessel displacement could possibly fail containment by tearing out mechanical penetrations for piping connected to the reactor coolant system.

The vessel is supported by four seats in the primary shield wall under two hot leg and two cold leg nozzles. The shield wall is anchored to the containment basemat (i.e., the floor of lower containment, as opposed to the floor of the excavated cavity). This suggests that the cavity provides little, if any, structural support to the RPV. Given that the cavity floor is significantly lower than the bottom of the containment basemat, radial ablation in the cavity would not directly affect the containment basemat. From a structural analysis perspective, there would need to be significant damage not only to the cavity wall but also the soil beneath the containment basemat to fail the vessel supports. Significant damage to the engineered backfill beneath the basemat would not occur for many hours following basemat melt-through. Because modeling the interaction between molten corium and the soil backing the reactor cavity is well beyond the state-of-practice (and state-of-the-art), failure of vessel supports due to significant radial erosion is not modeled in the MELCOR model or the L3PRA Project Level 2 PRA.

12. Reactor Cavity and Containment Response at the Time of Vessel Rupture

The reactor cavity communicates with the rest of containment following a vessel breach. There are several facets to this issue, as described below.

12.1. Gas flow-paths

The reactor cavity has an active ventilation system that is assumed to be isolated or to have failed due to harsh conditions (i.e., the fans are off, and the backdraft dampers are closed) by the time of vessel breach..[14] This provides several passive communication pathways, which are described below and depicted in **Figure 12-1**.

Loop cutouts - The cavity is adjacent to the annular region around the reactor vessel, which in turn has loop cutouts where the 4 hot legs and 4 cold legs pass through the cavity wall. These cutouts (which are partially insulated) connect this region to the remainder of lower containment. The MELCOR model does not consider any changes in the cutout flow area caused by dislodgement of the insulation during vessel blowdown in to the cavity. The MELCOR model assumes that all air being forced into the cavity by the ventilation system during normal operation passes through the loop cutouts.

Vessel flange - The vessel flange contains 8 ventilation ports, summing to a total flow area of approximately 20 ft^2. These ports have covers that are normally on, with vertical spacers to allow airflow but inhibit water intrusion. During normal operation, these ports can potentially allow for the intrusion of some water downward in to the cavity. These ports are treated in the MELCOR model in a manner that captures air flow but discounts downward water intrusion.

Instrument tunnel - The cavity is also connected to the reactor vessel instrumentation tunnel that leads to the seal table room. While the seal table room connects (via a normally closed door) to a stairwell that leads up to the operating floor (upper containment) and down to other elevations of lower containment, there is also an open pathway above a platform near the room's mid-height. This pathway leads to a portion of the lower containment directly below the operating deck. The seal table itself does not form a seal at the floor of the seal table room. Little communication is expected between the instrumentation tunnel and the seal table room if the seal table is in place. The above characterization is represented in the MELCOR model. The likelihood of vessel breach blowdown forces displacing the seal table was not further investigated (but this is assumed to occur in the vessel rocketing analysis done for the probabilistic model).

Cavity ventilation system - The ventilation system (which is designed to pull air from upper containment and force it in to the cavity, relying on passive pathways to equilibrate pressure) includes check valves and ductwork that are not designed to withstand high pressure differentials.

Failed RPV - The cavity will communicate with the reactor pressure vessel and reactor coolant system through the failed lower head. If there is already a LOCA, SGTR, or stuck-open SRV, then the cavity will also communicate with the lower containment, secondary side of the steam generator, and lower containment, respectively.

[14] There is some potential that the backdraft dampers would have failed in the reverse direction due to the pressurization of the cavity at the time of vessel breach. This is not explicitly considered here but would lead to enhanced communication between the cavity and the remainder of containment.

The reactor MELCOR model captures these connections to a reasonable level of accuracy. More generally, the MELCOR model assumes relatively free communication between the lower and upper containment.

12.2. Water intrusion

Water can enter the cavity prior to, or around the time of, vessel breach. The MELCOR model assumes that no water in the containment reaches the cavity prior to the deliberate injection of more than approximately 1.3M gallons of water. Nevertheless, there is the possibility that water enters the cavity in one of three ways.

First, a LOCA at the hot or cold leg nozzles, as well as any water in the lower head at the time of vessel rupture, would directly enter the cavity. This pathway is handled within the MELCOR model.

Second, regarding the potential for water to pass through the vessel flange ventilation ports and enter the cavity, the vessel flange design is engineered to funnel containment spray falling on the RPV head/flange to drains that lead to the lower containment sump. As mentioned previously, this water intrusion pathway is not considered in the MELCOR model.

Third, there are other penetrations of the cavity wall that can result in potential leakage in to the cavity if lower containment is flooded (e.g., if the RWST has been injected and then leaked out of an RCS break in to lower containment / the ECCS sump. **Table 12-1** presents these possibilities. These are not considered in the L3PRA Project MELCOR model.

Table 12-1: Cavity In-leakage Paths

Water leakage path in to the reactor cavity	Estimated flow rate (ft^3/min)	Notes
Vessel flange ventilation ports (described above)	2.4	Only relevant when containment sprays are running
Reactor vessel support cooling ducts	Sealed	This assumption has a large effect on the timing to flood the cavity
Sump discharge piping**	7.2	Roughly equal to the highest possible level for LOCAs with no additional SAMG-based containment flooding
Upper ex-core neutron detector positioning rods*	3.3	A LOCA resulting in containment spray actuation would lead to a water level above this point
Lower ex-core neutron detector positioning rods*	6.1	Any LOCA leading to filling of the ECCS sump would result in a water level above this point.

* The flow-rate is assumed to be zero.

** This pathway is above the static water height resulting from the injection of the entire RWST.

12.3. Summary

In summary, there are several different uncertainties with respect to the flow of gas and water within containment. The MELCOR model generally depicts the flow of gasses between the cavity, lower containment, and upper containment reasonably well. The MELCOR model also models the flow of water within containment reasonably well; however, it neglects several potential paths for water to slowly intrude into the cavity. This has a minor effect on ex-vessel coolability due to modest surface area available for melt spreading and the resulting high melt depth.

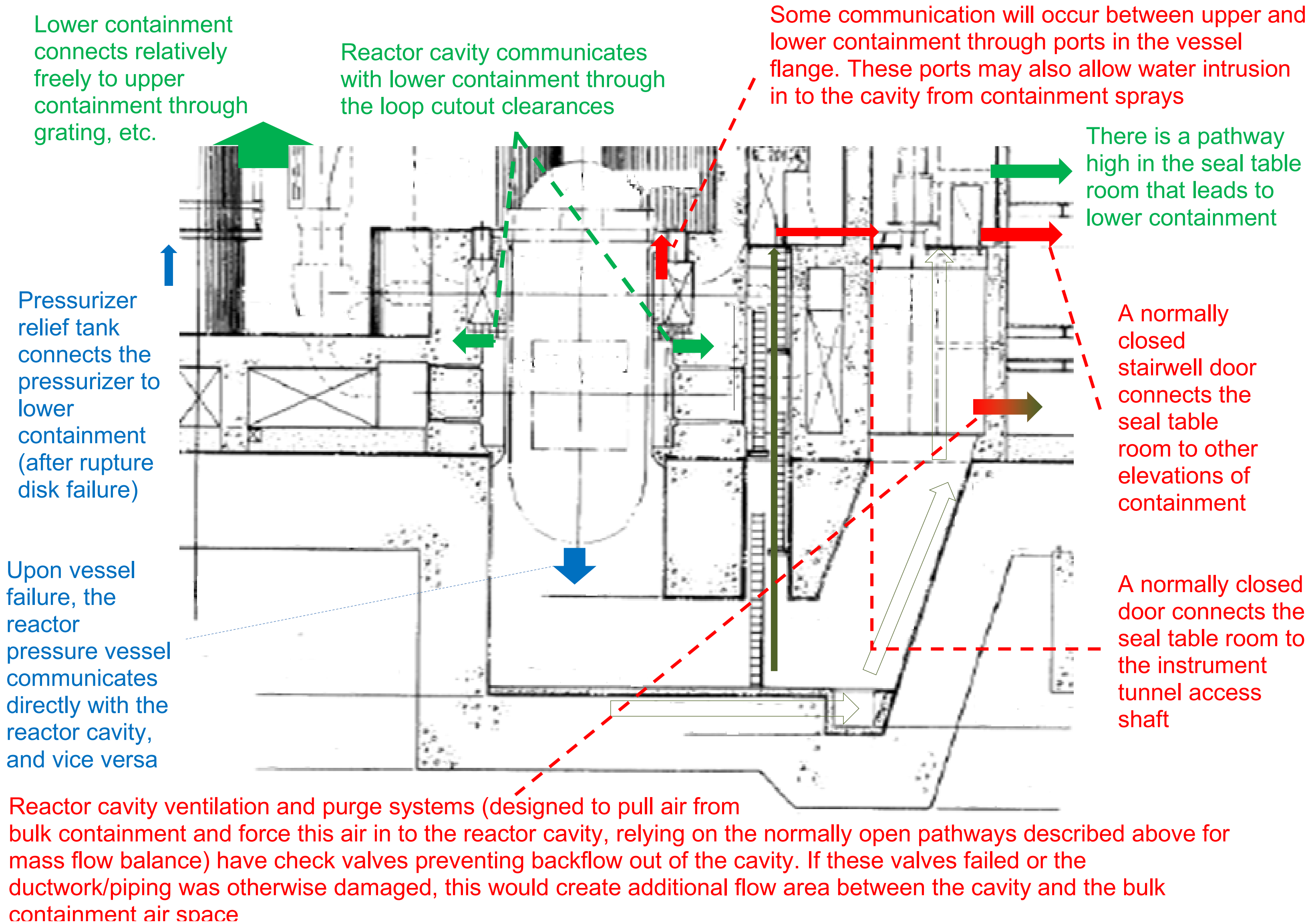


Figure 12-1: Major Reactor Cavity Communications Pathways

13. Risk Metric Surrogate Definitions (LERF, LRF, CCFP)

In the L3PRA Project, risk metric surrogates are used. Namely, large early release frequency (LERF), large release frequency (LRF), and conditional containment failure probability (CCFP). In the following paragraphs, the development and definitions of these terms as used in the L3PRA Project is provided.

13.1. Large early release frequency (LERF)

NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking," (NRC, 2012) and the Level 2 PRA trial use and pilot application (TUPA) standard (ASME, 2014) define large early release frequency (LERF) as the frequency of a "rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before effective implementation of offsite emergency response, and protective actions, such that there is a potential for early health effects." To specify a quantitative value for LERF for a given set of Level 2 PRA results, it is necessary to specify the source term specifications that will be used to delineate release category frequencies as either LERF or non-LERF. This is most commonly done using a combination of warning time and release magnitude. Using the R01_L2 results, after processing through the MACCS offsite consequence analysis,[15] a definition was developed based on when early health effects[16] were predicted as a function of the following attributes:

- The warning time, defined here to be the time at which the cumulative environmental Iodine release fraction exceeds 1% ($3.2 \cdot 10^{17}$ Bq)[17], minus the time that General Emergency conditions are met; and
- The cumulative release fraction of the Iodine chemical class.

The results are not very sensitive to these choices (relative to similar metrics related to Cesium or noble gas release timing and magnitude), but this combination provides the best discrimination. **Figure 13-1** shows these results, with cases where early health effects are predicted to occur shown in blue and those where they are not shown in red. From this, it can be seen that cases with early health effects generally meet the following criteria:

- Warning time (based on Iodine release exceeding 1%) < 20 hours; and
- Cumulative environmental Iodine release fraction > 4% ($1.3 \cdot 10^{18}$ Bq of I and $1.4 \cdot 10^{17}$ Bq of I-131).[18]

These criteria appear as a box in the figure. One source term meets this criteria without having predicted early health effects. Meanwhile, there is one notable outlier, which is that Case 3A1

[15] When used in the context of defining LERF, it is important to note that these MACCS results do include EP modeling. However, a non-evacuating cohort is defined, which is generally representative of a no-EP situation with two key exceptions: (i) the non-evacuating cohort is hot-spot relocated, if projected doses will exceed specified limits and (ii) the non-evacuating cohort has shielding factors that are indicative of normal activity rather than evacuation (evacuation-related shielding factors are based on the lower level of protection afforded to an individual who is in a vehicle relative to an individual engaging in normal activity who is likely to be inside a structure). Despite these two caveats, the non-evacuating cohort is still a reasonable "back stop" for capturing the avoided effects that the EP modeling of other cohorts will mask.

[16] Here, prodromal vomiting is the MACCS output used to distinguish cases where early health effects are predicted to occur.

[17] This is 1% of the sum of the un-decayed (i.e., at reactor trip) I-131 through I-135 initial inventory for middle-of-cycle used in the Level 2 and Level 3 PRA. Note that MACCS does account for radiological decay, its simply neglected here for ease in identifying a radiological release magnitude that is not time-dependent.

[18] Activity of all I and I-131, respectively, based upon un-decayed (i.e., at reactor trip) initial inventories for middle-of-cycle used in the Level 2 and Level 3 PRA.

(from the R01 results) predicted early health effects despite having a warning time and source term well outside of the above parameters. This outlier was investigated and found to be a combination of code bugs (since fixed) and unintended (but plausible) modeling.

Based on the above, release categories are defined to contribute to LERF if their representative source term has a warning time (based on Iodine release exceeding 1%) less than 20 hours simultaneous with the cumulative Iodine release fraction being greater than 4%.

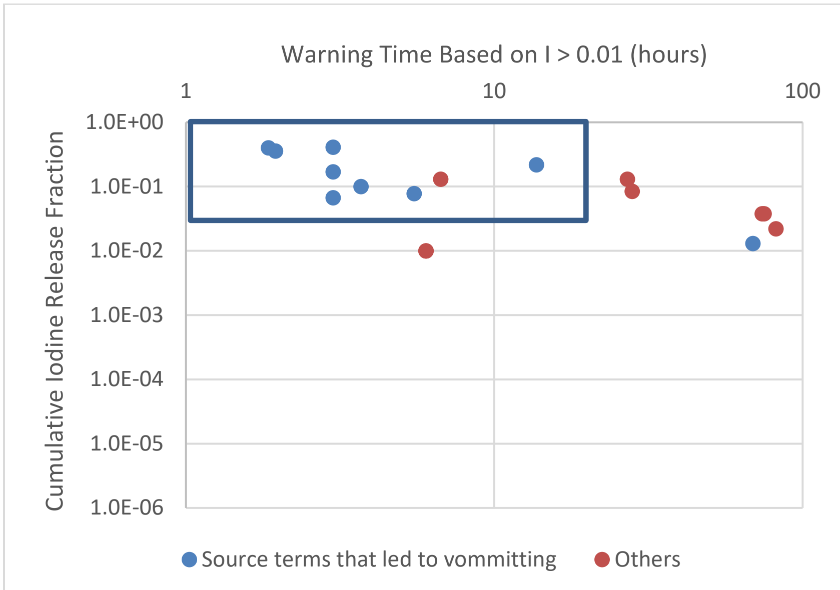


Figure 13-1: Early Health Effects Based on R01 Level 3 Analysis

While defining LERF based upon early injuries is an appropriate metric and consistent with the wording of the definition of LERF, it is not necessarily consistent when using LERF as a risk surrogate directly in conjunction with 51 FR 28044 (and the Quantitative Health Objectives therein) (NRC, 1986) in which early fatality serves as the defining metric.

An alternative definition for LERF is therefore investigated here considering the potential for early fatalities rather than early non-fatal health effects. Once again, using the R01_L2 results (Helton, 2014), a definition based on early fatalities was predicted based on warning time and cumulative Iodine release fraction. **Figure 13-2** demonstrates the R01_L2 results. The cases with early fatalities were encompassed by the following criteria:

- Warning time (based on Iodine release exceeding 1%) < 3.5 hours; and
- Cumulative environmental Iodine release fraction > 4% (1.3×10^{18} Bq of I and 1.4×10^{17} Bq of I-131) – somewhat arbitrarily chosen to match the previous definition since there is insufficient data to directly infer this.

In summary, release categories are defined to contribute to this alternative LERF definition if their representative source term has a warning time (based on Iodine release exceeding 1%) less than 3.5 hours simultaneous with the cumulative Iodine release fraction being greater than 4%.

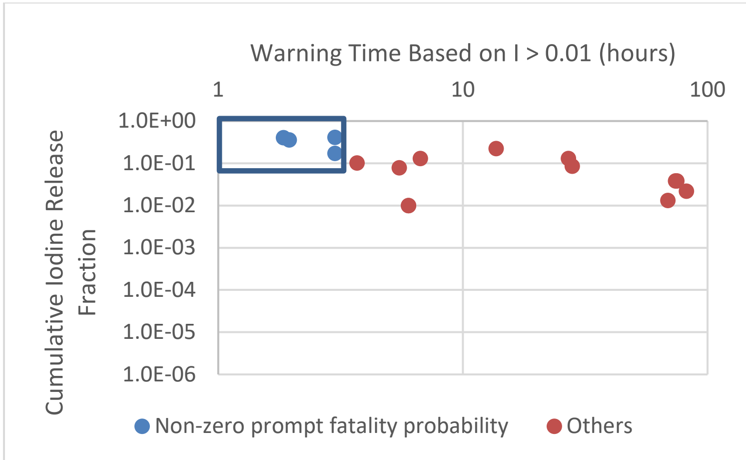


Figure 13-2: Early Fatalities Based on R01 Level 3 Analysis

To inform the appropriateness of these definitions, the R01_L2 results were also applied to several alternative definitions that are described below. **Table 13-1** provides the release category assignment to each of these definitions as well as the total release frequency. Note that the definitions previously introduced in this section generate similar results to the alternative definitions.

- *Definition Used to Assert Model Convergence for the R01_L2 model (LRF/LERF)* – For the sake of determining model convergence in the R01_L2 PRA, release categories that had a source term that includes a cumulative Cesium class release of 0.01 were assumed to have "large" releases (and thus contribute to LRF, and potentially to LERF).[19] Regarding the timing of release, it was assumed that a delta-t of less than 8 hours between the onset of a significant release (as discussed in Section 2.5.2 of the main report) and the provisional time for conditions corresponding to a General Emergency (as also discussed in Section 2.5.2 of the main report) were classified as early. So, release categories that met both the timing and magnitude criteria were considered LERF.
- *NUREG/CR-6595 LERF Definition* - In this study, a set of calculations sought to determine source terms which would lead to an early fatality within 1 mile of the plant (Pratt, 1999). It was determined that for early releases, an iodine or tellurium release fraction of around 2.5% was sufficient to cause a fatality. Here, "early release" was defined to be within four hours of accident initiation. The criteria for this metric applied here is as follows: warning time (based on Iodine release exceeding 1%) < 4 hours; and a cumulative environmental Iodine OR Tellurium release fraction > 2.5%
- *Electric Power Research Institute (EPRI) LERF Definition* - In 1995, EPRI issued its "PSA Applications Guide" as an aid to utilities in formulating a PRA (EPRI, 1995). Contained are two definitions for LERF: 1) unscrubbed containment failure pathway of sufficient size to release the contents of containment within one hour, which occurs before or within four hours of vessel breach, or 2) unscrubbed containment bypass (e.g., SGTR) pathway occurring with core damage.

[19] Note that the CIF and CIF-SC release categories were inclusively assumed to be "large" even though they did not meet this criterion because these release categories have a high degree of uncertainty due to the assumed location of the leak, which in the R01_L2 model was (potentially) non-conservatively assumed to be preferentially into the auxiliary building.

Table 13-1: Release Category Assignment to LERF Definitions Using the Previous (2014; R01_L2) Results

		Based on accident termination 48 hours after SAMG entry					Based on no accident termination assumption				
		New health eff.	New fatalities	R01_L2 Converge	N/CR-6595	EPRI '95	New health eff.	New fatalities	R01_L2 Converge	N/CR-6595	EPRI '95
Release Category	Frequency (/year)	LERF?	LERF?	LERF?	LERF?	LERF?	LERF?	LERF?	LERF?	LERF?	LERF?
V-F	1.129E-06	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
V-F-SC	2.258E-06	Yes	Yes	Yes	Yes	-	Yes	Yes	Yes	Yes	-
V	1.270E-07	-	-	Yes	-	Yes	-	-	Yes	-	Yes
V-SC	2.541E-07	-	-	-	-	-	-	-	-	-	-
SGTR-O	1.096E-08	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
SGTR-O-SC	2.956E-08	Yes	-	Yes	Yes	-	Yes	-	Yes	Yes	-
SGTR-C	7.598E-10	-	-	Yes	-	Yes	-	-	Yes	-	Yes
SGTR-C-SC	2.115E-09	-	-	-	-	-	-	-	-	-	-
ISGTR	2.416E-10	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
CIF	3.461E-08	-	-	Yes	-	-	-	-	Yes	-	-
CIF-SC	2.120E-08	-	-	Yes	-	-	-	-	Yes	-	-
ECF	9.143E-09	Yes	-	-	-	-	Yes	-	-	-	-
ECF-SC	0	-	-	Yes	-	-	-	-	Yes	-	-
LCF	1.158E-05	-	-	-	-	-	-	-	-	-	-
LCF-SC	3.395E-06	-	-	-	-	-	-	-	-	-	-
BMT	2.438E-05	-	-	-	-	-	-	-	-	-	-
BMT-SC	6.992E-06	-	-	-	-	-	-	-	-	-	-
NOCF	1.161E-06	-	-	-	-	-	-	-	-	-	-
NOCF-SC	3.595E-09	-	-	-	-	-	-	-	-	-	-
None	6.952E-07	-	-	-		-	-	-	-	-	-
Total	5.21E-05	3.44E-06	3.40E-06	3.61E-06	3.43E-06	1.27E-06	3.44E-06	3.40E-06	3.61E-06	3.43E-	1.27E-06
Fraction of total release frequency	1.0	0.066	0.065	0.069	0.066	0.024	0.066	0.065	0.069	0.066	0.024

13.2. Large release frequency (LRF)

NUREG-2122 (NRC, 2012) describes some of the considerations associated with this term under entries for “frequency” and “large release,” but like the Level 2 PRA TUPA standard, a definition of the term is not provided.

Separately, SECY-13-0029, “History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission,” (NRC, 2013) provides:

- A history of the term’s origin in the 1986 Safety Goal Policy Statement,
- Subsequent efforts by the NRC staff to define the term during the late 1980s and early 1990s,
- The origin and promulgation of core damage frequency and large early release frequency as surrogates for the Safety Goal Policy’s quantitative health objections (QHOs),
- The use of a definition prescribed in the Electric Power Research Institute’s Advanced Light Water Reactor Utility Requirement Document during certification of the first three advanced light water reactor designs,
- Discussion about complications in defining the term when it is viewed as a quantitative health objective (QHO) surrogate,
- The intent to transition ALWRs to core damage frequency/LERF-based decision-making prior to first fuel load.

Of note is the fact that the early 1990s definition attempts were pre-disposed to focus on LRF as a surrogate for the prompt fatality QHO.

For the purposes of the L3PRA Project the following definitions (and supporting quantitative measure) were used:

- Large release frequency - the summation of those release category frequencies that involve large releases; e.g., those that have an associated source term magnitude (in terms of cumulative cesium release) that is significantly greater than the source term for the largest “intact containment” release category.
- Large release - a release of airborne fission products to the environment that is of sufficient magnitude to cause a substantial increase in calculated offsite impacts, above those from accident classes where the containment fission product barrier has remained intact, regardless of its timing.
- Intact containment – a situation wherein the containment successfully isolates, is not bypassed, and does not experience an increase in the effective leakage area (i.e., an induced failure) to the airborne pathway.

In brief, large release frequency becomes the summation of the frequency of all release categories that include containment bypass or containment failure, excluding those where fission product scrubbing (or other mechanisms) result in a source term comparable to (quantified below), or smaller than, the remainder of the (intact containment) source terms.

The term “comparable to” here refers to a cumulative Cs release fraction that is within two orders of magnitude of the maximum intact containment source term. For the results assuming no accident termination of the simulation,[20] the cutoff value given by this definition is 2.9×10^{-4},

[20] Assuming termination of the simulation 48 hours after SAMG entry, the cutoff value given by this definition is $9.4 \cdot 10^{-3}$ which translates to a release of $6.5 \cdot 10^{11}$ Bq of total Cs and $2.5 \cdot 10^{11}$ Bq of Cs-137

which translates to a release of 2.0x10^{14} Bq of total Cs and 7.8x10^{13} Bq of Cs-137.[21] **Figure 13-3** and **Figure 13-4** offer support for this selection in cutoff value by plotting population dose and the economic cost versus the associated cumulative Cs release for each of the R01_L2 cases. Note that there is a modest degree of clustering in the results below 2.9x10^{-4} (dashed line).

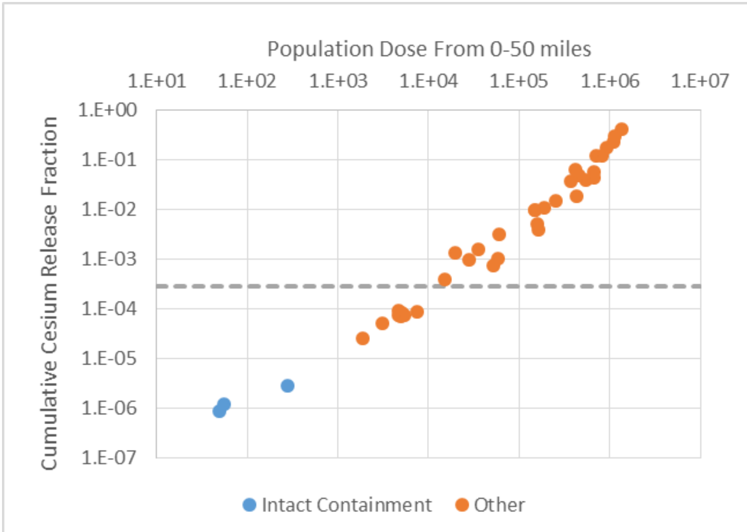


Figure 13-3: Population Dose Based on R01 Level 3 Analysis

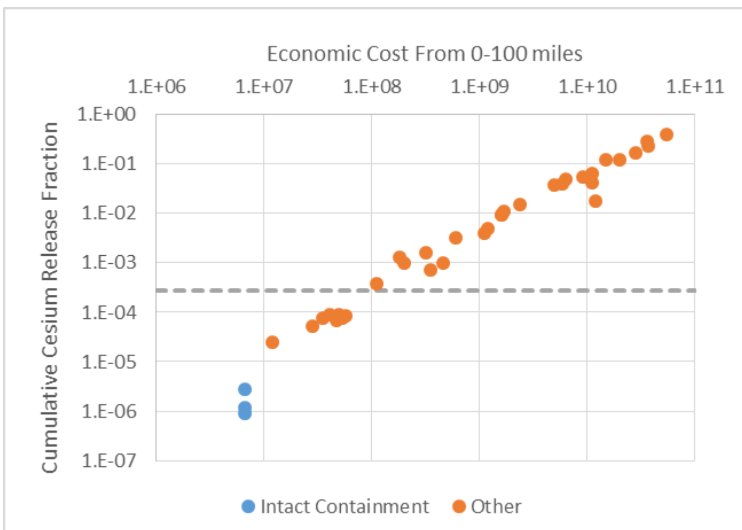


Figure 13-4: Economic Impact

To inform the appropriateness of the approach used for the R01_L2 in the L3PRA Project the results were also applied to several alternative definitions described below. **Table 13-2** provides the release category assignment to each of these definitions as well as the total release frequency (except for the EPRI definition). Note that the definition previously introduced in this section generates similar results to the alternative definitions.

[21] All MELCOR calculations for the Level 2 PRA are performed assuming middle-of-cycle inventories at the time of reactor trip. For reference, the beginning-of-cycle Cs-137 inventory is 57% of the middle-of-cycle inventory, while the end-of-cycle inventory is 40% higher than the middle-of-cycle inventory.

- *Definitions Used to Assert Model Convergence for the L3PRA Project R01_L2 model (LRF/LERF)* – For the sake of determining model convergence in the R01_L2 PRA, release categories that had a source term that includes a cumulative cesium class release of 0.01 were assumed to have "large" releases (and thus contribute to LRF, and potentially to LERF).[22] Regarding the timing of release, it was assumed that a delta-t of less than 8 hours between the onset of a significant release (as discussed in Section 2.5.2 of the main report) and the provisional time for conditions corresponding to a General Emergency (as also discussed in Section 2.5.2 of the main report) were classified as early. So, release categories that met both the timing and magnitude criteria were considered LERF.
- *NUREG/ CR-6094 LRF Definition* - Several calculations were performed to identify the characteristics of a release that would result in an early fatality (Hanson, 1994). A no-evacuation case with 260 MCi of noble gases and 12 MCi of iodine released had the potential for an early fatality. The criteria for this metric applied here is as follows: a "large" release is defined to be one in which the release fraction of Iodine is more than 12 MCi.
- *Finnish LRF Definition* - Guide YVL A.7, requirement 306 provides the Finnish definition of a Large Release Fraction (STUK, 2013) as "the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq." Hence, a "large" release is taken to be one in which more than 10^{14} Bq of Cs-137 is released.
- *Swiss LRF Definition* - ENSI-A05/e provides the Swiss definition of a large release fraction (ENSI, 2009) as "the expected number of events per calendar year with a release of more than 2×10^{14} Bq of Cs-137 per calendar year." For this case, a "large" release is one in which more than 2×10^{14} Bq of Cs-137 is released
- *EPRI LRF Definition* – In EPRI's "Advanced Light Water Reactor Utility Requirements Document" (EPRI, 1999), LRF is defined as the cumulative frequency of all sequences with a dose greater than 25 rem whole body at a half mile from the reactor assuming exposure to the plume for the first 24 hours after core damage begins. Application of this definition requires either dedicated analyses (e.g., establishing the peak dose location and calculating dose for the fixed exposure duration), or in the case of typical MACCS analyses, manipulation and extension of existing analyses. This definition was applied to the L3PRA Project R01_L2 results using the latter approach. The non-evacuating cohort within the 0.5-0.7-mile ring results were scrutinized (accounting for shielding factors and exposure duration) to arrive at an estimate of the peak dose rate. This was multiplied over the 24-hour duration yielding a simplified estimate of the dose over the first 24 hours of plume exposure ("exposure to the plume for the first 24 hours" is being interpreted to mean the significant portion of the plume exposure). It is important to note that close-in dose results used in this manner are particularly subject to modeling uncertainties in the MACCS framework.

[22] Note that the CIF and CIF-SC release categories were inclusively assumed to be "large" even though they did not meet this criteria. The reason for this was because these release categories have a high degree of uncertainty due to the assumed location of the leak, which in the R01_L2 model was (potentially) non-conservatively assumed to be preferentially into the auxiliary building.

Table 13-2: Release Category Assignment to LRF Definitions Using the L3PRA Project Results

RC	Frequency (/year)	Based on accident termination 48 hours after SAMG entry						Based on no accident termination assumption					
		New	R01_L2 L3PRA Project Converge	N\CR-6094	Swiss	Finnish	EPRI	New	R01_L2 L3PRA Project Converge	N/CR-6094	Swiss	Finnish	EPRI
		LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?	LRF?
V-F	1.129E-06	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
V-F-SC	2.258E-06	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
V	1.270E-07	-	Yes	-	Yes	Yes	Yes	Yes	Yes	-	Yes	Yes	Yes
V-SC	2.541E-07	-	-	-	-	Yes	Yes	Yes	-	-	-	Yes	Yes
SGTR-O	1.096E-08	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
SGTR-O-SC	2.956E-08	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
SGTR-C	7.598E-10	Yes	Yes	-	Yes	Yes	Yes	Yes	Yes	-	Yes	Yes	Yes
SGTR-C-SC	2.115E-09	-	-	-	Yes	Yes	Yes	Yes	-	-	Yes	Yes	Yes
ISGTR	2.416E-10	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
CIF	3.461E-08	-	Yes	-	Yes	Yes	-	Yes	Yes	-	Yes	Yes	-
CIF-SC	2.120E-08	-	Yes	-	-	-	-	-	Yes	-	-	-	-
ECF	9.143E-09	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
ECF-SC	0	-	Yes	-	Yes	Yes	-	Yes	Yes	-	Yes	Yes	-
LCF	1.158E-05	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
LCF-SC	3.395E-06	-	-	-	-	-	-	-	-	-	-	-	-
BMT	2.438E-05	-	-	-	-	-	-	-	-	-	-	-	-
BMT-SC	6.992E-06	-	-	-	-	-	-	-	-	-	-	-	-
NOCF	1.161E-06	-	-	-	-	-	-	-	-	-	-	-	-
NOCF-SC	3.595E-09	-	-	-	-	-	-	-	-	-	-	-	-
None	6.952E-07	-	-	-	-	-	-	-	-	-	-	-	-
Total	5.21E-05	1.50E-05	1.52E-05	1.50E-05	1.52E-05	1.54E-05	1.54E-05	1.54E-05	1.52E-05	1.50E-05	1.52E-05	1.54E-05	1.54E-05
Fraction of total frequency	1.0	0.29	0.29	0.29	0.29	0.30	0.30	0.30	0.29	0.29	0.29	0.30	0.30

13.3. Conditional containment failure probability (CCFP)

NUREG-2122 defines conditional containment failure probability as "the likelihood that the containment structure fails to perform its function of retaining fission products." A key point here is whether the containment in this context includes containment systems (e.g., containment sprays) and surrounding structures. Here, it is strictly assumed that the containment is either intact (and thus does not contribute to CCFP) or it has failed or been bypassed (and thus does contribute to CCFP), irrespective of the source term magnitude and role of scrubbing. Thus, CCFP here is simply the ratio of the release categories involving a failed or bypassed containment to the overall release frequency.

13.4. Application of these definitions to the R01_L2 Results

The above definitions were retroactively applied to the R01_L2 L3PRA Project Level 2 results as a means of showing their effect. The R01_L2 L3PRA Project Level 2 results presented in Section 2.6 of the main report were used, which were the results solved by sequence, as opposed to those gathered by end state. **Table 13-3** provides the release category assignment to the risk metric types introduced in this section.

Table 13-3: Release Category Assignment to Risk Metric Types Using the Previous (2014; R01_L2) L3PRA Project Results

		Based on accident termination 48 hours after SAMG entry[23]				Based on no accident termination assumption			
RC	Frequency (/year)	LERF?*	LERF?**	LRF?	CCFP?	LERF?*	LERF?**	LRF?	CCFP?
V-F	1.129E-06	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
V-F-SC	2.258E-06	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
V	1.270E-07	-	-	-	Yes	-	-	Yes	Yes
V-SC	2.541E-07	-	-	-	Yes	-	-	Yes	Yes
SGTR-O	1.096E-08	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
SGTR-O-SC	2.956E-08	Yes	-	Yes	Yes	Yes	-	Yes	Yes
SGTR-C	7.598E-10	-	-	Yes	Yes	-	-	Yes	Yes
SGTR-C-SC	2.115E-09	-	-	-	Yes	-	-	Yes	Yes
ISGTR	2.416E-10	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
CIF	3.461E-08	-	-	-	Yes	-	-	Yes	Yes
CIF-SC	2.120E-08	-	-	-	Yes	-	-	-	Yes
ECF	9.143E-09	Yes	-	Yes	Yes	Yes	-	Yes	Yes
ECF-SC	0	-	-	-	Yes	-	-	Yes	Yes
LCF	1.158E-05	-	-	Yes	Yes	-	-	Yes	Yes
LCF-SC	3.395E-06	-	-	-	-	-	-	-	Yes

[23] Here, the source term is truncated at 48 hours after SAMG entry; however, the MACCS results used to develop the quantitative definition use the untruncated source terms.

Table 13-3: Release Category Assignment to Risk Metric Types Using the Previous (2014; R01_L2) L3PRA Project Results

		Based on accident termination 48 hours after SAMG entry[23]				Based on no accident termination assumption			
RC	Frequency (/year)	LERF?*	LERF?**	LRF?	CCFP?	LERF?*	LERF?**	LRF?	CCFP?
BMT	2.438E-05	-	-	-	-	-	-	-	Yes
BMT-SC	6.992E-06	-	-	-	-	-	-	-	Yes
NOCF	1.161E-06	-	-	-	-	-	-	-	-
NOCF-SC	3.595E-09	-	-	-	-	-	-	-	-
None	6.952E-07	-	-	-	-	-	-	-	-
Total	5.21E-05	3.44E-06	3.40E-06	1.50E-05	1.55E-05	3.44E-06	3.40E-06	1.54E-05	5.02E-05
Fraction of total frequency	1.0	0.07	0.07	0.29	0.30	0.07	0.07	0.30	0.96

* Refers to the LERF definition based upon early injuries.
** Refers to the LERF definition based upon early fatalities.

References:

ASME, 2014 — ASME, *Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)*, ASME/ANS RA-S-1.2-2014, 2014

ENSI, 2009 — Swiss Federal Nuclear Safety Inspectorate (ENSI), *Probabilistic Safety Analysis (PSA): Quality and Scope,* ENSI-A05/E, March 2009.

EPRI, 1995 — Electric Power Research Institute, PSA Applications Guide, EPRI TR-105396, August 1995.

EPRI, 1999 — Electric Power Research Institute, Advanced Light Water Reactor Utility Requirements Document, EPRI TR-016780, March 1999.

Hanson, 1994 — Hanson, A. et al., *Calculations in Support of a Potential Definition of Large Release,* NUREG/CR-6094, May 1994. [ML 12191A004]

NRC, 1986 — NRC, *Safety Goals for the Operation of Nuclear Power Plants,* 51 FR 28044, August 21, 1986. [ML 011210381]

NRC, 2012 — NRC, *Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking,* NUREG-2122, June 5, 2012. [ML121570620]

NRC, 2013 — NRC, *History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission*, SECY-13-0029, March 22, 2013. [ML13022A207]

Pratt, 1999 — Pratt, W. et al., *An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events,* NUREG/CR-6595, January 1999. [ML072710052]

STUK, 2013 Finnish Radiation and Nuclear Safety Authority (STUK), *Probabilistic Risk Assessment and Risk Management of a Nuclear Power Plant,* Guide YVL A.7, November 15, 2013.

14. SAMG Navigation, Special Considerations in Modeling

Predicting how the plant staff will navigate through the SAMGs is based primarily on MELCOR results. **Figure 14-1** provides a list of the SAMGs referenced in the L3PRA Project. **Table 14-1** provides the thresholds specified in the SAMG DFC and severe challenge status tree for plant conditions that can be determined from MELCOR simulations for the purposes of informing human reliability analysis (see Section 2.4.5 of the main report for additional discussions on this topic). The two notable exceptions to using the MELCOR results are:

- Entrance to SCG-1: Site releases >1R total effective dose equivalent (TEDE) or 5R committed dose equivalent (CDE) Thyroid
- Entrance to SAG-5: Site releases >100 mrem TEDE of 500 mrem Thyroid CDE

SACRGs	**Severe Accident Control Room Guidelines**		
SACRG-1	Severe Accident Control Room Guideline Initial Response		
SACRG-2	Severe Accident Control Room Guideline for Transients After the TSC is Functional		
DFC	**TSC Diagnostic Flow Chart**		
SAGs	**Severe Accident Guidelines**	SAG-1	Inject into the Steam Generators
		SAG-2	Depressurize the RCS
		SAG-3	Inject into the RCS
		SAG-4	Inject into Containment
		SAG-5	Reduce Fission Product Releases
		SAG-6	Control Containment Conditions
		SAG-7	Reduce Containment Hydrogen
		SAG-8	Flood Containment
SCST	**TSC Severe Challenge Status Tree**		
SCGs	Severe Challenge Guidelines	SCG-1	Mitigate Fission Product Releases
		SCG-2	Depressurize Containment
		SCG-3	Control Hydrogen Flammability
		SCG-4	Control Containment Vacuum
SAEGs	**Severe Accident Exit Guidelines**	SAEG-1	TSC Long Term Monitoring Activities
		SAEG-2	SAMG Termination
CAs	**Computational Aids**	CA-1	RCS Injection to Recover Core
		CA-2	Injection Rate for Long Term Decay Heat Removal
		CA-3	Hydrogen Flammability in Containment
		CA-4	Volumetric Release Rate from Vent
		CA-5	Containment Water Level and Volume
		CA-6	RWST Gravity Drain
		CA-7	Hydrogen Impact when Depressurizing Containment

Figure 14-1: Severe Accident Management Guidelines (SAMGs)

Entry in to SCG-1 is the highest priority action (when conditions prompt) in the Westinghouse SAMG hierarchy. In both cases, the SAMGs point the TSC to offsite dose assessment as the method of estimation. The two thresholds above correspond to Emergency Action Level criteria for all reactor modes of operation for declaration of a General Emergency and Site Area Emergency, respectively. More specifically:

- The SCG-1 thresholds correspond to an Emergency Action Level (EAL) for a General Emergency (GE)

- The SAG-5 thresholds correspond to an EAL for a Site Area Emergency (SAE)

Unlike the SAMGs, the Emergency Plan includes alternate means of establishing GE and SAE, via either specified radiation monitor readings or field survey results that indicate closed window dose rates exceeding the above dose thresholds and are expected to persist for one hour. Since the TSC and EOF will be in communication (and in fact, dose projections are initially done by the TSC), it is reasonable to expect that the same alternative information sources may influence the execution of the SAMGs. Since entrance in to SCG-1 and SAG-5 cannot be determined based on MELCOR results, it is necessary to develop a basis for entry based on the scenario characterization.

This issue was investigated and resolved, including the conduct of supporting analysis using the NRC's RASCAL code by considering the full background in terms of the entry requirements for SCG-1 and SAG-5, in concert with the related Emergency Plan activities (as synopsized above). Scenarios to be investigated were identified to develop semi-generic criteria for SCG-1 and SAG-5 entry. Boundary condition variations were then prescribed to these scenarios, and the results of roughly 100 RASCAL calculations were supplied. The results of these calculations were used to diagnose SCG-1 and SAG-5 entry for each of the three weather conditions considered. Since the Level 2 HRA, by its nature, does not consider a particular weather condition when assessing SAMG navigation, this was further distilled into the notional scheme shown below in **Figure 14-2** that was weather independent.

There are also complexities related to the entry in to SCG-3 that require discussion. SCG-3 is the only guideline that has a 2-step entry process. The first step queries containment hydrogen concentration. A second entry condition is whether Computational Aid (CA)-3 indicates that a "Severe Hydrogen Challenge" exists. CA-3 includes a series of 5 figures for determining this, and there are some assumptions that must be made here to facilitate evaluation of this.

It is expected that MELCOR will predict combustible gas concentrations significantly higher than those envisioned by default curves/guidance in CA-3. This is due to the oxidation of non-zircaloy material in the core, and the production of other combustible gases during MCCI. Combustible gas generation in severe accident codes is affected by temperature profiles, the availability of steam or air for oxidation (i.e., starvation), and the material response (e.g., oxidation layer formation and breakaway). (EPRI, 2012) retains the same basic rules-of-thumb for combustible gas generation from zircaloy, in part owing to a different perspective on some of the above issues[24]. MELCOR's use here will highlight those differences in perspective, in terms of the evaluation of hydrogen concerns in the SAMGs.

Regarding the evaluation of SCG-3 entry, the following assumptions were made:

- From the time of SAMG entrance to vessel breach, hydrogen monitoring/sampling is assumed to be available, so long as DC power is available.
- From the time of vessel breach to the end of the accident, hydrogen monitoring/sampling may be compromised by plugging.
    a. Sample lines are likely to become plugged after vessel failure based on post-vessel failure debris generation and on the small size of the sampling lines.
    b. In cases where a lack of hydrogen monitoring/sampling (post vessel breach) would lead to different determinations in CA-3 than using the default oxidation curves, the default

[24] A key point related to model uncertainty is the difference in the way that MELCOR and MAAP treat core degradation. Due to different handling of melt porosity, the codes calculate significantly different degrees of zircaloy oxidation during core melt.

oxidation curves will be used on the assumption that hydrogen monitoring/sampling is unavailable.

c. In some cases, pre-vessel breach behavior may lead to hydrogen concentrations significantly different that those in the default CA-3 curves (e.g., the 50% zircaloy oxidation curve). This situation may lead the TSC to place less emphasis on these default curves even after the hydrogen measurement instrumentation is saturated or failed (due to plugging). This situation is acknowledged in an identified HRA-related (SCG-3 entry) model uncertainty.

- In terms of SCG-3 entry, no distinction will be made based on how far from the flammability line the conditions are (i.e., just above the line will be treated the same as well above the line).

In developing the L3PRA Project Level 2 model, assumptions were made pertaining to SAG-3 entry after vessel-breach. SAG-3 entry is based on core exit thermocouple (CETC) temperature greater than 711°F. Prior to vessel breach, this condition will always be met, unless the core is substantively re-covered, because the SAMG entry condition is CETC greater than 1200°F. CETC temperature will remain high while the core is uncovered and relocating to the lower head. Regarding survivability of the CETCs during this time (core melt through vessel breach):

- MELCOR generally predicts core exit temperatures as high as 3000°F, though the CETC readings from several sampled MELCOR calculations peaked at ~2300°F. The exact conditions the CETCs will see is dependent on their precise mounting characteristics, both axially and radially. Temporary errors occur above 2200°F, and permanent errors occur above 2500°F. The CETCs are not reliable after significant core relocation. This, in combination with the experience from Three Mile Island, suggests that some CETCs will fail, some will produce unreliable information, some will produce seemingly reliable information even though they have failed, and some will continue to produce reliable information.
- The other means of estimating SAG-3 entry are either unreliable after core damage or are unhelpful (e.g., the nuclear instrumentation system monitors). The cold leg RTDs may be the best source of corroborating information.

For surviving CETCs, post vessel breach temperatures, as predicted by MELCOR, generally remain high (above 711°F). MELCORs strengths and limitations in this regard are:

- MELCOR models the bulk uni-directional flow of gasses between the cavity and the vessel following vessel breach. It also models conduction heat transfer through the vessel walls, as well as convective heat transfer to/from the vessel walls. MELCOR also models transport and deposition of core material to the upper plenum. It will also model steaming effects if the cold-leg accumulators dump water (or other injection occurs), as well as the effect of isothermal expansion if the vessel is pressurized prior to breach.
- MELCOR does not model radiation or counter-current convection through the failed lower head[25]. It also does not model radiative heat transfer from the upper internals to the vessel walls.

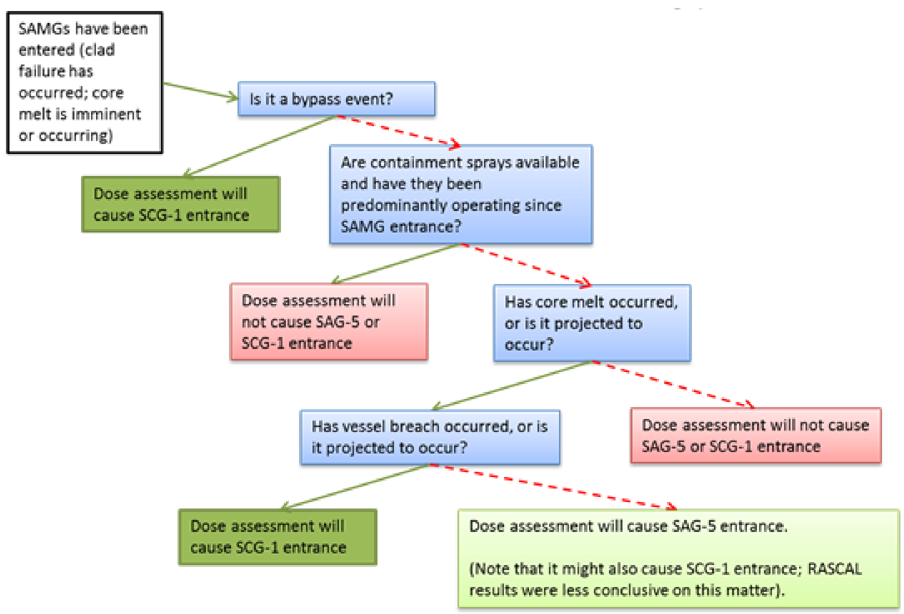
SAG-3 is relatively silent on the issue of its entrance being dependent upon vessel breach, though in stating potential negatives in Step 3, it invokes containment overpressure challenge if MCCI is occurring (a clear indication of the potential that vessel breach has occurred).

Given this information, in the L3PRA Project SAG-3 is assumed to be entered if conditions dictate, the MELCOR modeling of these conditions is reasonable, and sufficient information

[25] For reference, the size of the lower head failure for a sampled MELCOR calculation was 150 cm^2.

would be available from un-failed CETCs (acknowledging that operators may receive spurious information from CETCs that have failed in a manner that produces potentially plausible readings). (NRC, 2013a) and (NRC, 2013b) provide an agency perspective that the CETCs will provide adequate indication for SAMG entrance and subsidiary guideline assessment, in the context of deliberating on a potential regulatory enhancement to pressurized water reactor (PWR) core temperature instrumentation.

The main effect of this assumption is that SAG-3 will cause lower-priority SAGs to not meet the screening HRA factors related to being the 1st or 2nd priority. This generally means that SAG-3 was factored into the model (post-vessel-breach, and in cases in which it is the 1st or 2nd priority) rather than SAG-4 (when the scenario does not include injection of the RWST in to containment) or SAG-5 (in most other cases).



Caution: This assessment is subject to significant uncertainties related to the assumptions in representation of the underlying licensee dose assessment practices (e.g., role of field monitoring input), decision-making (e.g., projection of future degradations), plant conditions (e.g., precipitation), and computational model (e.g., RASCAL versus MIDAS).

Figure 14-2: Notional Scheme for Determining SCG-1 and SAG-3 Entrance

Table 14-1: Mapping of SAMG Parameters to MELCOR Outputs

Strategy / CA	Parameter / value	SAMG setpoint	Instrument range
Entrance	Core exit thermocouple (CETC) readout	1200F	200 to 2300F
SCG-1	Site doses	> 1R TEDE or	n/a

Table 14-1: Mapping of SAMG Parameters to MELCOR Outputs

Strategy / CA	Parameter / value	SAMG setpoint	Instrument range
		5R CDE Thyroid	
SCG-2	Containment pressure (extended range)	>102 psig	-5 to 160 psig
SCG-3	Containment hydrogen	>6% and severe challenge per CA-3	0 to 10% partial pressure
SCG-4	Containment pressure (extended range)	< -5 psig	-5 to 160 psig
SAG-1	SG NR level	<38% in any SG	0 to 100% of span
SAG-2	RCS pressure	>150psig	0 to 3,000 psi
SAG-3	Core temperature	> 711F	200 to 2300F
SAG-4	Containment water level	<23"	0 to 48" (narrow range)
SAG-5	Site doses	> 1R TEDE or 5R CDE Thyroid	n/a
SAG-6	Containment pressure	>3.8 psig	0 to 75 psig
SAG-7	Containment hydrogen	>4%	0 to 10% partial pressure
SAG-8	Containment water volume	< 1.3 million gallons	0 to 120in. (wide range) – note that top of range = ~950,000 gallons
CA-1	1. RCS pressure 2. # of Pressurizer PORVs opened 3. ECCS pump flow rates	n/a	1. 0 to 3,000 psi 2. n/a 3. n/a
CA-2	1. Time since shutdown 2. Injection flow rate	n/a	n/a

Table 14-1: Mapping of SAMG Parameters to MELCOR Outputs

Strategy / CA	Parameter / value	SAMG setpoint	Instrument range
CA-3	1. Containment pressure 2. Containment temperature 3. Containment hydrogen concentration	n/a	1. 0 to 75 psig 2. 0 to 300F 3. 0-10%
CA-4	1. Containment pressure 2. Vent flow rate	n/a	1. 0 to 75 psig 2. n/a
CA-5	Containment water level	n/a	0 to 120in. (wide range) – note that top of range = ~950,000 gallons
CA-6	1. Containment pressure 2. RWST water level	n/a	3. 0 to 75 psig 4. 0-100%
CA-7	1. Containment pressure 2. Containment hydrogen concentration	n/a	3. 0 to 75 psig 4. 0-10%

References:

EPRI, 2012 Electric Power Research Institute, *Severe Accident Management Guidance Technical Basis Report*, EPRI-TR-1025295, October 2012.

NRC, 2013a U.S. Nuclear Regulatory Commission, *Denial of Petition for Rulemaking PRM-50-105 Requesting Amendments Regarding In-Core Thermocouples at Different Elevations and Radial Positions Throughout the Reactor Core*, SECY-13-0063, June 2013.

NRC, 2013b U.S. Nuclear Regulatory Commission, *Staff Requirements – SECY-13-0063 - Denial of Petition for Rulemaking PRM-50-105 Requesting Amendments Regarding In-Core Thermocouples at Different Elevations and Radial Positions Throughout the Reactor Core*, SRM-SECY-13-0063, August 2013.

15. SAPHIRE, Solving the Level 2 PRA Logic Model

The following is a brief overview of the process to solve the Level 2 PRA logic model, which is followed by a detailed description. The L3PRA Project Level 2 (R02) model applies the SAPHIRE Level 1/2 PRA Interface using a switch represented by the 1-CD-XFER event tree that has four branches as a transfer mechanism. To solve the Level 2 PRA logic model to generate cut sets in the 1-CET event tree end states, the linkage rule for the 1-CD-XFER event tree was configured to activate the second of the following three quantification paths:

- For core damage quantification:

  Level 1 ETs → 1-CD-XFER >> 1-CD end-state

- For release category quantification:

  Level 1 ETs → 1-CD-XFER → 1-L1E-BRIDGE-N → 1-PDS → 1-CET >> RELEASE end states

- For PDS quantification:

  Level 1 ETs → 1-CD-XFER → 1-PDS-Q (which includes the containment systems fault trees) >> PDS end-states

- For PDS linkage rule confirmation:

  Level 1 ETs → 1-CD-XFER → 1-PDS-CHECK >> Dummy PDS end-states

Once the 1-CD-XFER event tree linkage rule was configured, the internal event and internal floods Level 1 event trees were linked. Linking the event trees directs SAPHIRE to generate the sequence logic associated with the Level 1 and Level 2 PRA models, as dictated by the structure of, and linkage rules associated with, the relevant event trees. As shown in the table below, the indicated portions of sequence logic were generated by SAPHIRE as it transferred through each event tree for a given sequence.

Event Trees	Sequence Logic Generated
Level 1 event trees	Level 1 accident sequences
1-L1E-BRIDGE-N	Containment systems status
1-PDS	Plant damage state categorization
1-CET	Level 2 accident progression sequences

The above was accomplished with the following four steps, each of which is discussed further below.

1. Configure the 1-CD-XFER event tree linkage rules
2. Link all event trees
3. Generate the Level 2 cut sets
4. Gather the Level 2 cut sets into the 1-REL-* end states

15.1. Configuring the 1-CD-XFER event tree linkage rules

The 1-CD-XFER event tree linkage rules consist of only one logical element that is used to activate or deactivate the different branches of the event tree by substituting either ZV-TRUE or SKIP(1-PHASE-CD-L2), respectively. To solve the Level 2 PRA logic model through the 1-CET to the REL end states, the 1-CD-XFER tree linkage rules are modified as shown below to activate the second branch (i.e., 1-PHASE-CD-L2[1]).

```
IF ALWAYS THEN
   /1-PHASE-CD-L2    = SKIP(1-PHASE-CD-L2);
    1-PHASE-CD-L2[1] = ZV-TRUE;
    1-PHASE-CD-L2[2] = SKIP(1-PHASE-CD-L2);
    1-PHASE-CD-L2[3] = SKIP(1-PHASE-CD-L2);
ENDIF
```

After the linkage rules have been modified, the changes are saved.

15.2. Linking the Level 1 event trees

To link the Level 1 internal event and internal flood event trees, the “Main Trees” display option in the Event Tree viewing pane in SAPHIRE is selected. The event trees with the 1-FLI- and 1-FPI- designators in the event tree name are selected. Then, any one of the selected event trees can be linked.

> ***NOTE:*** Given the size and complexity of the Level 2 PRA logic model and depending on the hardware performance of the computer being used, a long period of time may be needed to complete the event tree linking process, which can range from several tens of minutes to multiple hours. Because these Level 1 event trees now transfer to the Level 2 trees as a result of completing Step 1, SAPHIRE will also link all of the relevant Level 2 event trees.

15.3. Generating Level 2 cut set results

After the sequence logic has been generated by linking the Level 1 event trees through the 1-L1E-BRIDGE-N, 1-PDS, and 1-CET event trees, the model can be solved to generate the Level 2 cut set results. To generate the Level 2 cut set results, the “Main Trees” display option in the Event Tree viewing pane in SAPHIRE is selected, as well as the same event trees that were linked in the previous step. Then, the model can be solved from the pop-up menu.

> ***NOTE:*** The current results for the Level 2 PRA logic model were generated by solving at a truncation of 1E-11/yr. Again, given the size and complexity of the model and depending on the hardware performance of the computer being used, keep in mind that an extensive amount of time may be required to solve the model at this truncation, which can range from tens of hours to a few days. The recommended number of threads to use is two times the number of CPUs. During the quantification of the R02 model in 2017, it was necessary to perform the Solve stage in batches (5-10 initiators at a time) to avoid SAPHIRE crashes.

15.4. Gathering Level 2 cut sets into the 1-CET end states

Once the Level 2 cut sets have been generated, they need to be gathered into their respective release categories. To do this, all of the end states with the designator “1-REL-“ are selected and can be gathered. The indicated truncation should be the same as that used for solving the model. After the cut sets have been gathered, they can be viewed by selecting one or more 1-REL end states of interest and selecting “View Summary Results.”

16. SAPHIRE Level 1/2 PRA Interface

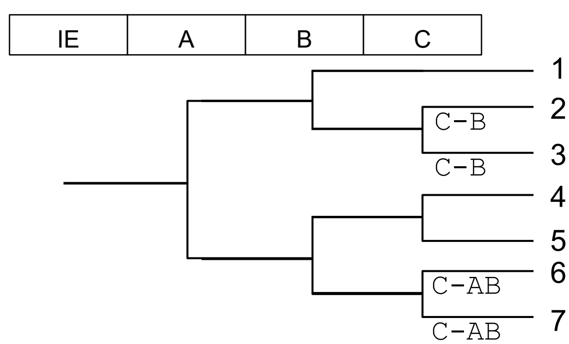
The documentation of the PDS binning process in Section 2.1.2 of the main report presumes a certain level of SAPHIRE familiarity. The discussion in this section provides additional information on how SAPHIRE runs the model. SAPHIRE uses the 1-PDS-Q event tree to sort the results via linkage rules into specific plant damage state (PDS) bins and append the appropriate logic from the containment system models, resulting in plant damage states with the nomenclature “PDS-xx-y,” where ‘xx’ is the sequential PDS number and ‘y’ is the associated combination state for the containment systems shown in the table below.

<table>
<tr><th>PDS sub-bin</th><th>Containment Isolation</th><th>Containment Sprays</th><th>Containment Coolers</th></tr>
<tr><td>PDS-xx-1</td><td rowspan="4">Success</td><td rowspan="2">Available</td><td>Available</td></tr>
<tr><td>PDS-xx-2</td><td>Unavailable</td></tr>
<tr><td>PDS-xx-3</td><td rowspan="2">Unavailable</td><td>Available</td></tr>
<tr><td>PDS-xx-4</td><td>Unavailable</td></tr>
<tr><td>PDS-xx-5</td><td rowspan="2">Failure</td><td>Available</td><td rowspan="2">Not Applicable</td></tr>
<tr><td>PDS-xx-6</td><td>Unavailable</td></tr>
</table>

Quantifying the PDS in this order (PDS rules, followed by containment systems failure combinations) facilitates the breakdown of PDS in terms of both Level 1 similarity and containment system functionality.

SAPHIRE can modify the logic for individual event tree sequences based on a set of predefined rules referred to as linkage rules. The event tree linkage rules are a set of user-defined instructions that search for and modify elements of the logic for individual event tree sequences based on specified criterion. A search criterion for a given linkage rule consists of one or more logical elements, such as an event tree top event or an initiating event. A given search criterion may include several logical elements and, which can be assigned to a single user-defined identifier called a macro. The use of macros enhances the readability of the linkage rules by replacing lengthy search criterion strings with a single identifier. When SAPHIRE evaluates sequences for possible linkage rule substitutions, the code applies the first linkage rule substitution that satisfies the search criterion for a given sequence. For example, the application of the following linkage rule to event tree shown would result in the indicated substitutions:

```
IF A * B THEN
   /C = C-AB
    C = C-AB
ELSEIF B THEN
   /C = C-B
    C = C-B
ENDIF
```



The 1-PDS-Q event tree contains a total of seven top events. Together with the 1-PDS-Q linkage rules, the first four top events operate as logic switches (i.e., they do not introduce any basic events to the Level 1 cut sets) and query the Level 1 sequence logic to determine the following conditions:

- accident type (1-ACCTYPE)
- status of steam generator cooling (1-SGCOOL)
- availability of the refueling water storage tank (1-RWSTAV), and

- availability of the emergency core cooling systems for injection (1-ECCSAV)

The last three top events are system top events with fault tree logic that is applied to the Level 1 cut sets. These top events represent the following containment systems:

- containment isolation system (1-CISOL-H)
- containment spray system (1-CONTSPRAY-H)
- containment cooling units (1-CONTCOOL-H)

The following provides a step-by-step example of how a Level 1 cut set is processed through the 1-PDS-Q event tree into a PDS end state. This example is based on the R02 version of the L3PRA Project model using version 8.1.4.6 of SAPHIRE. This example looks at the dominant Level 1 cut set, which is a loss of nuclear service cooling water (NSCW) precipitated by a common cause failure of all six NSCW motor-driven pumps and results in a failure of the reactor coolant pump (RCP) stage 2 seal, as shown in detail in the following table:

Basic Event Identifier	Description
1-IE-LONSCW	This loss of NSCW initiating event is appended to the cut set because of the initiating event in the next row.
1-IE-SWS-MDP-CR-123456	Common cause failure to run of all NSCW motor-driven pumps.
1-RCS-MDP-LK-BP2	RCP seal stage 2 integrity fails (binding/popping open).

The following is the Level 1 sequence logic associated with this cut set:

Sequence Logic Identifier	Description
/1-RPS	Successful reactor trip
/1-TT	Successful turbine trip
/1-SVC	Successful closure of secondary relief valves (ARVs and SRVs)
/1-PVC	Successful closure of primary relief valves (PORVs and SVs)
1-RCPS-BP	Failure of RCP seal integrity – binding/popping

Lines 42 through 47 of the 1-PDS-Q event tree linkage rules define the following macro relevant to many LONSCW sequences (though it will be seen later that this macro is not used in binning the sequence that manifests the dominant cut set:

```
(42)   PDSM_TRANS_ALL = PDSM_TRANS_A +
(43)                PDSM_TRANS_B +
(44)                INIT(1-IE-ISINJ) +
(45)                INIT(1-IE-LONSCW) +
(46)                INIT(1-IE-LOSING) +
(47)                INIT(1-IE-RTRIP);
```

The following table illustrates step-by-step how the dominant cut set is sorted through the 1-PDS-Q event tree. The relevant linkage rules are shown in the left column, including the related line numbers in parentheses, and the result of applying the linkage rule is described in the right column. The relevant macro identifiers and other search criteria are highlighted.

Relevant 1-PDS-Q Event Tree Linkage Rule	Result
(95) **IF** (120) (**INIT**(1-IE-LONSCW) + (121) **INIT**(1-IE-LOACCW)) * 1-RCPS-BP + (122) PDSM_TRANS_B * (/1-PVC * 1-RCPSC * 1-FAB-SCLOCA) + (123) PDSM_TRANS_B * ((1-BP1 + (124) 1-BP2) * (/1-OPR-02H + (125) /1-OPR-01H)) (139) **THEN** (141) /1-ACCTYPE = **FALSE**-SLOCA; (142) 1-ACCTYPE[1] = **SKIP**(1-ACCTYPE); (143) 1-ACCTYPE[2] = **SKIP**(1-ACCTYPE); … (151) **ENDIF**	The cut set passes through the first branch of the 1-ACCTYPE branch point that is designated for small LOCA accident types (due to the RCP seal LOCA).
(349) **IF** (371) **INIT**(1-IE-LONSCW) * (1-RCPS-BP + (372) 1-PVC) (376) **THEN** (378) /1-SGCOOL = **FALSE**-SGCOOL;[26] (379) 1-SGCOOL = **SKIP**(1-SGCOOL);	The cut set passes through the success branch of the 1-SGCOOL branch point in the event tree indicating that steam generator cooling is available. This is a special case, wherein feedwater is never queried by the Level 1 PRA and is assumed to be available in the PDS binning because its independent failure would greatly reduce the sequence frequency.
(443) **IF** (456) (/1-ACCTYPE + (457) 1-ACCTYPE[6]) * **INIT**(1-IE-LONSCW) (466) **THEN** (468) /1-RWSTAV = **FALSE**-RWSTAV; (469) 1-RWSTAV = **SKIP**(1-RWSTAV);	The cut set passes through the success branch of the 1-RWSTAV branch point in the event tree indicating that the RWST is available (ECCS injection has not occurred due to the loss of NSCW).
(527) **IF** (528) (/1-ACCTYPE + (529) 1-ACCTYPE[4] + (530) 1-ACCTYPE[6]) * **INIT**(1-IE-LONSCW) (542) **THEN** (543) /1-ECCSAV = **SKIP**(1-ECCSAV); (544) 1-ECCSAV = **TRUE**-ECCSAV;	The cut set passes through the failure branch of the 1-ECCSAV branch point in the event tree indicating that ECCS is not available (ECCS is unavailable due to the loss of NSCW).
(616) **IF ALWAYS THEN** (618) /1-CISOL-H = 1-FT-CISOL-F; (619) 1-CISOL-H = 1-FT-CISOL-F;	This linkage rule directs SAPHIRE to always use the fault tree 1-FT-CISOL-F for both branches of this branch point.
(627) **IF ALWAYS THEN** (629) /1-CONTSPRAY-H = 1-FT-CONTSPRAY-F; (630) 1-CONTSPRAY-H = 1-FT-CONTSPRAY-F;	This linkage rule directs SAPHIRE to always use the fault tree 1-FT-CONTSPRAY-F for both branches of this branch point.
(638) **IF ALWAYS THEN** (640) /1-CONTCOOL-H = 1-FT-CONTCOOL-F; (641) 1-CONTCOOL-H = 1-FT-CONTCOOL-F;	This linkage rule directs SAPHIRE to always use the fault tree 1-FT-CONTCOOL-F for both branches of this branch point.

[26] In SAPHIRE linkage rules, the logic of the top branch of an event tree (i.e., success branch) is always complimented. As such, because FALSE-SGCOOL is a house event with a value of FALSE, the use of FALSE-SGCOOL for the top branch results in a complimented value of TRUE.

17. SOARCA, Key Severe Accident Modeling Differences

In most cases, the deterministic modeling in the L3PRA Project and in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project (documented in [NRC, 2012] and [SNL, 2013]) is identical or analogous. However, some key differences are identified here to facilitate an understanding of why, in some cases, the two projects use reasonably different assumptions. This list is not intended to be comprehensive.

1. Regarding reactor coolant pump (RCP) seal leakage, the L3PRA Project relies on the Westinghouse Owners Group (WOG) 2000 RCP seal leakage model (as modified in the NRC's associated Safety Evaluation (NRC, 2003)), which is the PRA consensus model. The original SOARCA project elected to rely on a variation of this model that predicted enhanced seal leakage (for the sequences where enhanced seal leakage occurs) based on RCS conditions, rather than a prescribed time of 13 minutes. The WOG 2000 model leads to earlier RCP seal failure timings. Note that the more recent Surry SOARCA Uncertainty Analysis (SNL, 2016) effort relies on the WOG 2000 model.
2. For many (but not all) of the L3PRA Project accident progression MELCOR calculations, steam generator tube, relief valve cycling/seizure, and/or in-core instrument tube failure via creep rupture are suppressed. Separate analysis was performed to look at the failure likelihoods for these components or they were identified as model uncertainties, and the combination of information is used in the PRA logic model's treatment of induced RCS failures. In the SOARCA project, induced RCS failures (other than in-core instrument tube failure) are modeled to occur or not occur directly in the MELCOR analysis, since there is no accompanying PRA model.
3. There are several modeling differences related to interfacing systems LOCA: turbulent deposition in the RHR piping, assumptions about break submergence, and assumptions about auxiliary building failure. In general, the SOARCA study gives more credit for fission product scrubbing.
4. The SOARCA project assumes an accident termination time of 48 hours after the start of the accident (except in one Surry case that continued to 72 hours and in displaying of some results for longer timeframes in [SNL, 2013]), based on canvassing onsite and regional capabilities, and concluding that accident management personnel would be likely to flood containment and cover an ex-vessel melt by 48 hours. Implicit in this is the assumption that MCCI can be suppressed by covering the melt, even for later cavity flooding times. For the L3PRA Project, MELCOR calculations were performed to mechanistically assess the likelihood of terminating MCCI, and it was concluded that on a plant-specific basis (for the Reference Plant), MCCI was not likely to be suppressed by the overlying pool. The L3PRA Project provides results at multiple accident termination times.

References:

NRC, 2003 U.S. Nuclear Regulatory Commission, *Safety Evaluation of Topical Report WCAP-15603, Revision 1, WOG 2000 Reactor Coolant Pump Seal Leakage Model for Westinghouse PWRs*, May 2003. [ML031400376]

NRC, 2012 U.S. Nuclear Regulatory Commission, *State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*, NUREG-1935, November 2012. [ML12332A057]

SNL, 2013 Sandia National Laboratories, *State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis*, NUREG/CR-7110, Volume 2, August 2013. [ML13240A242]

SNL, 2016 Ross, K., et al., State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station, DRAFT report, Sandia National Laboratories, January 2016. ML15224A0001

18. SOARCA Surry Uncertainty Analysis, Relationship to

The L3PRA Project Level 2 PRA model leveraged the results from the State-of-the-Art Reactor Consequence Analysis (SOARCA) Surry Uncertainty Analysis (UA) (SNL, 2016). First, it is important to understand the focus of the SOARCA Surry UA, relative to the scope of the L3PRA Project, because the two projects represent very different points in the spectrum of completeness versus detail. The SOARCA Surry UA analyzes, in detail, the phenomenological modeling uncertainty associated with a seismically-induced station blackout without any form of feedwater available and including the potential of severe accident-induced (consequential) steam generator tube rupture. This means that the study considers uncertainties related to things like reactor coolant pump leakage, relief valve cycling and seizure, etc. Conversely, it means that the study doesnot consider system-related uncertainties like battery depletion, does not consider situations in which feedwater is available, and does not look more broadly at other initiating events.

This scope means that the SOARCA Surry UA has the most potential to inform the station blackout and consequential steam generator tube rupture modeling for the L3PRA Project severe accident modeling, though some insights were more broadly applicable. These insights can affect both the level of confidence in the L3PRA Project baseline modeling, as well as the level of confidence that L3PRA Project-related uncertainty activities accurately characterized the key uncertainties.

In the planning stages of both projects, there was coordination to compare the sources of uncertainty that were being considered. The draft findings of the SOARCA Surry UA became available after the reactor at-power internal events and floods Level 2 PRA MELCOR analysis was completed (for both the R01_L2 and R02_L2 models), but prior to completion of the associated L3PRA Project Level 2 PRA uncertainty characterization. As such, the outcome of the SOARCA Surry uncertainty analysis was used to retrospectively understand the potential importance of some of the L3PRA Project MELCOR baseline results, while the L3PRA Project results were used for answering questions about the significance of items not covered within the scope of the Surry SOARCA modeling, to the extent that they were generalizable from the L3PRA Project to Surry. In addition, the draft Surry SOARCA UA findings were used in developing some aspects of the uncertainty characterization for the L3PRA Project Level 2 PRA (e.g., specific parameter distributions in the PRA model, designing of specific sensitivity analyses).

As part of the overall effort, detailed qualitative and quantitative comparisons were developed to demonstrate key differences and similarities between the two project's input assignments, modeling assumptions, and results. These comparisons suggested broad similarity in the various analyses' results. Differences do exist, but they appear to be explainable based on differences in the two projects' scope and assumptions, or in differences in the subject plants' design.

In conclusion, the L3PRA Project Level 2 PRA and the Surry SOARCA uncertainty analysis are complimentary activities, and they were worked on with cognizance of the other. The results of each were beneficial in providing context for the other and helped to broadly probe at the effects of taking a more comprehensive approach (L3PRA Project) versus a very detailed approach (Surry SOARCA).

References:

SNL, 2016 Ross, K., et al., State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station, DRAFT report, Sandia National Laboratories, January 2016. ML15224A0001

19. Steamline Flooding

Scenarios that lead to flooding the steamline, including the extreme case where the section of pipe between the steam generator outlet and a closed main steam isolation valve completely fills with water,[27] could result in structural damage to the steamline. This issue was looked at from a structural perspective, focusing on whether the dead weight of a filled steamline would be likely to lead to deformation of the steamline significant enough to result in support failure, pipe damage, or containment penetration failure. It is of interest for steam generator tube rupture scenarios that might lead to overfill of a steam generator, and wherein secondary-side pipe failures can have a significant effect on the nature of the radiological release. The orientation of one steamline within containment is shown in **Figure 19-1**, and calculations were performed with ANSYS.

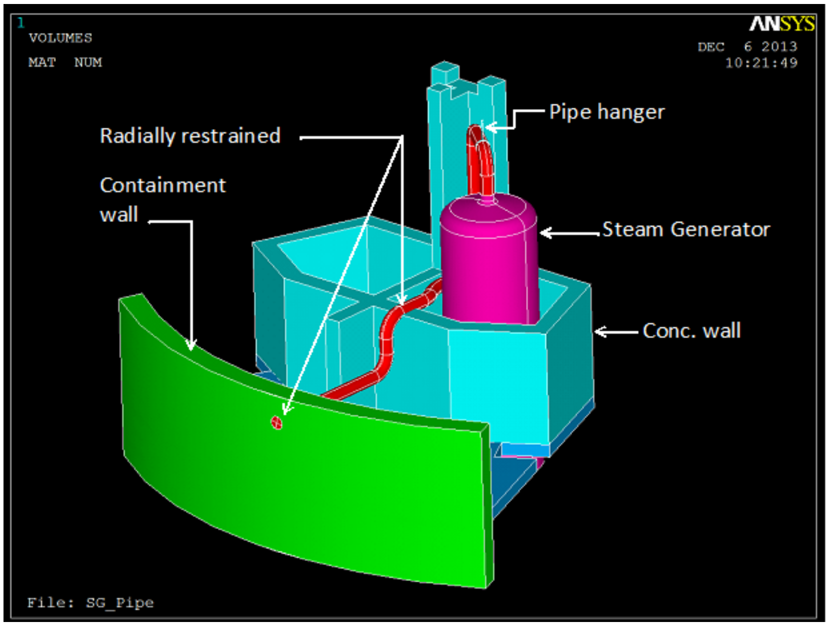


Figure 19-1: ANSYS Model Used to Investigate Steamline Flooding Effects

The dead weight of the pipe, when filled with water, was calculated to be 44.6 kips. The maximum principle stress was calculated to be about 6 ksi, which is about 34% of the maximum allowable value of 17.5 ksi. The maximum vertical deflection was estimated to be 0.9 mm (0.036 inches). It was conservatively assumed that there was no restraint at the concrete support, and with this assumption, the pipe was estimated to slide approximately 1.2 cm (0.49 in.). Based on these stresses, deformations, and translations, it is judged unlikely that any damage would occur to the steamline, its support, or its associated containment penetration.

27 Note that the discharge point for the steamline atmospheric and safety relief valves is at the elevation of the top of the SG, so a stuck-open relief valve does not in and of itself resolve the flooding concern.

20. Tendon Gallery Release Pathway

Uncertainty exists with respect to the tendon gallery as a release pathway, relative to other potential release pathways. **Table 20-1** provides an overview of the possible release pathways. If the containment fails due to very elevated static over-pressure, one of the potential failure locations is the junction at the containment basemat (i.e., the location where the horizontal and the vertical portions of containment meet). Within this 360-degree junction, structural analyses indicate that the containment may be more likely to fail near the 3 buttresses, and in a manner that would cause the through-liner/wall opening to be in an area where the containment wall is backed by the tendon gallery (rather than dirt).

The tendon gallery goes all around the containment perimeter (under the vertical containment wall). It is a 360-degree gallery under the basemat that follows the wall footprint. It is of constant cross-section. There are three access shafts to the tendon gallery. They are oriented 120-degrees from each other with the equipment hatch in between two of them, but closer to the outside one. At the bottom of each shaft, there is a small room (offset to the outside of the gallery) with two doors to the tendon gallery itself. One access shaft opens to the equipment building, adjacent (but separated by the equipment building wall) to the fuel handling building and auxiliary building (there is a wall right on the middle of the shaft opening). Another access shaft traverses (vertically) through a portion of the equipment building that is adjacent to the control building, near the personnel airlock. This area is poured concrete at all elevations, and the shaft access door is located on the control building roof. The final access shaft opens to the outside (again, near where the equipment hatch is located). This latter access has two manhole-sized openings in the grade-level deck plating causing the access shaft airspace to communicate freely with the atmosphere. No curb/cover is used that would prevent rainwater from entering the access shaft. The tendon gallery free volume is judged to be sufficiently small relative to the bulk containment free volume, such that it would significantly pressurize following containment failure, potentially failing the access shaft doors.

Table 20-1: Release Pathway Possibilities

Failure location	*Flow area fraction estimates and connections*
Containment basemat junction in to tendon gallery *[this is the failure mode modeled in the MELCOR model]*	1.0 from lower containment to tendon gallery • 0.67 to the environment; • 0.33 to the auxiliary building Note that actual flows may not follow this split, since the auxiliary building can pressurize, and the environment cannot.
Equipment hatch	1.0 from upper containment to the yard at grade elevation (Level 1)
Personnel airlock	1.0 from upper containment to the equipment building near a corridor leading in to the Control Building (Level 1)
Emergency airlock	1.0 from lower containment (Level B) to the tendon gallery access shaft that exits to the yard

The modeled failure location (and the details of the tendon gallery treatment when applicable) represent a tradeoff between maximizing direct environmental releases versus maximizing habitability and survivability concerns in the surrounding structures. The treatment was chosen as a compromise between those two, with the idea that the leakage would disperse through three pathways in relatively equal parts. The first two thirds open to the environment (FL-VELVAP.844) and final one third opens to the auxiliary building (FL-VELVAP.843).

Contrary to this expectation, analysis (see **Figure 20-1** below) shows reverse flow occurring in the path from the tendon gallery to the auxiliary building. As a result, at the time of containment overpressurization, there is not a general increase in fission products into the auxiliary building via the tendon gallery. This can be further seen from the auxiliary building retentions in **Figure 20-2**. The release to the auxiliary building is dominated by the normal leakage flowpath (primarily aqueous) from containment to the auxiliary building, for most fission products, rather than the overpressure failure flowpath (opening at 56 hours). The leveling off of the releases at 72 hours corresponds to the time when the containment water level drops below the leak pathway.

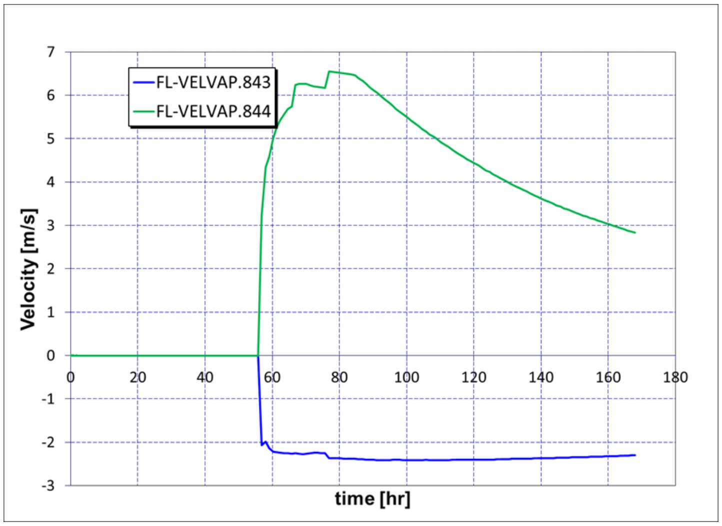


Figure 20-1: Flows for each of the containment overpressure failure pathways.

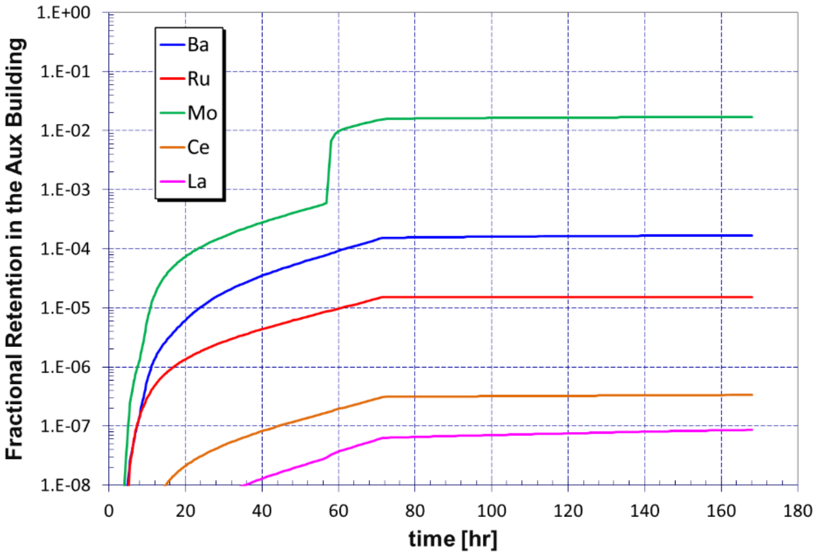
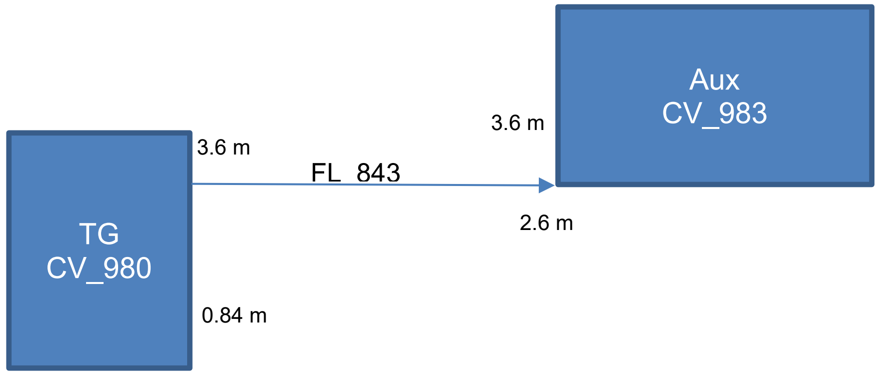


Figure 20-2: Fission Product Retentions in the Auxiliary Building

The reverse flow seen in FL-VELVAP.843 here is due to the gravitational head between the tendon gallery and auxiliary building control volumes. The dP is calculated based upon the altitude of the flowpath inlet and outlet.



Following containment failure, the pressure difference between the two control volumes remains within a few pascals[28] (**Figure 20-3**). However, MELCOR assumes the pressure is at the pool surface and, since both CVs do not fill with any significant amount of water, the pressure is taken to be at the bottom of the CV. As the figure above demonstrates, the bottom of the tendon gallery is much lower (0.84m) than that of the auxiliary building control volume (2.596m). When calculating the flow rate through FL-VELVAP.843 at a height of 3.6m, the differential pressure is calculated using the equation $P_i - P_k + (\rho g \Delta z)_j$ where P_i and P_k are the respective pressures in the "from" and "to" volumes and $(\rho g \Delta z)_j$ is the net gravitational head. It is in turn calculated using the equation.

$$(\rho g \Delta z)_j = \rho_i g \left(z_{p,i} - z_{J,i}\right) - \rho_j g \left(z_{p,k} - z_{J,k}\right)$$

In the case of 1B2, using values at 84 hours, this becomes.

$$(\rho g \Delta z)_j = 0.59g(3.6 - 0.84) - 1.16g(3.6 - 2.6)$$
$$= -4.54\, Pa$$

This negative gravitational head causes a reverse flow in the velocity equation. Using the simplified Bernoulli equation ½ $\rho k v^2 = P_i - P_k + (\rho g \Delta z)_j$, we get a reverse flow with magnitude

$$v = \sqrt{2\left(\frac{-1.1+4.54}{1.16}\right)} = 2.4\,\frac{m}{s}.$$

This is approximately the flow seen in **Figure 20-1** from the auxiliary building to the tendon gallery and exposes the cause of this counter flow seen in the simulation. From this it is important to note that, while not anticipated, MELCOR is modeling the flow as it should, given the current model and boundary conditions.

As a result, the L3PRA Project MELCOR modeling reflects that flow will preferentially go to the environment (since the auxiliary building can sustain some minimal over-pressure), and likely over-estimates the environmental releases for the late containment over-pressure failure cases.

[28] 1 Pa = 9.86923×10^{-6} atm

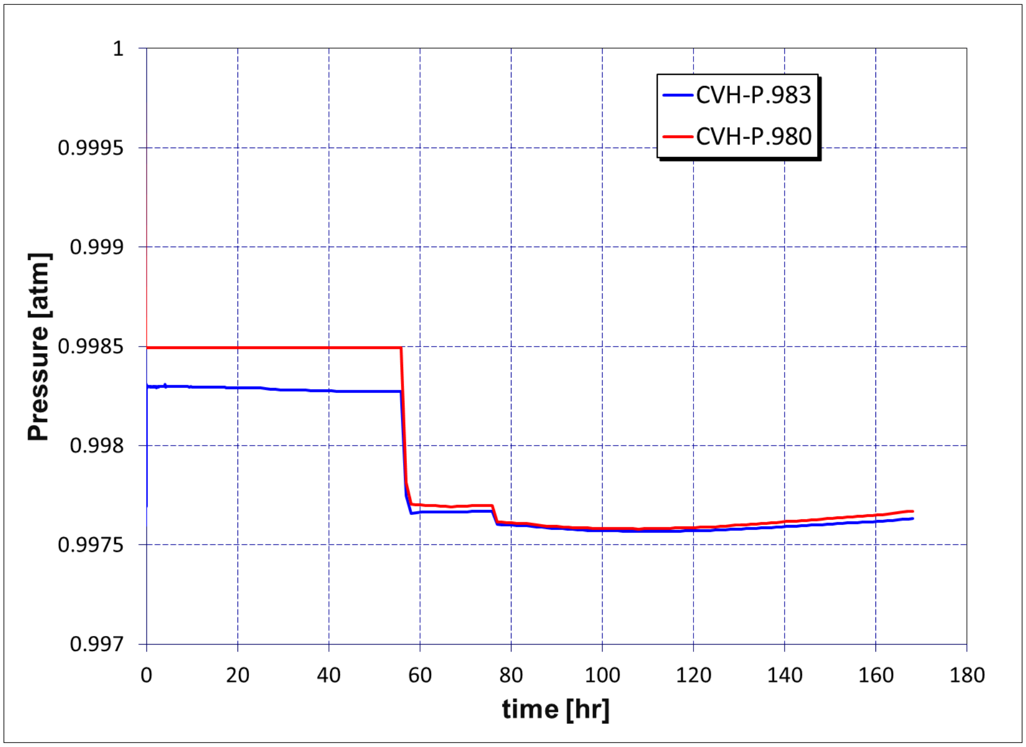


Figure 20-3: Pressures in the Tendon Gallery and the Auxiliary Building

21. Termination of Radiological Releases

Historically, PRA studies have not explicitly modeled the role of long-term onsite, or offsite, resources in terminating accidents after core damage has occurred. Underscoring this is the assumption in the L3PRA Project simplified Level 2 PRA model that all accidents resulting in vessel breach lead to eventual basemat melt-through or other containment failure. Such assumptions are becoming progressively more challenging to assert given:

- the maturity of onsite accident management and EP
- the slowness (relative to studies like NUREG-1150 [NRC, 1990]) with which accident progression proceeds for some types of scenarios
- the desire to characterize realistic accident outcomes
- the inclusion in this overall project of spent fuel pool (SFP) accidents that are slowly-evolving by nature (but may have significant personnel access constraints well prior to fuel damage)

NUREG-1935 (NRC, 2012) took the approach of canvasing the local and regional infrastructure for the two sites studied and asserting (via a few paragraphs in the report) that containment could be flooded by 48 hours, effectively terminating radiological releases (with the exception of one scenario that was carried out to 72 hours, because releases were in the midst of ramping up at 48 hours). NUREG-2161 (NRC, 2013) took a similar approach, ultimately relying on a 72-hour termination time. This issue affects the reactor and SFP Level 2 analysis and is particularly difficult as it is a hybrid of an onsite accident management, EP, and HRA issue.

The major considerations in terminating accident sequences for the reactor Level 2 PRA for this project are:

1. For station blackout scenarios (which were very significant contributors to the model):
    a. The Level 1 PRA assumes that turbine building batteries will deplete 2 hours after loss-of-AC, and in-plant safety-related batteries will deplete 4 hours after loss-of-AC. It further assumes that, following loss of the turbine building batteries, AC power cannot be restored (no power to close critical breakers).
    b. For most combinations of Level 1 PRA failures, the timing of core damage and subsequent TSC SAMG activity would be such that instrumentation will be lost prior to carrying out the post core-damage, pre-vessel breach SAMG action(s). Without instrumentation to guide detection, understanding, decisionmaking, and action, the Level 2 HRA does not have a basis to credit operator actions.
2. Only limited treatment of onsite accident management is modeled to keep the scope of work manageable, and consistent with the state-of-practice in Level 2 PRA. Carrying out sequences for extended periods of time with limited consideration of accident management may produce an unrealistically pessimistic representation of reality.
3. The role of off-site resources in supporting onsite accident management is generally not modeled.[29] This is due to the same reasons as above and results in the same concern as above.

[29] There is some consideration of offsite fire department resources being available for providing alternate capabilities to onsite resources.

4. The hierarchy of the SAMGs is focused on limiting offsite airborne radiological releases to protect public health and safety. This leads to a different set of priorities than if the hierarchy were focused on preventing basemat melt-through. Partial flooding of areas of containment other than the cavity may occur from implementing high-priority strategies (e.g., operating containment sprays in SCG-1 for reducing fission product release), but this will likely result in flooding to a similar elevation as that which would result from full injection of the RWST in to containment (e.g., during a design-basis LOCA).[30] To effectively terminate radiological releases after vessel rupture, the reactor cavity would need to be flooded. The action to deliberately flood containment to a height that results in spillover in to the cavity (roughly two RWST volumes worth of water) is SAG-8, which is the lowest priority strategy in the circa 2012 WOG SAMGs.
5. Accident progression modeling generally becomes more speculative as simulation time passes because of items such as:
    a. un-modeled low probability phenomena (e.g., detonation)
    b. uncertainty in longer-term processes (e.g., lack of modeling in core-concrete interaction of the thermal front beyond the immediate vicinity of the contact with the melt, smearing of cavity rebar and its effect on exothermic chemical reactions in core-concrete interaction)
    c. a generally weaker experimental basis for the empirical models
    d. weaknesses in the model specification that are not important earlier in the accident (e.g., thermally-induced containment seal degradation)
6. The Level 2 PRA is very limited in its treatment of equipment repair and recovery for use in accident management.
7. The Level 2 PRA relies on information from the Level 1 PRA that has its own constraining assumptions, such as:
    a. simplifications in operator response modeling (e.g., not modeling late actions that would not avert core damage)
    b. success criteria and sequence timing assumptions that envelope varying accident responses (e.g., injection criteria covering a range in break sizes)
    c. mission times when computing failure-to-run probabilities
8. Threshold effects (e.g., containment failure due to long-term over-pressurization) can occur late in an accident and affect the magnitude of radiological releases.
9. Once core melt has gone ex-vessel, it may be difficult to stop core-concrete interaction, in that (i) flooding the reactor cavity may not be a high priority, as discussed above, and (ii) PWR designs like the Reference Plant have a very tight cavity, leading to a higher debris depth, making the debris inherently harder to cool from above.

Performing a study-specific, site-specific, and scenario-specific assessment of these factors to arrive at a justified timing-based release termination criterion is beyond the state-of-practice. Instead, this issue is treated as a global model uncertainty, similar to how the selection of a

[30] Many PWR containments are designed to divert water to the ECCS sumps so that it is available for ECCS recirculation. As such, there are only certain pathways that lead to water entering the cavity, as discussed further in **Section 12** of this report. Otherwise, introducing water into the cavity can only occur by deliberately flooding containment to the point that water spills over through the loop cutouts, which requires there to be ~1.3 million gallons of water in lower containment.

dose response model is treated in some offsite consequence analyses. As such, and considering this issue's importance, this global modeling uncertainty approach is reflected in the following:

- environmental radiological release results are provided at three different accident termination times
- risk surrogates (LERF, LRF, CCFP) are presented at three different accident termination times (see Section 2.6 of the main report)
- this issue is flagged in a recommendation regarding the Level 2 / Level 3 PRA interface (see Section 2.6 of the main report)
- a model uncertainty sensitivity analysis investigated generally how reliable recovery actions would need to be in order to counteract the detrimental effects of using longer simulation end-times without accompanying recovery modeling (see Section 4.3 of Appendix C)

More background information

Initially, a more detailed treatment of accident termination time was envisioned, on the presumption that the planned evaluation would show that ex-vessel cooling was likely, or at least likely enough to significantly affect the results (and thus warrant more rigorous treatment). In actuality, ex-vessel cooling was estimated to be very unlikely. The determination of phenomenological accident termination split fractions (i.e., probability that cavity flooding halts core-concrete interaction) was based on deterministic (MELCOR) calculations and empirical experience, that resulted in a low subjective probability (0.1) that flooding of the cavity would halt significant additional MCCI and avert eventual basemat melt-through. This decreased the benefit of parsing sequences more finely in this regard. As such, the remainder of this section retains details that would be helpful if this were to be addressed in the future.

Regarding the anchoring around SAMG entrance, it is acknowledged that there are other suitable anchor points (e.g., activation of the Emergency Operations Facility or declaration of a General Emergency). The use here of SAMG entrance is intended to denote the point where the comprehensive (onsite and offsite) response to the accident has fundamentally changed from preventing core melt to terminating an imminent or ongoing radiological release. More specifically, a notional timeline supporting termination at 36 hours after SAMG entry is provided in **Table 21-1** below. Similar timelines could be portrayed supporting other termination times, and this timeline is only intended to be illustrative.

Table 21-1: Notional accident management phases

Timing	Focus	Notes
Prior to SAMG entry	Execute EOPs; prevent core damage	Level 1 PRA
0 to 6 hours after SAMG entry	Execute SAMGs as is; prevent vessel breach and minimize fission product release	Pre-vessel rupture modeled action
6 to 12 hours after SAMG entry	Execute SAMGs as is; minimize fission product release	Post-vessel rupture modeled action

Table 21-1: Notional accident management phases

Timing	Focus	Notes
12 to 18 hours after SAMG entry	Focus transition to lower-priority SAMGs on flooding containment to terminate core-concrete interaction	1. Consideration of installed and portable onsite equipment 2. Assessment of viability, including habitability and equipment/instrument survivability 3. Assessment of the positives of the action relative to the specific negatives identified in the SAMGs (e.g., causing a deflagration or detonation, loss of equipment and instrumentation in lower containment) 4. Reaching out for offsite support as needed
18 to 24 hours after SAMG entry	Stage equipment for flooding containment	5. Establish prolonged suction source, including refill capability as necessary 6. Establish pumping capability (unless RWST gravity drain is used) 7. Establish necessary hoses and/or installed-pipe alignment (these can be complex, and can require accessing locations that will be difficult to access during an ongoing accident)
24 to 36 hours after SAMG entry	Flooding of containment starting at 24 hours, leading to first introduction of new water to cavity at 36 hours*	8. Initiate injection 9. Monitor response 10. Adjust as necessary to attain desired injection rate and to react to changing plant conditions (such as rising containment pressure or combustion)

* The time before water first enters the cavity after injection into containment begins can vary widely, as shown in the corresponding **Table 21-2** below (0 to 72 hours); a value of 36 hours following the onset of injection is selected here as an illustration.

The timing for the implementation of the containment flooding strategy (leading to intrusion of water into the cavity) can vary widely as illustrated in **Table 21-2**. Note that for many scenarios the results are not expected to be very sensitive to the timing assumptions, so long as the diagnosis and implementation period falls within the timeframe between vessel failure and long-term containment failure due to static over-pressure or basemat melt-through. The MELCOR analysis supporting the L3PRA Project Level 2 PRA indicates that late containment failure is likely to happen between 36-60 hours if long-term containment heat removal is unavailable, or much later otherwise. Sequences where large global deflagrations or detonations are predicted to fail the containment during this timeframe are a key exception. Note that flooding of the lower containment has competing effects on rising containment pressure, prior to water spilling over into the cavity (it may help to cool containment and condense steam, while it will also reduce the free volume).

Table 21-2: Summary of containment flooding options

<table>
<tr><th colspan="2" rowspan="2">Strategy (based on SAG-8 and EDMG Appendix C)</th><th rowspan="2">Comments on implementation</th><th colspan="2">Sample time before water enters cavity (hrs)***</th></tr>
<tr><th>Previously dry lower containment</th><th>Equivalent of one RWST already in containment</th></tr>
<tr><td colspan="2">Notional timings</td><td>6 hours</td><td colspan="2">18 hours</td></tr>
<tr><td rowspan="2">Containment spray pumps</td><td>1 train</td><td rowspan="2">• Undamaged pump(s), ac and dc power available
• System procedure conditions met (e.g., NSCW)
• Align per EOP (4 valves)
• Will require RWST fill/refill</td><td>7</td><td>4</td></tr>
<tr><td>2 trains</td><td>4</td><td>2</td></tr>
<tr><td rowspan="2">Containment spray using EDMG pump (NMP-EP-404, Appendix C)</td><td>300 gpm (value cited in guidance)</td><td rowspan="2">• Staging of equipment from storage
• Suction from fire water storage tanks (FWSTs) in yard and discharge through air test connections of cont. sprays (8-step connection process)
• Setup/operation of trailer-mounted pump (20-step process)
• Requires access to A level of Aux. and/or Fuel Handling Building
• Will require FWST/demineralized water storage tank fill/refill</td><td>72</td><td>36</td></tr>
<tr><td>1000 gpm****</td><td>22</td><td>11</td></tr>
<tr><td>Containment spray using Offsite Fire Department pumping truck</td><td>1500 gpm</td><td>• Includes request and transit time
• Utilize connection strategy above, but with pumper truck as source
• Will require drafting (and potential refill) from FWST, NSCW basin, or another source</td><td>14</td><td>7</td></tr>
</table>

Table 21-2: Summary of containment flooding options

<table>
<tr><th colspan="2" rowspan="2">Strategy (based on SAG-8 and EDMG Appendix C)</th><th rowspan="2">Comments on implementation</th><th colspan="2">Sample time before water enters cavity (hrs)***</th></tr>
<tr><th>Previously dry lower containment</th><th>Equivalent of one RWST already in containment</th></tr>
<tr><td>Gravity drain to containment spray suction lines</td><td></td><td rowspan="4">Supply valve alignment per Abnormal Operating Procedure for Loss of RHR during shutdown
Injection path valve alignment using SACRG-1, AOP or EOP
Will require RWST refill; some sources of RWST refill would constrain injection rate; specific refill valve alignments not provided in SAMGs
Requires containment pressure to stay below ~35 psig</td><td>4 to 36*</td><td>2 to 18*</td></tr>
<tr><td>Gravity drain to intact RCS (other than lower head vessel failure)</td><td></td><td>0</td><td>0</td></tr>
<tr><td rowspan="2">Gravity drain to RCS broken outside of the cavity wall (e.g., RCS seal)</td><td>half of water going to vessel</td><td>0**</td><td>0**</td></tr>
<tr><td>no water going to vessel</td><td>4 to 36*</td><td>2 to 18*</td></tr>
</table>

* A theoretical range of 0 to 6,000 gpm exists based on CA-6 (actual injection rates are dependent on RWST level relative to containment water level, injection pathway, and containment pressure). Here, the timing range is based on the 10^{th} to 90^{th} percentile of the flow rate range assuming a uniform distribution, which is 600 to 5,400 gpm.

** Note that at the lower end of the injection range this split may not provide adequate water to the cavity to cool the core; nevertheless, this is not addressed explicitly here.

*** Consistent with the modeling of cavity water intrusion elsewhere in the Level 2 PRA, these values neglect the potential for earlier water intrusion through the vessel flange ventilation ports, the ex-core neutron detector positioning rods, etc., as discussed at length in Section 12 of this report.

**** A range is used that encompasses the minimum flow rate of 300 gpm cited in NMP-EP-404 and the theoretical maximum flow rate of 1,000 gpm for the EDMG pump.

At a generic (rather than scenario-specific) level, applying the screening HEP criteria used elsewhere in the Level 2 HRA arguably results in an HEP of 0.5. This is based on an argument that neither the high (0.9) nor low (0.1) HEP criteria apply. For the criteria related to an HEP = 0.9: (i) SAG-8 would eventually become the highest priority once all other (SCG-1 through SAG-7) relevant actions have been pursued and (ii) the likely accident-altering event to occur would be a combustion event induced by de-inerting of containment, which would likely occur after completion of the action setup. For the criteria related to an HEP = 0.1, containment flooding is not similar to an action in the EOPs and survivability/habitability may be of concern. Many of the screening HRA criteria do not directly apply at a generic level, because they require knowledge about the status of DC power, prior human failures, etc.

This broad issue (of accident termination time selection) was raised early in the project and received much attention during discussions with various internal stakeholders and reviewers during the formulation and review of the initial Level 2 PRA model in 2014. It was generally agreed that the following actions could improve the realism and utility of the modeling, should future opportunities permit additional work in this area:

- explicit consideration (perhaps via task simulation) of the involvement of off-site resources
- more rigorous treatment of the human performance aspects using the Level 2 HRA model
- development of scenario-specific timing estimates for flooding containment
- explicit treatment of variability in timing
- greater acknowledgment of the speculative nature of the source terms, given uncertainties in the underlying deterministic model and limitations of the underlying human reliability analysis

References:

NRC, 1990 US Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,* NUREG-1150, October 1990.

NRC, 2012 U.S. Nuclear Regulatory Commission, *State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*, NUREG-1935, November 2012.

NRC, 2013 U.S. Nuclear Regulatory Commission, *Consequence Study of a Beyond-Design Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor*, NUREG-2161, October 2013. [ML12332A057]

Appendix E: Phenomenological Issues Regarding Treatment of Containment Failure or Bypass Events in the Level 2 Reactor, At-Power, Internal Event and Flood PRA

1. Purpose

This document describes some of the important phenomenological issues that affect containment failure or bypass considered in the Level 3 Probabilistic Risk Assessment (PRA) Project (L3PRA Project), particularly those events listed in Table 4.5-8 of the Level 2 Trial Use and Pilot Application probabilistic risk assessment (PRA) Standard (ASME, 2014). This report also explains briefly how the L3PRA Project internal events/floods model addresses these containment failure and bypass events. Finally, this document identifies several sources that provide more information about the severe accident phenomena that are summarized here.

2. Containment Isolation Failure

Containment isolation failure is included as part of the bridge tree between the Level 1 event tree and the containment event tree. Only "large" containment isolation failures are considered in the model, where a large containment isolation failure is defined as i) containment pressure cannot increase significantly above atmospheric conditions, and ii) there is negligible residence time for radionuclides in containment. Based on MELCOR code calculations, the cutoff isolation failure size is a two-inch equivalent diameter hole, except pre-existing failures which use a smaller size, as discussed in Section 6 of Appendix D.

Containment isolation failure is considered in one of the eight representative sequences for deterministic analysis in the 7-series cases (see Section 7 of Appendix B).

3. Interfacing System Loss of Coolant Accident

Interfacing system loss of coolant accidents (ISLOCA) as an initiator is considered in the Level 1 and Level 2 PRA models. From the Level 1 model (R02), ISLOCA as an initiator accounts for ~0.5% of the total core damage frequency (CDF) for L3PRA Project (NRC, 2022), and the highest-contributing pathway is the residual heat removal (RHR) hot leg suction line. Even though its contribution to CDF is very low, ISLOCA is a concern because it bypasses containment, which may lead to very large fission product releases to the environment.

Because this is a containment bypass event, there is the potential for a large, early release of radioactive material to the environment. The timing and extent of the release are affected by several assumptions used to define the accident sequence. For example, the break size affects the time to core uncovery and thus the time of the release. The break location affects whether the break would be submerged by an overlying water pool in the Auxiliary Building, which in turn affects the magnitude of the release. Assumptions about building integrity during the initial blowdown or following potential hydrogen combustion events, and assumptions about the operability of the piping penetration area filtration and exhaust system (PPAFES), also affect the magnitude of the release.

Several MELCOR calculations have been performed to analyze the effects of the above uncertainties and to provide source terms for the ISLOCA release categories (i.e. V, V-F, and V-F-SC). The calculations are described in detail in Section 5 of Appendix B.

4. Steam Generator Tube Rupture

Steam generator tube rupture (SGTR) as an initiator is considered in the Level 1 and Level 2 PRA models. From the Level 1 model (R02), SGTR as an initiator accounts for ~0.2% of the total core damage frequency (CDF) for the L3PRA Project (NRC, 2022). Even though its contribution to CDF is very low, SGTR is a concern because it bypasses containment, which may lead to very large fission product releases to the environment.

One important uncertainty in analyzing SGTR is the number of tubes that fail. If leakage is less than the effective cross-sectional area of one tube, the primary system loses coolant at a relatively slow rate. Normal charging is sufficient to make up losses from the primary system, so SGTR will not proceed to core damage unless there are additional independent system failures (e.g. failure of normal charging and emergency core cooling systems or loss of feedwater). Even without injection to the primary system, the accident proceeds very slowly, such that core damage is not expected until tens of hours after accident initiation.

If a larger rupture occurs, then the primary system would lose coolant at a more rapid rate, though the leakage rate would likely still be within the capabilities of high pressure injection systems (but recirculation would not be possible since leakage would be leaving containment). If injection is unavailable, the accident would progress to core damage more quickly.

Rupture of a single steam generator tube (or leakage from multiple tubes summing to the equivalent) with failure of high pressure injection is one of the eight representative scenarios for which deterministic accident progression analysis (i.e. MELCOR calculations) has been performed (see Section 8 of Appendix B).

5. Induced SGTR

Induced steam generator tube rupture, or consequential steam generator tube rupture (C-SGTR), refers to the failure of one or more steam generator tubes under accident conditions. C-SGTR can occur because of high primary-to-secondary differential pressure and/or high temperatures during accident sequences. Cases involving creep damage to the tubes from pressure and temperature effects are commonly referred to as thermally-induced SGTR (TI-SGTR), while cases where tube damage is caused solely by pressure differential are referred to as pressure-induced SGTR (PI-SGTR).

Situations where the primary-side pressure is high, the steam generators are dry, and secondary side pressure is low (a.k.a., high-dry-low conditions) can lead to TI-SGTR. Situations where primary-side pressure is high and secondary-side pressure is low (e.g., anticipated transient without scram, secondary-side breaks) can lead to PI-SGTR. Both are potentially risk significant because they result in a containment bypass event that can lead to large release of fission products to the environment.

5.1 TI-SGTR

There are several factors that affect whether the steam generator tubes will fail due to creep rupture or whether another location (e.g. hot leg nozzle, surge line nozzle, reactor pressure vessel [RPV] lower head) will fail first. These factors are discussed in Section 5 of Appendix D, which summarizes the approach that has been adopted for the L3PRA Project and how that approach relates to recent Nuclear Regulatory Commission (NRC) consequential steam generator tube rupture analysis described in (NRC, 2017).

The L3 PRA Project considers induced SGTR as part of the 1-L2-DET-CONTVE decomposition event tree in the Level 2 PRA model. This event tree identifies those sequences that are susceptible to TI-SGTR, based on Level 1 PRA information and early post-core damage actions/events. The conditional probability of TI-SGTR for high-dry-low situations is ~2% for the L3PRA Project.

Regarding the reasonableness of the above estimate (2%), it is very similar to the 1.3% value cited in (NRC, 2017) for Westinghouse plants with the tube material used in the Reference Plant for the L3PRA Project (Inconel 600). It is also within the range of conditional probabilities used in the Reference Plant's Level 2 PRA for "pristine tubes" (see below for more explanation of why this is the applicable case for the L3PRA Project). Given this, the value used in the NRC Level 2 PRA is believed to be reasonable.

5.2 PI-SGTR

PI-SGTR was added to the Level 1 PRA model prior to finalizing the Level 2 PRA in 2017. Thus, PI-SGTR occurring prior to core damage is modeled in the Level 1 PRA and is therefore carried through the Level 1 / Level 2 PRA interface. PI-SGTR accounts for ~1% of the current CDF (NRC, 2022), and this contribution is spread across the many initiators that can lead to PI-SGTR. In the L3PRA Project Level 2 PRA, PI-SGTR occurring prior to core damage is carried through similarly to SGTR initiators, ultimately being categorized in the SGTR-C, SGTR-O, or SGTR-O-SC release categories.

5.3 Source Terms for Induced SGTR

Note that MELCOR calculations have been performed to provide source terms for C-SGTR scenarios; these calculations are described in Appendix B, Section 3 for TI-SGTR and Section 8 for SGTR initiator as a surrogate for PI-SGTR cases.

6. Induced ISLOCA

An induced ISLOCA refers to the failure of an interfacing system during accident conditions. An induced ISLOCA can occur due to a high differential pressure or a high temperature in the reactor coolant system (RCS). The components given in **Table 6-1** are the first valves upstream of the RCS which, if subsequent failures were to occur, may lead to an ISLOCA. Note that Steam Generator Tube Ruptures (SGTR) are handled explicitly in the model elsewhere (see Section 5).

Table 6-1: First upstream components of the RCS for given induced ISLOCA pathway

	Component Name	Component Type	Pipe Diam	Relative Location	Induced ISLOCA path (assuming subsequent failures)
1	CV 083 (084,085,086)	Check Valve	10"	Between the cold leg injection sight and the Accumulator/ safety injection (SI)/RHR; typically, the first is a few feet from the cold leg and the other is near the containment wall	• RHR cold leg injection • SI cold leg injection path
2	CV 126 (125)	Check Valve	6"	Between the hot leg injection sight and the SI/RHR piping junction; typically, the first is a few feet from the hot leg and the other is near the containment wall	• RHR hot leg injection • SI hot leg 1 or 4 injection path

Table 6-1: First upstream components of the RCS for given induced ISLOCA pathway

	Component Name	Component Type	Pipe Diam	Relative Location	Induced ISLOCA path (assuming subsequent failures)
3	MOV HV8701B (MOV HV8702B)	Motor Operated Valve	12”	Between the hot leg suction sight and containment boundary; typically, the first is a few feet from the hot leg and the other is near the containment wall	• RHR suction from hot legs
4	CV 124 (127)	Check Valve	6”	Should be relatively the same location as CV 126	• SI hot leg 2 or 3 injection path
5	Thermal barrier heat exchanger tube rupture of reactor coolant pump (RCP) 1,2,3, or 4	Thermal Barrier heat exchanger tube	N/A	In the RCP	• Auxiliary component cooling water to RCP thermal barrier heat exchanger
6	HV8095A	Solenoid Operated Globe Valve	1”	On top of the reactor vessel head	• RCS vent path A
7	HV8095B	Solenoid Operated Globe Valve	1”	On top of the reactor vessel head	• RCS vent path B
8	HV8154	Air Operated Globe Valve	1”	Downstream of loop 4 Intermediate Leg, excess letdown, typically near the containment wall	• Excess letdown
9	HV8141A	Air Operated Globe Valve	0.75”	Between the RCP and the containment boundary, typically, near the containment wall	• RCP seal leak off lines
10	CV 437 (438,439,440)	Check Valve	1”	Between the RCP and MOV8103A, typically, near the containment wall	• RCP seal injection line
11	CV026 (027,028,029)	Check Valve	1.5”	Downstream of the boron injection tank injection line; typically, the first is a few feet from the hot leg and the other is near the containment wall	• Bit injection line
12	CV036	Check Valve	3”	Between the Normal Charging injection sight and the regenerative heat exchanger; typically, a few feet from the cold leg	• Normal charging line
13	CV038	Check Valve	3”	Between the Alternate Charging injection sight and the regenerative heat exchanger; typically, a few feet from the cold leg	• Alternate charging line

Table 6-1: First upstream components of the RCS for given induced ISLOCA pathway

	Component Name	Component Type	Pipe Diam	Relative Location	Induced ISLOCA path (assuming subsequent failures)
14	CV033	Check Valve	3"	Between the pressurizer auxiliary spray and the regenerative heat exchanger	• Auxiliary spray line

To determine the possible thermal-hydraulic conditions these components may experience during a severe accident, a survey was made of results from the MELCOR calculations described in Section 2 of Appendix B. MELCOR does not directly model these interfacing systems. However, in a case where the pressurizer valves do not cycle, the pressurizer surge line serves a good approximation of the gas temperature gradient in a branch pipe that dead-ends in a closed valve.

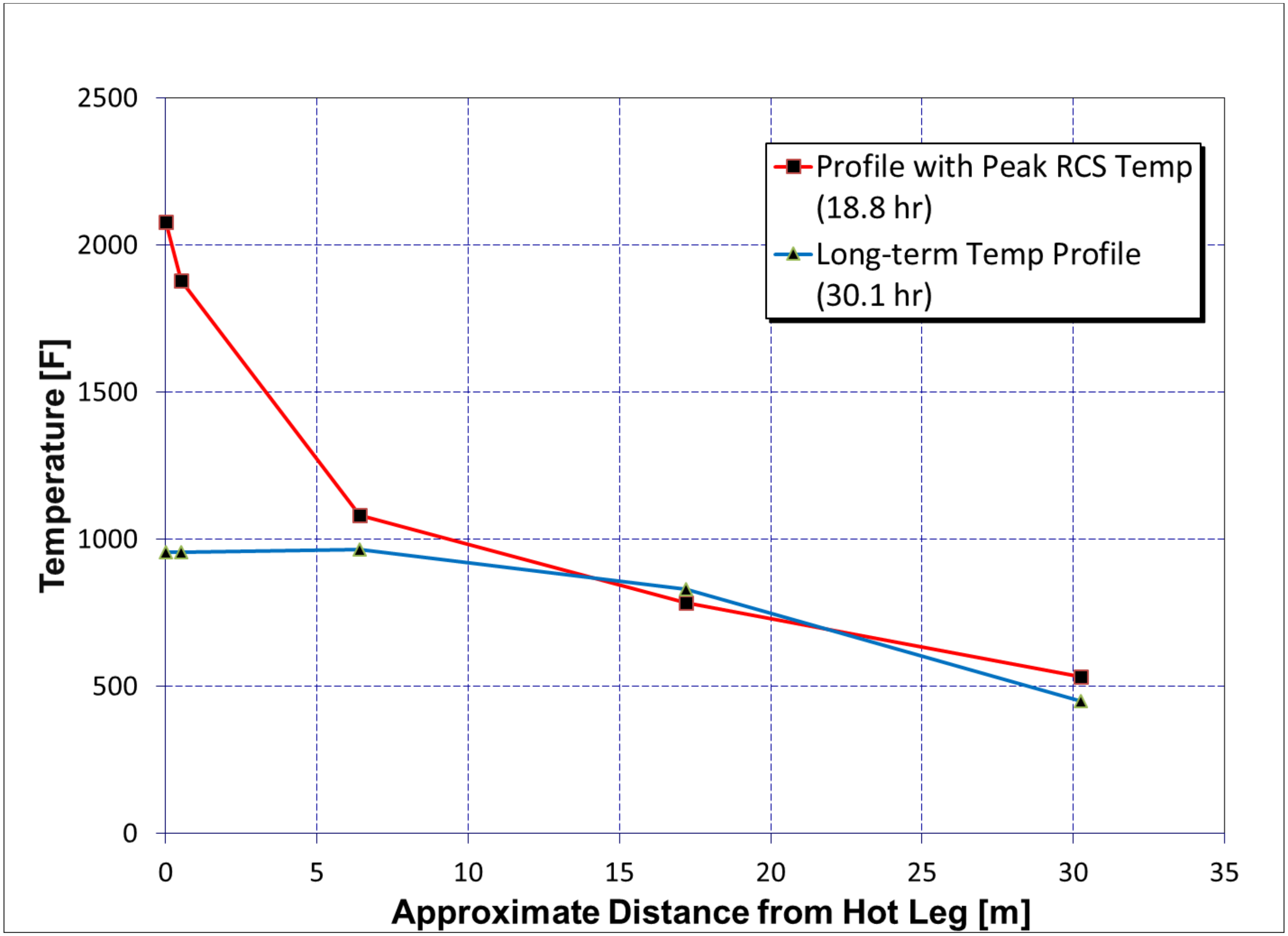
The 14" surge line is broken up into three control volumes (CV 302, 303, and 304) of lengths 1.0, 10.8, and 10.8 meters respectively. The pressurizer itself (CV 305) is ~15.3 m tall and the vertical distance between the hot leg and the bottom of the pressurizer is ~2.7 m.

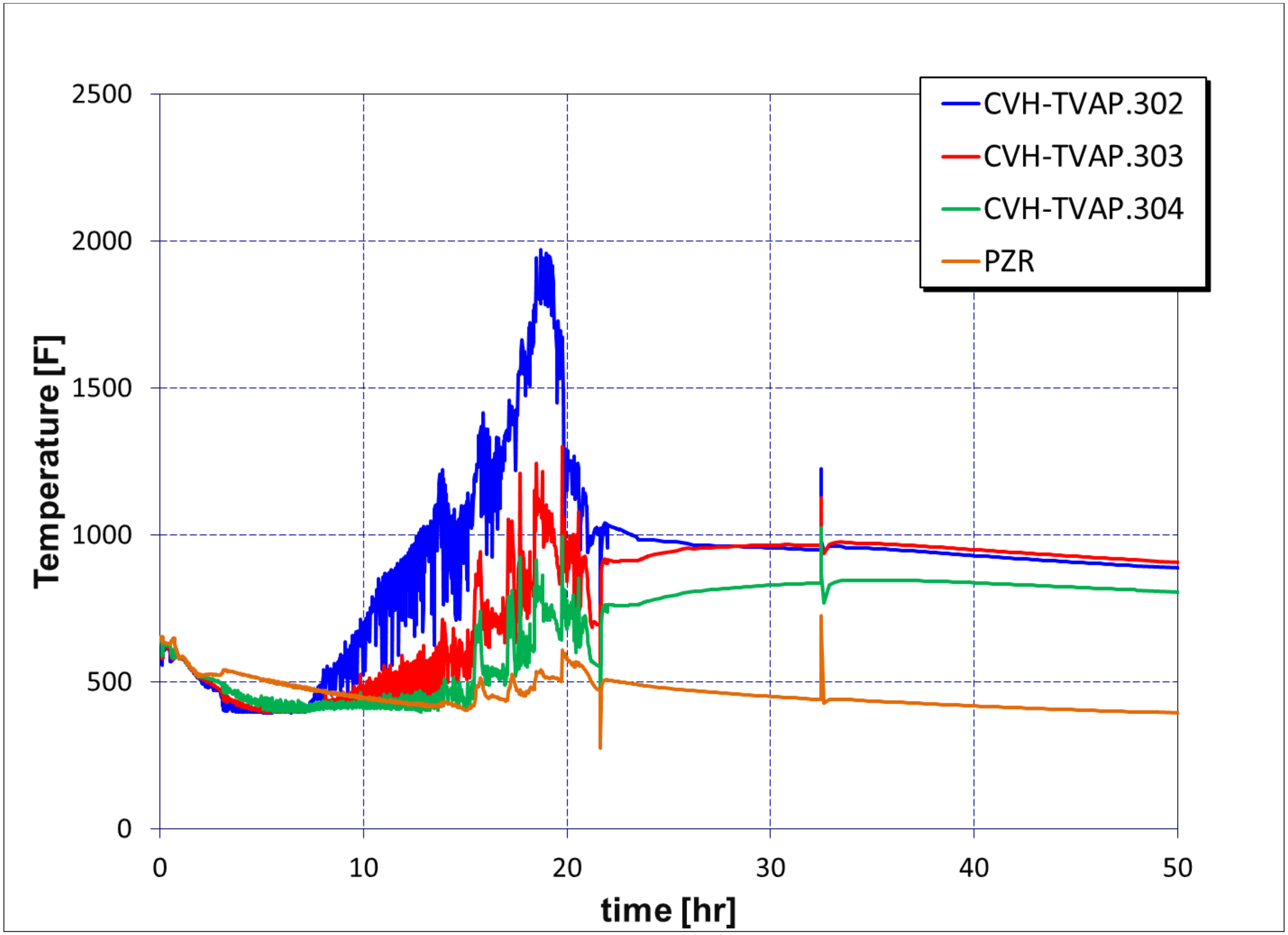
For Case 2 (Section 2 of Appendix B), **Table 6-2** provides the average, peak and lowest temperature attained in each of these control volumes after the start of core uncovery. The distances reflect the midpoints of each control volume. **Figure 6-1** provides the overall temperature profile of the surge line at 18.8 and 32.5 hours when the RCS and pressurizer reach their respective maximum temperatures. As seen from **Figure 6-2**, the actual temperatures in the line fluctuate greatly.

Table 6-2: Average temperatures in the hot leg, surge line, and pressurizer in Case 2

	Hot leg	**Surge line**			**Pressurizer**
	CV 310	CV 302	CV 303	CV 304	CV 305
Average Temperature[1] (°F)	1015	979	731	612	461
Peak Temperature[1] (°F)	2079	1971	1302	1027	726
Minimum Temperature[1] (°F)	389	391	382	323	274

[1] Taken from the start of core uncovery to the end of the simulation.


Profile with Peak RCS Temp (18.8 hr)
Long-term Temp Profile (30.1 hr)
Temperature [F]
Approximate Distance from Hot Leg [m]
2500
2000
1500
1000
500
0
0
5
10
15
20
25
30
35


2500
2000
1500
1000
500
0
0
10
20
30
40
50
Temperature [F]
time [hr]
CVH-TVAP.302
CVH-TVAP.303
CVH-TVAP.304
PZR

A vapor temperature gradient from the inlet to the pressurizer of 15.2°F/m can be matched to these results, though the initial drop-off near the hot leg is much larger than this average. Care should be taken in comparing results from the surge line to another branching line. Factors such as pipe diameter, length, and elevation change can greatly affect the heat transfer within a pipe.

The paths identified in **Table 6-1** are divided into four groups based upon their relative proximity to the RCS and the maximal temperatures the associated valve could experience. For each, a representative heat structure temperature history from Case 2 is provided.

1. The valve is on a part of the system that typically sees much lower temperatures (even during accident conditions) (items 1, 5, 12, and 13). Representative temperature is taken from the cold leg segment adjacent to the RCS (HS 34801 off of CV 348).
2. The valve is on a part of the system that is a few feet from the hot leg and will likely see relatively higher temperatures (items 2, 3, 4, and 11). Representative temperature is taken from the surge line segment adjacent to the hot leg (HS 30201 off of CV 302).
3. The valve is relatively distant from the RCS (items 8, 9, 10, and 14). The temperature is conservatively taken to be from the top of the pressurizer (HS 30503 off of CV 305).
4. The valve is on top of the vessel head (items 6 and 7). The representative temperature is taken to be that of the vessel head (HS 19501 off of CV 195).[1]

Figure 6-3, **Figure 6-4**, **Figure 6-5**, and **Figure 6-6** provide the temperature profile of each of these groups for Case 2. Between each of the categories, the pressure is not significantly different so a single representative pressure is provided in **Figure 6-7**.

A look at these figures suggests that, based on the somewhat limited characterizations of the thermal-hydraulic conditions, temperatures and pressures are approaching the ranges associated with creep damage but are not extremely challenged in this regard. Adding confidence to this conclusion is that the MELCOR analysis used surge line creep failure, but it is not predicted in any of the cases (unlike hot leg nozzle creep rupture). In this regard, the valves listed above would arguable not be expected to fail by creep rupture, since they experience similar pressures and generally lower temperatures, albeit with different material and structural properties. Thus, the more relevant failure mode would be deformation or melting of the valve components. Note that the treatment of valve leakage prior to core damage is considered in the ISLOCA initiating event portion of the Level 1 PRA and not considered here.

Case 2 was selected for the preceding discussion because of the lack of pressurizer valve cycling, which allowed an approximation of the surge lie as a dead-ended branch pipe. However, Case 2 also serves as a reasonable representative of the general thermal-hydraulic conditions in the RCS during a severe accident. **Figure 6-8** and **Figure 6-9** compare the surge line structural temperatures and pressures for a selection of other accident simulations. While scenarios S1A and S3 (see Section 1.1 and Section 3 of Appendix B) show higher surge line pressures with increasing temperature, the creep failure of the hot leg nozzles precludes surge line creep failure (as previously discussed). Scenarios S2 and S4 (see Section 2 and Section 4 of Appendix B) has a relatively higher peak and sustained temperature causing more long-term stress on the heat structures.

[1] This is a conservative choice, since the valve is located on a pipe that branches off the vessel head and is relatively stagnant during the accident. As demonstrated in the **Figure 6-1** and **Figure 6-2**, the temperature should drop significantly from the inlet to the valve.

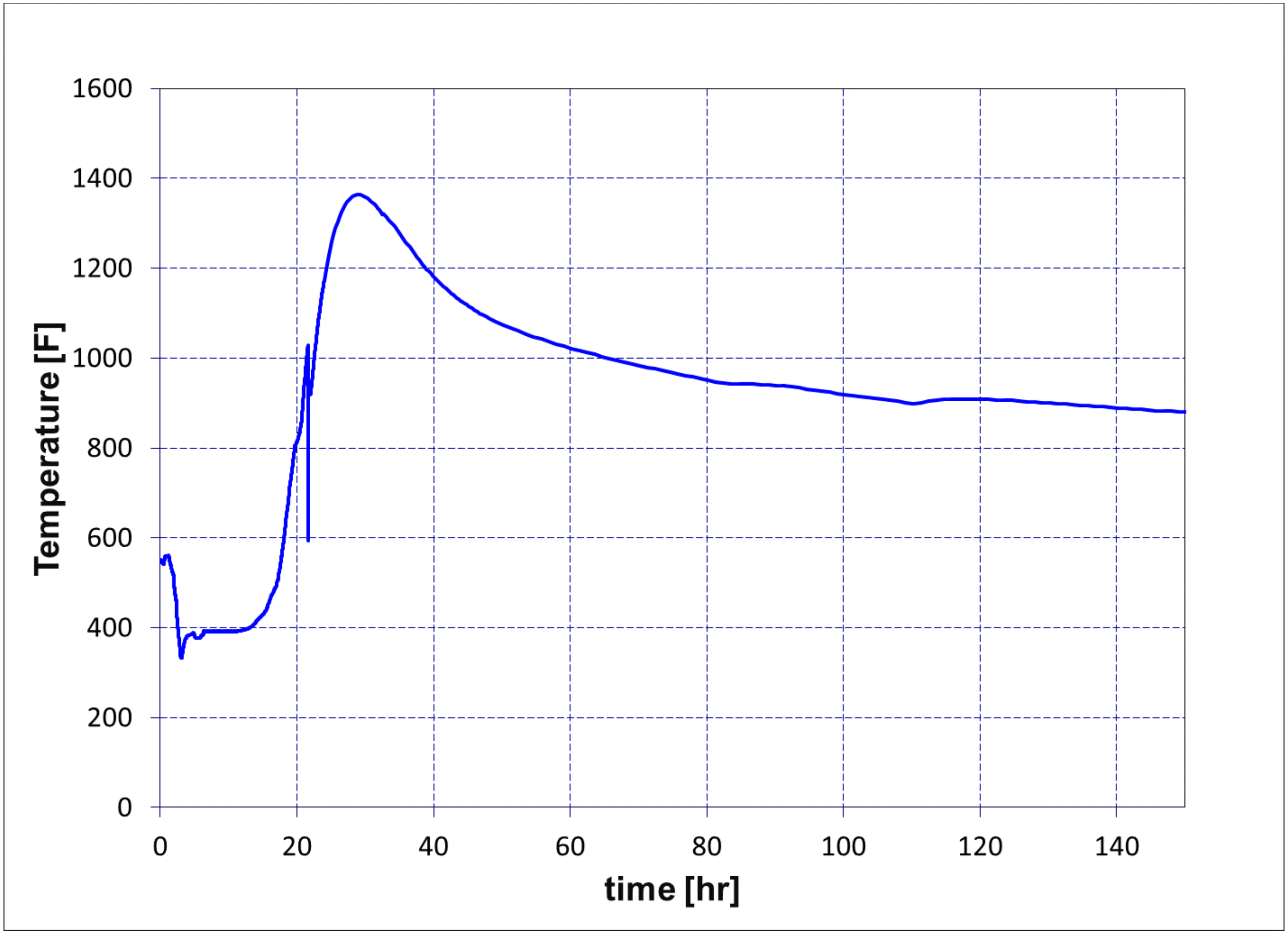


Figure 6-3: Heat Structure Temperature of the Cold Leg - Representative of Temperature Experienced by the "Category 1" Valves Branching Off the Cold Leg

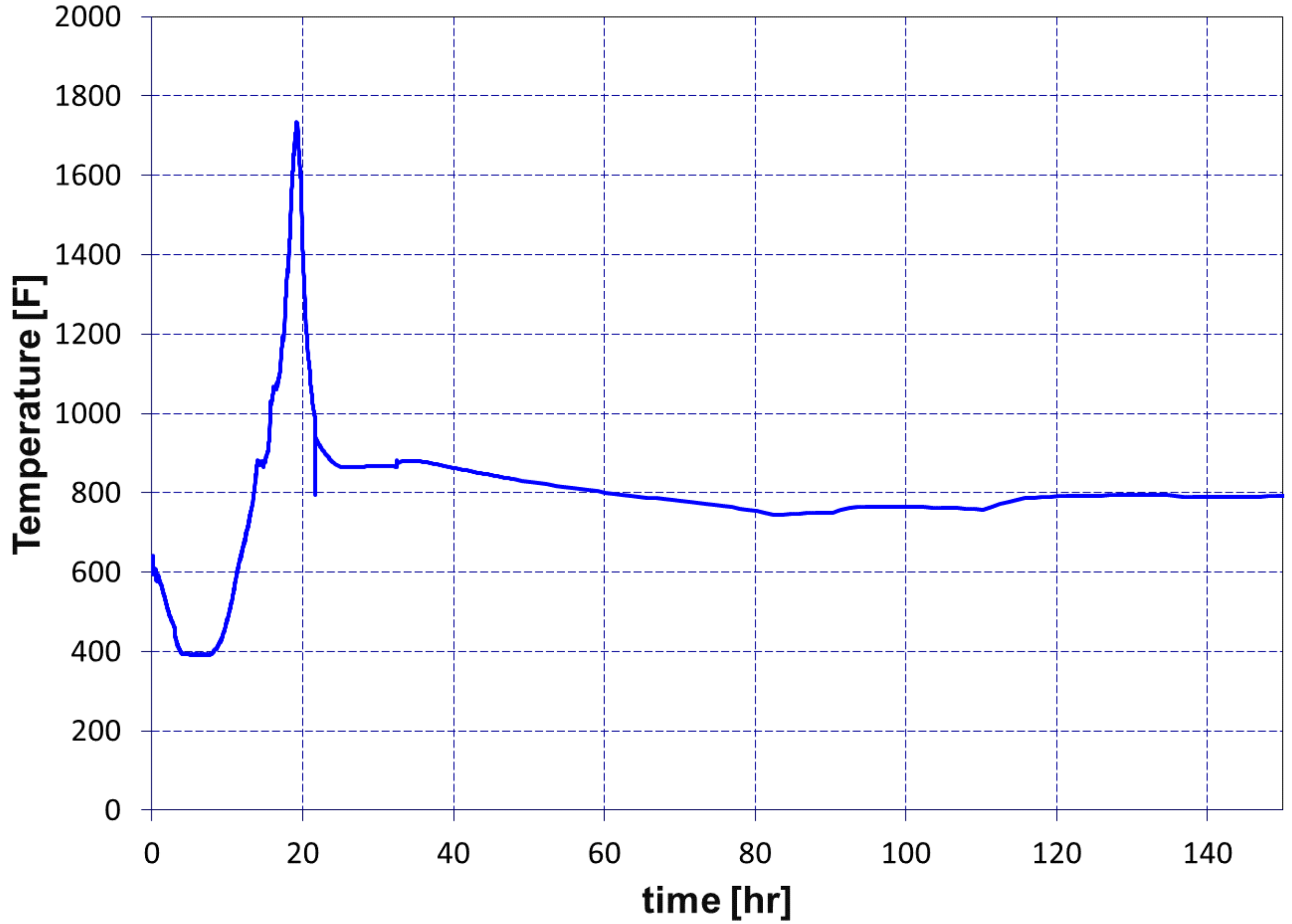


Figure 6-4: Heat Structure Temperature of the First Segment of the Pressurizer Surge Line - Representative of Temperature Experienced by the "Category 2" Valves Branching Off the Hot Leg

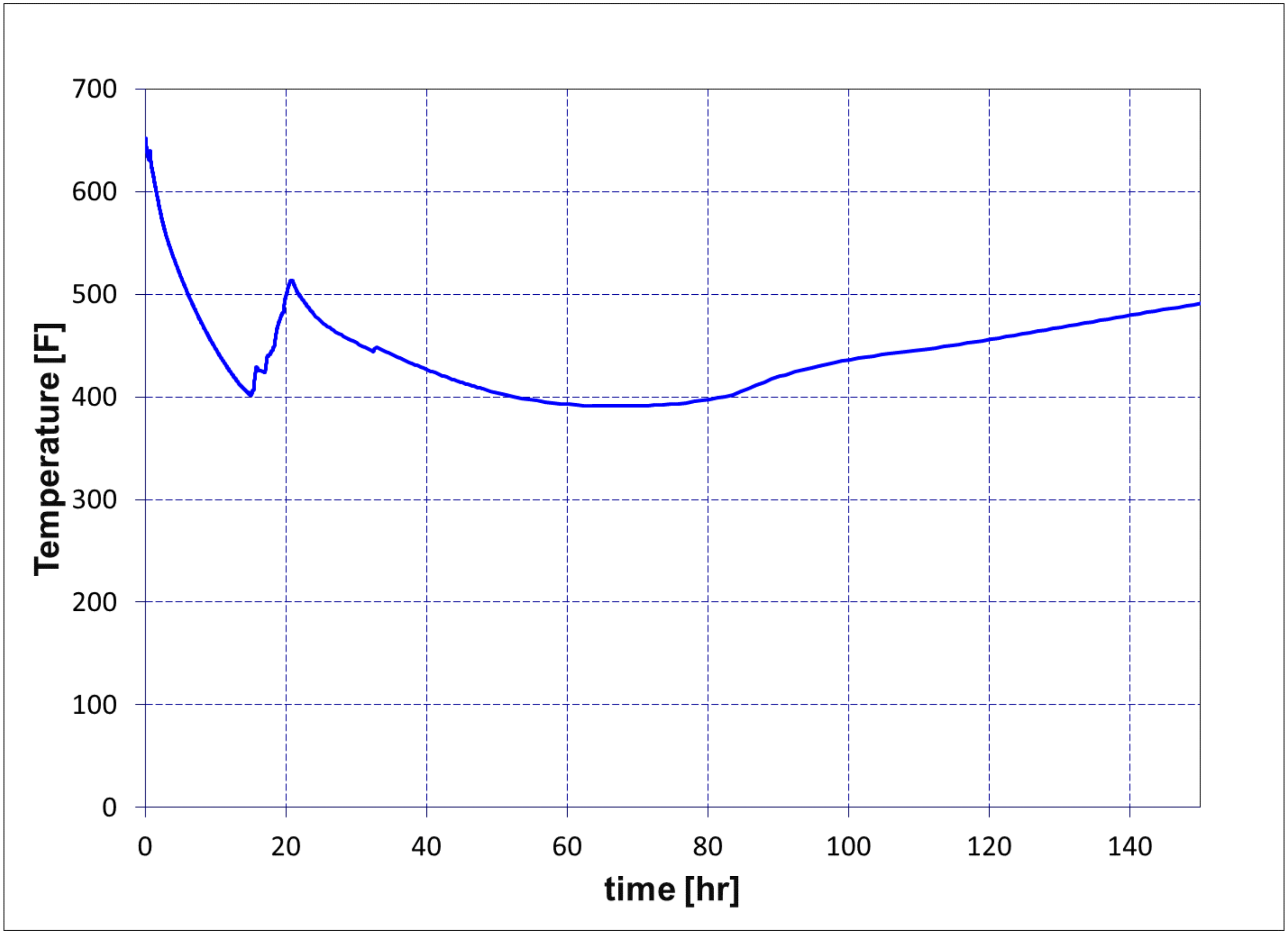


Figure 6-5: Heat Structure Temperature of the Top of the Pressurizer - Representative of Temperature Experienced by the "Category 3" Valves Distant From the RCS

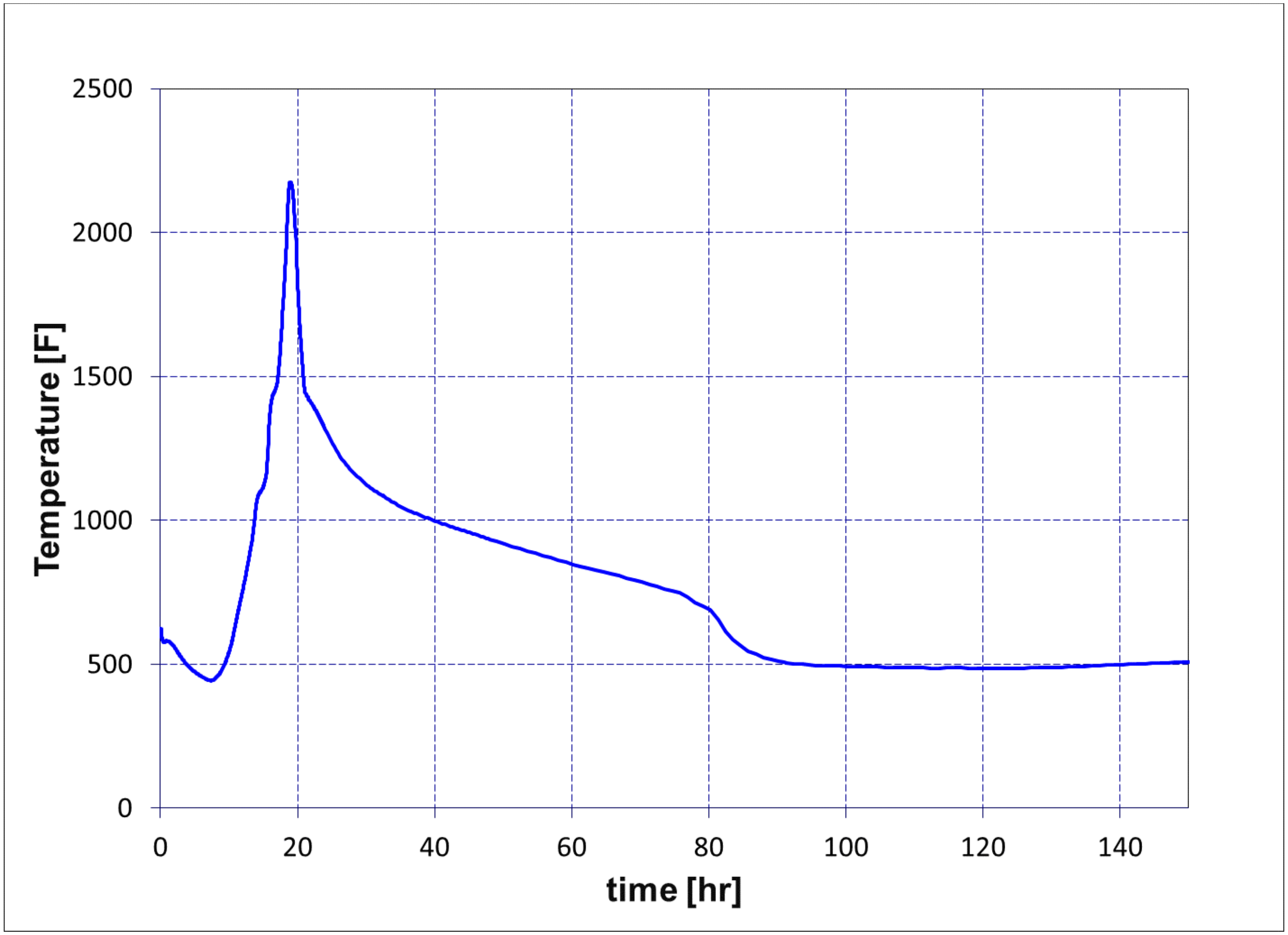


Figure 6-6: Heat Structure Temperature of the Vessel Head - Representative of Temperature Experienced by the "Category 4" Valves of the Reactor Vessel Head Vents

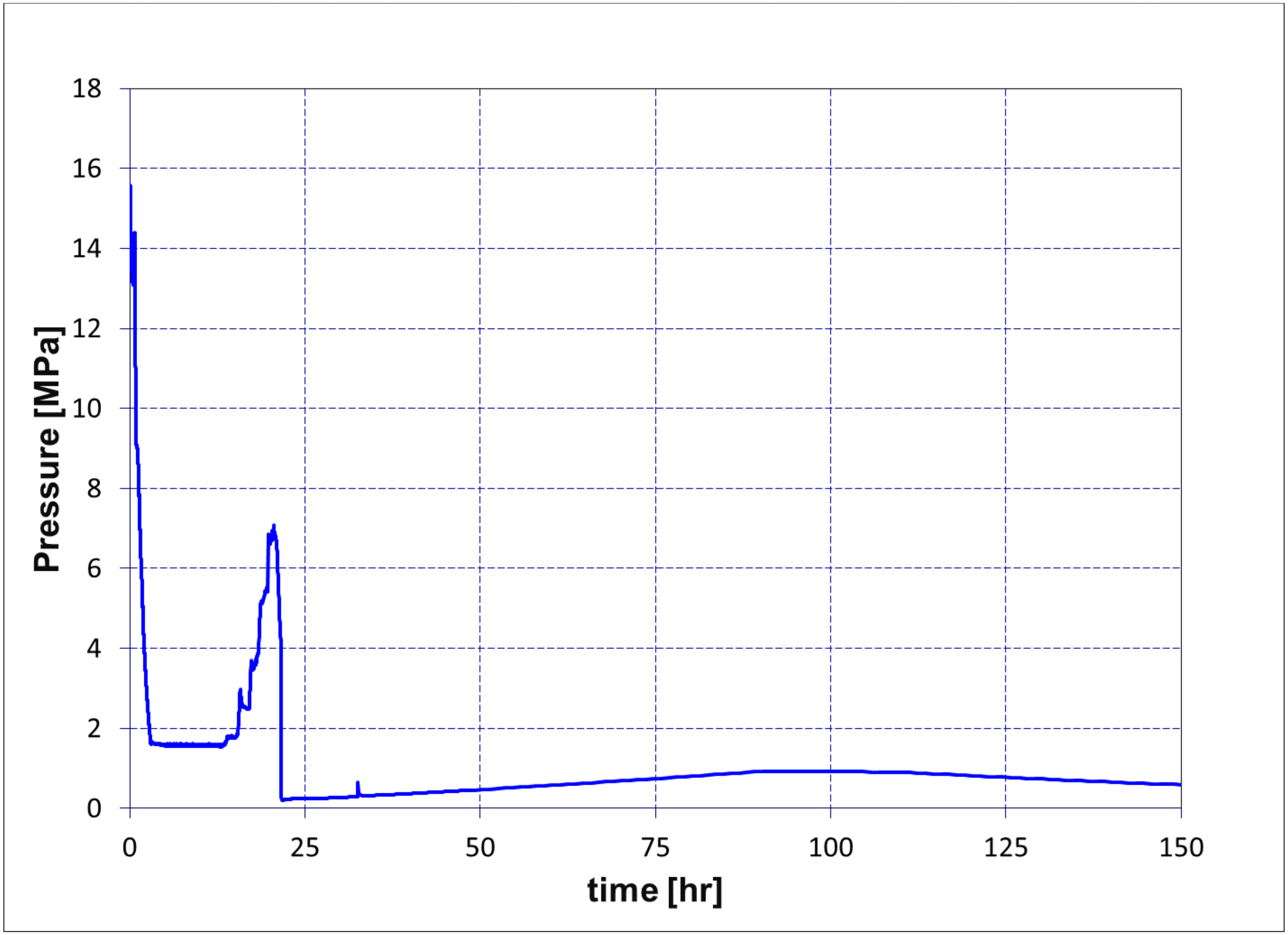


Figure 6-7: Representative Pressure Experienced by the Pathways in Table 6-1

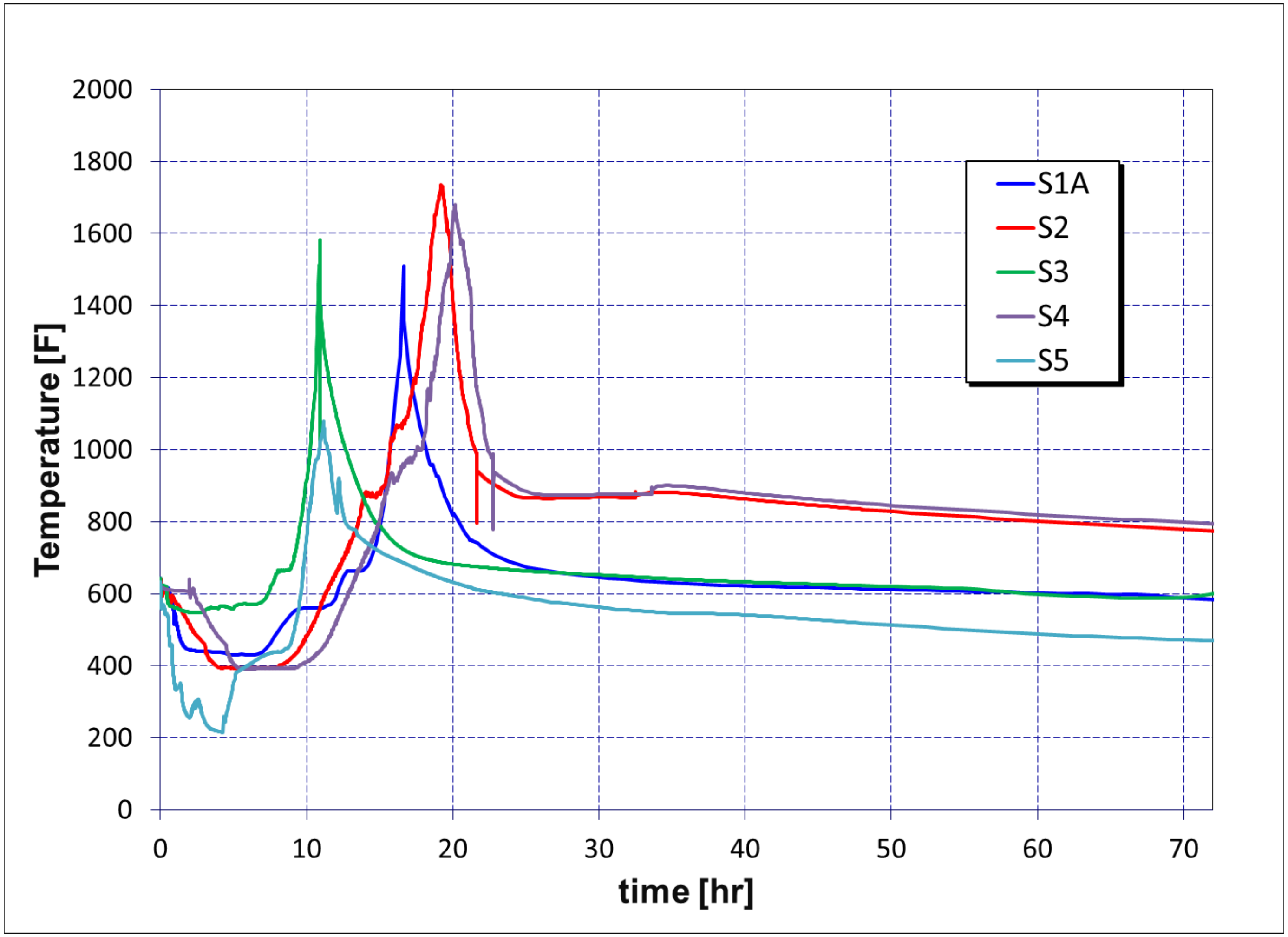


Figure 6-8: Temperature of the Surge Line Heat Structure for Various MELCOR Simulations.

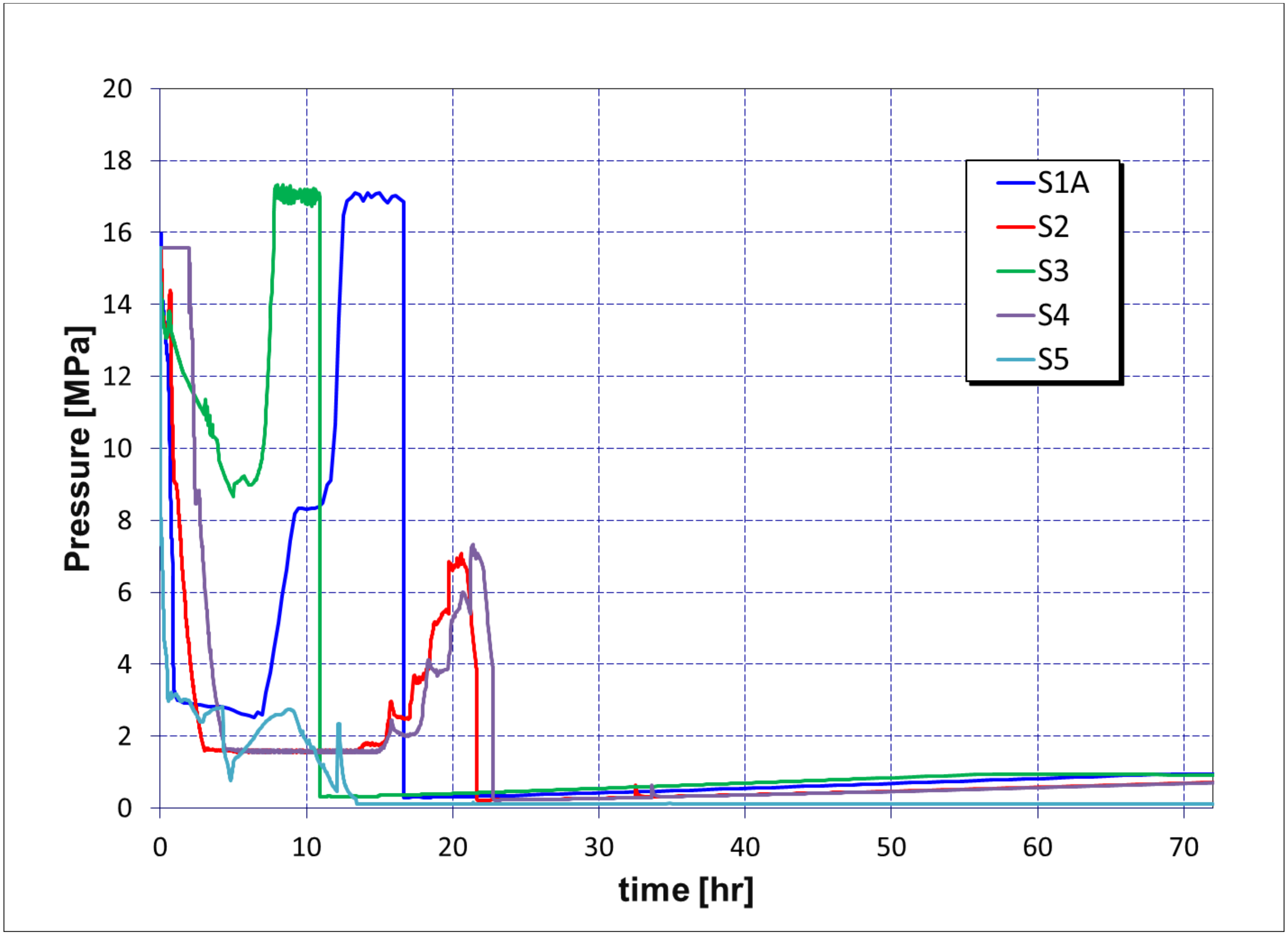


Figure 6-9: Pressure of the RCS for various MELCOR simulations.

While the temperature seen by a particular valve may vary in a given accident, the paths identified in **Table 6-1** are generally not susceptible to induced ISLOCAs because either:

- There are multiple normally closed valves in series, with the second valve being distant from the RCS (items 1, 2, 3, 4, and 11);
- The valve is on a part of the system that typically sees much lower temperatures (even during accident conditions), such as the cold leg (items 5, 12, and 13);
- The valve is on a part of the system that is relatively stagnant during most accident conditions, and thus not likely to see the peak hot leg flows depicted in the figures above (items 6, 7, and 14);
- The valve is distant from the RCS (items 8, 9, and 10);

The above discussion lends some confidence that an induced ISLOCA is not likely for this design; however, to dismiss this would require more investigation and be beyond the current state-of-practice in Level 2 PRA.

7. High-pressure melt ejection

High-pressure melt ejection occurs when the RPV lower head fails with the reactor coolant system at elevated pressure. The blowdown of high-pressure steam and hydrogen following vessel failure entrains molten core debris and transports that debris to the containment atmosphere, leading to containment pressurization, and possibly containment failure. The mechanisms that may cause the rapid increase in containment pressure and temperature are the blowdown of the RCS, heat transfer from debris to the containment atmosphere, exothermic metal/steam and metal/oxygen reactions, and hydrogen combustion events triggered by the molten debris. These phenomena are collectively known as direct containment heating. It is important to address the factors that enhance or mitigate direct containment heating (DCH) because it could lead to early containment failure (Pilch, 1996).

The Reference Plant is a Westinghouse plant with a large, dry containment, and past NRC studies (e.g., (Pilch, 1994) (Pilch, 1995) and (Pilch, 1996)) have concluded that the probability of containment failure due to DCH is very low for this plant type. With that said, the amount of debris entrainment and transport is highly plant- and scenario-specific. The cavity and instrument tunnel configuration and the amount of water in the cavity (plant parameters), as well as the melt composition, the containment, and the RCS pressure at the time of vessel failure (sequence-specific parameters), are some of the parameters that have an important influence on DCH phenomena.

The Reference Plant cavity arrangement was categorized as a “Type C” containment during the IDCOR program, but was re-classified as a “Type M” containment in (Pilch, 1996).[2] Plant-specific considerations for the Reference Plant suggest that this cavity arrangement would permit less direct entrainment of core debris (EPRI, 2012) than other Westinghouse large, dry containment configurations. There is limited potential for debris entrainment in the containment atmosphere, this is due in part to the arrangement of the instrument tunnel and seal table, which do not provide a clear path from the cavity to lower containment (unlike the Zion containment analyzed in (Pilch, 1994)). Instead, the seal table room floor blocks most of the flow pathway between the instrument tunnel and the seal table room. The primary vent paths are through the

[2] See Appendix C of (Pilch, 1996), Section C.2. For both containment types, debris retention in the cavity is expected to be higher, and debris transport to the dome is expected to be lower, compared to the Zion cavity arrangement described in (Pilch, 1994).

loop cutouts and through small vessel instrumentation ports around the vessel flange, neither of which provide the same potential for direct debris entrainment as the Zion cavity arrangement. There are additional vent pathways to lower containment through the manway leading to the instrument tunnel and through the cavity purge system ducting, which could fail due to overpressure following vessel breach and debris ejection. The instrument tunnel manway would likely enhance debris retention in the cavity (Pilch, 1996). The cavity purge ducting connects the cavity to lower containment, so there is limited potential for debris entrainment through this pathway if backflow dampers in the ducts fail.

Scenario-specific considerations are addressed by the accident progression calculations performed with MELCOR. The MELCOR calculations provide insight into whether or not induced hot leg creep rupture would depressurize the RCS before lower head failure, the melt composition at the time of vessel failure, and the containment pressure at vessel failure.[3] These plant- and scenario-specific considerations have been factored into DCH split fraction estimates in the probabilistic model and reaffirm the conclusion from (Pilch, 1996) that DCH is unlikely.

8. Hydrogen combustion

Containment failure due to hydrogen combustion is included in the Level 2 PRA model. Note that much of the discussion that follows is based on information in (NRC, 1983). Please refer to that document for a broader discussion of hydrogen issues in reactor containments.

During a severe reactor accident, hydrogen is produced when zirconium – and, to a lesser extent, steel – in the reactor vessel reacts with steam at very high temperatures. Hydrogen and carbon monoxide are also generated ex-vessel by the reaction between molten metals and concrete components released during molten core concrete interactions. Both hydrogen and carbon monoxide are combustible and may pose a challenge to containment integrity in the form of increased static or dynamic pressure. The high temperatures and pressure loads resulting from combustion may also challenge the containment penetration seals and equipment needed to mitigate the accident progression.

Hydrogen combustion in hydrogen:oxygen:nitrogen mixtures is possible at concentrations as low as 4%, provided there is sufficient oxygen (< 5%). The lower flammability limit increases as carbon dioxide or steam are added to the atmosphere. Above approximately 60% steam or carbon dioxide, the atmosphere is inerted. (The diagram of Shapiro and Moffette – reproduced as Figure 2-16 in (NRC, 1983) – shows the regions of flammability and detonability for hydrogen:air:steam mixtures.) Containment systems such as fan coolers and sprays can condense steam in containment, de-inerting the atmosphere. Thus, containment heat removal systems have a significant influence on whether the containment atmosphere is flammable. For this reason, there are numerous cautions in the (EPRI, 2012) about starting containment sprays and fan coolers because these systems could de-inert containment.

The amount of energy needed to ignite a combustible mixture decreases with increasing hydrogen concentration and increases with increasing steam or carbon dioxide concentration. For example, high temperatures or discharges of static electricity are sufficient sources of ignition for combustible mixtures that are well within the flammability limits, while a larger source of energy (e.g., sparks from electrical equipment in containment, very high gas temperatures in the cavity) is needed to ignite a mixture near the flammability limits.

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[3] Note that the DCH issue resolution analysis (Pilch, 1996) assumes that the containment would be at or near atmospheric pressure, which may not be the case for a sequence with delayed lower head failure and with containment heat removal systems unavailable. Higher containment pressures would increase the probability of containment failure due to DCH.

Hydrogen combustion may propagate throughout flammable regions of containment. Combustion propagates upward more easily than downward; see for example (NRC, 1983) for upward, downward, and horizontal propagation limits. (Note that the default propagation parameters in MELCOR are taken from this table.) For lean (i.e. low combustible gas concentration) mixtures, experiments have shown that not all the combustible gas burns. In general, combustion completeness increases with increasing hydrogen concentration and is nearly complete at about 8-10% hydrogen. Turbulence – which may be induced by containment fan coolers or containment sprays – significantly improves combustion completeness. This is important to consider when predicting the pressure rise from a combustion event, which in turn is crucial for determining whether the resulting pressure spike would fail containment.

Combustion may be classified as either deflagration – in which flames travel at subsonic speeds relative to the unburned gas – or detonation – in which flames travel at supersonic speeds. Detonations produce both static and dynamic pressure loads that pose a severe challenge for even large, dry containments like the Reference Plant. Deflagrations generate quasi-static loads that may be high enough to fail containment in the event of a global deflagration with a high initial hydrogen concentration. The transition from deflagration to detonation is poorly understood and is generally not modeled in severe accident codes, such as MELCOR. With that said, deflagration is much more likely than detonation.

One important consideration is whether the containment atmosphere is well-mixed. If containment is well-mixed, then there is a lower likelihood that there would be pockets of high hydrogen concentration where hydrogen combustion may begin when the global containment atmosphere is not flammable. In general, mixing delays or prevents hydrogen combustion. On the other hand, if the entire containment atmosphere is flammable, combustion may propagate throughout containment. If such a global deflagration occurs, the energy released would be much greater than if combustion is confined to a small region of containment.

For the Reference Plant, hydrogen and steam are generally released low in containment. Because hydrogen and steam are less dense than air, combined with the relatively open lower-to-upper containment communication[4], natural circulation tends to promote mixing of the containment atmosphere. Thus, it is expected that the hydrogen concentration will be relatively homogenous in the upper containment atmosphere (i.e. above the operating floor), with higher concentrations near hydrogen sources in lower containment and in the cavity. This hydrogen concentration gradient increases with increasing hydrogen release rates and decreases if the containment sprays or fan coolers are operating. Because the hydrogen concentration is higher in the cavity and in lower containment, and because there are more possible ignition sources in these regions (i.e. mechanical and electrical equipment, very high temperatures near molten material in the cavity and lower containment), combustion tends to begin in the lower regions of containment and propagates to upper containment – provided that the upper containment atmospheric conditions are sufficient for combustion. Note that the MELCOR analysis performed for this project largely shows this combustion behavior.

MELCOR and adiabatic isochoric complete combustion (AICC) calculations suggest that containment failure due to hydrogen combustion is very unlikely before vessel failure because hydrogen concentrations are too low to generate quasi-static pressure loads high enough to fail

[4] Note that statements about open communication between lower and upper containment may seem contradictory to statements that core material is unlikely to reach the containment liner/penetrations. The two differences are geometry and time-scale. The geometry of the containment is such that most of the open communication between lower and upper containment is occurring through the vertical grates that make up part of the operating deck, while lower containment itself is somewhat compartmentalized (in the radial direction) by the shield wall. Meanwhile, vessel blowdown phenomena occur over a matter of seconds or minutes, while containment mixing occurs over time-scales of tens of minutes or hours.

containment. Combustion-induced containment failure is more likely around the time of vessel failure due to the large amounts of hydrogen released to containment during vessel blowdown, though the probability of containment failure is still low. During the later term of the accident, after vessel failure, molten core-concrete interaction (MCCI) is a driving force for generation of additional combustible gases. Conditions could be conducive to combustion during this period depending on the progression of containtment conditions (e.g., if containment is steam-inerted) and whether prior combustion events have occurred.

In considering the potential for energetic combustion events that could challenge containment, a deterministic modeling approach, such as MELCOR, may not be sufficient for quantifying combustion issues, since the code by nature is deterministic; it uses fixed values for various parameters governing inherently uncertain issues. Furthermore, ignition is generally assumed to occur at a fixed, specified combustible gas concentration (10% by default), even though random ignition is a highly stochastic phenomenon. Random ignition may occur at lower or higher values. In the case of many large dry PWR containments, a global deflagration at 10% hydrogen concentration would not be capable of failing the containment, and using this result in a PRA may be falsely optimistic in implicitly assuming it is impossible for ignition to be delayed to slightly later in time when combustible gas concentrations are higher.The probabilistic ERPRA-BURN code is intended to overcome these limitations by simulating the combustion-related event progression of a severe accident in a Monte Carlo framework. For each combustion history, the values of various uncertain input parameters (such as the amount of hydrogen generated by in-vessel Zr oxidation) are randomly sampled. And, importantly, the timing of ignition is randomly sampled in such a way that (1) the rate of ignition sources per unit of time in a given time frame (very early, early, late) is assumed to be constant, and (2) the cumulative probability of an ignition source in a given time frame is consistent with a user-specified value (e.g., cumulative probability of a late ignition source is 0.99 for scenarios with AC power available). Whether or not ignition actually occurs at the moment when an ignition source is randomly determined to be present depends on the flammability of the gas mixture at that point in time, among other considerations. Burns, when they occur, alter the gas mixture going forward in that particular history, for example by depleting hydrogen and oxygen. In this manner, each combustion history consists of a random sequence of potential burns with a range of combustible gas concentrations, which may be less than or greater than 10%. Sorting and collating the results over all Monte Carlo combustion histories provides information from which one can determine cumulative containment failure probabilities (CCFPs) and other basic event quantifications for the PRA, including uncertainty distributions in these parameters.

Because ERPRA-BURN allows for potential combustion at gas concentrations higher than 10%, the results are less optimistic than one would obtain from a deterministic MELCOR calculation. For example, if a burn at 12% or greater hydrogen concentration is required in order to have any chance of failing the containment, MELCOR results would indicate there is a CCFP of zero, but ERPRA-BURN would predict a non-zero result due to a finite probability of random ignition events occurring at 12% or greater.

Using the ERPRA-BURN approach, the combustion event progression is differentiated into at least three time phases, on the basis of the fact that each is governed by different phenomena and prevailing conditions:

- Very Early (also called “phase I” in ERPRA-BURN): This is the time frame from the start of the accident until just before the instant of vessel breach. (If there is no breach of the vessel for a particular sequence, then the very early time frame prevails until the end of the accident.) In this time frame of a severe accident, hydrogen is generated by metal oxidation in the reactor core, which is transported into the containment where it can potentially burn.

- Early (also called “phase II” in ERPRA-BURN): This is a short time frame beginning at the moment of vessel breach and ending, defined somewhat vaguely, at the end of whatever brief containment transient accompanies vessel breach and the start of more quasi-static containment phenomena (typically less than one hour). The failure of the vessel expels hot or molten core elements into the containment, potentially along with some additional hydrogen that had been held up inside the RCS.

- Late (also called “phase III” in ERPRA-BURN): The late time frame applies from the end of the brief containment transient accompanying vessel breach and until the end of the accident as defined in the particular scenario. It is characterized by base containment atmosphere conditions that are either constant or changing slowly and quasi-statically – for example, a gradual increase in containment pressure and steam concentration due to decay heat from the core. In addition, MCCI is a driving force in this time frame for generation of additional combustible or non-condensable gases, and ex-vessel oxidation of any remaining core Zr can contribute as well.

For purposes of the PRA logic model, the relevant portions of the containment event tree and its decomposition event trees, decompose the question of combustion into three top events for each of the three time frames. The event tree top events query: 1) whether or not there is a source of ignition in the containment in a given time phase, 2) whether or not ignition and combustion actually occur, conditional on there being a source of ignition in this time phase of the accident, and 3) whether or not containment fails as the result of combustion in a particular time frame. Combustion-induced containment failure is a function of the peak pressures generated by the combustion event(s) that take place during the time frame, as compared against the containment’s static fragility.

The decomposition for the late time frame adds one extra top event in between the questions of ignition and containment failure. The top event asks, given ignition in the late time frame, whether or not the burn undergoes a deflagration-to-detonation transition (DDT). Including this top event aids the analyst in determining which sequences are containment failures as a result of detonation versus deflagration. This is a complex phenomenological issue, which is however addressed in ERPRA-BURN in rather simplistic fashion dependent only on the combustible gas concentration in containment at the time of ignition. In order to explore the possible impact of detonation on containment failure within the ERPRA-BURN probabilistic framework, cases were run using the code’s “Minimum Detonable Concentration” feature. Normally 1.0 by default, this parameter was changed to 0.15 for cases considering potential detonation (i.e., ERPRA-BURN will assume any burn at 15% or higher combustible gas concentration will be a detonation and will fail the containment with a probability of 1.0).

The compilation of history and history set results from ERPRA-BURN are used to arrive at final probabilities of ignition or containment failure. Point values of these probabilities are generally the weighted average of the results over all histories. In the case of CCFPs, the results also can be sorted in increasing order of CCFP and the weighted probabilities integrated to obtain an uncertainty distribution for containment failure probability.

Generally, the scenario-dependent ERPRA-BURN input parameter values are set by extracting results from a corresponding MELCOR calculation, while some input parameters are based on assumptions informed by the scenario attributes (e.g., probabilities of an ignition source in containment in each time phase). Times and durations from the start of the accident are extracted from MELCOR and input on a scenario-specific basis. The pre-accident containment conditions, including pressure, temperature, and humidity of the containment atmosphere, are obtained from the MELCOR base input deck initial conditions. The ERPRA-BURN modeling considers changes to the containment pressure and relative humidity during the different phases of the accident scenario. These values are extracted from the MELCOR results. The amounts of combustible gases generated throughout the accident are obtained from the corresponding MELCOR scenario. The containment fragility uncertainty distribution, used for determining the probability of containment failure, is taken from a study by Sandia National Laboratories on resolving direct containment heating issues for large dry containments (Pilch, 1996).

Several ERPRA-BURN calculations were carried out for the two representative MELCOR scenarios:

- A general transient scenario (MELCOR Case 6B) with long-term containment heat removal available as a result of continued functioning of the containment sprays and fan coolers. Since AC power is available in this scenario, cumulative ignition source probabilities with point values of 0.99 for all phases were assumed. Associated uncertainty distributions are assigned based on qualitative judgment.
- An SBO (MELCOR Case 1A1) with loss of AFW and ARVs after a short duration. Since AC power is unavailable in this scenario, cumulative ignition source probabilities with point values of 0.1, 0.5 and 0.3 were assumed for phases I, II and III, respectively.

Two factors were found to be important to formulating the scenarios and sensitivity cases on those scenarios.

- Whether AC power is available or unavailable can influence combustion behavior through the probability that there is an ignition source in the containment (higher with power than without). However, the ignition source probabilities are a direct input to ERPRA-BURN that is generally independent of a particular MELCOR scenario. Each MELCOR scenario can be used as the basis for two ERPRA-BURN scenarios with and without AC power assumed.

- The main time phase of interest is the late time frame after vessel breach, since it is there that there is a potentially very large cumulative amount of combustible gas available, especially for very long accident scenarios. There is often a sharp difference in behavior between scenarios with late containment heat removal (CHR) – with low, stable base pressures and low steam concentrations – and ones without CHR, where pressures and steam concentrations are much higher and rising (and potentially inert the atmosphere to combustion late in the accident). Therefore, each set of scenarios includes at least one with late CHR (containment sprays and/or fan coolers) and one without. The analyzed scenarios only consider heat removal by fan coolers when AC power is available. Containment sprays are not considered. (Later analyses for reactor-at-shutdown scenarios consider containment sprays.)

Some high-level observations concerning the predicted combustion behavior are as follows:

- Across all scenarios and assumptions, there is zero or negligible chance of containment failure in the very early to early time frames from combustion, even though there is a high probability of combustion taking place. There simply is not enough hydrogen from in-vessel oxidation alone to pose a challenge to the containment's high structural capacity.

- In the late time frame, there is ample quantity of combustible gas generated to burn given that an ignition source is present. Therefore, across all scenarios, the conditional probability of a late burn given an ignition source ranges from very high (0.80) to certain (1.0).

- In the late time frame of station blackouts, there is less likely than not to be a source of ignition in the containment, making burns somewhat unlikely. But by the same token, the infrequency of ignition events permits more hydrogen to build up before a burn takes place, making them potentially more consequential. The mean conditional probability of containment failure given a late burn in an SBO scenario ranges from about 0.33 to 0.67; however, owing to the low probability of an ignition event, the net containment failure probability is on the order of 0.1 to 0.2.

- As compared with SBOs, transients with AC power and no late CHR generally indicate a slightly lower late CCFP, since the more frequent sources of ignition permit less combustible gas to build up between burns. However, the increased frequency of these burns roughly balances this (i.e., the net containment failure probability for full-power transients is slightly higher than for SBOs). The late CCFP with prior combustion events occuring is about 0.28.

- The main risk of containment failure from combustion is posed by scenarios with operating late containment heat removal. In these scenarios, while the base pressure is small, the low steam concentration amplifies the relative concentrations of oxygen and combustible gas, and the containment never becomes steam-inerted. This permits the concentration of combustible gas to reach potentially very high levels, very often in the range assumed to be detonable (15% or higher). ERPRA-BURN results for CHR scenarios indicate that the conditional probability of detonation in the late time frame is roughly 0.8, and the current model simply assumes that such detonations as might occur always fail the containment. From the AICC calculations for global deflagrations, it turns out that gas mixtures capable of failing the containment in these cool conditions require at least 15% or very slightly less; therefore, almost all containment failures for CHR scenarios are detonations, with very few deflagrations challenging the containment.

- Some additional considerations on the importance of detonations is provided here. In the general transient with CHR scenario, for example (Case 6B), the base number of kg-moles of initial air in the containment is about 2770, and total in- and ex-vessel hydrogen generation is 3582 kg or 1791 kg-moles. So, if all cumulative hydrogen were dumped into the containment at once, it would be a concentration of about 39%, the combustion of which would require about 20% concentration of oxygen. Note that the initial concentration of oxygen is 21%, and combustion in the model is assumed to stop at 5% oxygen as the mixture becomes no longer flammable, so only about 16% is actually available for reaction; therefore, combustion in the late time frame of CHR scenarios is generally limited by the quantity of available oxygen rather than combustible gases. Furthermore, the sheer quantity of gases available through the end of the scenario can fuel several very large burns. Since the model assumes that even a single burn at 15% or greater combustible gas concentration will result in a detonation, the sequence of combustion events that could prevent build up of such concentrations in the late time frame may not be very likely to occur. The late time frame as simulated in ERPRA-BURN considers whether or not the burns are all equally spaced out enough to burn at individually small concentrations; or whether there are one or a few burns early at small concentrations and then no ignition sources later on. It seems as if these sequences of events would be more unlikely than not, and therefore ERPRA-BURN's prediction of a conditional detonation probability in the range of 0.6 to 0.8 seems appropriate.

9. RPV vertical displacement due to blowdown forces

During a high-pressure melt ejection, thrust generated by the ejection of molten corium, steam and hydrogen gases has the potential to lift the RPV. The concern is that vertical displacement of the RPV could tear out containment penetrations (i.e. for piping connected to the RCS). This possibility was evaluated as part of the Reference Plant Individual Plant Examination (IPE) and was deemed to be unlikely based on the fact that the maximum jet thrust force that could be expected during high-pressure melt ejection (1×10^6 lb) is less than the lower-bound RPV dead weight (estimated as 1.3×10^6 lb for the IPE). The lower-bound RPV weight excludes the weight of the fuel, cladding, control rods, and lower core support plate.

The maximum thrust generated during high-pressure melt ejection (HPME) has been evaluated using input from MELCOR code calculations. The supplemental calculations verify that containment failure due to vessel rocketing is unlikely.

10. Core debris impingement

Core debris impingement involves direct contact of corium with the steel containment liner. For corium to contact the containment liner, the molten debris must be ejected from the reactor cavity. Such a situation is unlikely given the arrangement of the reactor cavity and instrument tunnel at the Reference Plant.

Core debris impingement is related to the issue of high-pressure melt ejection, which is described in Section 7. As stated above, the Reference Plant cavity design has limited potential for debris entrainment and relocation to the containment atmosphere. Furthermore, the location of the primary cavity blowdown pathways, and the numerous structures in the pathways between the cavity and the containment liner, significantly limit the amount of debris that can contact the penetration seals and the containment liner. For example, the seal table room floor blocks most of the flow between the instrument tunnel and the seal table room, a concrete platform in the seal table room obstructs flow to the annular region of lower containment, and the door to the seal table room connects to an internal stairwell. Also, the loop cutouts and the instrument ports around the vessel flange lead to areas near the center of the Reference Plant containment, well away from the containment liner.

This phenomenon is discussed briefly in the Reference Plant IPE in the context of thermal failure of containment penetrations. In the IPE, thermal failure of penetrations due to direct contact with debris was considered to be an unlikely mode of containment failure, for the reasons discussed above.

Because there are no direct pathways by which hot core debris may contact the containment liner or containment penetrations, containment failure due to debris impingement is deemed to be extremely unlikely and is not included in the Level 2 PRA model.

11. In-vessel steam explosion

During a severe reactor accident, molten fuel may relocate to a water-filled lower plenum and interact with the water to form steam. The interaction between the molten fuel and the liquid water is known as fuel-coolant interaction. If the interaction time scale is on the order of milliseconds, it could create an energetic steam explosion. A steam explosion is a process in which steam production occurs at a rate greater than the surrounding medium can relieve, leading to the formation of a shock wave (EPRI, 2012). Large, energetic explosions can only occur if the premixture of fuel, coolant, and vapor is adequately dispersed and if a trigger (i.e. a pressure and/or flow perturbation that may cause destabilization of film boiling and fragmentation of melt into small drops that come into contact with the liquid water is available. It has been postulated (e.g., (NRC, 1975)) that under certain conditions, an energetic steam explosion could generate a shock wave with sufficient force to fail the reactor pressure vessel upper head and to generate a missile that would subsequently fail containment. This event is known as alpha-mode containment failure.

The issue of alpha-mode containment failure was addressed and resolved by the Second Steam Explosion Review Group (SERG-2) Workshop, which was sponsored by the NRC in 1995 (NRC, 1995). The expert panel concluded that alpha-mode containment failure is very unlikely, and that the issue is therefore resolved from a risk perspective. One reason for this conclusion is that a highly fragmented melt would result in high steam production rates that would drive water from the interaction zone. On the other hand, a poorly fragmented melt would limit the contact area between water and melt, thereby limiting the amount of melt that could participate in an explosion. In both cases, the premixture is not conducive to a large steam explosion that could fail the reactor vessel. Some experts also stated that melt relocation scenarios limit the amount of melt that could be involved in a steam explosion to less than that needed to fail the reactor vessel (Basu, 1996).

It must be noted that steam voiding of water from the premixture is a low-pressure phenomenon and cannot be relied upon to prevent a steam explosion at pressures above approximately 1 MPa (Basu, 1996). However, experimental and analytical studies suggest that it is difficult to trigger a steam explosion above approximately 1 MPa, so alpha-mode failure is unlikely in these scenarios ((EPRI, 2012) and (Haskin, 2002)). Note that reactor vessel pressure is significantly higher than 1 MPa in many of the core damage sequences in the reactor-at-power PRA for the Reference Plant. Nevertheless, alpha-mode failure is still included in the Level 2 model, with split fractions based on expert guidance in (Basu, 1996).

12. Ex-vessel steam explosion

A steam explosion may also occur if the reactor cavity is flooded at the time of melt relocation from the reactor vessel to the cavity. The basic steam explosion phenomena (i.e. premixing, triggering, and propagation) described in the previous section also apply to ex-vessel steam explosions. Here, the concern is that an ex-vessel steam explosion will fail containment through either combined static and dynamic pressure loading; failure of vessel support structures, leading to movement of the vessel and failure of piping penetrations; or generation of missiles that impact the containment walls (Haskin, 2002).

For an ex-vessel steam explosion to occur, the reactor cavity must be flooded. The Reference Plant has a dry cavity design, which means that the cavity is not expected to be flooded at the time of vessel failure in most accident sequences. With that said, there are a few potential pathways for water to enter the cavity:

- In a loss of coolant accident (LOCA) scenario, water may be discharged to the cavity if the break is near the hot leg nozzles (i.e. inside of the cavity wall).
- If containment sprays operate, some spray water may collect in the refueling cavity and drain to the reactor cavity through instrumentation ports around the reactor vessel upper head flange. A calculation provided by the licensee estimates that a best-estimate flow

rate through the ventilation ports would be approximately 1 ft^3/min with both trains of containment sprays operating.

- Water from lower containment may leak to the reactor cavity through penetrations in the cavity wall. The height of these penetrations above the floor of lower containment is such that there would be some leakage to the cavity in any LOCA scenario that leads to filling of the ECCS sump. Leakage rates are on the order of 1 to 10 ft^3/min.

Note that small, medium, and large LOCAs make up a relatively small portion of the core damage frequency, so the cavity would be dry for most core damage sequences.

For a discussion focused more on the MELCOR modeling itself in this regard, see Section 12 of Appendix D.

13. RPV and/or containment venting

RPV venting is considered within the human reliability analysis, in situations where it would be likely to be pursued as part of severe accident mitigation guideline (SAMG) implementation (namely SAG-2). It is not discussed further here because it is not a unique containment failure or bypass phenomena, beyond its effect on (i) vessel pressure at the time of vessel failure, (ii) stresses on steam generator tubes, and (iii) changes to distribution of steam and hydrogen in containment (all of which are discussed elsewhere in this document).

Containment venting is included as part of the Reference Plant SAMGs, in particular SCG-2, which is entered if containment pressure is greater than 102 psig. Venting protects containment integrity; however, venting could potentially release a large amount of radioactive material to the environment. The time of venting relative to containment failure is crucial in determining potential releases. Very early venting (i.e. before substantial fuel degradation) would lead to relatively low releases. Similarly, very late venting (i.e. many hours after vessel failure) may also lead to low releases because fission product aerosols have time to agglomerate and settle in containment. In contrast, very high releases are likely if containment venting occurs shortly after vessel failure because there would be a very large amount of fission product material in the containment atmosphere.

Because the Reference Plant has a large, dry containment, venting is more likely during the late phase of the accident due to non-condensable gas addition from molten core concrete interaction (MCCI) and steam addition from evaporation of water in the sump and/or boiling of an overlying water pool in the reactor cavity. That said, there are other situations (e.g., hydrogen control) that could lead to venting in other timeframes.

MELCOR calculations do not explicitly account for containment venting; however, these calculations assume containment overpressure failure at 120 psig, with the failure mode being one that leads to gradual (as opposed to catastrophic) leakage. Because the SAMGs do not direct operators to vent for over-pressure protection until pressure exceeds 102 psig, the containment overpressure failure in the MELCOR calculations could be viewed as operator actions to vent containment. Likewise, containment venting is not explicitly included in the Level 2 PRA model, but containment venting releases are expected to be very similar to releases from late containment overpressure failure.

14. Overpressure failure due to increases in quasi-static pressure

During a severe accident, significant quantities of non-condensable gases and steam are produced during molten core concrete interactions in the cavity. Significant quantities of steam may also be produced by evaporation of water in the sump and lower containment or by boiling of an overlying water pool in the cavity. Heat transfer from core debris to the containment atmosphere also contributes to the increase in pressure in containment during a severe accident. Left unchecked for many hours, and in the absence of containment heat removal systems, these processes may increase containment internal pressure beyond its pressure retention capacity. This is generally referred to as containment overpressure failure. (Note that this failure mode is distinct from containment failure due to hydrogen combustion, which may also result in containment pressure exceeding its capacity.) This failure mode is of concern in the late stage of the accident, typically tens of hours after lower head failure for a large, dry, pre-stressed concrete containment with a steel liner like the Reference Plant.

Experiments have shown that a reinforced or pre-stressed concrete with a steel liner can withstand pressures well in excess of design pressure. Tests show that this containment design develops cracks in the concrete that increase with pressure. Strain concentrations in the steel liner eventually lead to small liner tears that rapidly expand with increasing pressure. At some point, leakage through these liner tears prevents further pressurization of containment, such that there is no global failure of the containment. Liner tears typically occur at discontinuities in containment, such as near hatches, penetrations, and the liner-basemat junction. The Reference Plant IPE identifies the basemat junction as the most likely failure location, with the equipment hatch as another likely failure location. Analysis performed for this project generally corroborates the likely failure locations and the pressure at which liner tears would develop for the Reference Plant. Section 6 of Appendix D also provides a brief discussion of over-pressure failure.

15. Mechanical and electrical penetration failure

Failure of a mechanical or electrical penetration may be caused by operator failure to isolate an administratively controlled penetration by thermal loading of nonmetallic components of penetration seals or by tearing of the penetration by displacement of the reactor pressure vessel. Each of these issues is discussed in the following paragraphs.

Operator failure to isolate an administratively controlled penetration could lead to a containment isolation failure that is not directly linked to the accident progression. This issue is addressed in the containment isolation failure fault tree in the bridge tree between the Level 1 and Level 2 PRA models. Section 2 of Appendix D includes additional information about treatment of containment isolation failure in the Level 2 PRA model.

Failure of penetration seals due to thermal loading may be caused by direct impingement of core debris on the seals during vessel blowdown following lower head failure, or by high gas temperatures in containment due to a hydrogen burn or prolonged MCCI. Direct debris impingement is discussed in Section 10. (It is unlikely due to the location of the mechanical and electrical penetrations with respect to the likely core debris dispersal pathways from the cavity.) The Reference Plant IPE concluded that the maximum gas temperatures predicted for lower containment would not challenge the limiting sealant material (i.e. polysulfene thermoplastic). The IPE’s conclusions about penetration seal failure due to thermal loading are adopted for the Level 2 PRA project, so this phenomenon is not included in the PRA model.

RPV vertical displacement due to blowdown forces is included in the Level 2 PRA model and is discussed in Section 9 (though gradation of failure types is not pursued). Weakening of RPV supports by erosion of the cavity walls is not included in the PRA model due to a lack of supporting structural analysis. With that said, the Reference Plant IPE argues that failure of RPV supports due to radial erosion will not occur prior to basemat melt-through. Cavity erosion is described in more detail in Section 17.

16. Leakage at hatches

All containment penetrations – including hatches such as the equipment hatch, personnel airlock, and emergency airlock – are sealed to limit leakage of radioactive material to the environment during a reactor accident. Nevertheless, there is always some leakage from containment to the environment. For this reason, there is a limit on allowable leakage from containment at design pressure, which is known as the design basis leakage rate. Licensees must perform integral leak rate tests (ILRTs) at specified intervals to demonstrate that their plants meet their design basis leakage criterion. For this project, deterministic accident progression calculations performed with MELCOR assume that containment leaks at the design basis rate, as explained in Section 6 of Appendix D.

Given that ILRTs are conducted at infrequent intervals (once every 15 years for the Reference Plant), there is some probability that leakage would exceed the design basis rate. This is true for leakage past containment penetrations and hatches as well as for leakage through cracks and tears in the containment structure and the containment liner. Leakage at hatches resulting in leakage above the design basis rate is included in the containment isolation failure fault tree in the bridge tree between the Level 1 and Level 2 PRA models, as described in Section 2 of Appendix D.

In addition, high temperatures could degrade hatch seals, thus increasing leakage from containment. This issue is described for mechanical and electrical penetrations in Section 15. That discussion applies as well to the seals around hatches. Again, the Level 2 PRA model does not address this issue, but this is reasonable, given that the licensee's analysis performed as part of the Reference Plant IPE concluded that thermal degradation of seals around penetrations is unlikely.

17. Basemat Melt-Through

Basemat melt-through is not included in Table 3.5-8 of the draft Level 2 PRA standard (ASME, 2014), but it is included as a failure mode in the Level 2 PRA model. Basemat melt-through refers to erosion of the concrete in the cavity such that the melt comes into contact with the soil beneath the plant. This failure mode is inevitable if the molten pool in the cavity is not coolable. The concern with basemat melt-through is that it would release a large amount of radioactive material to the local water table, which poses significant challenges for long-term health effects and environment remediation efforts.

If the lower head of the reactor pressure vessel fails, hot core debris will relocate to the space beneath the vessel and come in contact with the concrete floor and walls (for an excavated cavity like the Reference Plant) of the reactor cavity. Heat transfer from the debris to the concrete can chemically decompose the concrete, releasing water vapor and carbon dioxide, and melt the residual oxides and the metal reinforcing bars in the concrete. These processes are referred to as molten core concrete interaction (MCCI). MCCI dilutes the molten pool and reduces the volumetric heat generation rate and simultaneously increases the surface area of the debris pool. In time, heat transfer from the debris pool to the concrete and to the cavity atmosphere may be sufficient to remove the (decay and chemical) heat and reduce the debris temperature such that MCCI is terminated (Haskin, 2002). However, it is possible that basemat melt-through would occur before cessation of MCCI.

There are several factors that influence the rate of concrete erosion. These factors include the amount of debris in the cavity, the debris composition, and the concrete composition. Larger amounts of debris ablate the concrete at a more rapid rate, thus reducing the prospects of cooling the debris and preventing basemat melt-through. Significant quantities of chemical heat may be added to pools with high metal contents, whereas highly oxidic pools will experience less chemical heating. Concrete composition determines both the rate of ablation and the quantity of gases (both water vapor and non-condensables) produced. Limestone concrete erodes much more rapidly and releases a larger quantity of gases than siliceous concrete. Note that the Reference Plant uses siliceous (basaltic) concrete.

The floor area of the cavity also plays a significant role. A larger floor area allows the debris to spread, thus increasing the surface area and enhancing the prospects for debris coolability. The Reference Plant has a relatively tight cavity, so there is little room for the debris to spread.

Addition of water on top of the debris pool enhances heat transfer and can substantially increase the prospects for arresting MCCI before basemat melt-through. During top-cooling by an overlying water pool, a solid crust forms between the molten pool and the overlying water pool. Molten material erupts through cracks in the crust, forming a debris bed on top of the crust. Laboratory experiments show that the top crust is mechanically weak and likely would not be able to support the weight of the overlying pool. Crust failure allows water intrusion to the molten pool, which further enhances coolability (EPRI, 2012). However, there is significant uncertainty as to whether these mechanisms observed in the laboratory would be sufficient to cool the debris in a real accident scenario, especially for reactors with tight cavity designs.

For the Reference Plant, if the lower head fails when the reactor is at high pressure, much of the core debris will be expelled from the cavity and deposit in the instrument tunnel and throughout containment (see Section 7). In this situation, the debris is much more likely to be coolable because the debris has a significantly larger surface area. Thus, the Level 2 PRA model views debris coolability following HPME as very likely. On the other hand, if the lower head fails at low pressure, the debris will end up in the relatively tight cavity beneath the vessel and will likely not be coolable unless operators take immediate action to flood the cavity. Even if operators were to flood containment early, there is still significant uncertainty as to whether top-cooling would be effective in cooling the debris and halting MCCI because the debris forms a deep molten pool in the cavity following lower head failure. Thus, sequences that do not undergo HPME and that do not result in some other containment failure or bypass are likely to result in basemat melt-through.

18. References

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