

U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 5b: Reactor, At-Shutdown, Level 2 PRA for Internal Events

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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 5b: Reactor, At-Shutdown, Level 2 PRA for Internal Events

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ABSTRACT

The U.S. Nuclear Regulatory Commission performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. 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. 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. This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-shutdown, Level 2 PRA model for internal events 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.¹

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, 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 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 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 Coping Strategies [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 assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (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 decision-making. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to 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.

To provide results and insights better aligned with the current design and operation of the reference plant, the overview reports also provide the results of a parametric sensitivity analysis based on a set of new plant equipment and PRA model assumptions for all three PRA levels. The sensitivity analysis reflects the current reactor coolant pump shutdown seal design at the reference plant, as well as the potential impact of FLEX strategies,² 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 decision-making 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 L3PRA 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 [ML12202B170]):

- 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

² FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. 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 off site.

advanced reactor designs, a multi-unit accident, or an accident involving spent fuel; and using PRA information to inform emergency planning)

- 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 L3PRA project model can be exercised to provide insights regarding 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 endorsed via Regulatory Guide 1.233 in June 2020. 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 L3PRA project, will be incorporated into a summary report to be published after all technical work for the L3PRA project has been completed.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

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

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data, that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (ML040140729), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown 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.

This report documents the single-unit, reactor at-shutdown, Level 2 PRA for internal events that supports the L3PRA project. 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 reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.³) 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. Finally, it should be noted that the project team does not consider the reference plant’s Severe Accident Management Guidelines (SAMGs) to be applicable to the reactor at-shutdown Level 2 PRA.⁴ **As such, this report will not be the sole basis for any regulatory decisions specific to the reference plant.**

³ FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. 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 off site.

⁴ The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available, and thus not considered, at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

As a part of the L3PRA study, the NRC has constructed a Level 1, at-shutdown, PRA model for internal events using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software. The shutdown Level 1 PRA model was developed to assess the risk associated with the shutdown plant operating states (POSSs) of a refueling outage. Six categories of initiating events are assessed: loss of offsite power, loss of RHR cooling, loss of inventory, loss of power to a 4,160V AC safety-related bus, overdraining entering mid-loop, and loss of nuclear service cooling water.

The shutdown Level 1 PRA model (NRC, 2025) is used as the starting point for the shutdown Level 2 PRA model. The core damage frequency (CDF) accident sequences and results from the Level 1 PRA are carried forward in the Level 2 PRA to characterize radiological releases and their frequencies. 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.

The Level 2 PRA covers a broad spectrum of severe accident sequences, and considers a variety of different systems, phenomena, and operator actions. The shutdown Level 2 PRA report is structured to align with the technical elements as outlined in the requirements of the Level 2 PRA Standard for trial use and pilot application (ASME, 2014). Each technical element has been addressed, and a summary of results and observations is provided below.

Eleven release categories are defined to characterize releases in terms of their release timing, energy, location, magnitude, and frequency. Notable observations from the release category frequency results include:

- The largest contribution to release frequency (about 49 percent) is associated with the intact containment end state.
- Containment overpressure failure due to boiling of refueling cavity inventory contributes about 17 percent to the total release frequency. This category represents a condition unique to refueling outages where a loss of all AC power and failure to restore cooling results in gradual heatup, evaporation, and boiling of refueling cavity inventory.
- Containment isolation failures contribute about 14 percent to the total release frequency. The containment isolation failures are predominately associated with human failures to isolate open penetrations at the time of the event occurrence. Most of the release category frequency is associated with station blackout sequences.
- Combustion-induced containment failure occurring after vessel breach contributes about 14 percent to the total release frequency. An important feature of these accident sequences is the availability of containment heat removal (primarily via containment spray) contributing to combustible conditions. The result is also associated with the use of a stochastic combustion modeling approach, which was also used in the at-power Level 2 PRA portion of this study (NRC, 2022a).
- Three release categories associated with large containment openings (i.e., failure to close the open containment equipment hatch) collectively account for about four percent of the total release frequency. While these categories are not among the highest frequency contributors, they lead to the largest release magnitudes and thus are expected to be an important contributor to the reactor at-shutdown risk.

- The initiating events and POSs contributing to the significant release categories generally reflect the results of the Level 1 CDF model. Losses of offsite power and station blackout sequences are significant contributors for several release categories. Other notable shutdown initiating events include loss of RHR cooling due to overdraining entering mid-loop and loss of inventory events with RCS intact. Significant accident progression sequences involve events initiated during POSs with the refueling cavity flooded prior to refueling, pressurizer solid operation, and mid-loop operation. These attributes reflect the dominant accident progression sequence frequency contributions, and they are also used to inform the representative accident scenarios for developing release category source terms.

Three surrogate risk metrics are defined and assessed for the shutdown Level 2 PRA. The surrogate risk metrics and calculated results are: large early release frequency (LERF) of 5.3×10^{-7} per year, large release frequency (LRF) of 6.1×10^{-6} per year, and conditional containment failure probability (CCFP) of 50.6 percent.

The LERF contribution is approximately four percent of the total release frequency of 1.2×10^{-5} per year. Three release categories are identified as contributors to LERF, which include an interfacing system loss-of-coolant accident release category and two release categories with large, unisolated containment openings. The LRF release categories contribute about 49 percent of the total release frequency. Note that the basemat melt-through and late containment failure release categories involve containment failures but are excluded from LRF due to the magnitude of the airborne environmental releases being comparable to the intact containment release category. The CCFP is just over 50 percent of the total release frequency with the intact containment category accounting for just over 49 percent of the frequency.

Sources of model uncertainty for the shutdown Level 2 PRA are identified and assessed. Several model uncertainties were identified that can have significant impacts on the Level 2 PRA results. Sensitivity studies were performed for selected model uncertainties to quantify the impacts on the Level 2 PRA results. The sensitivity studies show the impacts in terms of changes to the release frequencies or the source term release magnitudes.

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- Structural analysis to support the containment capacity analysis was performed by Hernando Candra² and Jose Pires.²
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- Don Helton² has provided significant contributions to this work and served as the initial project technical leader for the shutdown Level 2 PRA.

¹ Energy Research, Inc.

² Formerly of U.S. Nuclear Regulatory Commission

³ Oak Ridge National Laboratory

⁴ Formerly of Sandia National Laboratories

⁵ U.S. Nuclear Regulatory Commission

⁶ Idaho National Laboratory

⁷ Formerly of Idaho National Laboratory

⁸ Formerly of Westinghouse Electric Company

ABBREVIATIONS AND ACRONYMS

AC	Alternating current
ADAMS	Agencywide Documents and Management System
AFW	Auxiliary feedwater
AICC	Adiabatic isochoric complete combustion
ALWR	Advanced light water reactor
ANS	American Nuclear Society
AOP	Abnormal operating procedure
ARV	Atmospheric relief valve
ASME	American Society of Mechanical Engineers
BMT	Basemat melt-through
BOC	Beginning-of-cycle
CCDP	Conditional core damage probability
CCFP	Conditional containment failure probability
CCU	Containment cooling unit
CD	Core damage
CDF	Core damage frequency
CET	Containment event tree
CHR	Containment heat removal
CIS	Containment isolation system
CST	Condensate storage tank
DC	Direct current
DCH	Direct containment heating
DDT	Deflagration-to-detonation transition
DET	Decomposition event tree
EAL	Emergency action level
ECCS	Emergency core cooling system
ED	Emergency director
EDG	Emergency diesel generator
EDMG	Extensive damage mitigation guideline
EF	Error factor
EOC	End-of-cycle
EOF	Emergency operations facility
EOP	Emergency operating procedure
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
EVSE	Ex-vessel steam explosion
FLEX	Diverse and Flexible Coping Strategies
FV	Fussell-Vesely
GE	General emergency

HD	High dependence
HEP	Human error probability
HFE	Human failure event
HPME	High-pressure melt ejection
HRA	Human reliability analysis
IE	Initiating event
ISGTR	Induced steam generator tube rupture
ISLOCA	Interfacing system loss-of-coolant accident
IVSE	In-vessel steam explosion
kV	Kilovolt
L3PRA	Level 3 Probabilistic Risk Assessment (project)
LERF	Large early release frequency
LMP	Licensing Modernization Project
LO4160V	Loss of power to 4,160V AC bus
LOCA	Loss-of-coolant accident
LOINV	Loss of inventory
LOI-OC	Loss of inventory outside of containment
LOOP	Loss of offsite power
LORHR	Loss of residual heat removal
LPSD	Low-power and shutdown
LRF	Large release frequency
MCCI	Molten core-concrete interaction
MCR	Main control room
MD	Moderate dependence
MDP	Motor-driven pump
MOC	Middle-of-cycle
MOV	Motor-operated valve
NLWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NSCW	Nuclear service cooling water
ODML	Overdrain entering mid-loop
ORNL	Oak Ridge National Laboratory
OSC	Operational support center
PDS	Plant damage state
PORV	Power-operated relief valve
POS	Plant operating state
PRA	Probabilistic risk assessment
PRT	Pressurizer relief tank
PWR	Pressurized-water reactor
RAT	Reserve auxiliary transformer
RCP	Reactor coolant pump

RCS	Reactor coolant system
RHR	Residual heat removal
RPV	Reactor pressure vessel
RVLIS	Reactor vessel level indication system
RWST	Refueling water storage tank
SAE	Site area emergency
SAG	Severe accident guidelines
SAMG	Severe accident management guideline
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SAT	Standby auxiliary transformer
SBO	Station blackout
SCG	Severe challenge guideline
SD	Shutdown
SFP	Spent fuel pool
SG	Steam generator
SGTR	Steam generator tube rupture
SI	Safety injection
SNL	Sandia National Laboratories
SOV	Solenoid-operated valves
SPAR	Standardized plant analysis risk
SSC	Structure, system, or component
TAG	Technical Advisory Group
TDAFW	Turbine-driven auxiliary feedwater
TSC	Technical support center
V	Volt

1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

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

- 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 (NRC, 1990), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown 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.

This report documents the single-unit, reactor at-shutdown, Level 2 PRA for internal events that supports the L3PRA project.

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 reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.²) 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. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

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 FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. 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 off site.

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: Final summary report

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

Volume 3: Reactor, at-power, internal event and flood PRA (overview report)

Volume 3a: Level 1 PRA for internal events

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 (overview report)

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, at-shutdown, internal event PRA (overview report)

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 (no overview report)

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

1.1 Modeling Assumptions and Limitations

The starting point of this Level 2 PRA is the corresponding Level 1 PRA, documented in (NRC, 2022a). That probabilistic logic model is extended (in an integrated, single-model fashion) to carry sequences beyond core damage, to their ultimate end-state in terms of radiological release to the environment. The Level 2 PRA's primary purpose is to provide a characterization of accident progression and radiological release suitable for use by the offsite consequence

analysis team in completing the third level of the Level 3 PRA. This situation is depicted simplistically in Figure 1-1. The middle three (black) boxes in Figure 1-1 are the domain of this report. 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?” (i.e., the third component of the “risk triplet”). To do this, the Level 2 PRA generally considers a broader range of phenomena, system failures, and operator actions.

The reactor at-shutdown Level 2 PRA heavily leverages the work previously done to develop the reactor at-power Level 2 PRA (NRC, 2022a). The Level 2 PRA follows the state-of-practice paradigm of the entire project, except in areas where there is no mature state-of-practice (e.g., post core damage human reliability analysis).

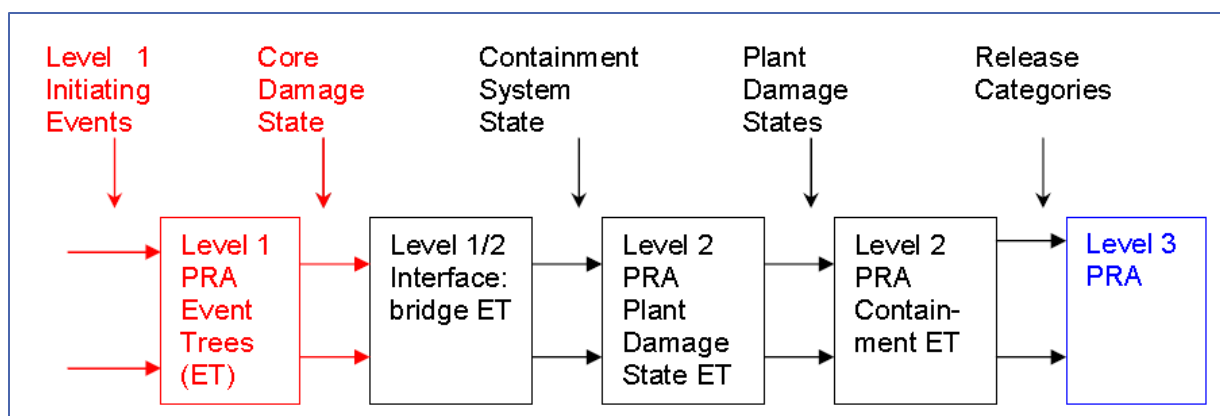


Figure 1-1 Flow of Information from Level 1 to Level 3 PRA

As of the time that the reactor-at-shutdown Level 1 and Level 2 PRAs were developed by NRC, the reference plant did not have a PRA with equivalent scope. However, the reference plant had previously performed some developmental work on a reactor-at-shutdown Level 1 PRA and provided a significant amount of supporting information (e.g., outage reports), and these information sources were leveraged.

The results of the Level 2 PRA presented in this report characterize the radiological releases for internal events for shutdown plant operating states (POSS) during a refueling outage. The scope of the model is limited to certain hazard categories, accident types, POSS, and outage types consistent with the corresponding shutdown Level 1 PRA (NRC, 2022a). Areas that are not addressed are identified and discussed in Appendix A. Table 1-1 summarizes the key modeling assumptions, scope areas, and limitations.

Table 1-1 Key Modeling Assumptions and Limitations

Item	Description
Low power and hot standby states	Consistent with the Level 1 PRA, low power and hot standby operating states are not modeled. This model has the potential to under-estimate radiological release frequency from this standpoint. The estimated CDF contributions from low power and hot standby states are provided as a sensitivity study in the shutdown Level 1 PRA (NRC, 2022a).
Fires, floods, and external hazards	Consistent with the Level 1 PRA, only internal events are modeled. Hazards such as fire, flooding, seismic, and other hazards while at shutdown are not captured. These are potentially important for the subject site, and this model may under-estimate the actual radiological release frequencies from this standpoint.
Sitewide impacts	The other operating unit (i.e., Unit 2) and the two operating spent fuel pools are not modeled. These are potentially important for the subject site due to coupling of accident management resources, and to a lesser degree, potential spatial interactions. The other operating unit and the two spent fuel pools are modeled as part of the L3PRA project integrated site risk study, which is documented in a separate report.
Airborne pathway	The focus of this model is airborne radiological releases only, and modeling decisions (e.g., release category selection) are made as such. In any cases where surface aqueous releases are estimated (e.g., spilling when the lower containment is flooded and the containment rupture point is low), only airborne releases are passed to the offsite consequence analysis.
Outage types	Consistent with the Level 1 PRA, only a refueling outage is modeled. Other outage types (e.g., emergent maintenance outage) would share many of the same characteristics as the modeled generic refueling outage but could introduce other un-modeled considerations. The estimated CDF contributions from other outage types are provided as sensitivity studies in the shutdown Level 1 PRA (NRC, 2022a).
Surrounding structures outside containment	Modeling of the surrounding structures is 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 and modeling of internal structures) of their treatment within the MELCOR model.
System modeling for severe accident mitigation	Detailed fault tree and equipment reliability modeling is not performed for some systems and components used to support post-core-damage accident mitigation, as discussed further in Section 2.4.1. This is due to the lack of routinely collected data for these systems and components and the expectation that the overall system or component reliability is dominated by the human reliability associated with its deployment. This limits the use of the PRA in assessing the importance of these systems and components, such as the impacts of maintenance unavailabilities.
Survivability	Survivability of equipment is handled qualitatively, as discussed in Section 2.3.6. As such, the model may be limited in assessing the merit of activities related to hardening equipment.
Recriticality	The PRA does not include consideration of inadvertent criticality during core reflood, which is relevant when the core is reflooded with unborated water following significant heatup of the fuel but prior to relocation. ³ Credit is assigned in some sequences for core reflood to arrest core damage, which might use a borated or unborated water source. Potential injection of unborated water is assumed to not present any criticality issues.

³ The current accident management approach is to inject water regardless of re-criticality potential, though this potential might affect the flow rate used.

Table 1-1 Key Modeling Assumptions and Limitations (cont.)

Item	Description
Numerous modeling assumptions	There are many modeling assumptions made that can influence the analysis results. These are identified throughout the documentation. The associated model uncertainties are identified, and their impacts are assessed qualitatively and in some cases quantitative sensitivity studies are performed. Section 2.4.7 describes the uncertainty characterization of the probabilistic treatment of accident progression, and Section 2.5.3 describes uncertainties associated with radiological source terms. An overall summary of model uncertainties is presented in Section 2.6.
FLEX strategies	This study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The implementation of FLEX is expected to have a positive impact on reducing release frequencies. The impacts of FLEX strategies are explored in the Level 3 PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

1.2 Model Computer Codes and Revisions

Table 1-2 below identifies the major models and computer codes used in the development of the Level 2 PRA, provides information about the code and model versions used, and provides other relevant information about where the analyses are documented.

Table 1-2 Model and Computer Code Versions

Item	Model version-of-record	Code version-of-record
ORIGEN analysis	ORIGEN model as of April 2013	SCALE 6.1.2 with updated ENDF/B-VII.1 nuclear data
MELCOR accident sequence analysis, source term development, and phenomenological investigations	Shutdown MELCOR Model Revision 2 (which is built from At-Power MELCOR Model Revision 7 and shutdown accident analyses described in Section 2.3.3)	MELCOR version 2.2.9607
SAPHIRE probabilistic risk assessment software	Final quantification with model version 421	SAPHIRE 8.2.6

The two computer codes used most prominently in this report are MELCOR and SAPHIRE. MELCOR's use is discussed predominantly in Sections 2.3.2 through 2.3.5, but there are a few key concepts worth introducing here. 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 models specify the geometry of major structures, systems, and components; the nominal operating conditions; and the response of major automatic control and protection systems. Initial and boundary conditions are specified for each calculation to define the particular plant configuration (including equipment and operator successes and failures) that is 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, and is used here to construct and solve the bridge tree (Section 2.1.1), plant damage state tree (Section 2.1.2), containment event tree (Section 2.4.2), and decomposition event trees (Section 2.4.3). This is all done in the same SAPHIRE project as the Level 1 PRA, resulting in a linked Level 1 and Level 2 PRA model. As discussed later, the plant damage state tree is strictly a sorting tool to group similar sequences, whereas the bridge and containment trees are akin to Level 1 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 is assigned to one of a few end-states, each of which in turn 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 are used to define how branch assignments are 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 are used to define what assumptions SAPHIRE makes when quantifying fault trees.

1.3 Structure of Documentation

The report is structured to align with the technical elements as outlined in the requirements of the Level 2 PRA Standard for trial use and pilot application (ASME, 2014). Each technical element is addressed in Section 2. A summary of the project results, insights, and conclusions is provided in Section 3. Section 4 contains the references.

Several appendices are included with the report to document additional details of the analysis. Appendix A describes areas that are considered candidates for future work beyond this study. Appendix B includes tables of results, including cutsets and importance measures. Appendix C contains descriptions of the uncertainty distributions used in the parameter uncertainty analysis. Appendix D describes the human reliability analysis (HRA) performed for the Level 1/Level 2 bridge event tree. Appendix E describes the HRA performed for post-core-damage actions. Appendix F presents the approach for modeling combustion in containment.

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, 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.

2 LEVEL 2 PRA TECHNICAL ELEMENTS

The Level 2 PRA Standard for trial use and pilot application (ASME, 2014) was consulted to ensure the complete set of requirements for a technically acceptable Level 2 PRA are addressed in this study. The report sections are structured to address the technical elements, as follows:

- Section 2.1 Level 1/Level 2 PRA Interface
- Section 2.2 Containment Capacity Analysis
- Section 2.3 Severe Accident Progression Analysis
- Section 2.4 Probabilistic Treatment of Accident Progression
- Section 2.5 Radiological Source Term Analysis
- Section 2.6 Evaluation and Presentation of Results
- Section 2.7 Level 2/Level 3 PRA Interface

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

The objective of the Level 1/Level 2 PRA interface is to develop the methods for transferring information between the Level 1 PRA evaluation of core damage frequency (CDF) and the accident progression analysis that is treated in the Level 2 PRA. The Level 1/Level 2 PRA interface consists of five interrelated steps:

1. Development of the bridge tree (Section 2.1.1)
2. Development of plant damage state (PDS) binning (Section 2.1.2)
3. Review of the resulting plant damage states (Section 2.1.3)
4. Iteration on the Level 1 PRA modeling as necessary (Section 2.1.4)
5. Development of criteria for, and selection of, representative sequences (Section 2.1.5)

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 that will be used to identify a manageable number of sequences for deterministic investigation. The objective of the third step is to review the resulting PDSs to ensure adequate transfer of information across the Level 1/Level 2 PRA interface, such that information important to the Level 2 PRA (e.g., initiator and support system dependencies and operator action dependencies) is transferred and that credit is not being given for equipment or operator actions that are not appropriate for each PDS. The objective of the fourth step is to re-visit and refine any Level 1 PRA modeling assumptions which adversely affect the PDS binning (e.g., lack of adequate distinction in different accident evolutions arising from failure-to-run of auxiliary feedwater for a particular sequence(s) where that largely affects the timing of core damage). The objective of the fifth step is to establish the criteria that will be used for the selection of representative sequences and selecting those sequences for each PDS.

Table 2-1 shows the results of the Level 1 PRA, as reproduced from (NRC, 2022a), in terms of the contribution of the various initiating events to the shutdown CDF. The CDF results are calculated in units per year (i.e., they account for the expected annual fraction of time spent in each POS). The Level 1 PRA abbreviated names are defined below. The probabilistic logic model that generated these results is used in the Level 2 PRA quantification, though as will be discussed later, provisional PDS quantification was performed using a predecessor to the final model.

Shutdown Level 1 PRA initiating event definitions:

- LORHR – loss of residual heat removal (RHR) cooling
- LOINV – loss of inventory, which includes the following subcategories:
 - LOINV-OP – loss of inventory due to reactor coolant system (RCS) pressure excursion resulting in lifting of pressure relief valves. These events are applicable only in POSs with the RCS intact. This scenario uses a representative loss of inventory flow rate of 500 gpm.
 - LOINV-SP – loss of inventory due to a small break or pressure boundary failure resulting in a spray event with break flow rate up to 100 gpm.
 - LOINV – loss of inventory due to a break or pressure boundary failure resulting in a break flow in the range of 100 to 2000 gpm.
 - LOINV-LG – loss of inventory due to a large break or pressure boundary failure resulting in break flow greater than 2000 gpm.
- ODML – overdraining RCS level, which includes the following subcategories:
 - LOINV-DS – loss of inventory due to human errors resulting in inadvertent flow diversions and reducing the RCS inventory level available for cooling the core.
 - ODML-EARLY – Overdraining the RCS level when entering mid-loop condition prior to refueling.
 - ODML-LATE – Overdraining the RCS level when entering mid-loop condition after refueling.
- LO4160V – loss of power to 4,160V alternating current (AC) bus
- LOOP – loss of offsite power
- LONSCW – loss of nuclear service cooling water (NSCW)

Table 2-1 Level 1 PRA CDF Results by Plant Operating State

IE Categories: IE Frequency: (per shutdown- year) Shutdown POS	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr) (Note 10)
	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
POS 3 Mode 4 cooldown	< 1E-12	(Note 9)	N/A	(Note 9)	2.3E-08	(Note 9)	2.3E-08	0.2%	N/A
POS 4 Mode 5 cooldown	2.4E-12	(Note 9)	N/A	(Note 9)	1.7E-07	(Note 9)	1.7E-07	1.3%	N/A
POS 5A Pressurizer solid	3.9E-11	7.8E-07	N/A	(Note 9)	1.3E-06	(Note 9)	2.0E-06	16.3%	27
POS 5B Draining RCS	9.9E-08	2.2E-08	1.2E-08	2.1E-08	4.9E-07	1.1E-07	7.5E-07	5.9%	5.3
POS 6 Mid-loop (pre-refuel)	1.3E-07	7.7E-09	1.2E-06	1.4E-08	2.2E-07	3.7E-08	1.6E-06	13.0%	3.3
POS 7 Filling refuel cavity	2.7E-08	1.2E-11	N/A	1.7E-08	1.6E-07	(Note 9)	2.0E-07	1.6%	52
POS 8E Refuel cavity flooded (pre-refuel)	1.4E-07	1.1E-10	N/A	1.8E-07	2.7E-06	1.9E-07	3.3E-06	26.0%	90
POS 8L Refuel cavity flooded (post-refuel)	1.0E-07	1.3E-11	N/A	1.0E-07	2.1E-06	1.5E-07	2.4E-06	19.3%	214
POS 9 Draining refuel cavity	5.6E-08	1.7E-08	8.6E-09	1.6E-08	2.8E-07	8.5E-08	4.7E-07	3.7%	15
POS 10 Mid-loop (post- refuel)	7.4E-08	1.4E-08	2.3E-07	2.5E-08	3.1E-07	7.1E-08	7.2E-07	5.8%	11
POS 11 Vacuum fill RCS	4.6E-08	(Note 9)	N/A	(Note 9)	2.3E-07	(Note 9)	2.7E-07	2.2%	N/A
POS 12 Mode 5 heat-up	6.4E-12	(Note 9)	N/A	(Note 9)	4.2E-07	(Note 9)	4.2E-07	3.4%	177

Table 2-1 Level 1 PRA CDF Results by Plant Operating State (cont.)

IE Categories: IE Frequency: (per shutdown- year) Shutdown POS	LORHR	LOINV	ODML / LOINV-DS	LO4160V	LOOP	LONSCW	Total POS CDF (per year)	% Contr. to CDF	Time to CD (hr) (Note 10)
	3.7E-1	6.1E-1 ⁽¹⁾ 3.0E-3 ⁽²⁾ 6.4E-5 ⁽³⁾ 1.2E-5 ⁽⁴⁾	2.4E-3 ⁽⁵⁾ 2.1E-3 ⁽⁶⁾ 2.1E-2 ⁽⁷⁾	3.4E-1	1.6E-1	3.5E-5 ⁽⁸⁾			
POS 13 Mode 4 heat-up	1.3E-12	(Note 9)	N/A	(Note 9)	1.6E-07	(Note 9)	1.6E-07	1.2%	N/A
Total:	6.7E-07	8.4E-07	1.5E-06	3.7E-07	8.5E-06	6.4E-07	1.3E-05		
Note (1) Loss of inventory overpressure event frequency 6.1×10^{-1}/shutdown-year is applied in POS 5A. Note (2) Loss of inventory spray event frequency 3.0×10^{-3}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (3) Loss of inventory event frequency 6.4×10^{-5}/shutdown-year is applied in POS 5B, 6, 9, and 10. Note (4) Loss of inventory initiating event frequency 1.2×10^{-5}/shutdown-year is applied in POS 7, 8E, and 8L. Note (5) Frequency of mid-loop entry in POS 6 multiplied by conditional probability of overdrain. Note (6) Frequency of mid-loop entry in POS 10 multiplied by conditional probability of overdrain. Note (7) Loss of inventory due to displacement frequency 2.1×10^{-2}/shutdown-year is applied in POS 5B and 9. Note (8) Frequency calculated by fault tree is approximately 3.5×10^{-5}/shutdown-year, but varies by POS. Note (9) This accident scenario was not modeled. Note (10) Time to core damage is estimated for a LORHR scenario with no operator intervention. The time to core damage was not evaluated for all POS conditions, as noted by "N/A" in the table.									

2.1.1 Development of the Bridge Tree

This step functionally adds additional top events associated with containment systems onto the end of the Level 1 PRA event trees, forming the first part of the bridge between the Level 1 and Level 2 PRAs. This is achieved by developing a transfer event tree, which is commonly referred to as a bridge tree. While this tree is a distinct event tree, use of SAPHIRE's transfer option functionally makes it a continuation of the Level 1 PRA event trees. The top events for the containment systems define the initial availability or unavailability or success or failure of these systems at the time of core damage, based on the support system information resident in the Level 1 PRA model.⁴ Addressing this prior to plant damage state binning allows for consideration of containment system functionality prior to parsing cutsets into bins.

Unlike the at-power model wherein containment systems would typically be available barring their unavailability or unreliability, during shutdown, plant technical specifications allow for these systems to be out of service. As such, the reactor at-shutdown bridge tree addresses both the availability state of the systems, as well as (when relevant) their reliability. The other major difference from the at-power equivalent modeling is the addition of a top event related to containment closure, which includes manual actions to close containment. These actions occur prior to core damage but are not within the scope of the Level 1 PRA modeling. For this reason, the Level 1 PRA team was extensively involved with the development of the containment closure modeling, while the Level 2 PRA team extended and implemented the modeling.

The following top events are treated in the bridge tree (which is shown in Figure 2-1):

1. containment closure (addressing closure of equipment hatch and personnel airlock)
2. containment isolation system (CIS) (addressing isolation of other containment penetrations)
3. containment spray system
4. containment cooling unit (CCU) system (also known as containment fan coolers)

The basic approach for modeling these items involved: (i) defining the availability of these systems during the various POSs, (ii) leveraging the reference plant's model and documentation for the three engineered systems, and (iii) incorporating the manual containment closure work performed by the NRC Level 1 PRA team. The treatment of data, human error, and common-cause failure aspects of porting over the reference plant's modeling for these systems was handled in the same manner as for the at-power Level 1 PRA (NRC, 2022b).

During shutdown operations, the containment configuration can change throughout the outage. The model accounts for the POS-specific initial conditions and human actions required for containment closure and isolation. The human failure events and dependency between those

⁴ The terminology used to describe the up and down branches 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 its function 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 was not yet demanded but it will be and will succeed or fail; and sometimes it never will be demanded).

events are the dominant contributors to failure for the majority of shutdown sequences. Refer to Appendix D for the detailed assessment of the bridge tree human failure events and dependencies. The dependency analysis involves replacing the nominal human error probability (HEP) values for dependent human failure events (HFEs) in the cutsets with dependent HEP values, some of which are fairly large (e.g., some dependent failures to isolate or close containment have a failure probability of approximately 0.5). Key aspects of the modeling of the bridge tree top events include:

Containment closure

- The bridge tree linkage rules use Level 1 PRA sequence information to categorize accident sequences in terms of whether containment would already be closed, and if not, the time available to close containment. Based on this information, category-specific fault trees are used to assign placeholder basic events. Post-processing rules are then used to replace these placeholder events with the actual applicable basic event.

Containment isolation

- The bridge tree linkage rules use the CIS model (fault tree) for POSs where the system would be in-service, which is limited to only a small portion of shutdown operations (i.e., POS 3 and POS 13).
- POS-specific fault trees are developed to account for the status of the containment purge system. The containment purge lines are the largest openings that are considered for potential containment isolation failure pathways. The main purge lines include two 24-inch diameter containment penetrations, and the mini-purge lines include two 14-inch diameter containment penetrations. Based on review of the reference plant's procedures and operating history, only the mini-purge system is assumed to be operated during portions of the outage. A human failure event is developed for isolating the mini-purge lines for POSs when the system is in operation, and a pre-initiator failure to restore the purge system to a closed state is modeled for POSs when the system was previously operating.
- Several small containment penetrations (e.g., equivalent diameter less than 2 in.) may be opened during shutdown POSs. All containment penetrations are tracked using procedural guidance and checklists. The containment isolation fault trees include human failures and valve failures that can result in a small containment isolation failure. These are the dominant failure modes for the POS-specific fault trees.
- The containment isolation fault trees also include the pre-existing containment tear passive failure that was modeled in the at-power Level 2 PRA (NRC, 2022a).

Containment spray

- The bridge tree linkage rules use the relevant system model for the few POSs where the sprays would be in-service, and otherwise assume sprays are unavailable.

Containment coolers

- The bridge tree linkage rules use the relevant system model, with varying assumptions about the availability of one, both, or neither train depending on the POS. Both trains are initially operating during POS 3. Neither train is operating in POSs 4–9. One train is operating in POSs 10–13. For POSs 4–9, recovery of coolers is considered.

Note that the bridge tree excludes some containment systems; namely, electric hydrogen recombiners and containment ventilation and purge systems. The former is excluded because they are retired in place at the reference plant. The latter are excluded because they are 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. While these systems can potentially have a positive impact on mitigating releases, their applicability and availability during severe accidents is expected to be very limited.⁵ In the current full Level 2 PRA and human reliability analysis (HRA), the actions related to containment venting do not end up being prompted (and thus modeled). This is partially a reflection of simplifications in the post-core-damage accident management modeling and the associated human reliability modeling is discussed later in Sections 2.4.4 and 2.4.5.

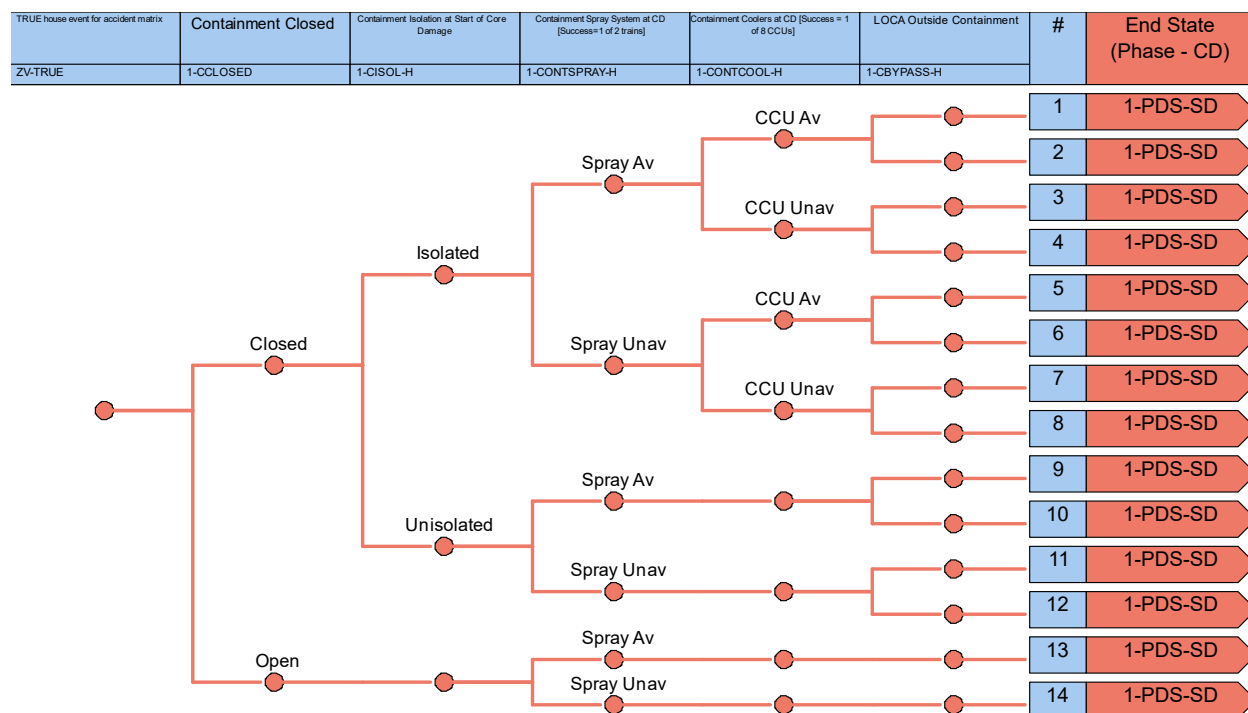


Figure 2-1 Shutdown Bridge Tree (1-L1E-BRIDGE-N-SD)

⁵ As discussed in SECY-16-0041 (ML16049A088), the NRC's post-Fukushima Near-Term Task Force recommended assessing requirements associated with containment venting (for other than Mark I and II containments) and hydrogen control. The staff's analyses support the conclusions that additional capabilities in these areas would not provide substantial safety enhancements and additional regulatory actions are not warranted.

2.1.2 Development of Plant Damage State Binning

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

Figure 2-2 Schematic of the Role of PDS Binning

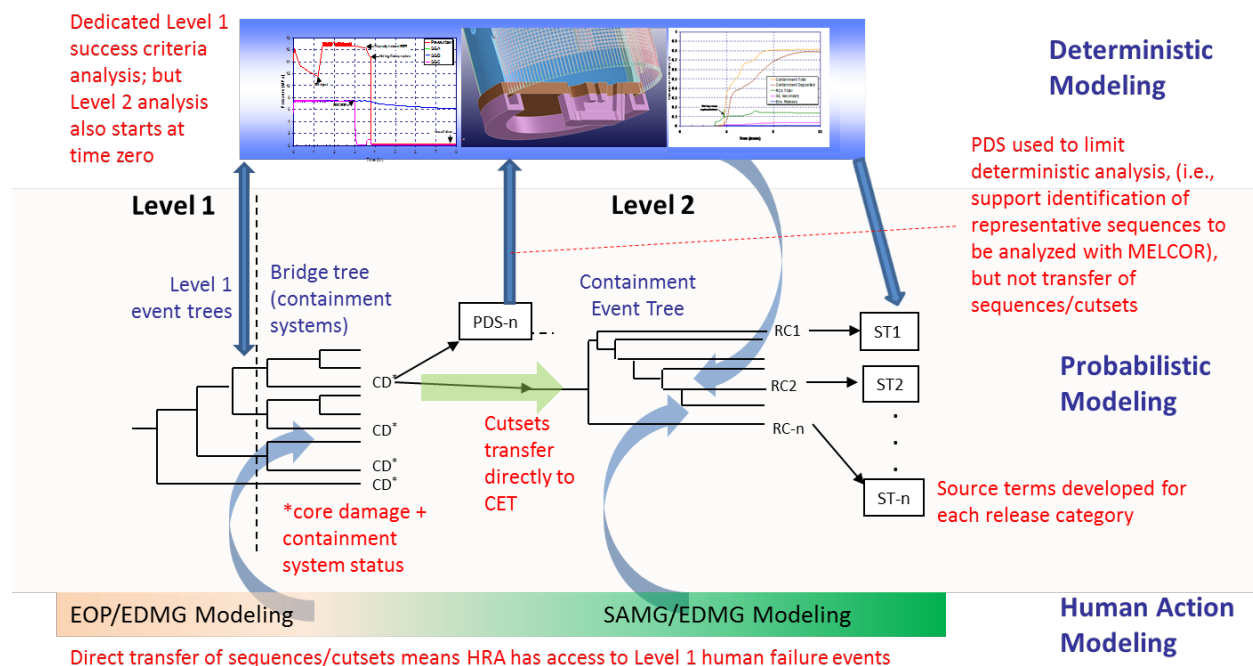


Figure 2-2 Schematic of the Role of PDS Binning

Specifically, the PDSs are used to prioritize and select representative scenarios, which in turn provide the underlying basis for the following:

- narrative understanding of post core-damage response
- Level 2 PRA sequence timing
- phenomenological evaluations
- survivability and habitability determinations
- HRA
- source terms characteristics
- starting points for some model uncertainty sensitivity analyses

All shutdown initiators modeled in the Level 1 PRA are propagated to the PDSs, and ultimately to the containment event tree.

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

- RCS status
- timing relative to refueling (pre- versus post-)
- accident type
- steam generator cooling availability/success
- refueling water storage tank (RWST) availability/success
- emergency core cooling system (ECCS) availability/success
- sump availability/success
- containment closure status –*included in the PDS quantification tree*
- containment isolation status – *included in the PDS quantification tree*
- containment sprays status –*included in the PDS quantification tree*
- containment cooling units status –*included in the PDS quantification tree*

This PDS scheme includes the four PDS top events and the three bridge tree top events from the at-power model. It also includes the addition of the containment closure status in the reactor at-shutdown bridge tree. Additional top events of importance to shutdown are included; namely, the RCS configuration at the time of the initiating event (intact versus vented versus vented and flooded), timing relative to refueling (which affects decay heat load and the amount of used fuel in the reactor pressure vessel [RPV]), and sump availability.

When quantifying the model for PDS frequencies, the 1-PDS-Q-SD tree is used, which explicitly merges the 1-PDS-SD tree and the bridge tree (1-L1E-BRIDGE-N-SD) and has end-states associated with the PDS bins. When quantifying the model for release frequency, core damage sequences are routed to the bridge tree, from there to the 1-PDS-SD tree, and from there to the containment event tree. For illustration, the “Not Vented” portion of the 1-PDS-SD tree (end-states 1 through 50) is shown Figure 2-3. The remaining two sections of the tree (“Vented” and “Flooded”), covering end-states 51-100 and 101-150, have an identical structure. A transfer tree (1-CD-SD-XFER) is used within the current model to readily switch between core damage quantification, PDS quantification, PDS linkage rule debugging, and release frequency quantification.

Note that PDS binning, while represented as an event tree, does not have probabilities associated with the pathways (i.e., it sorts cutsets into bins rather than adding additional conditional probabilities). On the other hand, the containment status top events addressed in the bridge tree (1-L1E-BRIDGE-N-SD) do introduce success and failure probabilities. The PDS quantification tree (1-PDS-Q-SD) combines the Level 1 event trees, containment status probabilities, and the PDS binning logic to quantify the PDS end states.

Table 2-2 provides a crosswalk of the modeling aspects covered by Supporting Requirements L1-A1 through L1-A3a of the Level 2 PRA trial use standard (ASME, 2014).

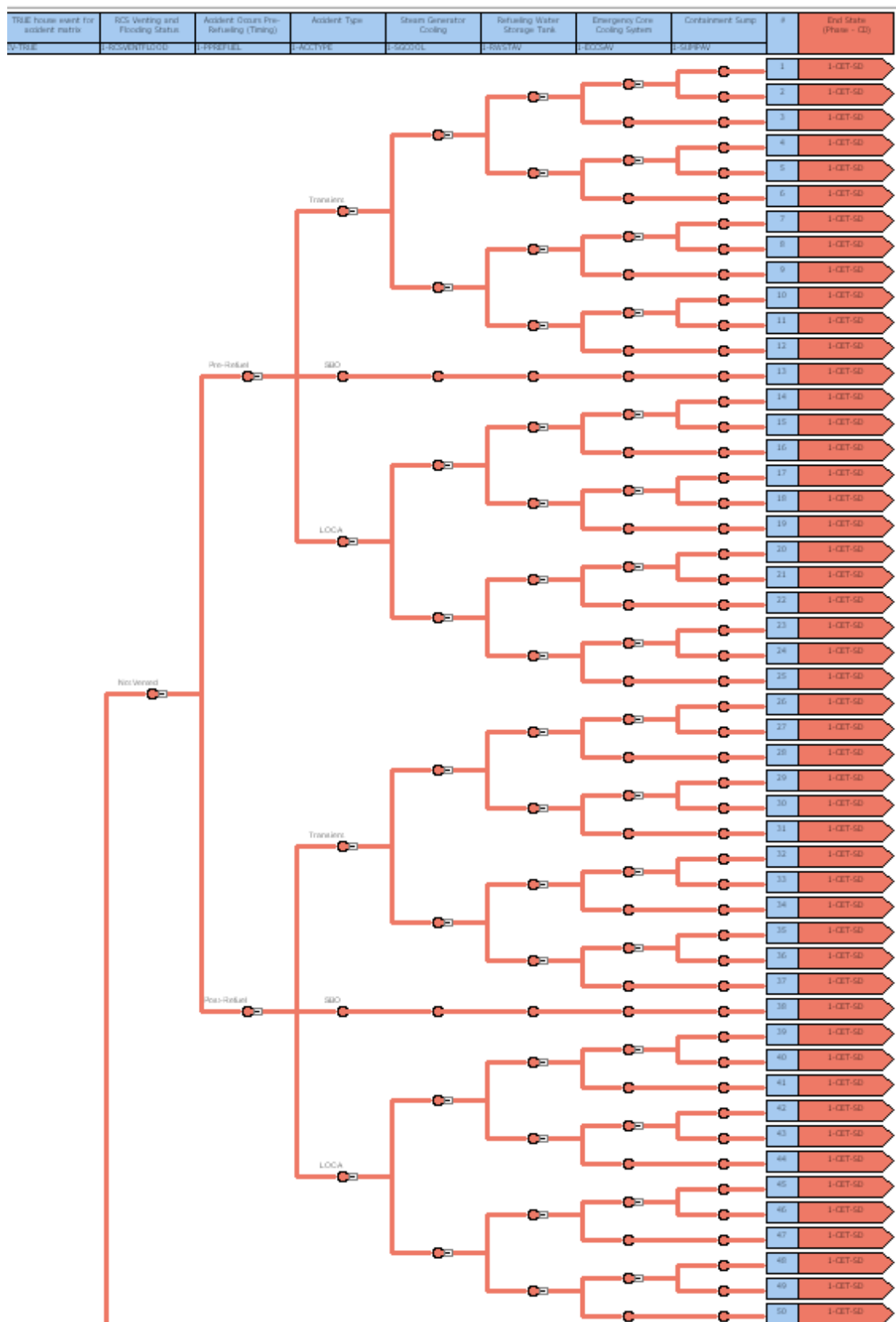


Figure 2-3 The “Not Vented” portion of the Plant Damage State Tree (1-PDS-SD)

Table 2-2 Cross-Walk of Physical and Accident Sequence Characteristics from HLR L1-A

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)
RCS pressure	Type of initiating event and subsequent accident sequence characteristics (i.e., accident type, operator depressurization, and availability of steam generator cooling).	The RCS is always de-pressurized during the POSs modeled. The RCS status top in the PDS tree (namely whether the RCS is vented) addresses whether the system has the potential to re-pressurize.
RCS configuration	RCS status in terms of being vented or not vented, status of head removal, and status of flooding the in-containment refueling cavity, all of which are determinable from the POS.	This aspect is directly addressed by the PDS top event associated with RCS status.
ECCS status	General availability is determinable from the POS; sequence-specific availability is generally determinable from the 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	• POS • System dependencies on supporting systems • System availability	The containment isolation system availability, dependencies, and status (i.e., system component failures) are modeled in the supporting fault tree logic for the containment isolation top event in the bridge tree.
Containment heat removal status	• POS • System dependencies on supporting systems • System availability	The containment cooling unit system availability, dependencies, and status (i.e., system component failures) are modeled in the supporting fault tree logic for the CCU top event in the bridge tree.
Containment integrity	• POS and any subsequent actions to close a previously open containment • Induced containment failure is within the scope of the containment event tree (CET). Containment isolation is covered above.	This is addressed in the containment closure top event in the bridge tree, and containment integrity is addressed in the CET linkage rules for various top events.
Steam generator pressure	POS	SG status is addressed in the PDS top event for SG cooling.
Steam generator water level	POS	SG status is addressed in the PDS top event for SG cooling.

Table 2-2 Cross-Walk of Physical and Accident Sequence Characteristics from HLR L1-A (cont.)

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	The potential for induced steam generator tube rupture is modeled in a manner consistent with the at-power PRA (NRC, 2022a), but there are only limited states where this is applicable (POS 3 and POS 13).	Since the model only deals with shutdown states with limited SG availability during a routine refueling outage, SG rupture does not contribute to any significant sequences.
Containment pressure	Containment pressure is addressed indirectly by the characteristics governing containment integrity/closure and containment heat removal (see above).	Containment pressure is handled indirectly via the containment status and containment systems status.
Availability/accessibility of mitigating equipment	No information on EDMG equipment is passed from the Level 1 PRA since EDMGs are not generally addressed in the Level 1 PRA.	This is handled within the CET (and the supporting HRA).
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 if LOOP initiating event occurs or status of NSCW system is unavailable if an SBO sequence or LONSCW initiating event occurs). The CET indirectly receives this information because the CET linkage rules query the PDS tree sequence logic. Detailed information about support system availability is passed on to the Level 2 PRA via the Level 1 PRA cutsets; however, this information is not available for the Level 2 probabilistic sequence modeling. Meanwhile, support system information is manually addressed in the Level 2 PRA deterministic analysis (e.g., assuming a support system success or failure for a particular representative sequence by scrutinizing the Level 1 PRA cutsets).
Time of core damage	The timing of core damage is a function of the POS, the initiating event, and the subsequent Level 1 PRA successes and failures—no explicit core damage timing is passed through the Level 1 to Level 2 interface.	The role of core damage timing in the Level 2 PRA model is handled via the CET's use of the Level 2-specific deterministic accident progression modeling, which 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 (when relevant).

Table 2-2 Cross-Walk of Physical and Accident Sequence Characteristics from HLR L1-A (cont.)

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)
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	Not considered because the current model does not include any accident sequences which involve the auxiliary building or other surrounding structures.	Not treated.
Physical effects of flooding	Flooding is not relevant because the current scope of the shutdown model does not include internal flooding.	Not treated.

2.1.3 Review of the Resulting Plant Damage States

Two phases of PDS binning are performed and these are discussed in the two sub-sections that follow. The first, termed the provisional binning, is done early in the PRA development in order to guide selection of representative sequences which are in turn used to guide evaluations using MELCOR, human reliability analysis, etc. Inevitably, changes to the Level 1 PRA, bridge tree, and PDS tree occur during subsequent model development, and so PDS bins are also quantified at the end in order to illustrate to what extent the final model deviates from the version used for the provisional binning. Section 2.4 provides additional detail about these intervening changes.

Provisional PDS Binning to Facilitate Model Build-Out

The provisional PDS binning was performed using SAPHIRE version 8.1.5 and a modified version of the version 316 model. A truncation of $10^{-12}/\text{yr}$ was used for the quantification. In the quantification, 76 PDS bins (of a possible 1,200) had a non-zero frequency. Of these, the top 32 PDS bins cumulatively contributed 99.9 percent of the total PDS frequency, with 12 bins having an individual frequency greater than $10^{-7}/\text{yr}$. This situation is depicted in Figure 2-4.

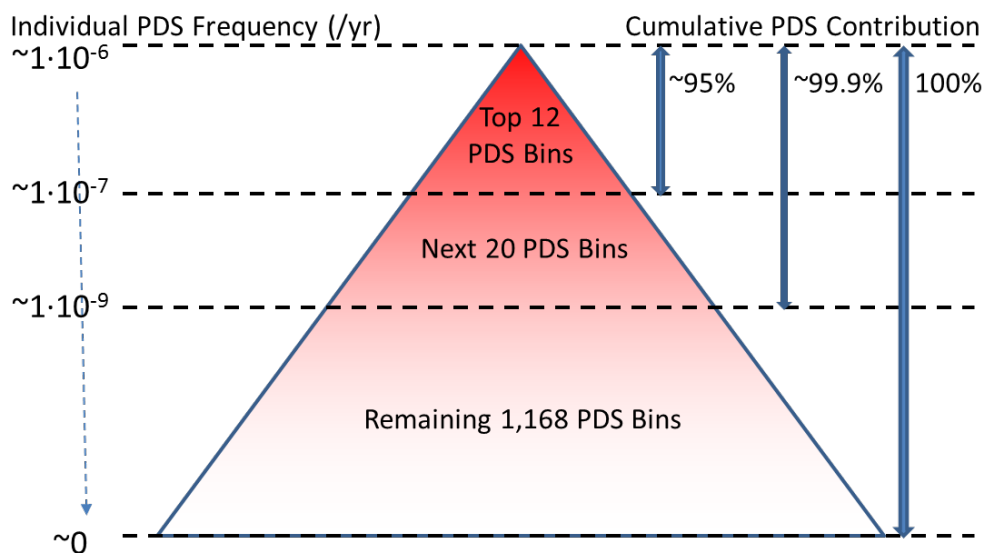


Figure 2-4 Schematic Showing Breakdown of PDS Frequency

The top contributing provisional PDS bins had the following features:

- The containment hatch is open (i.e., open prior to the accident and not successfully closed prior to core damage) for approximately 8 percent of the total shutdown CDF.
- The containment is unisolated (i.e., hatches closed but one or more penetrations open) for approximately 0.1 percent of CDF.
- Containment sprays are available for only approximately 0.1 percent of CDF.
- Containment coolers are available for approximately 6 percent of CDF.

- The RCS has the reactor vessel head removed and is flooded for 59 percent, vented with level ranging from mid-loop to the pressurizer for approximately 19 percent, and intact for the remainder of the shutdown CDF.
- The contribution to core damage is approximately 64 percent during pre-refueling and approximately 36 percent during post-refueling periods.
- Station blackout (SBO) accidents and loss-of-coolant accidents (LOCAs), as these are defined in the PDS accident type, make up approximately 77 percent and 23 percent of low-power and shutdown (LPSD) CDF, respectively. The LOCA PDS designation includes loss of shutdown cooling transients with successful RCS injection and an RCS opening (i.e., RCS vented, vessel head removed or failed open relief valve). Non-LOCA transients comprise the small remainder of shutdown CDF.
- Note that since LOINV initiators contribute only approximately 7 percent to the total CDF, most of the LOCA contribution here is due to a vented RCS or removed RPV head rather than an actual pipe break. While these accidents are classified as LOCAs in the PDS scheme, the representative sequence development considers these as loss of heat removal transients in defining the sequence attributes.
- Steam generator (SG) cooling is unavailable for 100 percent of non-SBO CDF.
- The RWST is unavailable for approximately 90 percent of non-SBO CDF.
- The ECCS is unavailable for approximately 9 percent of non-SBO CDF.
- The sump is unavailable for 100 percent of non-SBO core damage accidents in which the ECCS is available.
- Sequences in which the ECCS may operate following core damage (i.e., non-SBO sequences with ECCS available and either RWST or sump available) make up only approximately 0.2 percent of CDF. While later revisions to the model show a much high availability of ECCS for non-SBO sequences, this is still limited due to the predominance of successful ECCS injection sequences in the Level 1 PRA, making injection unavailable following core damage after the RWST inventory is expended.

Note that these results were based on an early version of the shutdown PRA. The contributions do not correspond directly to the final model version results. However, many of the dominant contributions and insights remain unchanged. The quantitative PDS results and the observations above were used to define the initial set of accident progression sequences to be analyzed deterministically. Additional accident sequence cases and sensitivity cases were analyzed as model revisions and revealed additional scenarios that were important to the results. The accident sequences are described in Section 2.1.5. The PDS binning is performed as a convenient pinch-point for making modeling assumptions and guiding deterministic analysis, but all Level 1 PRA sequences and cutsets are carried forward to the containment event tree (CET).

2.1.4 Iteration on the Level 1 PRA Modeling as Necessary

Development of the reactor-at-shutdown Level 1 and Level 2 PRA models required some iteration. This iteration had largely to do with:

- modifications to the Level 1 PRA accident sequence logic's use of TRUE/FALSE branches so as to produce sequence logic consistent with the PDS binning linkage rule writing
- translating ambiguities in the Level 1 PRA accident sequence logic to PDS binning assumptions and boundary conditions for the representative sequence MELCOR analysis (e.g., assumptions about core damage occurrence for slowly developing sequences, particularly during post-refueling POSs)
- resolution of reviewer comments on the Level 2 PRA that led to changes in the Level 1 PRA (e.g., inclusion of additional initiating event categories in the Level 1 PRA model)

Iteration of the models and results can change the risk insights and reveal areas where additional study is needed, or not, due to increased, or lowered, risk contributions after model revisions. For example, the revisions to the reactor at-shutdown Level 1 PRA involved improvements to post-refueling HRA and other conditions that resulted in reductions to the CDF contributions for post-refueling POSs. Had this information been known earlier in the process, the results could have led to a different initial selection of accident sequences to be analyzed with MELCOR. Having a stable, peer reviewed Level 1 PRA as a starting point for the Level 2 PRA can help to avoid inefficiencies in the development process, even though some level of iteration may be unavoidable.

2.1.5 Criteria for, and Selection of, Representative Sequences

The previous steps take thousands of cutsets and bin them into a smaller group (hundreds) of plant damage states. From each of the significant provisional plant damage states (as defined below), one or more "representative" sequences are 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, 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 (e.g., to provide a narrative of post-core-damage behavior and support HRA) is discussed further in Section 2.1.2

While the representative sequences are intended to encompass the individual traits associated with the Level 1 PRA cutsets, it is often difficult to anticipate the effects of a particular assumption across the full scope of a Level 2 PRA. Therefore, Level 2 PRA team judgment is necessarily a part of representative sequence selection. In some cases, variability is addressed through the identification of new sources of model uncertainty or identification of sensitivity cases (see Section 2.6.3 for discussion of model uncertainties and sensitivity studies).

Provisional plant damage states were selected for deterministic evaluation (i.e., were translated to representative sequences or sensitivities) if they met at least one of the following conditions:

- comprise an important portion of PDS frequency

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

The detailed deterministic specification (covered 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 particular PDS, conditions that would generally be expected to pessimistically affect radiological source term were chosen, or else the modeling selection was specifically identified as a model uncertainty. In most cases, however, the boundary conditions were able to be defined in a way to realistically represent the PDS conditions without bias in a conservative direction.

A total of 13 base representative sequences, several permutations of those sequences, and several sensitivity analyses were simulated. The representative sequences and sensitivities capture the scope of information as dictated by the probabilistic model results and provide a population of candidate release category representative source terms. The full list of cases is provided in Table 2-3 below, while Table 2-4 provides a high-level mapping of these calculations to key PDS attributes (POS, initiator type, and containment status at the onset of core damage). As can be seen, the calculations performed span a broad range of the operating states, initiator types, and containment status.

PDS Binning Associated with the Final Level 2 PRA Model

After the provisional PDS results were generated, several model changes were made to both the Level 1 and Level 2 PRA models in finalizing the CDF and release frequency results. The final PDS results were quantified with the final shutdown Level 2 PRA model version 421. Key differences between the final PDS results versus the provisional PDS results are:

- The PDS event tree was modified to include a PDS to account for sequences involving losses of RCS inventory outside of containment (i.e., interfacing system LOCAs [ISLOCAs]), which accounts for 1 percent of the total PDS frequency.
- Station blackout accidents and LOCAs (as defined in the PDS accident type) make up approximately 65 percent and 33 percent of total PDS frequency, respectively. These two accident types remain the dominant contributors to the PDS frequency, though their relative contributions have changed compared to the provisional PDS results.
- The containment hatch is open for approximately 4 percent of the total PDS frequency. Though this is slightly lower than the provisional PDS results, this attribute remains an important input for defining the representative sequences.
- The containment is unisolated for approximately 20 percent of the PDS frequency, which is significantly higher than the provisional PDS result.
- PDSs with containment coolers (i.e., CCUs) available increased from 6 percent to approximately 34 percent of total PDS frequency, and PDSs with containment sprays available increased from approximately 0.1 percent to approximately 2 percent of the total PDS frequency.

- The profile of RCS status at the time of core damage has shifted, with PDSs occurring with the RCS vented now contributing approximately 24 percent to total PDS frequency (previously 19 percent) and PDSs occurring with the RPV head removed and the vessel flooded now contributing approximately 30 percent to total PDS frequency (previously 59 percent). PDSs occurring with the RCS intact at the time of core damage increased to 45 percent of the total PDS frequency (previously approximately 22 percent).
- The fraction of the CDF during pre-refueling and post-refueling periods remained similar to the provisional PDS results. However, the development of release categories has shown many of the post-refueling sequences contribute to the intact containment release category.
- The availability of ECCS and RWST for injection in non-SBO scenarios generally increased with respect to the provisional PDS results.

As the model results have shifted, additional representative sequences and sensitivity analyses were developed to ensure the key sequence attributes were represented in the deterministic analysis. Despite these differences in the PDS profile, the initial set of scenarios selected for deterministic analysis, along with the additional analyses, were found to be sufficiently broad to serve the needs for event progression and source terms for the final Level 2 PRA.

The final PDS frequency results are presented in Table 2-5. The table shows the PDS results contributing to the top 95 percent of the total frequency. The table includes a description of the attributes of each PDS and the representative sequence cases that align with those attributes.

Table 2-3 Representative Sequences and Permutations

Case #	Base Scenario	Permutations
SD01	Transient in plant operating state (POS) 6 with forced feed	1. Suppression of deflagration events 2. Energetic containment failure at the time of vessel breach 3. Energetic containment failure 4 hours after vessel breach 4. EDMG-based containment spray 2 hours after vessel breach with energetic containment failure 4 hours after vessel breach 5. Containment overpressure failure with alternate containment failure location 6. Containment spray with energetic containment failure 35 hours after vessel breach, alternate containment failure location 7. Containment spray with energetic containment failure 35 hours after vessel breach, failure through the tendon gallery 8. Containment spray (with low spray inventory) with energetic containment failure 35 hours after vessel breach, failure through the tendon gallery 9. Containment spray (with high spray inventory) with energetic containment failure 35 hours after vessel breach, failure through the tendon gallery 10. Containment isolation failure through mini-purge line with credit for filters 11. Containment isolation failure through mini-purge line with no credit for filters 12. Containment isolation failure due to undetected liner tear at low containment elevation 13. Containment isolation failure due to undetected liner tear at high containment elevation
SD02	Transient in POS 10 with gravity feed	None
SD03	SBO in POS 8E with gravity feed and unisolated containment	1. EDMG-based containment spray 2 hours after vessel breach
SD04	SBO in POS 8L with gravity feed	None
SD05	SBO in POS 8E with equipment hatch open	None
SD06	Transient in POS 11 with forced feed and equipment hatch open	1. Stuck-open RHR relief valve with EDMG-based containment spray 2 hours after vessel breach
SD07	SBO in POS 5A with equipment hatch open	1. Open containment by open personnel airlock 2. EDMG-based containment spray 2 hours after vessel breach
SD08	Transient in POS 3 with containment sprays	1. Suppression of creep-induced failure of RCS components 2. Two RCPs operate until consequential trip
SD09	SBO in POS 8E following gravity feed	None
SD10	Loss of Inventory in POS 8E with forced feed	1. Suppression of deflagrations
SD11	Loss of Inventory in POS 5A with forced feed	1. Interfacing-systems LOCA

Table 2-3 Representative Sequences and Permutations (cont.)

Case #	Base Scenario	Permutations
SD12	Loss of Inventory in POS 10 with forced feed	1. Fan coolers failed by the first large deflagration in containment
SD13	SBO in POS 5B with equipment hatch open	None

Table 2-4 Mapping of Representative Sequences to Key Attributes

Plant Oper. State	Initiating Event Type			Cont. Status at Onset of Core Damage
	SBO	Transient	Loss of Inventory	
POS 3		SD08, SD08.1, SD08.2		Containment Closed
				Containment Open
POS 4				Containment Closed
				Containment Open
POS 5A			SD11, SD11.1	Containment Closed
	SD07, SD07.1, SD07.2			Containment Open
POS 5B				Containment Closed
	SD13			Containment Open
POS 6		SD01, SD01.1, SD01.2, SD01.3, SD01.4, SD01.5, SD01.6, SD01.7, SD01.8, SD01.9, SD01.10, SD01.11, SD01.12, SD01.13		Containment Closed
				Containment Open
POS 7				Containment Closed
				Containment Open
POS 8E	SD09		SD10, SD10.1	Containment Closed
	SD03, SD03.1, SD05			Containment Open
POS 8L	SD04			Containment Closed
				Containment Open
POS 9				Containment Closed
				Containment Open
POS 10		SD02	SD12, SD12.1	Containment Closed
				Containment Open
POS 11				Containment Closed
		SD06	SD06.1 ¹	Containment Open
POS 12				Containment Closed
				Containment Open
POS 13				Containment Closed
				Containment Open

¹ The loss-of-inventory through a stuck-open relief valve occurs as a result of a transient event.

Table 2-5 Final PDS Results and Descriptions

PDS Identifier	Frequency (per year)	% Contr.	Cumulative Contr.	PDS Description¹	Representative Cases with Similar Attributes
PDS-SD-074-3	1.3E-06	10.3%	10.3%	POS 6, loss of heat removal transient, successful cont. isolation, successful CCU	SD01.4, SD01.5, SD01.6, SD01.7, SD01.8, SD01.9 with successful sprays, rather than CCU
PDS-SD-113-3	1.2E-06	9.3%	19.6%	POS 8E, SBO, successful cont. isolation, failure of CCU and sprays	SD09 POS8E SBO with intact cont.
PDS-SD-113-6	1.1E-06	8.6%	28.2%	POS 8E, SBO, failure of cont. isolation, failure of CCU and sprays	SD03 and SD03.1 POS8E SBO with cont. unisolated
PDS-SD-138-3	8.8E-07	7.1%	35.2%	POS 8L (post-refuel), SBO, successful cont. isolation, failure of CCU and sprays	SD04 POS8L SBO with intact cont.
PDS-SD-138-6	8.2E-07	6.5%	41.8%	POS 8L (post-refuel), SBO, failure of cont. isolation, failure of CCU and sprays	SD03 and SD03.1 are similar but with pre-refueling conditions
PDS-SD-013-3	6.6E-07	5.3%	47.0%	POS 5A, SBO, successful cont. isolation, failure of CCU and sprays	SD08 and sensitivity cases and SD11
PDS-SD-063-4	6.5E-07	5.2%	52.2%	POS 5B, SBO, successful cont. isolation, failure of CCU and sprays	SD01, SD01.1, SD01.2, and SD01.3 without cont. sprays and CCUs
PDS-SD-024-3	6.1E-07	4.9%	57.1%	POS 5A, loss of inventory, successful cont. isolation, successful CCU	SD11 POS5A loss of inventory
PDS-SD-013-6	5.2E-07	4.1%	61.2%	POS 5A, SBO, failure of cont. isolation, failure of CCU and sprays	SD07 and sensitivity cases with cont. open
PDS-SD-072-3	4.3E-07	3.4%	64.6%	POS 6, loss of heat removal transient, successful cont. isolation, successful CCU	SD01.4, SD01.5, SD01.6, SD01.7, SD01.8, SD01.9 with successful sprays, rather than CCU
PDS-SD-038-3	3.7E-07	3.0%	67.6%	POS 12 (post-refuel), SBO, successful cont. isolation, failure of CCU and sprays	SD04, SD12, and SD12.1 are relevant though different POS conditions
PDS-SD-097-3	3.6E-07	2.9%	70.5%	POS 9 (post-refuel), loss of heat removal transient, successful cont. isolation, successful CCU	SD02, SD12, and SD12.1 are relevant though different POS conditions
PDS-SD-113-4	3.5E-07	2.8%	73.2%	POS 8E, SBO, successful cont. isolation, failure of CCU and sprays	SD09 POS8E SBO with intact cont.
PDS-SD-124-3	3.4E-07	2.7%	78.5%	POS 8E, loss of heat removal transient, successful cont. isolation, successful CCU	SD10 and SD10.1 POS8E loss of inventory with intact cont.
PDS-SD-088-4	3.2E-07	2.5%	78.5%	POS 10 (post-refuel), SBO, successful cont. isolation, failure of CCU	SD02, SD12, and SD12.1 POS 10 transients with intact cont.

Table 2-5 Final PDS Results and Descriptions (cont.)

PDS Identifier	Frequency (per year)	% Contr.	Cumulative Contr.	PDS Description¹	Representative Cases with Similar Attributes
PDS-SD-147-3	2.8E-07	2.2%	80.7%	POS 8L (post-refuel), loss of heat removal transient, successful cont. isolation, successful CCU	SD04 is relevant but without CCU
PDS-SD-138-4	2.6E-07	2.1%	82.8%	POS 8L (post-refuel), SBO, successful cont. isolation, failure of CCU	SD04 POS8L SBO with intact cont.
PDS-SD-099-3	2.4E-07	1.9%	84.7%	POS 10 (post-refuel), loss of heat removal transient, successful cont. isolation, successful CCU	SD02 POS10 transient with intact cont.
PDS-SD-122-3	2.0E-07	1.6%	86.2%	POS 8E, loss of heat removal transient, successful cont. isolation, successful CCU	SD10 and SD10.1 POS8E loss of inventory with intact cont.
PDS-SD-013-8	1.9E-07	1.5%	87.7%	POS 5A, SBO, failure to close cont., failure of CCU and sprays	SD07 and sensitivities POS5A SBO with equipment hatch open
PDS-SD-038-4	1.8E-07	1.4%	89.2%	POS 12 (post-refuel), SBO, successful cont. isolation, failure of CCU and sprays	SD04, SD12, and SD12.1 are relevant though different POS conditions
PDS-SD-088-3	1.6E-07	1.3%	90.5%	POS 9 (post-refuel), SBO, successful cont. isolation, failure of CCU and sprays	SD04, SD12, and SD12.1 are relevant though different POS conditions
PDS-SD-151	1.4E-07	1.1%	91.6%	POS5A, LOCA, containment bypass	SD11.1 POS5A loss of inventory with ISLOCA
PDS-SD-038-1	1.1E-07	0.9%	92.5%	POS 13 (post-refuel), SBO, successful cont. isolation, failure of CCU and sprays	SD04, SD12, and SD12.1 are relevant though different POS conditions
PDS-SD-149-3	9.1E-08	0.7%	93.2%	POS 8L (post-refuel), loss of heat removal transient, successful cont. isolation, successful CCU	SD10 and SD10.1 are similar but with pre-refueling conditions
PDS-SD-013-4	8.5E-08	0.7%	93.9%	POS 5A, SBO, successful cont. isolation, failure of CCU and sprays	SD08 and sensitivity cases and SD11
PDS-SD-038-8	8.1E-08	0.6%	94.5%	POS 12 (post-refuel), SBO, failure to close cont., failure of CCU and sprays	SD06, SD07 and sensitivities are relevant but with different POS conditions
PDS-SD-113-8	7.8E-08	0.6%	95.1%	POS 8E, SBO, failure to close cont., failure of CCU and sprays	SD05 POS8E SBO with equipment hatch open

Table 2-5 Final PDS Results and Descriptions (cont.)

PDS Identifier	Frequency (per year)	% Contr.	Cumulative Contr.	PDS Description¹	Representative Cases with Similar Attributes
PDS-SD-063-8	7.0E-08	0.6%	95.7%	POS 5B, SBO, failure to close cont., failure of CCU and sprays	SD13 POS5B SBO with equipment hatch open
Total:	1.3E-05		100%		
¹ PDS description includes the POS that is the predominant contributor to the PDS frequency, accident type, containment status, success or failure of containment systems, and identification of post-refueling POSs.					

2.2 Containment Capacity Analysis

The purpose of the containment capacity analysis is to characterize how the containment structure (and surrounding structures) will respond to varying severe accident loads. The containment capacity analysis for the reactor-at-shutdown leverages the same analysis for the at-power situation.

The containment capacity analysis described in the at-power Level 2 PRA (NRC, 2022a) broadly applies to the shutdown. The failure mechanisms assessed in the at-power PRA that are also applicable to shutdown include failures due to:

- energetic combustion events
- high temperatures
- dynamic interactions between core debris and water
- direct core debris impingement
- concrete cracking and liner tearing due to over-pressurization
- vessel displacement due to molten core concrete interaction attack of the cavity
- displacement caused by vessel failure at high pressure

A detailed structural analysis was performed to characterize the reference plant's containment structural response to static over-pressurization. The containment capacity analysis involved review the containment fragility curve that was prepared for the reference plant Individual Plant Examination submittal to the NRC. Then, a three-dimensional finite element model was developed using the LS-DYNA software. The model was based plant-specific information from the reference plant. The full analysis is not reproduced here, but the outcomes are summarized below.

The structural analysis characterized the likely containment failure locations and failure sizes. The analysis identified failure locations considered to be equally likely at the containment basemat-wall junction, equipment hatch, emergency airlock, and personnel airlock. For both the at-power and shutdown analyses, the location near the containment basemat and wall junction was selected for modeling containment over-pressurization failure. However, an alternate failure location near the containment equipment hatch is considered in a shutdown-specific sensitivity analysis described in Section 2.6.3.13.

The structural analysis characterized the tearing and opening of containment at increasing pressures. The analysis is interpreted by the following criteria that are used for modeling the containment response in the severe accident progression analysis:

- Failure flow area is closed when pressure less than 120 psig.
- Failure flow area is 0.1 ft² once maximum pressure exceeds 120 psig.
- Failure flow area is 0.3 ft² once maximum pressure exceeds 127 psig.
- Failure flow area is 1.0 ft² once maximum pressure exceeds 130 psig.

After reaching the maximum opening failure size, the pressurization rates due to severe accident phenomena are not expected to be large enough to continue to increase the containment pressure. If pressurization did occur at a rate where the failure opening did not prevent continued pressurization, then more catastrophic containment failure modes would be reached at roughly 160 psig.

The at-power Level 2 PRA (NRC, 2022a) provides additional details and a summary of all of the containment failure mechanisms that were considered in the containment capacity analysis. While the at-power containment capacity analysis is broadly applicable during shutdown, there is one potentially shutdown-specific situation involving an emergency closure of the containment equipment hatch during the accident sequence. In this case, only a small number of hatch bolts may be in place, thus potentially changing the response of the containment to elevated loads. Some sites have reported that using only the minimum number of bolts was not sufficient for hatch sealing (See Sections 6.9.2 and 6.9.3 of (NRC, 1993)). For the reference plant, once the hatch is in place, the number of bolts used is not viewed to be important because the pressure-sealing hatch design would tend to improve the seal as pressure increased. Nevertheless, this concern is documented as a model uncertainty in Section 2.6.3.8.

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 (Section 2.3.1)
2. Development of a plant-specific MELCOR model (Section 2.3.2)
3. Accident progression modeling for the representative Level 2 sequences (Section 2.3.3)
4. Phenomenological evaluations for split fraction assignment and logic model construction (Section 2.3.4)
5. Evaluation of the impact of post-core-damage recovery actions (Section 2.3.5)
6. Evaluation of equipment survivability (Section 2.3.6)

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 (as well as the Level 3 MACCS model). The objective of the second step is to develop a plant-specific MELCOR model of the RCS, SGs and steam lines, ECCS, containment and containment systems, reactor protection system and engineered safety features actuation 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 associated with key plant damage states, for informing the development of the accident progression event tree and providing timelines for use by the HRA. 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 PRA 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 term. The objective of the sixth step is to evaluate what equipment will be adversely affected by the conditions (temperature, pressure, humidity, radiation, energetic events) associated with the accident.

2.3.1 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, in order to calculate the quantity and composition of various radionuclides in each fuel 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 MACCS). Data sets were developed for the reactor core (both from at-power trip conditions and a planned shutdown leading to a refueling outage), as well as for all spent fuel in the Unit 1 and Unit 2 spent fuel pools (SFPs). The ORNL analysis was performed on an assembly-by-assembly basis for the 2,885 assemblies processed (193 for the Unit 1 core, 1,129 for the Unit 1 SFP, and 1,563 for the Unit 2 SFP). The results are grouped in a manner convenient for their use by MELCOR and MACCS.

The coastdown to refueling conditions were chosen to mimic an actual September 2012 unit outage, and this is the data set used in the reactor at-shutdown MELCOR model. However, if other types of outages are considered in future studies, then the assumptions about the time during the cycle when shutdown occurs can influence the decay heat and radionuclide inventories. The selection of timing within a fuel cycle is identified as a modeling uncertainty in Section 2.6.3.14. Nevertheless, the end-of-cycle coastdown to refueling conditions used for the SCALE analysis are appropriate for the refueling outage analyzed in this study.

In using the SCALE coastdown results, static chemical class masses (also known as inventories) were used to simplify the MELCOR modeling, although time-dependent masses are available. This introduces small effects (either higher or lower, depending on the evaluation time and chemical class in question), that can have minor effects on fission product transport behaviors. This is identified as a source of model uncertainty in Section 2.6.3.14.

2.3.2 Development of a Plant-Specific MELCOR Model

2.3.2.1 Use of MELCOR

Severe accident progression calculations were 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 uncover, 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

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 (NRC, 2012) and studies to support new reactor licensing (e.g., the AP1000 design certification review (Esmaili, 2004)). This validation basis is discussed in more detail in the reactor-at-power Level 2 PRA main report, and the references therein (NRC, 2022a). However, there are some phenomena unique to reactor-at-shutdown conditions for which MELCOR has less extensive validation. Examples include counter-current flow in the pressurizer surge line, air ingress into the RPV, and containment release flow path modeling with the containment equipment hatch open. These and other aspects of the Level 2 PRA may be subject to greater uncertainty. Modeling uncertainties are identified, and select sensitivity analyses are documented, in Section 2.6.3.

2.3.2.2 *MELCOR Model*

The existing reference plant reactor at-power MELCOR model was taken as a starting point and was heavily modified to represent the plant in a shutdown condition. This included modifications related to having an open equipment hatch or personnel air lock, flooding of the in-containment refueling cavity, modeling of the cold overpressure protection system, modeling of the closed-loop decay heat removal mode of RHR, potential presence of SG dams, and removal of the RPV head. The model and supporting documentation were prepared by Energy Research Inc. The MELCOR model incorporates numerous sources of reference plant-specific information (e.g., system geometries, system flow rates, set points), and those details are not reproduced here. Numerous shakedown calculations were performed during model development to assist in identifying modeling needs and model limitations.

2.3.3 Accident Progression Modeling for the Representative Level 2 PRA Sequences

The discussion here focuses on the deterministic thermal-hydraulic work performed for the Level 2 PRA. For the Level 2 PRA, 13 representative sequences were previously defined in Section 2.1.5 (Table 2-3), based on the provisional plant damage state quantification. In addition, several variations were also analyzed to provide related information needed for development of the Level 2 PRA. The MELCOR model described in the previous section was used to analyze these representative sequences and variations.

Throughout the remaining sections of this report, the specific MELCOR results used to achieve that portion of the analysis are distilled. A summary of boundary conditions and accident progression results is presented in Table 2-6, Table 2-7, and Table 2-8. Some of the noteworthy modeling aspects are described here:

- Battery depletion time for the safety-related batteries is assumed to be four hours, which is consistent with the Level 1 shutdown PRA and the at-power PRA.
- Case 2 and Case 12 use two different evaluation times for POS 10 (501 hours and 507 hours, respectively). This is not intended to be a meaningful distinction but rather reflects minor shifts in the underlying POS characterization during the Level 2 PRA development.

- For the loss of inventory outside of containment (LOI-OC) (also known as ISLOCA) sensitivity calculation, turbulent deposition in the connected piping is not modeled. Other retention uncertainties related to auxiliary building failure, break coverage, and filter plugging were previously investigated via sensitivity studies in the at-power PRA.
- In all cases (except where noted otherwise), creep failure of the hot leg and surge line are tracked based on the Larson-Miller correlation, and if failure conditions are predicted then a failure is modeled. Conversely, creep-induced SG tube rupture (SGTR) is tracked, but not invoked. (The creep-induced SG tube failure is addressed in the probabilistic modeling, and the model parameters are provided in Section C.2 of Appendix C.)
- 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 performed assuming valve failure. (Failure of relief valves is addressed in the probabilistic modeling, and the model parameters are provided in Section C.2 of Appendix C.)
- Except where otherwise noted, combustion is enabled, including random ignition. This means that if combustible gas and inert gas compositions support combustion, it is predicted to occur. (The probabilistic modeling of combustion events is discussed as part of the modeling uncertainty analyses in Section 2.6.3.7. The model parameters used for the probabilistic combustion modeling are provided in Section C.2 with additional details on the uncertainty distributions in Section C.3 of Appendix C.)
- The containment overpressure failure pathway is assumed to be into the tendon gallery, which in turn connects to both the environment and the auxiliary building, with the release pathway area apportioned through the three access shafts (two to the environment and one to the auxiliary building).^{6, 7}
- The time of initiation of gravity or forced feed prior to core damage considers both the abnormal operating procedure (AOP) guidance and the HRA for the Level 1 PRA. That said, timings are generally determined by the AOP-specified plant conditions (e.g., gravity drain begins when bulk water temperature equals 100°C). In some cases, the timing has a large influence on the calculation. In particular, gravity injection in SD09 quickly dead-heads due to rising containment pressure at that point in the accident. In general, earlier or later injection timing may have led to a different accident progression.
- Since fission product retention in secondary structures (meant here to include 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

⁶ The apportionment of leakage to the environment and auxiliary building is 70 percent and 30 percent, respectively.

⁷ In viewing auxiliary building retentions for cases with containment overpressure failure, 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 (as observed in the MELCOR calculations), due to small pressure differences between the environment (atmospheric pressure) and the auxiliary building (whose normal leakage area is not large enough to prevent some pressurization).

resolution and state-of-knowledge limitations, it is worth pointing out cases with the highest attributed retention (cases with greater than 10 percent of the initial core inventory of cesium/iodine retained) in these secondary structures. These are:

- Case SD01.2 and SD01.3, discussed further in the evaluation of fission product release modeling uncertainties in Section 2.6.3.13 and Section 2.6.3.14.
- Case SD09, which involves containment overpressure failure (which models pathways to both the environment and the auxiliary building) prior to core damage.
- Case SD11.1, involving a simulated LOI-OC (ISLOCA) into the auxiliary building.
- The default MELCOR molten core-concrete interaction (MCCI) and cavity modeling is 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 is not mechanistically predicted; nevertheless, the report cites the time when the radial or axial ablation exceeds the known thickness of the plant cavity wall or basemat.
- The termination time in almost all simulations is seven days after the initiating event occurs. This selection is discussed further in the Section 2.6.3.15 discussion of accident termination model uncertainty.

The results from the MELCOR analysis can be generally summarized as follows:

- Core damage is not predicted to occur within 7 days of the initiating event for three cases.
- Significant air oxidation during in-vessel melt progression was not predicted, recognizing that there are some limitations in the MELCOR model's ability to predict air ingress.
- Vessel breach occurs for all cases in which core damage occurs (noting that these simulations do not involve mitigating actions that would be expected to lead to in-vessel melt arrest). Vessel breach is indicated by through-wall failure of the lower vessel after relocation of damaged fuel.
- Combustions are predicted to occur in several different simulations.
- Containment long-term overpressure failure is predicted to occur in three simulations and basemat melt-through is predicted to occur in two simulations.
- The magnitudes of the releases correlate directly with the status of containment, with fractional releases of the volatile classes (e.g., cesium) being very low in cases where containment integrity is maintained (i.e., intact/isolated) and being generally quite high in cases where containment is either open, un-isolated, or compromised.

Another key location for material related to the treatment, uncertainty, and impact of various modeling assumptions is in the Level 2 PRA's treatment of modeling uncertainty and sensitivity analyses (see Section 2.6.3). Beyond the consideration of shutdown-unique modeling issues in the MELCOR model development and the aforementioned treatment of uncertainty, the general issue of comparison to other studies relies on the at-power comparison documented in the at-power Level 2 PRA in the Appendix D entry entitled, "Comparison of MELCOR Results to Other Relevant Analyses" (NRC, 2022a).

Table 2-6 Summary of MELCOR Analysis Boundary Conditions and Accident Progression Results for Case 1 and Sensitivities

	SD01	SD01.1	SD01.2	SD01.3	SD01.4	SD01.5	SD01.6	SD01.7	SD01.8	SD01.9	SD01.10	SD01.11
Initiator	Loss of RHR (LORHR)											
Tech Spec (TS) Mode	Mode 5 – Cold Shutdown											
Plant Oper. State (POS)	6 – Hot mid-loop											
Initial time (hrs)	82 hours (after start of outage)											
RCS status	Vented											
Cont. status at core damage	Closed / isolated										Un-isolated	
Cont. systems	Unavailable											
Early Injection?	Yes											
Post-core-damage action?	No				EDMG spray	No	EDMG spray				No	
Core damage (hr) ¹	15	15	15	15	15	15	15	15	15	15	15	15
Vessel breach (hr)	24	24	24	24	24	24	25	25	25	25	25	23
Combustions (y/n)	Yes	No ²	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Cause of contain. failure ³	OP	n/a	F:VB	F:VB+4	F:VB+4	OP	F:VB +35	F:VB +35	F:VB +35	F:VB +35	Un-isol. ⁴	Un-isol.
Time of contain. failure (hr)	168	>168	24	28	28	168	60	60	60	60	Early	Early
Cumulative noble gas release (%)	1%	1%	95% ⁵	98% ⁵	100%	2%	100%	100%	100%	100%	100% ⁴	100%
Cumulative Cs release (%)	<1%	<1%	1% ⁵	1% ⁵	3%	<1%	<1%	1%	1%	1%	<1% ⁴	9%
Cumulative I release (%)	<1%	<1%	3% ⁵	4% ⁵	17%	<1%	1%	1%	3%	2%	<1% ⁴	33%
End of calc. (hr)	168	168	168	168	168	192	168	168	168	168	168	168

¹ The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.

² Combustion is deliberately suppressed in this calculation.

³ OP = Late overpressure; F:VB = Forced at vessel breach; F:VB+4 = Forced 4 hours after vessel breach; F:VB+35 = Forced 35 hours after vessel breach

⁴ Case SD01.10 considers full functionality of filters on the isolation opening, though several mechanisms are likely to challenge the filters and cause failure.

⁵ These estimates may be under-predicted due to modeling of retentions in connected structures—see model uncertainty MU-14.2 in Section 2.6.3.14.

Table 2-7 Summary of MELCOR Analysis Boundary Conditions and Accident Progression Results for Cases 1–7

	SD01.12	SD01.13	SD02	SD03	SD03.1	SD04	SD05	SD06	SD06.1	SD07	SD07.1	SD07.2
Initiator	LORHR		LORHR	Station Blackout (SBO)		SBO	SBO	LORHR		SBO		
Tech Spec (TS) Mode	5		5	6 - Refueling		6	6	5		5		
Plant Oper. State (POS)	6 – Hot mid-loop		10 – Cold mid-loop	8E – Early refueling		8L – Late refueling	8E	11 – Refill RCS		5A - Water solid		
Initial time (hrs)	82		501	141		329	141	526		12		
RCS status	Vented		Vented	Head off		Head off	Head off	Intact		Intact		
Cont. status at core damage	Un-isolated		Closed / isolated	Un-isolated		Closed / isolated	Open	Open		Open ²		
Cont. systems	Unavailable		Coolers	Unavailable		Unavail.	Unavail.	Unavailable		Unavailable		
Early Injection?	Yes		Yes	Yes		Yes	No	Yes		No		
Post-core-damage action?	No		No	No	EDMG spray	No	No	No	EDMG spray	No		EDMG Spray
Core damage (hr) ¹	15	15	>168	125	125	>168	100	>168	~44	15	15	15
Vessel breach (hr)	24	23	n/a	137	137	n/a	110	n/a	74	21	21	21
Combustions (y/n)	Yes	Yes	n/a	No	No	n/a	No	n/a	Yes	Yes	Yes	Yes
Cause of contain. failure	Un-isol.	Un-isol.	n/a	Un-isol.	Un-isol.	n/a	Open	Open	Open	Open	Open	Open
Time of contain. failure (hr)	Early	Early	n/a	Early	Early	n/a	Early	Early	Early	Early	Early	Early
Cumulative noble gas release (%)	98%	99%	n/a	67%	26%	n/a	100%	n/a	100%	100%	100%	100%
Cumulative Cs release (%)	2%	1%	n/a	7%	5%	n/a	87%	n/a	46%	25%	14%	20%
Cumulative I release (%)	8%	1%	n/a	6%	5%	n/a	88%	n/a	50%	38%	20%	30%
End of calc. (hr)	168	168	168	168	168	168	168	168	168	168	168	168

¹ The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.

² For case SD07 and SD07.2 it is the hatch that is open; in SD07.1 it is the personnel airlock that is open.

Table 2-8 Summary of MELCOR Analysis Boundary Conditions and Accident Progression Results for Cases 8–13

	SD08	SD08.1	SD08.2	SD09	SD10	SD10.1	SD11	SD11.1	SD12	SD12.1	SD13
Initiator	LORHR			SBO	Loss of Inventory (LOINV)		LOINV	LOI-OC/ISLOCA	LOINV		SBO
TS Mode	4 - Hot Shutdown			6	6		5		5		5
POS	3 – Cooldown to 200F			8E	8E		5A		10		5B
Initial time (hrs)	5			141	141		12		507		55
RCS status	Intact			Head off	Head off		Intact (prior to the LOINV)		Vented		Vented
Cont. status at core damage	Closed / isolated			Closed / isolated	Closed / isolated		Closed / isolated		Closed / isolated		Open
Cont. systems	Available			Unavail.	Unavailable		Unavailable		Coolers ⁵		Unav.
Early Injection?	No			Yes	Yes		Yes		Yes		No
Post core-damage action?	No			No	No		No		No		No
Core damage (hr) ¹	28	28	28	125	9.5	9.5	21	21	24	24	7
Vessel breach (hr)	33	33 ³	32	135	20	20	28	30	56	54	14
Combustions (y/n)	Yes	-	Yes	No	Yes	No ⁴	Yes	Yes	Yes	Yes	
Cause of contain. failure ²	BMT	-	BMT	OP	n/a	n/a	OP	bypass	n/a	n/a	Open
Time of contain. failure (hr)	164	-	165	118	n/a	n/a	166	n/a	n/a	n/a	Early
Cumul. noble gas release (%)	1%	-	1%	95%	1%	1%	1%	99%	<1%	<1%	100%
Cumulative Cs release (%)	<1%	-	<1%	15%	<1%	<1%	<1%	3%	<1%	<1%	79%
Cumulative I release (%)	<1%	-	<1%	14%	<1%	<1%	<1%	5%	<1%	<1%	82%
End of calc. (hr)	168	33	168	168	168	168	168	168	168	168	168

¹ The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.

² OP = Late overpressure; BMT = basemat melt-through

³ Creep rupture of RCS components (other than the RPV lower head) during core damage is suppressed. The calculation is ended at vessel breach.

⁴ Combustion is deliberately suppressed in this calculation.

⁵ The fan coolers are assumed to fail upon the first large deflagration.

2.3.4 Phenomenological Evaluations for Split Fraction Assignments and Logic Model Construction

The phenomenological evaluations for split fraction assignment and logic model construction relied on: (i) the equivalent activity performed for the at-power Level 2 PRA in (NRC, 2022a), (ii) performance of the aforementioned sensitivity calculations based on the representative sequences, and (iii) stand-alone analytical investigations. In general, the discussion of phenomenological issues in the at-power Level 2 PRA in Appendix E of (NRC, 2022a) applies to the shutdown situation, with key exceptions related to ISLOCA modeling and SGTR-related issues (that are broadly not relevant at shutdown). As such, (NRC, 2022a) should be consulted in understanding how the various phenomenological items are treated.

The phenomena that were evaluated include:

- Basemat melt-through
- Consequential SGTR
- Containment overpressure
- Direct containment heating
- Early containment failure effects on fission product release
- Hydrogen and carbon monoxide combustion
- In-vessel and ex-vessel coolability
- Vessel rocketing

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 2.6.3, irrespective of the selection of their split fraction. In other words, the split fraction is an attempt to estimate the expected response; the model uncertainty or parameter distribution is an attempt to express the uncertainty about that estimate.

In performing the above phenomenological evaluations, there are cases where simplifying assumptions have been made. Sometimes these assumptions are made because the additional resources to go to the next degree of mechanistic modeling is 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 in this particular PRA. Similarly, it is also important to recall that these assumptions are in the context of a probabilistic risk assessment, which deliberately goes beyond the design-basis of a plant in answering the questions of what can go wrong, how likely it is to happen, and what are the consequences.

2.3.5 Assessment of Post-Core-Damage Recovery Actions

The analyses described in Section 2.3.3 simulate the accident progression for various scenarios largely in the absence of operator intervention. They are used as the basis for the HRA discussed further in Sections 2.4.4 and 2.4.5. The HRA is used to develop the set of most likely operator actions. Simulations are then re-run modeling the specific actions assumed for each scenario. Due to resource and timing considerations, only a few simulations were repeated with post-core-damage mitigative actions considered, specifically the modeling of extensive damage mitigation guideline (EDMG)-based spray after the onset of core damage in Cases SD01.4,

SD01.6, SD01.7, SD01.8, SD01.9, SD03.1, SD06.1, and SD07.2. The original motivation for focusing on EDMG sprays was to provide scrubbed versions of otherwise un-scrubbed source terms. In the end, the simulated source terms led to generally not making distinctions between scrubbed and un-scrubbed source terms due to the lack of a large difference relative to other factors affecting source term magnitude. Keep in mind that the EDMG-based spray capability is generally much less effective than the engineered spray capability. While the EDMG-based containment spray has shown limited effectiveness for scrubbing releases for these particular accident scenarios, the sprays do play an important role in cooling the containment and also influence containment conditions that can support combustion.

In-vessel coolability based on re-flooding of the core during in-vessel melt progression was not specifically investigated. In addition, calculations were run early in the project to investigate the effect of flooding the reactor cavity late in the accident in an effort to terminate the accident prior to basemat melt-through or long-term containment overpressurization. In general, they show that these late actions are likely to significantly reduce further radiological releases and prevent long-term containment overpressurization but are not likely to prevent basemat melt-through.

2.3.6 Evaluation of Equipment Survivability

Like the equivalent activity in the at-power Level 2 PRA (NRC, 2022a), the treatment of equipment survivability is simplistic, owing to constraints associated with the available plant-specific information, the level of detail of accident management modeling, and the level of project resources. The reader is referred to the analogous section of (NRC, 2022a) for further information about the formative work done in this area.

The Level 2 PRA post-core-damage HRA (described in Appendix E) defines several accident scenarios where post-core-damage actions are considered. A decision tree method was developed to estimate HEP values for the modeled actions. The HRA scenarios consider the viability of actions in the context of the scenarios, including consideration of severe accident phenomena, availability of information, procedural support, location of the action, complexity, environmental concerns, and other factors that can influence performance of post-core-damage actions. Equipment (including instrumentation) survivability is assessed qualitatively and used to inform the decision tree assessment. The following scenario-specific assumptions are made, overlying generic survivability considerations such as those previously considered in the at-power Level 2 PRA (NRC, 2022a):

- Scenarios #1 (pre-refueling) and #2 (post-refueling), loss of inventory or transient developing inventory loss through open RCS vent during mid-loop, prior to vessel breach
 - Survivability concerns: Core damage occurs with a vented RCS, containment closed, and with containment pressure steadily rising. Areas adjacent to containment will experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.

- Survivability assessment: Some habitability concerns near containment penetrations, but no equipment survivability concerns.
- Scenarios #1 (pre-refueling) and #2 (post-refueling), loss of inventory or transient developing inventory loss through open RCS vent during mid-loop, following vessel breach
 - Survivability concerns: In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.
 - Survivability assessment: While potential combustion could impact some equipment within containment, no impacts are expected for the credited EDMG-based actions.
- Scenarios #3, #4, #5, #7, #9 and #13, station blackout considering variations of initial POS conditions and containment status
 - Survivability concerns: Conditions will generally limit availability of equipment and likelihood of successfully implementing actions, but no specific survivability concerns were assessed.
 - Equipment used: None. No credit for post-core-damage human actions is considered for SBO scenarios.
 - Survivability assessment: No survivability assessment is performed since no equipment is credited.
- Scenario #6, loss of cooling transient, RCS not vented, containment open
 - Survivability concerns: This is a slowly developing scenario with some survivability concerns related to slowly increasing radiation and temperature within containment.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.
 - Survivability assessment: Some habitability concerns near containment opening and other penetrations, but no equipment survivability concerns.
- Scenario #8, loss of cooling transient, RCS not vented, containment closed, prior to vessel breach
 - Survivability concerns: The intermittent combustions inside containment have the potential to damage equipment or instrumentation.
 - Equipment used: Atmospheric relief valves for depressurizing SGs, fire water storage tanks or demineralized water storage tank for suction source for injection to SGs,

equipment for feedwater flange removal and connection, EDMG portable pump for SG injection, and portable gasoline generator and power supply for opening vent valves on the reactor head vent piping via the electrical connections located on lower level of the control building.

- Survivability assessment: No survivability concerns for equipment used. The only impacts are inside containment.
- Scenario #8, loss of cooling transient, RCS not vented, containment closed, following vessel breach
 - Survivability concerns: With containment pressure being retained low, and containment being intact, radiation levels outside containment would be expected to be reasonably low; the intermittent combustions inside containment have the potential to damage equipment.
 - Equipment used: Atmospheric relief valves for depressurizing SGs, fire water storage tanks or demineralized water storage tank for suction source for injection to SGs, equipment for feedwater flange removal and connection, EDMG portable pump for SG injection, and portable gasoline generator and power supply for opening vent valves on the reactor head vent piping via the electrical connections located on lower level of the control building.
 - Survivability assessment: No survivability concerns for equipment used. The only impacts are inside containment.
- Scenario #10, loss of inventory or transient developing inventory loss through RCS opening with refueling cavity flooded, prior to vessel breach
 - Survivability concerns: Core damage occurs with the vessel head removed, containment closed, and with containment pressure steadily rising. Areas adjacent to containment will experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.
 - Survivability assessment: Some habitability concerns near containment penetrations, but no equipment survivability concerns.
- Scenario #10, loss of inventory or transient developing inventory loss through RCS opening with refueling cavity flooded, following vessel breach
 - Survivability concerns: In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment

spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.

- Survivability assessment: While potential combustion could impact some equipment within containment, no impacts are expected for the credited EDMG-based actions.
- Scenario #11, loss of inventory, RCS initially intact with stuck open relief valve, prior to vessel breach
 - Survivability concerns: With an RCS leak path, containment closed, and with containment pressure steadily rising, areas adjacent to containment could experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.
 - Survivability assessment: Some habitability concerns near containment penetrations, but no equipment survivability concerns.
- Scenario #11, loss of inventory, RCS initially intact with stuck open relief valve, following vessel breach
 - Survivability concerns: In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.
 - Survivability assessment: While potential combustion could impact some equipment within containment, no impacts are expected for the credited EDMG-based actions.
- Scenario #12, loss of inventory due to inadvertent diversion of water
 - Survivability concerns: There is an RCS leak path and containment is closed, but containment pressure is being kept near atmospheric conditions, so ex-containment areas should be relatively unaffected relative to the situation in other cases analyzed in this report. However, some damage within-containment is possible from the combustions that are predicted to occur.
 - Equipment used: Fire water storage tanks or demineralized water storage tank for RWST refill, EDMG portable pump for RWST refill or EDMG-based containment spray, and available safety injection and charging pumps for recovery of ECCS injection after RWST refill.

- Survivability assessment: While potential combustion could impact some equipment within containment, no impacts are expected for the credited EDMG-based actions.

These general and scenario-specific considerations feed into one of the decision tree questions for each of diagnosis (Information Availability) and execution (Environmental Concerns), though these decision tree questions can also be impacted by other factors. These evaluations inform the modeling of post-core-damage actions, but there are limitations to this approach. There is a lack of detailed plant-specific information on the credited equipment. While the type of equipment and general locations of equipment use are known, details on the equipment qualifications, instrumentation routing, and connection points were not available for this study. Also, the environmental impacts of the severe accident scenarios are informed by the representative MELCOR scenarios, but these also have limitations. While general conditions in containment and connected structures can be estimated, detailed assessments of radiation levels, temperatures, pressures, and combustion effects are not performed for all equipment locations. Ultimately, a clear nexus was not judged to exist between the environmental challenges and equipment (including instrumentation) performance, and no specific severe accident environment-induced equipment failure mechanisms are modeled. Despite these limitations, the equipment survivability concerns provide qualitative insights to the overall assessment of post-core-damage operator actions. There are several uncertainties associated with the modeling of post-core-damage operator actions (see further discussion of HRA uncertainties in Section 2.6.3.3); however, the uncertainty associated with the simplified equipment survivability approach itself is not expected to significantly impact the release category frequency results.

2.4 Probabilistic Treatment of Accident Progression

This subtask consists of seven interrelated steps:

1. Data analysis for components/systems not considered in the Level 1 PRA (Section 2.4.1)
2. Construction of accident progression event tree (containment event tree) (Section 2.4.2)
3. Development of support trees (Section 2.4.3)
4. Human reliability model development (Section 2.4.4)
5. Human reliability analysis (Section 2.4.5)
6. Level 2 model quantification (Section 2.4.6)
7. Uncertainty characterization (Section 2.4.7)

The objective of the first step is to consider the reliability of structures, systems, and components not considered in the Level 1 PRA. The objective of the second step is to develop the set of accident progression event tree (also referred to as the containment event tree or 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 quantification of 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.

2.4.1 Data Analysis for Components/Systems Not Considered in the Level 1 PRA

The reliability of containment systems to perform their engineered, design-basis accident functions has already been captured by the inclusion of containment sprays, containment fan coolers, and containment isolation in the bridge tree. The underlying fault tree models in the bridge tree capture these systems' reliability, using a mix of information adopted from the licensee's PRA model and information developed by NRC, following the lead of the Level 1 PRA's treatment of data issues including common-cause failures (NRC, 2022b). This includes an embedded assumption that a 24-hour mission time for equipment reliability is sufficient, and that consideration of longer mission times would not significantly influence the results.

The only system modeled in the Level 2 PRA where the extension of the mission time assumption is debatable is the containment fan cooler system (which is credited with controlling containment conditions over longer timeframes). However, the containment fan cooler system failure is dominated by mechanisms that are unrelated to the mission time (e.g., unavailability of AC power in SBO sequences and human failures to initiate or restore containment cooling). Even if the mission time was extended resulting in higher containment fan cooler failure probabilities, the impact would just tend to shift the frequency contribution from the basemat melt-through release category (1-REL-SD-BMT) to the late containment failure due to overpressure release category (1-REL-SD-LCF). Both of these release categories have relatively low contributions to the overall release frequency (as shown in Section 2.4.6) and relatively low release magnitudes (as shown in Section 2.5.1). Therefore, extending the containment fan cooler system mission time is not expected to significantly impact the results.

With the containment systems addressed in the bridge tree, this leaves systems that are modeled (or otherwise considered) within the CET (or the supporting DETs). These systems are those that either actuate automatically based on their design or are manually operated as part of post-core-damage accident management (using the EDMGs). As with the at-power Level 2 PRA (NRC, 2022a), reliability models are 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 covered elsewhere in this document. 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, and the Level 2 sequence logic, include consideration of system/component availability (with some embedded limitations), and so system unavailability due to sequence-specific support system failures (e.g., AC power, NSCW cooling) has already been accounted for in the PDS event tree logic and linking rules.

- In most cases, data for these systems/components is not harvested as part of the NRC's data collection processes, and thus reliability evaluations are not routinely estimated as they are for other modeled system equipment.
- These systems require manual actions to use or actuate the system functions. Reliability of automatically actuated systems is modeled in the fault trees when applicable for the modeled POS.

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 less value/impact. Also, while equipment reliability is not explicitly modeled, the post-core-damage HRA considers the equipment availability, accessibility, and survivability given severe accident conditions, and these factors influence the HEP estimates (see Sections 2.4.4 and 2.4.5 for further discussion of the post-core-damage HRA model). The systems for which this decision applies are covered in Table 2-9.

A separate data-oriented question is treatment of offsite power recovery during loss-of-offsite power events. This issue is discussed in more detail in (NRC, 2022a), (NRC, 2022b), and (NRC, 2025), but the bottom line is that offsite power recovery is not credited following depletion of the station batteries.

Finally, on another related matter, the Level 2 PRA does not currently treat equipment repair or recovery (which is discussed in more detail in the aforementioned references). However, and unlike the at-power Level 2 PRA, the reactor at-shutdown Level 2 HRA does credit restoration of equipment that was out-of-service at the time of the accident, on a very limited basis. Presently, such credit is limited to manual containment isolation, emergency containment hatch closure, and the containment fan cooler system all of which are discussed in the Level 1/2 interface portions of this report in Section 2.1.1.

2.4.2 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 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 factors during this timeframe include relief valve cycling, operator actions, secondary-side pressure, induced piping or SG tube failure, and hydrogen combustion.
- *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 (DCH), hydrogen combustion, vessel rocketing, and ex-vessel steam explosions (EVSEs).

Table 2-9 Systems for Which Reliability Estimates Are Not Generated

System/component	Comments
Post-accident monitoring instrumentation	Recall that no credit is given for actions when no AC/DC power is available post core damage; this equipment may or may not be in-service during the outage.
Security diesel	The security diesel provides power to both the security system and the Technical Support Center (TSC) in the case of a LOOP. Recall that 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 emergency diesel generators function but the security diesel does not. Also understand that the HEPs for modeled post-core-damage actions are high, and that alternative locations (TSC) and plant access (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 utilizing the backup location during an actual emergency on human error probabilities was not assessed.
Operational Support Center (OSC) and Emergency Operations Facility (EOF) availability	As above, the impacts of availability of these on human error probabilities was not assessed.
Effluent radiation monitors (e.g., main plant stack monitor)	These are not currently 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	Only relevant for substantial leaks into the auxiliary building (namely, containment isolation failures and LOI-OC (ISLOCA)).
Containment hydrogen recombiners	These are not credited anywhere in the model; they are retired-in-place.
Portable equipment (e.g., diesel-driven trailer-mounted pump)	Such equipment is used for RWST refill, internal and external containment spray, and SG feed—see preceding discussion regarding the effect of high HEPs.
Valves used in accident management	This includes valves associated with use of the portable equipment referenced above, as well as opening of the reactor head vents—see preceding discussion regarding the effect of high HEPs.
Fire detection and suppression systems	These are not currently considered by the model.

- *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 is 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 EVSEs 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. Sequences are intended to be modeled to the point that further core degradation or containment cavity basemat attack has been terminated and releases have stabilized. A back-stop time of 7 days from the initiating event is used for all sequences to be consistent with the severe accident progression simulation time. The issue of accident termination modeling and associated uncertainties are discussed further in Section 2.6.3.15.

Again, a single CET is used to handle all Level 2 PRA sequences in the reactor-at-shutdown Level 2 PRA. The CET contains the following top events:

1. *Additional Level 1 Recovery Considerations (1-L2-REC):* This top event is used as a means of giving special treatment to extremely slowly developing scenarios. Currently, it captures a set of conditions where the event progression is so slow that core damage would not be anticipated within 7 days. Preliminary revisions of the Level 1 PRA included several sequences, particularly for post-refueling POSs, with conditions where core damage would not be anticipated within 7 days. Ultimately, the Level 1 PRA modeling of human actions was re-visited to apply a more realistic timing evaluation for actions during slowly developing sequences in post-refueling POSs. These changes in the final model revision resulted in reducing the core damage frequency for post-refueling POSs that is carried forward to the Level 2 PRA.
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). It also captures a situation unique to the reactor-at-shutdown modeling related to containment overpressure prior to core damage during particular accident sequences where sustained boiling of the upper refueling cavity occurs. This top event is necessary in order 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-SD):* 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 it is evaluated via a DET. It is similar to the at-power Level 2 PRA analogue, with the addition of the possibility that the system is, by definition, at low pressure due to shutdown activities.

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.
5. Containment status during in-vessel phase (1-L2-DET-CONTVE-SD): 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- or temperature-induced SGTR. It is evaluated via a DET.
6. Scrubbing of radionuclide release during in-vessel and vessel failure phase (1-L2-DET-SCRUBE-SD): This event questions whether the phase of the radiological release occurring during the time frame at or before vessel breach is mitigated by containment spray operation. It is evaluated via a DET.
7. Containment status at vessel breach (1-L2-DET-CONTE-SD): 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, DCH, or hydrogen combustion. It is evaluated via 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 that consider energetic events at vessel breach such as steam explosions that could disperse the debris, rendering it coolable, and prevent concentrated basemat attack.
9. Scrubbing of radioactive release after vessel failure (1-L2-DET-SCRUBL-SD): 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 1-L2-DET-SCRUBE-SD (above), with the additional factor that water in the reactor cavity may be present to scrub the ex-vessel release. It is evaluated via a DET.
10. Containment status well after vessel failure (1-L2-DET-CONTL-SD): 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 via a DET.
11. Auxiliary building status (1-L2-DET-ABF-SD) – *Not used*. This event evaluates whether the auxiliary building fails due to overpressure during the post-core-damage accident progression. It is retained from the full-power model but is not used. The only sequences in the shutdown model with potential impacts on the auxiliary building are ISLOCA sequences. In this case, auxiliary building failure is assumed to occur based on the severe accident scenario simulation (case SD11.1) that predicts a deflagration in the auxiliary building with a pressure spike exceeding the building capacity.
12. RCS status for RC binning (1-L2-SUM-RCSSTATUS): This event is used to differentiate the RCS status in cases where the containment is open, because this makes a large difference in the accident source term. The RCS status is either intact,

vented/compromised, or with the vessel head removed, and these are treated as separate release categories.

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 86 sequences mapping to 12 release categories. The CET is shown in Figure 2-5, Figure 2-6, and Figure 2-7.

2.4.3 Development of Support Trees

The support trees take the following forms:

- Fault trees to support the top events in the bridge tree – these are discussed in Section 2.1.1.
- Linkage rule logic to support the plant damage state tree – these are discussed conceptually in Sections 2.1.2 and 2.1.3.
- Linkage rules to support CET/DET top events – these are discussed 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-AB)

A DET functionally expands a single top event in a main tree 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 because they enable the main CET to be simplified into only the 12 top events directly determinative of release category, 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.

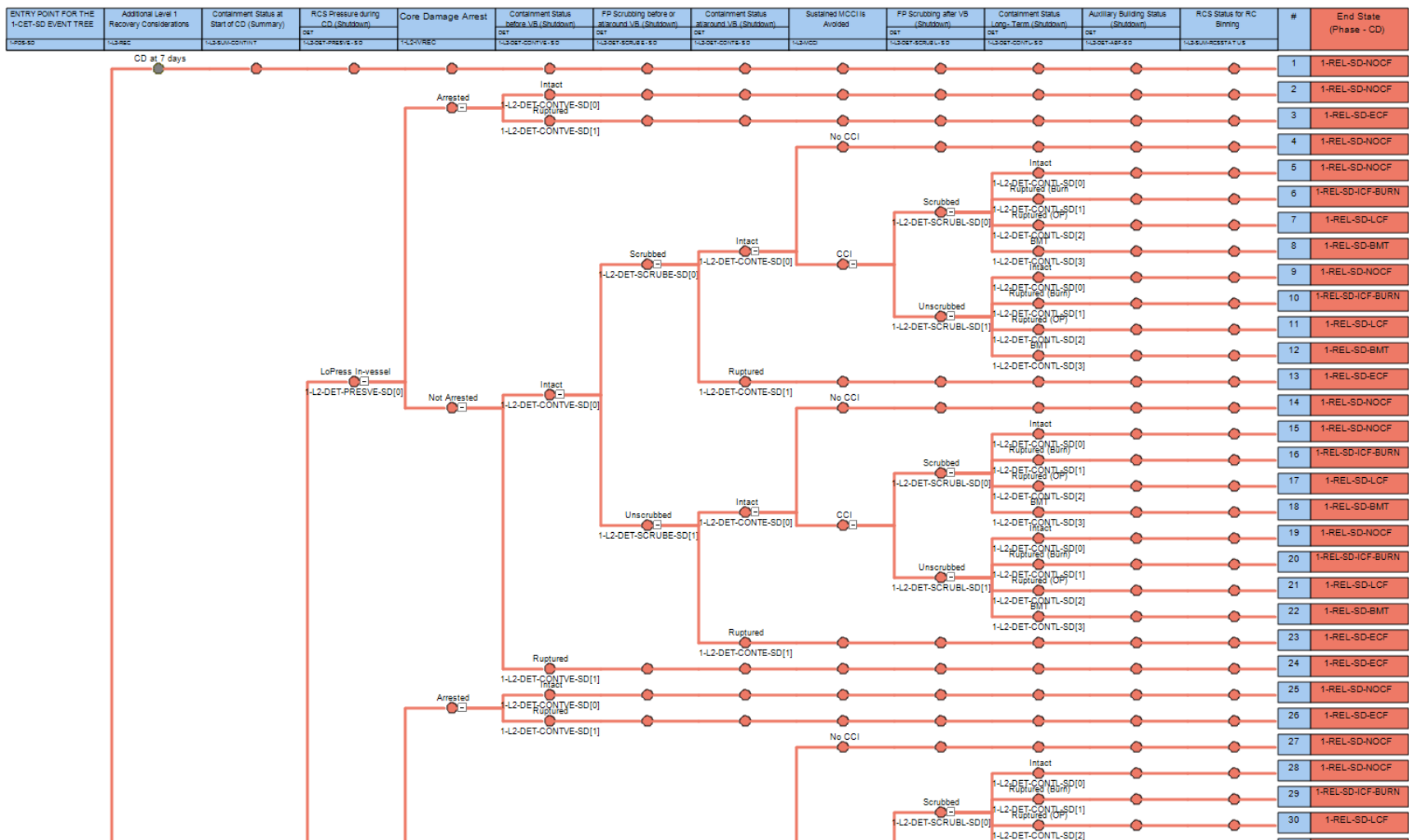


Figure 2-5 Containment Event Tree (1 of 3)

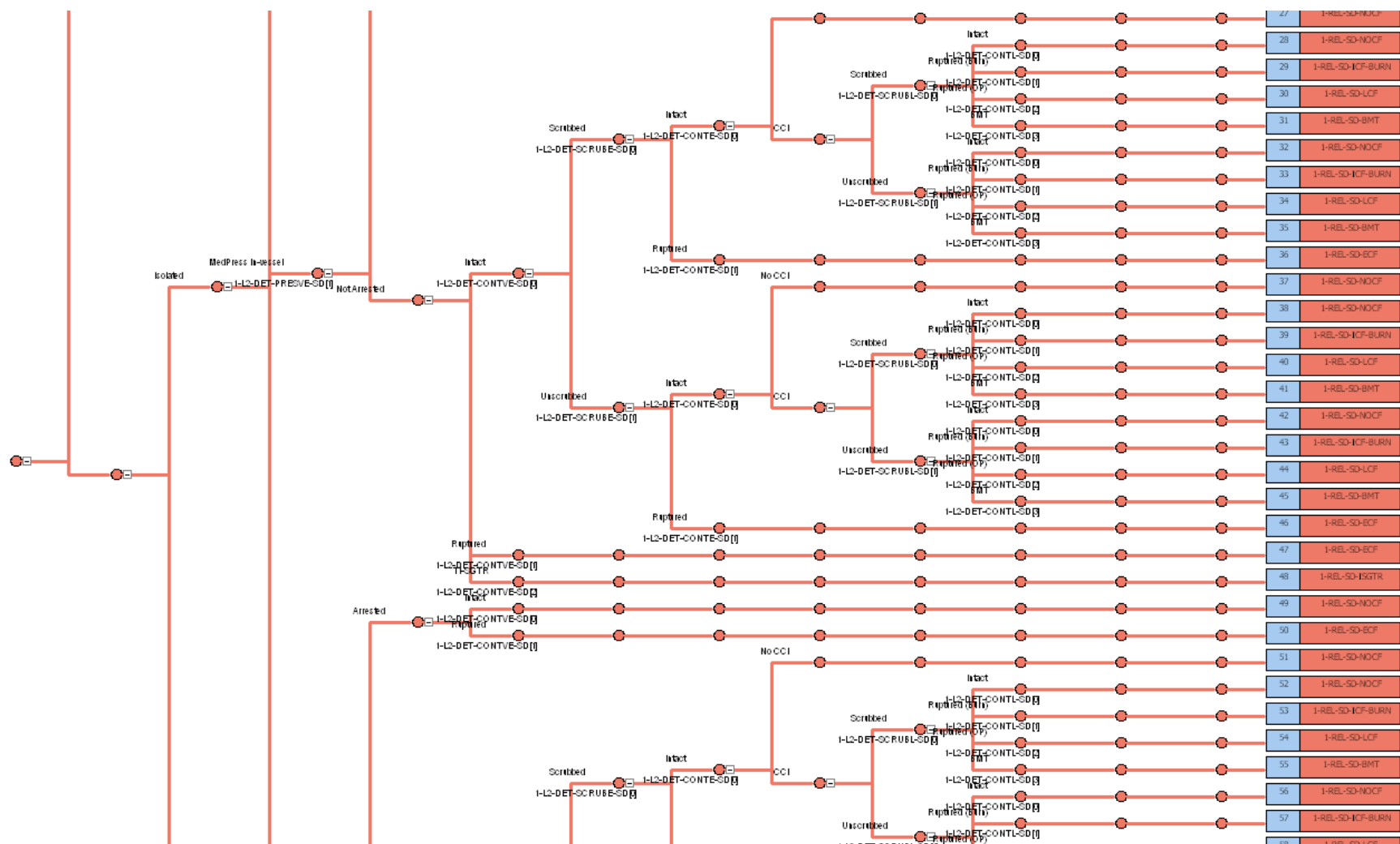


Figure 2-6 Containment Event Tree (2 of 3)

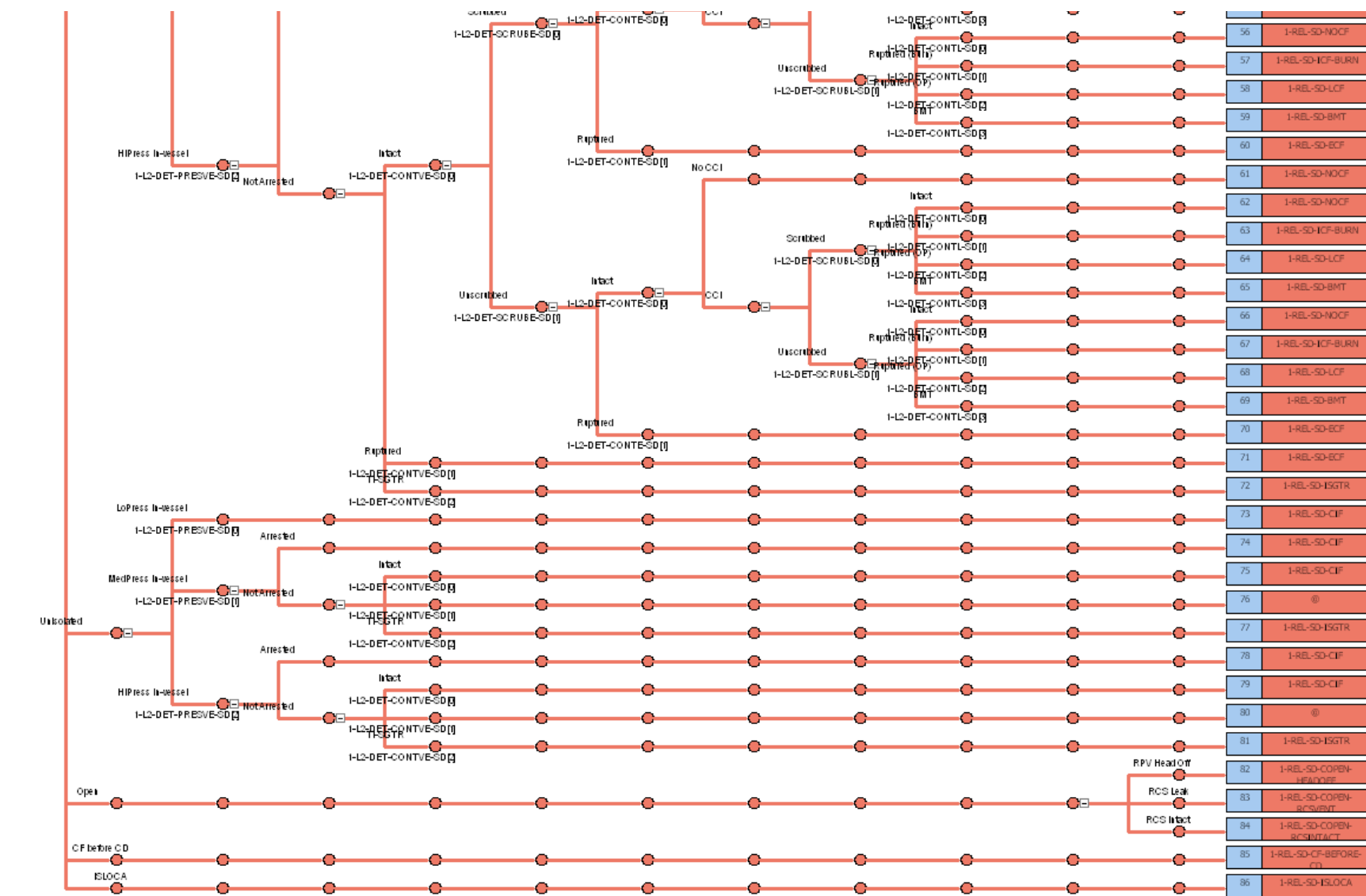


Figure 2-7 Containment Event Tree (3 of 3)

2.4.4 Human Reliability Model Development

Within the area of nuclear power risk assessment, 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 (EOPs). There are very few HRA applications that have supported PRA for post-core-damage accident sequences. Also, there has been very limited HRA method developments aimed at supporting such PRA studies (e.g., Baumont, 2000). Consequently, for the most part, Level 2 PRA HRA applications to-date have been performed using the existing HRA methods that were intended for use in supporting at-power, Level 1 internal events PRA. Additional background and discussion are available in the corresponding section of the reactor at-power Level 2 PRA main report (NRC, 2022a). The reactor at-power portion of this study developed a Level 2 PRA HRA quantification approach specifically for use in this project. This section and the discussion on shutdown post-core-damage HRA report in Appendix E discuss the extension of the approach used in the reactor at-shutdown Level 2 PRA.

Procedure/guidance scope

The first step is to determine the procedural/guidance scope of the reactor at-shutdown Level 2 PRA HRA, specifically regarding how the severe accident management guidelines (SAMGs) and EDMGs are, or are not, considered in the post-core-damage accident management modeling. In the timeframe at which this project is modeling the plant (circa 2012-2016), the SAMGs, EDMGs, and Emergency Plan all work concurrently, but are not integrated into a single governing framework. It is important to keep in mind that some of the same individuals (e.g., the Emergency Director [ED]) and organizational entities (e.g., the Technical Support Center [TSC]) have roles in the various processes outlined in these document sets.

The SAMGs are specifically focused on providing guidance to onsite accident management staff when core damage is imminent or has occurred, and thus when the EOPs no longer supply sufficient direction. The SAMGs that are used to inform the L3PRA project follow the contemporary Westinghouse SAMGs in place since 1998. The SAMGs provide guidance to the main control room crew and the TSC, and by extension to the Operations Support Center (OSC) and ED. The SAMGs (in place at the time) are specifically intended for reactor, at-power operation. They do not provide any guidance for reactor shutdown,⁸ or SFP, accident mitigation.

On this basis, the project team does not consider the SAMGs within the reactor at-shutdown Level 2 PRA HRA. Planned upgrades to the site-specific SAMGs and their process infrastructure based on changes to the vendor-generic SAMGs (and the associated SAMG technical basis revision in (EPRI, 2012a)) are specifically outside the scope of the Level 3 PRA project, based on application of the inclusion criteria found in the project's Technical Analysis Approach Plan (NRC, 2013a).

The EDMGs, on the other hand, are guidelines focused on responding to a loss of large area of the plant due to fire(s) or explosion(s), including situations in which normal command and control may be compromised. They focus on providing alternative means for (i) re-establishing

⁸ The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available, and thus not considered, at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

command and control; (ii) providing core, containment, and SFP cooling; and (iii) dealing with other emergency-related needs and responsibilities.

The EDMGs rely on a mix of pre-existing, installed equipment and portable equipment that was acquired specifically for this purpose. The EDMGs are not restricted to at-power reactor operation or post-core-damage conditions; however, some of the specific guidance and strategies are only applicable to at-power operations (the governing procedure is silent about their intended scope in this respect). Based on a detailed review of the reference plant situation and detailed discussions with plant staff, the project team has confidence that this guidance set would be in the forefront of responders' minds during a severe accident. On this basis, the project team considers the use of EDMG strategies in the reactor at-shutdown Level 2 HRA. To this end, Table 2-10 provides a mapping of the individual EDMG strategies to the conditions that may prompt their use during reactor-at-shutdown accident sequences.

Overall HRA process overview

The reactor at-power Level 2 PRA HRA approach, as described in Section 2.4.4 of (NRC, 2022a), was also used as the general framework for the reactor at-shutdown Level 2 PRA HRA. The analysis to extend the approach to the shutdown PRA is documented in Appendix E. The MELCOR analyses were used as the underlying basis for the accident progression and timings. As for the reactor at-power PRA, a screening-level HRA and a more detailed HRA were performed for each of the MELCOR representative sequences.

The lack of an over-arching SAMG “playbook” requires modifications to the Level 2 PRA HRA approach. It is assumed that the plant staff and TSC may continue to try those mitigation actions that failed to prevent core damage, which most typically would involve either forced or gravity feed into the RCS. The availability of additional time and additional plant staff may or may not warrant credit for these actions post core damage. Following that, the above EDMGs are expected to be consulted, the EDMGs relevant to the ongoing accident would need to be identified, and the available options would need to be prioritized. Here, it is assumed that the general hierarchy of the SAMGs (i.e., Severe Challenge Guidelines (SCGs) are higher priority than Severe Accident Guidelines (SAGs); SCG-1 is higher priority than SCG-2; etc.) would influence the TSC’s prioritization of the EDMGs. This provides a general set of priorities, which are then translated to the priority of upset conditions and the associated candidate actions, for varying RCS and containment status (which are expected to be the two high-level plant states with the most influence on identifying and prioritizing actions). This taxonomy is then re-cast to the relevant PDS and bridge tree top events, so that it can be employed in the PRA model. The resulting scheme is shown in Table 2-11.

It is important to keep in mind that, while based on a detailed knowledge of the plant’s EOP, SAMG, and EDMG guidance and framework, along with past discussions with plant operators and emergency response personnel regarding response to at-power accidents, this scheme represents a large departure from any actual codified response procedure or guidance. Rather, it is an attempt to acknowledge that plant staff will take actions to try and arrest the severe accident, and that there are relevant documents, training, and knowledge that they will rely on in doing so.

Table 2-10 Mapping of EDMG Strategies to Potential Shutdown Level 2 HRA Use

EDMG Strategy	Relevance to Reactor-At-Shutdown Level 2 HRA
Extensive Damage Mitigation Guideline	Situations where a loss of large area of the plant due to fire or explosion has compromised the MCR
RWST Makeup Using External Sources	Situations relying on injection from the RWST, for which regular RWST makeup means are unavailable
Containment Flooding Using Godwin ¹ Pump	Situations with (i) an intact containment and rising containment pressure or (ii) known high radiation levels within containment, and for which the engineered containment spray system is not restorable
Makeup to Condensate Storage Tank (CST)	Situations relying on injection from the CST, for which regular CST makeup means are unavailable
Fire System Strategies	General supporting function for other EDMGs strategies
Manual Operation of TDAFW Pump	n/a (no or insufficient steam source during shutdown)
Manually Depressurize SGs	Situations with rising RCS pressure, and for which normal means of opening SG atmospheric relief valves (ARVs) are unavailable
AFW Supply Using Godwin ¹ Pump	Situations relying on secondary-side heat removal, for which regular means of feeding the SGs are unavailable
Spraying Down Release Points	Situations with known, sustained fission product releases from a specific location (e.g., the equipment hatch)
Godwin ¹ Pump Setup and Operation	General supporting function for other EDMGs
Godwin ¹ Pump and Accessories	
SFP Makeup Using Internal Sources	n/a (except conceivably in situations where the fuel transfer tube is open, and the containment refuel cavity, and RCS can be re-flooded via the SFP)
SFP Makeup or Spray Using External Sources	
SFP Leakage Control Strategies	
SG Indication Measurement	Situations relying on secondary-side heat removal, for which regular means of monitoring SG level are unavailable
Rx Vessel Head Vent Strategy	Situations with rising RCS pressure, and for which normal means of controlling RCS pressure are unavailable
1. Godwin is the manufacturer's name for the portable trailer-mounted diesel-driven pump used by many plants (including the reference plant) 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."	

With the above scheme having been developed, the remainder of the reactor at-shutdown Level 2 PRA HRA approach is a variation on the at-power Level 2 PRA HRA. The generic timings used in the HRA are modified to account for shutdown-unique aspects, as are the screening criteria for HFE identification and HEP assignment. Similarly, minor modifications were needed for the diagnosis decision tree. No modification was required for the execution decision tree.

Table 2-11 Candidate Post-Core-Damage Actions Related to PDS and Bridge Tree Top Events

Cat.	ACCTYPE ¹	RCS Vent Flood	Containment	1 st Priority	2 nd Priority	3 rd Priority
I	Transient	Not vented	Closed & Isolated	Continued attempts to accomplish pre-core damage injection / recovery	Inject into SGs	Open head vents
II			Open or unisolated			Spray internally / externally in/on containment
III		Vented or Flooded	Closed & Isolated		Spray in containment	Spray internally / externally in/on containment
IV			Open or unisolated		Spray internally / externally in/on containment	
V	Station Blackout	All	All	No HRA credit is given for post-core-damage actions		
VI	Loss Of Coolant Accidents (LOCAs)	All	Closed & Isolated	Continued attempts to accomplish pre-core damage injection / recovery	Spray in containment	Spray internally / externally in/on containment
VII			Open or unisolated		Spray internally / externally in/on containment	

¹ This is the accident type based on plant conditions at the onset of core damage, as defined in the plant damage state binning. In particular, note that LOCA does not necessarily infer a pipe rupture (it may be a diversion of water or a deliberate opening of an RCS vent path) and transient does not necessarily infer a transient initiating event (a transient initiator can lead to a feed and bleed situation while a loss-of-inventory initiator can result in an isolated RCS).

2.4.5 Human Reliability Analysis of Post-Core-Damage Actions

With the approach in place, as described in the preceding section, the candidate actions were defined for each of the 13 representative MELCOR scenarios. This information is reproduced in Table 2-12. A screening and detailed HRA were then performed for the non-SBO sequences where core damage did occur, and this is described in Appendix E. This set of HEPs and HFEs, which only covers a sub-set of the categories previously shown in Table 2-11, was then extended to all categories so as to provide a more complete basis for incorporation of operator actions into the Level 2 PRA. This resulted in the assignments shown in Table 2-13. The basic events that are used to represent the HFEs, and where they are used in the Level 2 PRA event trees, are listed in Table 2-14.

A nuance of the Level 2 PRA HRA for reactor at-shutdown is the issue of equipment status during shutdown (i.e., being (or not being) in pull-to-lock). The at-power Level 1 HRA reflects a situation where virtually all equipment is in-service and will actuate upon receiving the relevant signal. The at-power Level 2 PRA HRA included an embedded assumption that operators successfully complete one of the first steps in the SAMGs which directs operators to place a broad range of non-operating equipment in pull-to-lock (i.e., will not actuate without operator intervention). The shutdown situation is more nuanced. For most plant configurations of interest to the reactor at-shutdown Level 1 PRA HRA, plant equipment will have been placed in pull-to-lock or out-of-service as part of plant cooldown and refueling outage entrance. In particular, the Level 1 PRA HRA credits manual operation of ECCS equipment (for forced and gravity feed)

that was previously in pull-to-lock. The Level 2 PRA HRA modeling does not currently address whether the equipment is placed back in pull-to-lock prior to core damage. Meanwhile, containment sprays are assumed to be initially out-of-service for the vast majority of plant conditions modeled in the Level 1 and Level 2 reactor at-shutdown PRAs (for all but POSs 3 and 13). Containment fan coolers are somewhat more variable in terms of their usage (0, 1, or 2 trains in-service at the time of the accident depending on the POS). Recovery of containment fan coolers is considered for a broader range of POSs relative to sprays. This situation is summarized in Table 2-15.

In general, since almost all uses of these systems are based on manual action to return equipment to operation, the existing treatment (though ambiguous) is reasonable. However, the lack of a SAMG-like direction to place all equipment in pull-to-lock, and ambiguities on the state of the equipment prior to core damage, have not been explicitly addressed.

Table 2-12 Mapping of MELCOR Scenario Plant Conditions to Candidate Actions

Scen. #	Plant/Accident Conditions	1 st Priority	2 nd Priority	3 rd Priority
1	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
2 ¹	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
3	Station blackout, containment un-isolated	No credit for post-core-damage human actions is considered		
4 ¹	Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
5	Station blackout, containment open	No credit for post-core-damage human actions is considered		
6 ^{1, 2}	Transient accident type, RPV not vented, containment open	Continued attempts to accomplish pre-core damage injection/recovery	Inject into SGs	Spray int/ext containment
7	Station blackout, containment open	No credit for post-core-damage human actions is considered		
8	Transient accident type, RPV not vented, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Inject into SGs	Open head vents
9	Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
10	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
11	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
12	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
13	Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		

Table 2-12 Mapping of MELCOR Scenario Plant Conditions to Candidate Actions (cont.)

Scen. #	Plant/Accident Conditions	1 st Priority	2 nd Priority	3 rd Priority
¹ These scenarios did not actually proceed to core damage in the representative sequence analysis for reasons related to either (a) details of their boundary condition specification (e.g., that a valve would not fail despite cycling an inordinate number of times) or (b) simplifications in the Level 1 PRA accident sequence modeling (e.g., outage symmetry despite decay heat differences or neglect of SG cooling when SGs are in wet layup). ² The MELCOR scenario SD06.1, as shown in Table 2-7, was considered as an alternative that could be used to generate HFEs for this scenario, since it proceeds to core damage. However, the reason it proceeds to core damage is the consideration of a relief valve failure which would actually make the simulated scenario a LOCA (rather than a transient) in the plant damage state ACCTYPE categorization. The transient-oriented HFEs, most notably injecting into the SGs, would thus not be the appropriate HFEs for this situation.				

Table 2-13 Final Set of Human Failure Events and Human Error Probabilities for Scenario Categories

Cat.	Description	HFE & HEP #1	HFE & HEP #2	HFE & HEP #3
I	Transient accident type, RPV not vented, containment closed & isolated	n/a	Inject into SGs ¹ – 0.24	Open head vents ² – 0.17
II	Transient accident type, RPV not vented, containment open	None of the MELCOR scenarios that proceeded to core damage covered this situation		
III	Transient accident type, RPV vented or flooded, containment closed and isolated	The Category VI results can be directly applied here.		
IV	Transient accident type, RPV vented or flooded, containment open or un-isolated	The Category VI results can be extended to cover this grouping, using the ISLOCA outcomes (similar radiological habitability concerns): • RWST refill and ECCS injection: HEP of 0.52 • Containment spray: HEP of 1.0		
V	SBO	n/a	n/a	n/a
VI	LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection – 0.14 ³	Spray in containment – 0.24 to 0.58 ⁴	Spray int/ext containment – n/a
VII	LOCA accident type, containment open or un-isolated	The Category VI results can be extended to cover this grouping, using the ISLOCA outcomes (similar radiological habitability concerns): • RWST refill and ECCS injection: HEP of 0.52 • Containment spray: HEP of 1.0		

¹ This action is only relevant prior to installation of the SG dams (which occurs during POS 6) or after removal of the SG dams (which occurs during POS 10).

² This action is only relevant for situations where hot leg nozzle creep rupture has not occurred.

³ This would increase to 0.52 for the corresponding ISLOCA variant.

⁴ The lower end of this range represents head-on conditions, and the upper end represents head-off conditions, with the distinction being the amount of ongoing containment leakage that might affect habitability near the connection point. This would increase to 1.0 for the corresponding ISLOCA variant.

Table 2-14 Basic Events Used to Represent Post-Core-Damage Operator Actions

Basic Event	Description	Used in Event Tree	Point Estimate Failure Probability
1-L2-OP-EDMGB-SD1	Operators refill RWST via EDMG pump (category VI scenario, post-refueling)	1-CET-SD (top event 1-L2-IVREC)	2.2E-1 ¹
1-L2-OP-EDMGB-SD2	Operators refill RWST via EDMG pump (category II, III, or IV scenario, post-refueling)	1-CET-SD (top event 1-L2-IVREC)	5.4E-1 ¹
1-L2-OP-EDMGB-SD3	Operators refill RWST via EDMG pump (category VI scenario, pre-refueling)	1-CET-SD (top event 1-L2-IVREC)	5.6E-1 ¹
1-L2-OP-EDMGB-SD4	Operators refill RWST via EDMG pump (category II, III, or IV scenario, pre-refueling)	1-CET-SD (top event 1-L2-IVREC)	7.4E-1 ¹
1-L2-OP-EDMGC-SD1	Operators spray containment (RPV head still on vessel)	1-L2-DET-SCRUBL-SD (top event 1-CONTSPRAY-H)	2.4E-1
1-L2-OP-EDMGC-SD2	Operators spray containment (RPV head removed from vessel)	1-L2-DET-SCRUBL-SD (top event 1-CONTSPRAY-H)	5.8E-1
1-L2-OP-EDMGH-SD1	Operators inject water to steam generators	1-L2-DET-PRESVE-SD (top event 1-L2-SUM-SGCOOL)	2.4E-1
1-L2-OP-EDMGP-SD1	Operators open reactor vessel head vents	1-L2-DET-PRESVE-SD (top event 1-L2-OP-PRIDEPRES-VE)	1.7E-1
¹ Failure probability includes phenomenological issues related to core coolability for the scenario-specific conditions, in addition to human error probability for carrying out the associated actions. For additional information on the approach for developing these probabilities, refer to Sections C.2 and C.3 in Appendix C.			

Table 2-15 Summary of Equipment Status Assumptions

System	Used in:	Assumption in internal events model	POS where available
ECCS	1-L2-IVREC 1-L2-MCCI 1-L2-CAVWATER-VE 1-L2-CAVWATER-L	In pull-to-lock	Potentially available in all POSs, but the system is assumed to be in pull-to-lock in POSs 4-12 by the L1 PRA (credit for ECCS in L1 is via manual action) – so they will be in pull-to-lock in these POSs unless L1 has re-instated them
Containment Coolers	1-L2-CCU-L	Not in pull-to-lock – assumed operating at CD	Potentially available in all POSs
Installed Containment Sprays	1-L2-CSS-E 1-L2-CSS-L	In pull-to-lock	Only in POSs 3 & 13

2.4.6 Level 2 PRA Model Quantification

The combined Level 1 and Level 2 PRA model was linked for all shutdown internal initiating events and solved using a truncation frequency of 10^{-12} per year. SAPHIRE version 8.2.6 and model version 421 were used. The total release frequency is shown in Table 2-16. The total release frequency is compared to the total PDS frequency and shutdown CDF results. Some differences in the numerical results can arise between the CDF and total release frequency due to issues that can influence the quantification accuracy, including large failure probabilities that are prevalent in the Level 2 PRA modeling, treatment of success terms, potential apportioning of non-minimal CDF cutsets to different release categories, frequency truncation, and application of post-processing substitution rules. These quantification issues can inflate, and in some cases reduce, the release frequency results with respect to the CDF. In this study, the frequency results are in close agreement with the CDF being approximately 1 percent higher than the total release frequency.

Table 2-17 provides the release category frequency contributions (the release category scheme itself is discussed in Section 2.5.1) for the reactor at-shutdown model. The frequencies are reported in terms of occurrence per calendar year, consistent with the frequencies reported in the Level 1 PRA for shutdown. Note that the frequency estimates incorporate factors accounting for the average duration spent in outage and in individual POSs per year. Figure 2-8 shows the same information in graphical form.

As can be seen, much of the release frequency (approximately 49 percent) is attributable to the intact containment end state (i.e., release category 1-REL-SD-NOCF). Other notable contributors include containment overpressure failure prior to core damage (approximately 17 percent), containment isolation failure (approximately 14 percent), and intermediate combustion-induced containment failure (approximately 14 percent). Three release categories represent releases with large containment openings (1-REL-SD-COPEN-HEADOFF, 1-REL-SD-COPEN-RCSINTACT, and 1-REL-SD-COPEN-RCSVENT). These collectively account for about 4 percent of the total release frequency. The shutdown PRA scope includes only limited exposure to plant states where SGs are available for heat removal (i.e., during POS 3 and POS 13). Accordingly, SGTR is not modeled as a shutdown initiating event, and while induced-SGTR is modeled, the frequency contribution is insignificant. The quantified frequency for the 1-REL-SD-ISGTR release category is below the 10^{-12} frequency truncation limit. Therefore, the 1-REL-SD-ISGTR release category is not analyzed further, and a representative source term is not developed for this release category.

The quantification results for all PRA models are impacted by the truncation limits used during the quantification process. The truncation limit used should demonstrate convergence towards a stable result. The trial use Level 2 PRA standard requirements related to quantification and truncation limits refers to corresponding requirements in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Level 1 PRA standard (ASME/ANS, 2009). The Level 1 PRA supporting requirements QU-B2 and QU-B3 dictate that:

- Accident sequences and associated system models are truncated at a sufficiently low cutoff value that dependencies associated with significant cutsets or accident sequences are not eliminated.

- An iterative process with convergence is considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF, and the final change is less than five percent.

The shutdown Level 2 PRA model was quantified at various truncation levels to ensure model release category frequency convergence. The results of these calculations are provided in Table 2-18. Based on these results, a truncation level of 10^{-12} is deemed sufficient to satisfy truncation requirements. While three release categories continue to show change in frequency greater than five percent, these categories are not significant contributors to the overall release or surrogate risk metrics. The significant release categories and surrogate risk metric results are discussed in Section 2.6.1.

These release category estimates represent the combination of contributions to release frequency across the modeled plant operating states (POSs). To this end, Table 2-19 provides a breakdown of the release category frequency for each release category by POS. This table shows the following:

- For 1-REL-SD-CF-BEFORE-CD and 1-REL-SD-COPEN-HEADOFF, all the frequency stems from POS 8E with the refueling cavity flooded.
- For 1-REL-SD-NOCF, the frequency is distributed across the range of modeled POSs with about 64 percent of the category frequency associated with POSs occurring after refueling.
- For all other release categories, the majority of the frequency stems from POSs prior to refueling.
- POS 6 (mid-loop operations prior to refueling) is the highest contributing POS for release categories 1-REL-SD-ICF-BURN, 1-REL-SD-BMT, and 1-REL-SD-ECF.
- Nearly all the ISLOCA release category frequency is attributed to POS 5A (pressurizer water solid operation). POS 5A is also a significant contributor to the open containment release category, 1-REL-SD-COPEN-RCSINTACT.

While each release category includes a mix of shutdown initiating events, POSs, and accident sequences, understanding the highest contributing modeling elements is helpful for characterizing the release attributes and assigning a representative accident simulation scenario and associated source term. The highest frequency contributing accident sequence for each release category is described below. See Appendix B for additional release category results including cutsets and importance measures.

1-REL-SD-BMT highest frequency accident sequence

A LOOP occurs during POS 5A. Onsite emergency AC power fails resulting in SBO. SG cooling with turbine-driven auxiliary feedwater (TDAFW) is successful, but core damage occurs due to failure to restore power and long-term shutdown cooling. Molten core-concrete interaction contributes to combustible gas generation, but an energetic combustion event that could challenge containment does not occur. Containment is breached due to gradual concrete erosion in the reactor cavity. The release is not scrubbed by sprays or water pools.

1-REL-SD-CF-BEFORE-CD highest frequency accident sequence

A LOOP occurs during POS 8E. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Containment penetrations are successfully isolated. Without containment heat removal systems available, containment gradually pressurizes due to evaporation and boiling of refueling cavity water, eventually resulting in containment overpressure failure. Core damage occurs a few hours after containment failure due to failure to restore power and RHR cooling.

1-REL-SD-CIF highest frequency accident sequence

A LOOP occurs during POS 8E. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Core damage occurs due to failure to restore power and RHR cooling. The containment equipment hatch and personnel airlock are successfully closed, but operators fail to isolate containment penetrations. The release is not scrubbed by sprays or water pools.

1-REL-SD-COPEN-HEADOFF highest frequency accident sequence

A LOOP occurs during POS 8E with the reactor vessel head removed and the refueling cavity flooded. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Core damage occurs due to failure to restore power and RHR cooling. Closure of the containment equipment hatch fails. The release has a direct pathway to the environment via the open equipment hatch.

1-REL-SD-COPEN-RCSINTACT highest frequency accident sequence

A LOOP occurs during POS 5A with the RCS intact. Onsite emergency AC power fails resulting in SBO. SG cooling with turbine-driven auxiliary feedwater is successful, but core damage occurs due to failure to restore power and long-term shutdown cooling. Closure of the containment equipment hatch fails. The release has a direct pathway to the environment via the open equipment hatch.

1-REL-SD-COPEN-RCSVENT highest frequency accident sequence

A LOOP occurs during POS 5B with the RCS vented via removal of three pressurizer safety valves. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Core damage occurs due to failure to restore power and RHR cooling. Closure of the containment equipment hatch fails. The release has a direct pathway to the environment via the open equipment hatch.

1-REL-SD-ECF highest frequency accident sequence

RHR cooling is lost due to overdraining when entering mid-loop during POS 6. Initial restoration of RHR cooling fails. Pumped injection is successfully initiated, but core damage occurs due to failure to restore long-term shutdown cooling. Containment penetrations are successfully isolated. Post-core-damage action to recover injection fails. In-vessel steam explosion (IVSE) occurs causing containment failure with 4-inch equivalent diameter at 24 hours after event initiation.

1-REL-SD-ICF-BURN highest frequency accident sequence

RHR cooling is lost due to overdraining when entering mid-loop during POS 6. Initial restoration of RHR cooling fails. Pumped injection is successfully initiated, but core damage occurs due to failure to restore long-term shutdown cooling. Containment penetrations are successfully isolated. Post-core-damage action to recover injection fails. Post-core-damage action to establish containment spray is successful. A detonation occurs during the period after vessel breach resulting in containment failure. The representative time to reach detonable conditions in containment is 60 hours after event initiation based on the stochastic combustion modeling.

1-REL-SD-ISLOCA highest frequency accident sequence

Loss of inventory occurs during POS 5A with failure of an RHR relief valve to reclose after opening. Operators fail to diagnose and isolate the loss before RCS makeup is needed. Pumped injection is successfully initiated, but core damage occurs due to failure to restore long-term shutdown cooling. Loss of inventory results in a containment bypass pathway to the auxiliary building through the stuck-open RHR relief valve.

1-REL-SD-LCF highest frequency accident sequence

A LOOP occurs during POS 5B. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Core damage occurs due to failure to restore power and RHR cooling. Containment penetrations are successfully isolated. Post-core-damage actions to recover injection and containment spray fail. Molten core-concrete interaction contributes to combustible gas generation, but an energetic combustion event that could challenge containment does not occur. Without containment heat removal systems available, containment gradually pressurizes resulting in containment overpressure failure. The release is not scrubbed by sprays or water pools.

1-REL-SD-NOCF highest frequency accident sequence

A LOOP occurs during POS 8L after refueling. Onsite emergency AC power fails resulting in SBO. Gravity-driven injection fails. Core damage occurs due to failure to restore power and RHR cooling. Containment penetrations are successfully isolated. Containment remains intact.

Table 2-16 Comparison of CDF to Total Release Frequency

Total Level 1 CDF	$1.25 \times 10^{-5}/\text{year}$
Total PDS Frequency	$1.25 \times 10^{-5}/\text{year}$
Total Release Category Frequency	$1.24 \times 10^{-5}/\text{year}$
CDF to Release Frequency Difference	~1%

Table 2-17 Calculated Release Category Frequencies

Release Category	Release Category Description	Frequency (per year)	Percentage Contribution
1-REL-SD-CIF	Containment isolation failure	1.7E-06	13.8%
1-REL-SD-COPEN- HEADOFF	Containment hatch/airlock open; RPV head off	7.8E-08	0.6%
1-REL-SD-COPEN- RCSINTACT	Containment hatch/airlock open; RCS intact	2.7E-07	2.2%
1-REL-SD-COPEN- RCSVENT	Containment hatch/airlock open; RCS vented	1.3E-07	1.1%
1-REL-SD-ISGTR	Temperature-induced SGTR	< 1E-12	< 0.1%
1-REL-SD-ISLOCA	Interfacing systems LOCA (loss of inventory outside of containment)	1.3E-07	1.1%
Subtotal	Containment bypass categories	2.3E-06	18.7%
1-REL-SD-CF- BEFORE-CD	Late containment overpressure failure due to boiling of refueling cavity inventory before core damage	2.1E-06	16.7%
1-REL-SD-ECF	Early containment failure at or around time of vessel breach due to an energetic event	5.1E-10	< 0.1%
1-REL-SD-ICF- BURN	Combustion-induced containment failure occurring some hours after vessel breach	1.7E-06	13.8%
1-REL-SD-LCF	Late containment failure due to overpressure	5.5E-08	0.4%
Subtotal	Containment failure categories	3.8E-06	30.9%
1-REL-SD-BMT	Basemat melt-through	1.2E-07	1.0%
1-REL-SD-NOCF	Containment intact	6.1E-06	49.4%
Subtotal	Containment intact and BMT	6.2E-06	50.4%
Total		1.2E-05	100%

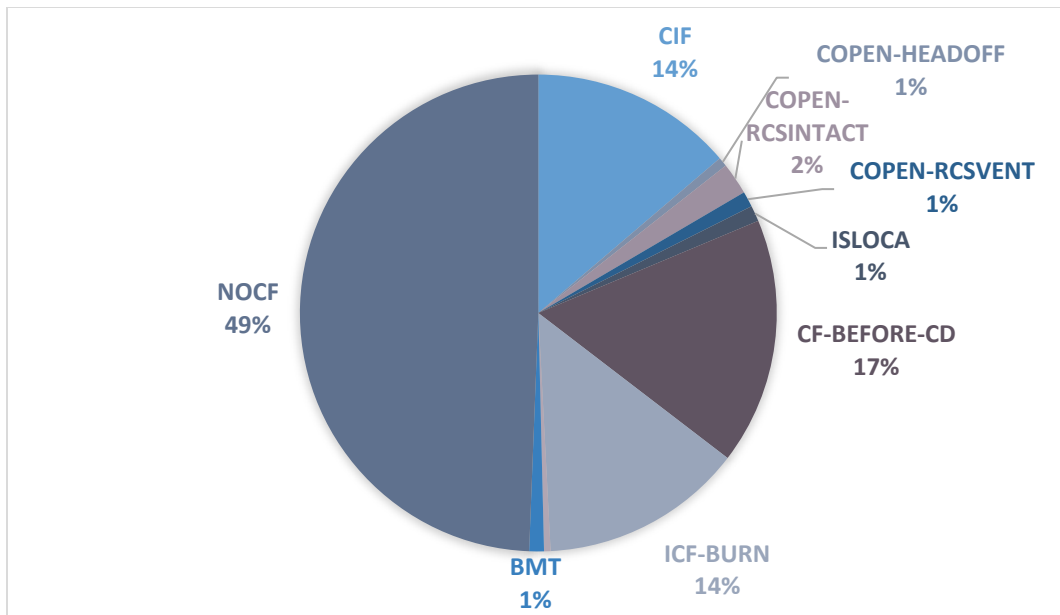


Figure 2-8 Relative Release Category Frequencies

Table 2-18 Release Frequency Truncation Analysis

	Truncation Frequency 1.0E-11 (per year)		Truncation Frequency 1.0E-12 (per year)			Truncation Frequency 1.0E-13 (pe year)		
Release Category	Frequency (per year)	Cut Set Count	Frequency (per year)	Cut Set Count	Freq. % Difference	Frequency (per year)	Cut Set Count	Freq. % Difference
Total Release	1.2E-05	21033	1.2E-05	66550	1.3%	1.3E-05	226475	1.4%
1-REL-SD-BMT	1.1E-07	1303	1.2E-07	5211	11.5%	1.3E-07	20839	6.4%
1-REL-SD-CF-BEFORE- CD	2.1E-06	1086	2.1E-06	2512	0.2%	2.1E-06	6310	0.4%
1-REL-SD-CIF	1.7E-06	869	1.7E-06	2331	0.3%	1.8E-06	10081	4.3%
1-REL-SD-COPEN- HEADOFF	7.7E-08	142	7.8E-08	445	1.3%	7.8E-08	1111	0.3%
1-REL-SD-COPEN- RCSINTACT	2.6E-07	448	2.7E-07	1410	1.2%	2.7E-07	3995	0.3%
1-REL-SD-COPEN- RCSVENT	1.3E-07	369	1.3E-07	1153	2.0%	1.3E-07	3811	0.8%
1-REL-SD-ECF	2.1E-10	11	5.1E-10	97	143.8%	7.2E-10	531	42.4%
1-REL-SD-ICF-BURN	1.7E-06	4750	1.7E-06	15117	2.2%	1.7E-06	51907	1.1%
1-REL-SD-ISGTR	< 1.0E-11	0	< 1.0E-12	0	N/A	4.8E-13	2	N/A
1-REL-SD-ISLOCA	1.3E-07	66	1.3E-07	131	0.2%	1.3E-07	383	0.3%
1-REL-SD-LCF	5.0E-08	466	5.5E-08	1902	9.7%	5.7E-08	7189	4.0%
1-REL-SD-NOCF	6.0E-06	11523	6.1E-06	36241	1.4%	6.2E-06	120316	0.9%
Surrogate Release Metric								
LERF	5.3E-07	883	5.3E-07	2694	1.1%	5.3E-07	8189	0.4%
LRF	6.0E-06	7741	6.1E-06	23196	0.9%	6.2E-06	78131	1.7%
Overall release	6.2E-06	9510	6.3E-06	30309	1.2%	6.4E-06	106159	1.8%

Table 2-19 Release Category Frequency Contribution by Plant Operating State

RC Bin	Plant Operating State Relative Contribution to RC (%)												
	3	4	5A	5B	6	7	8E	8L	9	10	11	12	13
SD-BMT	<1	2	21	17	25	5			8	11	3	5	3
SD-CF-BEFORE-CD							100						
SD-CIF	<1	3	26	<1	5	<1	63		<1	<1	<1	1	
SD-COPEN-HEADOFF							100						
SD-COPEN-RCSINTACT		8	62								10	20	
SD-COPEN-RCSVENT			4	56		3			37				
SD-ECF		<1	20	10	55	3			2	5	<1	4	1
SD-ICF-BURN	<1	<1	20	13	48	3	<1		6	7	1	1	<1
SD-ISGTR													
SD-ISLOCA			97		1	<1	<1		<1	2			
SD-LCF	2	<1	4	52	25	17							
SD-NOCF	<1	1	15	6	11	2	<1	39	5	10	3	5	2
Total	<1	1	16	6	13	1	26	19	4	6	2	3	1

2.4.7 Parameter Uncertainty Characterization

The reactor at-shutdown Level 2 PRA uncertainty characterization follows a similar approach as that developed for the at-power internal events and floods Level 2 PRA. The two broad categories of uncertainty that are addressed are model uncertainty and parameter uncertainty.

Model uncertainty is any uncertainty that impacts the model results related to modeling assumptions, model scope, or underlying supporting analyses. Model uncertainties are typically assessed using sensitivity studies. The sensitivity studies consider alternate assumptions and consider the impacts on the model results. A set of model uncertainty sensitivity studies for the reactor at-shutdown Level 2 PRA are described in Section 2.6.3.

Parameter uncertainty is any uncertainty that relates to a parameter in the PRA logic model that has a probability value. The uncertainty of each parameter is characterized by a probability distribution. The overall uncertainty is estimated by using a sampling approach to propagate the uncertainty of all the contributing parameters. The results of the parameter uncertainty assessment are presented below. Parameter uncertainty associated with Level 1 basic events are treated as described in the at-power Level 1 PRA (NRC, 2022b) and shutdown Level 1 PRA (NRC, 2022a). Refer to Appendix C for the uncertainty parameters that are developed for the reactor at-shutdown Level 2 PRA model basic events representing human failures, severe accident phenomena, and other containment event tree modeling.

The mean release frequencies for each release category are estimated using a Monte Carlo sampling approach with the SAPHIRE software. The state-of-knowledge correlations are addressed by assigning correlation classes to basic events with similar component types and failure modes that use the same operating experience data for determining the uncertainty distribution. The state-of-knowledge correlation treatment is described in the SAPHIRE technical reference manual (Smith, 2011). The point estimate frequency, mean values, and uncertainty parameter results are provided in Table 2-20.

The parameter uncertainty results are based on 5000 samples that are generated during the sampling process. The mean values of the sampled uncertainty distributions for each release category frequency show stability as the number of samples approaches 5000. The sampled mean frequency values approach stable values that are near to, but not equal to, the point estimate frequency values. The correlations between different model parameters that are derived from the same source data can inflate the sampled mean with respect to the point estimate.⁹ Another effect that can influence the mean value is discarding of invalid samples. In some cases, invalid samples are discarded if a probability with a value greater than 1.0 is selected. The model includes several parameters related to severe accident phenomena with failure probabilities near 1.0. Although care is taken in defining the parameter uncertainty distributions, samples from the tails of the distributions with values greater than 1.0 can occur. The discarding of probability samples with values greater than 1.0 can bias the sampled mean value to be less than the point estimate. Therefore, the sampled mean value and the point estimate value are not expected to exactly match.

⁹ See NUREG-1855, Volume 1, Appendix 4-A for discussion of the impacts of epistemic correlation.

Table 2-20 Release Category Frequency Contribution by Plant Operating State

Release Category	Point Estimate Freq.	Mean Value Freq.	5th Per-centile	Median	95th Per-centile	Standard Deviation
1-REL-SD-BMT	1.2E-07	1.2E-07	5.4E-09	6.4E-08	4.0E-07	1.8E-07
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.0E-06	3.3E-07	1.4E-06	5.3E-06	1.8E-06
1-REL-SD-CIF	1.7E-06	1.5E-06	1.4E-07	1.0E-06	4.7E-06	1.7E-06
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.0E-08	1.9E-09	2.8E-08	2.6E-07	1.4E-07
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.3E-07	7.1E-09	9.4E-08	8.8E-07	5.0E-07
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.1E-07	8.3E-09	5.0E-08	4.0E-07	1.9E-07
1-REL-SD-ECF	5.1E-10	4.7E-10	9.2E-13	2.8E-11	1.2E-09	4.6E-09
1-REL-SD-ICF-BURN	1.7E-06	1.7E-06	2.8E-08	1.1E-06	5.1E-06	2.3E-06
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	1.9E-09	1.9E-08	4.9E-07	5.7E-07
1-REL-SD-LCF	5.5E-08	4.4E-08	5.2E-10	1.2E-08	1.8E-07	1.1E-07
1-REL-SD-NOCF	6.1E-06	5.9E-06	1.2E-06	4.3E-06	1.6E-05	5.2E-06
Total release frequency (all release categories)	1.2E-05	1.2E-05	2.7E-06	8.9E-06	3.0E-05	9.9E-06
Surrogate risk metrics						
LERF	5.3E-07	5.0E-07	3.9E-08	2.4E-07	1.6E-06	2.2E-06
LRF	6.1E-06	5.7E-06	1.1E-06	4.4E-06	1.4E-05	5.0E-06

2.5 Radiological Source Term Analysis

The radiological source term analysis consists of three interrelated steps:

1. Definition of the release category binning logic (Section 2.5.1)
2. Development of source terms for the various release categories (Section 2.5.2)
3. Consideration of uncertainties in the source term development (Section 2.5.3)

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 an outcome of the PDS representative sequence analyses and sensitivity analyses and specify which source term(s) will be associated with each release category. The objective of the third step is to identify important source term uncertainties.

2.5.1 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 number 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 selected utilizes functional sequence characterizations related to:

- RCS configuration
- containment bypass pathways
- containment status at the time of core damage
- containment status following core damage

Unlike the at-power PRA, fission product scrubbing is not a key determiner of release magnitude for the at-shutdown results. This is due to a mix of different factors including the shutdown model having scenarios with containment failure prior to core damage and reduced likelihood of operators successfully implementing post-core-damage actions that lead to enhanced fission product scrubbing.

Table 2-21 shows how the release categories map to the containment and RCS status categories. Meanwhile, Table 2-22 provides a description of each release category.

Table 2-21 Mapping of Release Categories to Containment and RCS Status

Containment Status	RCS Status		
	Intact	Vented / Compromised	Head Off
Bypassed – ISLOCA	1-REL-SD-ISLOCA (also referred to as LOI-OC)		
Isolation failure	1-REL-SD-CIF		
Hatch/airlock open	1-REL-SD-COPEN-RCSINTACT	1-REL-SD-COPEN-RCSVENT	1-REL-SD-COPEN-HEADOFF
Overpressure failure before core damage	N/A	N/A	1-REL-SD-CF-BEFORE-CD
Bypassed - ISGTR	1-REL-SD-ISGTR	N/A	N/A
Fails at or around the time of vessel breach	1-REL-SD-ECF		
Fails hours (e.g., 35 hours) after vessel breach	1-REL-SD-ICF-BURN		
Overpressure failure late	1-REL-SD-LCF		
Basemat melt-through	1-REL-SD-BMT		
Containment remains intact	1-REL-SD-NOCF		

Table 2-22 Description of Release Categories

Name	Description
1-REL-SD-COPEN-HEADOFF	The containment hatch and/or personnel airlock are open at the time when the accident begins, and operators fail to reclose them. The release is considered to reach the environment directly via these large openings. Furthermore, the plant is in an operating state at the time of the accident where the reactor vessel head has been removed, so in-vessel release from the core benefits from little to no retention inside the RCS.
1-REL-SD-COPEN-RCSVENT	The containment hatch and/or personnel airlock are open at the time when the accident begins, and operators fail to reclose them. The release is considered to reach the environment directly via these large openings. Prior to any breach of the reactor vessel lower head by core debris, the RCS is vented or has a persistent leak not including normally cycling relief valves (e.g., pipe break or manual bleed), such that retention of aerosols inside the RCS may be intermediate in magnitude.
1-REL-SD-COPEN-RCSINTACT	The containment hatch and/or personnel airlock are open at the time when the accident begins, and operators fail to reclose them. The release is considered to reach the environment directly via these large openings. Prior to any breach of the reactor vessel lower head by core debris, the RCS is intact, such that releases from the RCS to the containment can occur only by normal cycling of the pressurizer and/or RHR relief valves; therefore, significant retention of aerosols inside the RCS is possible.
1-REL-SD-ISLOCA (also referred to as LOI-OC)	The accident consists of a pipe break to a location outside of the containment (i.e., into the auxiliary building). Releases reach the auxiliary building via this break from the RCS, and from there to the environment. The release category does not distinguish between cases where the auxiliary building remains intact and leaks relatively slowly to the environment, from cases where the auxiliary building fails due to steam overpressure or energetic event. Also, whether the release may be scrubbed by a water pool overlying the break is left unspecified.
1-REL-SD-ISGTR	One or more SG tubes fail after the start of core damage due to temperature-induced creep rupture. The release to the environment occurs via the break created in the damaged SG, and from there to the environment via atmospheric relief valves or secondary-side leaks. The release category does not distinguish between cases where relief valves cycle normally in response to steam line pressure versus cases where the valves have failed in the open position or been forced open by previous operator action.
1-REL-SD-CF-BEFORE-CD	The containment fails due to steam overpressure sometime in between the start of the accident and the start of core damage. This release may or may not benefit from any aerosol scrubbing.
1-REL-SD-ECF	The containment fails some time in between the start of core damage and the time frame at and around vessel breach due to an energetic event. This release may or may not benefit from any aerosol scrubbing.
1-REL-SD-CIF	Release from the containment to the environment occurs via a containment penetration that fails to be isolated by the containment isolation system or by manual action, or a pre-existing leakage path (not including open containment hatch or personnel airlock). This release may or may not benefit from any aerosol scrubbing. Also, depending on the type and location of the isolation failure, the release may benefit from some retention inside the auxiliary building.
1-REL-SD-ICF-BURN	The containment fails some hours after vessel breach due to a global deflagration or detonation. This release may or may not benefit from any aerosol scrubbing.
1-REL-SD-LCF	The containment fails at some time between tens of hours after the time of vessel breach, up to as late as seven days after the start of the accident, due to long-term quasi-static overpressure. Releases to the environment are not mitigated significantly by sprays or water pools.

Table 2-22 Description of Release Categories (cont.)

Name	Description
1-REL-SD-BMT	The containment eventually fails due to basemat ablation from sustained core-concrete interaction, at some point within seven days of the start of the accident. Only the airborne component of release to the environment (which stems from normal design-basis containment leakage while the containment is pressurized) is modeled here.
1-REL-SD-NOCF	Containment is not open, unisolated, 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.

2.5.2 Development of Source Terms for the Various Release Categories

This section provides the assignment of source terms to release categories. It starts by providing the criteria for this selection process. It then provides the emergency action level declarations for the various sequences. Following this, the source term mapping is provided, followed by additional subsections that each justify the mapping in cases where multiple candidate source terms exist.

Release Category Source Term Criteria

Criteria for assigning a particular 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-SD-LCF must result in predicted late containment failure).
- Among possible candidates, the chosen calculation should ideally be free from known, significant modeling inaccuracies.
- Among possible candidates, the chosen calculation's predicted accident progression should ideally be consistent with the assumptions of the Level 2 PRA logic model.
- Among possible candidates, the chosen calculation should ideally look more reflective in its particulars of the high-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 is more encompassing of the range of potential outcomes.

Emergency Action Level Declarations

Timely monitoring and declaration of Emergency Action Levels (EALs) is assumed. This is an extension of the analogous assumption from the at-power Level 2 PRA, for which the basis is provided in the Appendix D entry entitled, "Emergency Action Level Monitoring Assumptions," in the at-power Level 2 PRA (NRC, 2022a). As catalogued in the post-core-damage HRA for shutdown (in Appendix E of this report), the abnormal operating procedure paths relevant to the studied shutdown accidents have a callout to the EALs analogous to the emergency operating procedures. In addition, the over-arching Emergency Plan commitments discussed in the

aforementioned at-power PRA documentation entry also apply to the reactor at-shutdown accident situation.

During the development of the reactor at-shutdown Level 2 PRA, preliminary EAL timings were developed for 12 representative sequences during the screening HRA, as documented in Appendix E. These preliminary estimates were extended to apply to all the representative sequences and sensitivity cases, using the same EAL timing assumptions applied to similar accident scenarios. The EAL classification and timing assumptions are summarized in Table 2-23. In most cases, the sensitivity cases have the same preliminary EAL declarations as their associated representative sequences. The exceptions are case SD06.1 and case 13, which were developed after the preliminary EAL timing estimates. These cases followed a similar evaluation of the reference plant's guidelines, and the preliminary EAL declarations and bases are provided in Table 2-23.

These EAL estimates are based on the reference plant's guidelines defining EAL thresholds and bases. However, in some cases, exactly how the thresholds would be interpreted and applied at the reference plant is uncertain. The EAL timings for station blackouts are especially uncertain. The site area emergency (SAE) and general emergency (GE) initiating conditions rely on indications of reactor vessel level and radiation monitoring. For station blackout sequences, where all instrumentation is lost after four hours, these indications may not be readily identified. In these station blackout scenarios, ED discretion is expected to be invoked to declare SAE and GE prior to onset of a radiological release. Note that, unlike the at-power EALs, there are not SAE and GE declarations in Modes 5 and 6 associated with extended loss of all AC power. There are EALs associated with inability to monitor RPV level, but they refer to other forms of indication to corroborate the need to declare.

Mapping Source Terms to Release Categories

Table 2-24 shows (for each accident simulation) the relationship between the declaration of a GE versus timings in terms of the radiological release, in order to provide a sense of the timing considerations in selecting a representative source term. In general, these results show a significant amount of time between the provisional GE declaration and the onset of elevated environmental releases. Exceptions include the SD07-series, SD08.1, and SD11.1, so these cases are highlighted to note the potential impacts of the EAL timings on the consequence analysis.

Table 2-25 maps the source terms from the candidate MELCOR cases 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. Note that the at-power internal events and floods Level 2 PRA (NRC, 2022a) includes an iterative look at the source term binning judgments and concluded that release magnitude is somewhat more important than release timing when making source term selections. However, this may not be generally true and depends on the source terms characteristics and consequence metrics of interest.

Table 2-23 Emergency Action Level Classification and Timing Assumptions

Rep. Seq. No.	Alert Decl. ¹ (hr)	SAE Decl. ¹ (hr)	GE Decl. ¹ (hr)	Basis (Referencing Criteria from the Reference Plant EAL Thresholds and Bases Guidelines)
SD01	~5	11	11.5	- Alert: Loss of RCS inventory and RCS temperature exceeding 200 °F will apply starting at ~9 hours; based on the simulation results. Elevated RCS temperature conditions briefly apply around t=30 minutes, and temperatures are close to exceeding 200 °F thereafter; it is assumed that an ALERT is declared at the mid-point of this time window, based on ED discretion. - SAE: RPV level at top of active fuel with containment closure established - GE: Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig
SD01.1 through SD01.13				
SD02	~20	n/a	n/a	- Alert: RCS temperature > 200F for >20 minutes, containment closed, and RCS integrity not established. - SAE: Site Area conditions are never met. - GE: General Emergency conditions are never met.
SD03	<1	~26	~33	- Alert: [Loss of all offsite and onsite ac sources for greater than 15 minutes. - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 26 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under ED discretion. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared upon RWST depletion, at 33 hours. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
SD03.1				
SD04	<1	~91	>168	- Alert: Loss of all offsite and onsite ac sources for greater than 15 minutes. - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 91 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100 °C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under ED discretion. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is not declared prior to the end of the simulation.

Table 2-23 Emergency Action Level Classification and Timing Assumptions (cont.)

Rep. Seq. No.	Alert Decl. ¹ (hr)	SAE Decl. ¹ (hr)	GE Decl. ¹ (hr)	Basis (Referencing Criteria from the Reference Plant EAL Thresholds and Bases Guidelines)
SD05	<1	~6	~48	- Alert: Loss of all offsite and onsite ac sources for greater than 15 minutes - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at six hours, under ED discretion, when significant steam would be exiting the open hatch. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via ED discretion based on extended boiling with (i) no compensating injection and (ii) containment open. A time of 48 hours is assumed. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
SD06	~8	n/a	n/a	- Alert: RCS pressure increasing greater than 10 psig - SAE: Site Area conditions are never met. - GE: General Emergency conditions are never met.
SD06.1	~8	~25	~38	- Alert: RCS pressure increasing greater than 10 psig - SAE: RCS level <72% RVLIS with containment closure not established. - GE: Unexplained containment sump level rise, RPV level >63% for >30 minutes, and containment closure not established
SD07	<1	~10	~15	- Alert: Loss of all offsite and onsite ac sources for greater than 15 minutes - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at 10 hours, under ED discretion, when significant steam would be exiting the open hatch. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via ED discretion based on extended boiling with (i) no compensating injection and (ii) containment open. At around 15 hours, core damage is starting, which will also lead to increasing radiation doses near the open hatch. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
SD07.1				
SD07.2				
SD08	~20	~20	~28	Alert: Potential loss of the fuel clad barrier based on heat sink with SG narrow range level < 10%
SD08.1				

Table 2-23 Emergency Action Level Classification and Timing Assumptions (cont.)

Rep. Seq. No.	Alert Decl. ¹ (hr)	SAE Decl. ¹ (hr)	GE Decl. ¹ (hr)	Basis (Referencing Criteria from the Reference Plant EAL Thresholds and Bases Guidelines)
SD08.2				- SAE: Potential loss of the fuel clad barrier and RCS barrier based on heat sink with SG narrow range level < 10% - GE: Loss of fuel clad barrier based on core cooling with RVLIS < 41% and CETC > 711F; Loss of RCS Barrier based on high containment radiation readings due to ongoing fuel damage and cycling PORV; Potential loss of containment barrier based on core cooling conditions for > 15 minutes with RPV level < 63%
SD09	<1	~23	~24	- Alert: Loss of all offsite and onsite ac sources for greater than 15 minutes. - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 23 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100 °C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under ED discretion. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared when the RWST gravity-feed becomes dead-headed and under the presumption that RWST level can be obtained, at 24 hours. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
SD10	~2.5	8	8.5	- Alert: RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established. - SAE: RPV level at top of active fuel with containment closure established - GE: Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig.
SD10.1				
SD11	~6	17	18	- Alert: RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established. – Earlier declaration might occur due to fluctuating temperatures around 200 °F between two to six hours. - SAE: RPV level at top of active fuel with containment closure established. - GE: Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig.
SD11.1				
SD12	9	17	30	

Table 2-23 Emergency Action Level Classification and Timing Assumptions (cont.)

Rep. Seq. No.	Alert Decl. ¹ (hr)	SAE Decl. ¹ (hr)	GE Decl. ¹ (hr)	Basis (Referencing Criteria from the Reference Plant EAL Thresholds and Bases Guidelines)
SD12.1				- Alert: RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established. - SAE: RPV level at top of active fuel with containment closure established. - GE: Likely via a combination of conditions with significant uncertainty: Containment sump level rise and RPV level less than top of active fuel for >30 minutes are met by 18 hours, but a containment challenge is also required; containment pressure hovers around the 3 psig threshold after ~30 hours; meanwhile, the 6% H2 limit is probably met soon before the 2nd large deflagration at ~49 hours or ED Discretion. 30 hours is selected as the GE declaration time as it is the earliest threshold reached.
SD13	<1	1	~5	- Alert: Loss of all offsite and onsite ac sources for greater than 15 minutes. - SAE: Declaration of Site Area Emergency (SAE) is difficult to predict, station blackout conditions may challenge plant indications, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at one hour, under ED discretion, when significant steam would be exiting the open hatch. - GE: Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via ED discretion based on extended boiling with (i) no compensating injection and (ii) containment open. At around five hours, indications of core damage are starting, which will also lead to increasing radiation doses near the open hatch. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
¹ Timings are relative to the time of occurrence of the initiating event. Abbreviations: Rep. Seq. = representative sequence; Decl. = declaration; SAE = site area emergency; GE = general emergency; SD = shutdown; ED = Emergency Director; RPV = reactor pressure vessel; RVLIS = reactor vessel level indication system; RCS = reactor coolant system; RWST = refueling water storage tank; SG = steam generator; CETC = core-exit thermocouple; PORV = power-operated relief valve				

Table 2-24 Comparison of General Emergency Timing to Release Timing

	Time (hr)					
	Reactor Shutdown Relative to Initiator ¹	Time of Initiator	GE Declaration	Onset of Core Damage	Xe Release > 10% ²	I Release > 1% ²
SD01	-82	0	11.5	15	>168	>168
SD01.1	-82	0	11.5	15	>168	>168
SD01.2	-82	0	11.5	15	50	75
SD01.3	-82	0	11.5	15	40	80
SD01.4	-82	0	11.5	15	69	95
SD01.5	-82	0	11.5	15	>168	>168
SD01.6	-82	0	11.5	15	60	115
SD01.7	-82	0	11.5	15	70	155
SD01.8	-82	0	11.5	15	67	97
SD01.9	-82	0	11.5	15	72	110
SD01.10	-82	0	11.5	15	18	>168
SD01.11	-82	0	11.5	15	18	150
SD01.12	-82	0	11.5	15	30	55
SD01.13	-82	0	11.5	15	20	100
SD02	-501	0	>168	>168	>168	>168
SD03	-141	0	~33	125	132	128
SD03.1	-141	0	~33	125	133	128
SD04	-329	0	>168	>168	>168	>168
SD05	-141	0	~48	100	102	100
SD06	-526	0	>168	>168	>168	>168
SD06.1	-526	0	~38	~44	46	45
SD07	-12	0	~15	16	17	17
SD07.1	-12	0	~15	16	17	17
SD07.2	-12	0	~15	16	17	17
SD08	-5	0	~28	27	>168	>168
SD08.1	-5	0	~28	27	>33	>33
SD08.2	-5	0	~28	27	>168	>168
SD09	-141	0	~24	126	128	127
SD10	-141	0	8.5	10	>168	>168
SD10.1	-141	0	8.5	10	>168	>168
SD11	-12	0	18	21	>168	>168
SD11.1	-12	0	18	22	25	25
SD12	-507	0	30	24	>168	>168
SD12.1	-507	0	30	24	>168	>168
SD13	-55	0	~5	~5	7	7

¹ This refers to how long the reactor has been shut down with respect to the time the initiator occurs.

² The timing of xenon (Xe) and iodine (I) releases are estimated from the results of the MELCOR cases in Figure 2-9 through Figure 2-24. More precise values can be obtained from the MELCOR output files as needed. The greater than symbol (e.g., >168) indicates the release fraction threshold was not reached by the end of the simulated calculation time (168 hours for all calculations except case SD08.1, which ended at 33 hours).

Table 2-25 Mapping Source Terms to Release Categories

Release Category	Candidate [Cs/ Release Fractions] ^{1,2}	Chosen Source Term	Notes
1-REL-SD-ISLOCA	SD11.1 [3%, 5%]	SD11.1	
1-REL-SD-ISGTR	-	-	There are no source terms available for this release category, but there is also negligible frequency associated with it in the current model.
1-REL-SD-CIF	SD01.10 [<1%, <1%] SD01.11 [9%, 33%] SD01.12 [2%, 8%] SD01.13 [1%, 1%] SD03 [7%, 6%] SD03.1 [5%, 5%]	SD03	Dominant cutsets are more indicative of the SD03 accident sequence. The frequency is dominated by small area isolation failures (i.e., 2-inch diameter or less). The frequency is dominated by un-scrubbed sequences.
1-REL-SD-COPEN-RCSINTACT	SD07 [25%, 38%] SD07.1 [14%, 20%] SD07.2 [20%, 30%]	SD07	The frequency is dominated by un-scrubbed sequences.
1-REL-SD-COPEN-RCSVENT	SD06.1 [46%, 50%] SD13 [79%, 82%]	SD13	This RC frequency is dominated by RCS vented configuration but also includes other losses of RCS integrity (e.g., loss-of-inventory, stuck-open relief valve) other than removal of the vessel head.
1-REL-SD-COPEN-HEADOFF	SD05 [87%, 88%]	SD05	
1-REL-SD-ECF	SD01.2 [1%, 3%]	SD01.2	
1-REL-SD-CF-Before-CD	SD09 [15%, 14%]	SD09	
1-REL-SD-ICF-BURN	SD01.3 [1%, 4%] SD01.4 [3%, 17%] SD01.6 [<1%, 1%] SD01.7 [1%, 1%] SD01.8 [1%, 3%] SD01.9 [1%, 2%]	SD01.8	The frequency is dominated by scrubbed sequences.
1-REL-SD-LCF	SD01 [<1%, <1%] SD01.5 [<1%, <1%] SD11 [<1%, <1%]	SD01	Dominant cutsets are more indicative of the SD01 accident sequence.
1-REL-SD-BMT	SD08 [<1%, <1%] SD08.2 [<1%, <1%]	SD08.2	SD08.2 actually represents the expected plant response more faithfully.
1-REL-SD-NOCF	SD01.1 [<1%, <1%] ³ SD10 [<1%, <1%] SD10.1 [<1%, <1%] ³ SD12 [<1%, <1%] SD12.1 [<1%, <1%]	SD12	SD12 is most representative of the top cutsets in the model.
¹ Calculations SD02, SD04, and SD06 are not included because no core damage occurs within 7 days. ² Calculation SD08.1 is not included because it was stopped at the time of vessel breach. ³ These cases deliberately suppressed combustion to provide boundary conditions to other tools, so they should be used with caution.			

Selection of the Containment Isolation Failure (CIF) Representative Source Term

Six MELCOR calculations were performed for scenarios with closed containment hatch and personnel airlock but with the containment not successfully isolated, and which would therefore be characterized as a 1-REL-SD-CIF (unisolated containment) release. The boundary conditions for these six scenarios may be summarized as follows:

- SD01.10: LORHR in POS 6 (mid-loop, pre-refueling) with failure to isolate a 14-inch purge line, and with credit for filters in the release pathway.
- SD01.11: LORHR in POS 6 (mid-loop, pre-refueling) with failure to isolate a 14-inch purge line, and with no credit for filters.
- SD01.12: LORHR in POS 6 (mid-loop, pre-refueling) with failure to isolate a 2-inch containment penetration at a low elevation.
- SD01.13: LORHR in POS 6 (mid-loop, pre-refueling) with failure to isolate a 2-inch containment penetration at a high elevation.
- SD03: SBO in POS 8E (RPV head removed, pre-refueling) with gravity injection to the RPV and failure to isolate a 2-inch containment penetration at a low elevation.
- SD03.1: SBO in POS 8E (RPV head removed, pre-refueling) with gravity injection to the RPV, failure to isolate a 2-inch containment penetration at a low elevation, and EDMG-based containment spray beginning 2 hours after vessel breach.¹⁰

The following observations can be made regarding the sequences leading to release category 1-REL-SD-CIF in the final Level 2 PRA quantification results:

- Accidents beginning in POS 8E contribute 61 percent to containment isolation failure frequency, whereas accidents beginning in POS 6 contribute only about 5 percent.
- Small unisolated containment penetrations make up 96 percent of this release category, whereas failure of a large mini-purge line makes up only about 0.5 percent.
- Containment sprays do not operate for any appreciable fraction of the containment isolation failure release category frequency.

In consideration of the above factors, SD03 is the most representative scenario for this release category. As Figure 2-9 and Figure 2-10 indicate, SD03 is also bounding in terms of the cumulative magnitude of volatiles released to the environment of all the small containment isolation failure scenarios—of the large containment isolation failure scenarios (which, as noted above, turn out to be very unrepresentative of the Level 2 PRA frequency results), one is slightly higher in terms of cesium release magnitude, and they are either slightly or significantly higher in terms of iodine release magnitude. The significantly larger iodine release in SD01.11 is

¹⁰ Vessel breach refers to the through-wall failure of the lower vessel after core damage and relocation of damaged fuel. The initial conditions in scenario SD03.1 in POS 8E have the RPV integrity defeated by removal of the reactor vessel head for refueling operations. However, removal of the vessel head is not considered a vessel breach as the term is used in the accident progression modeling.

mainly the result of a large very late (about 6 days) revaporization contribution from RCS surfaces, which is not predicted in SD01.10.

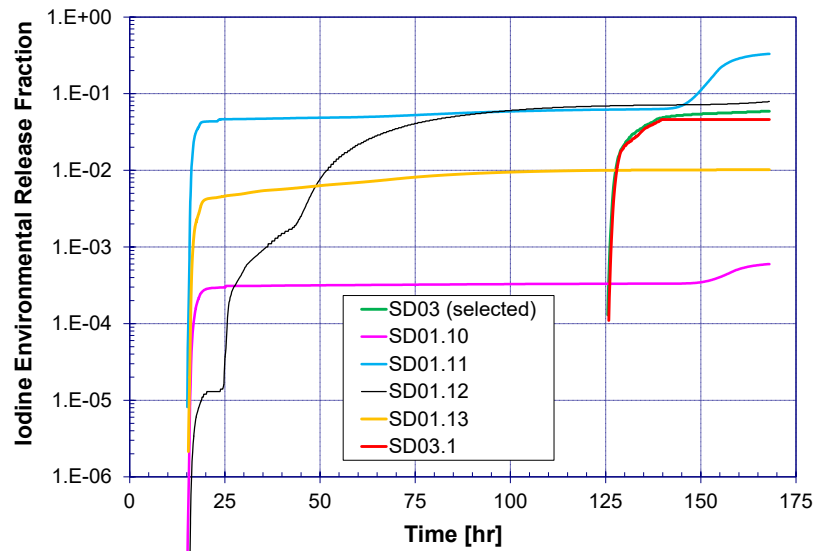


Figure 2-9 Iodine Releases for RC 1-REL-SD-CIF

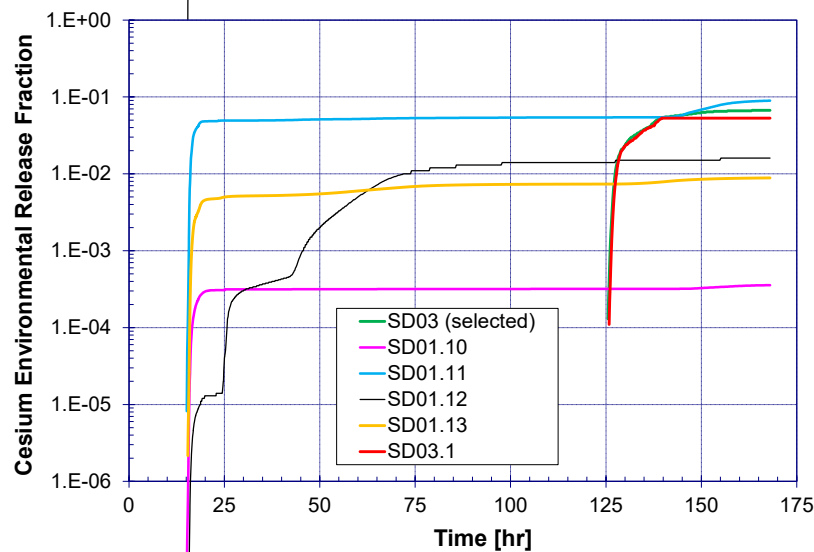


Figure 2-10 Cesium Releases for RC 1-REL-SD-CIF

Selection of the Open Containment / RCS Intact (COPEN-RCSINTACT) Representative Source Term

Among the MELCOR calculations performed for shutdown, three are candidates for the source term for open containment with an intact RCS, all of which are variants of SD07 (SBO in POS 5A). SD07.1 is a variant otherwise identical in boundary conditions, except that the containment opening is assumed to be via the personnel airlock instead of the equipment hatch. In SD07.2, EDMG-based spray operation is assumed for a period beginning two hours after vessel breach. Note that slices on the cutsets in 1-REL-SD-COPEN-RCSINTACT show that, by frequency, approximately 99.9 percent of releases in this category can be characterized as due to SBO, and 62 percent occur in POS 5A, so the boundary conditions of this calculation are representative in those respects. (Those not in POS 5A are mainly in POSs 11 and 12, so the selected source term is for the most part bounding in terms of decay time before release; only 8 percent of the frequency of this release category is represented by accidents earlier than in POS 5A.)

The iodine and cesium release fractions for the candidate source terms are shown in Figure 2-11 and Figure 2-12. The rationale for choosing SD07 as a more representative source term over SD07.1 is simply to choose the bounding release magnitude (larger opening size) in view of uncertainty as to which containment opening would fail to be closed. SD07.2 is eliminated as the source term due to the fact that there is no effective contribution from sequences with containment sprays in the results from this release category.

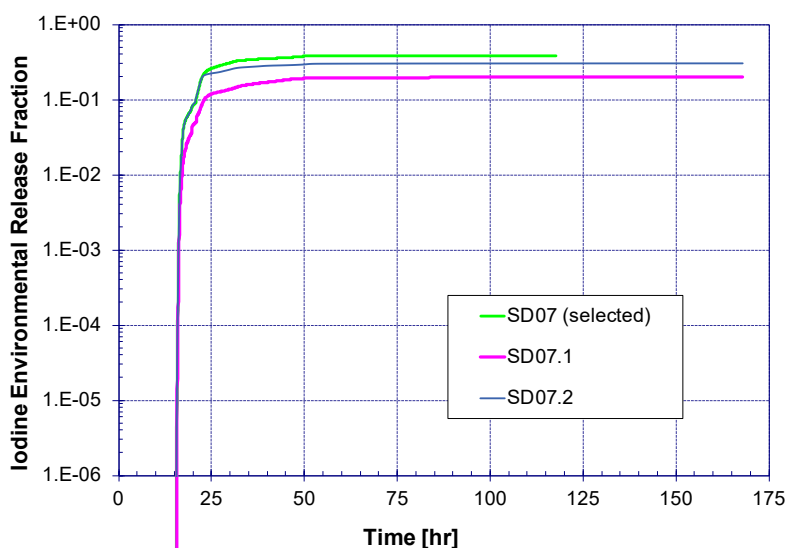


Figure 2-11 Iodine Releases for RC 1-REL-SD-COPEN-RCSINTACT

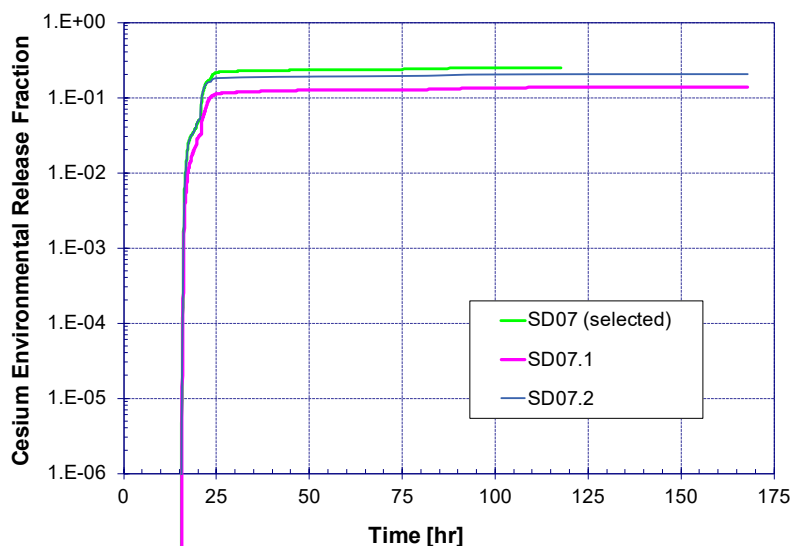


Figure 2-12 Cesium Releases for RC 1-REL-SD-COPEN-RCSINTACT

Selection of the Containment Open / RCS Vented (COPEN-RCSVENT) Representative Source Term

Two MELCOR calculations were performed for accidents in which the RCS is vented at the start of the accident and where the primary pathway for fission product release is via an open containment hatch or personnel airlock (release category 1-REL-SD-COPEN-RCSVENT):

- SD06.1: Transient in POS 11 (vacuum fill RCS) with ECCS available, RHR relief valve assumed to stick open during cycling, EDMG-based containment spray operation, and containment hatch open.
- SD13: SBO in POS 5B (RCS drained down) with containment hatch open.

The iodine and cesium release fractions for the candidate source terms are shown in Figure 2-13 and Figure 2-14. Of these two candidate source terms, SD13 was selected as the most representative. The case SD13 is more indicative of the dominant sequence results for the release category frequency (56 percent of the release category frequency is associated with POS 5B). Also, SD13 represents an accident scenario occurring prior to refueling, while SD06.1 occurs after refueling. The released activities and consequences are significantly influenced by the selection of a pre-refueling versus a post-refueling representative source term. Approximately 63 percent of the release category frequency is associated with pre-refueling accident scenarios.

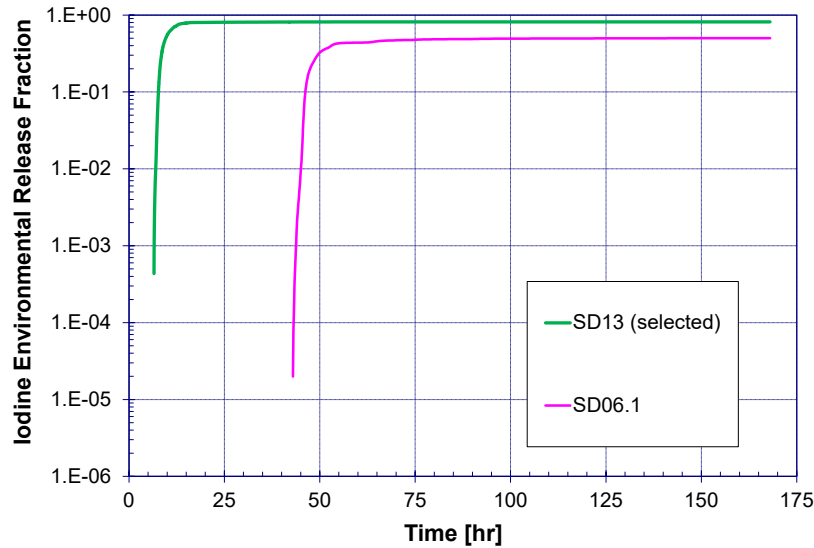


Figure 2-13 Iodine Releases for RC 1-REL-SD-COPEN-RCSVENT

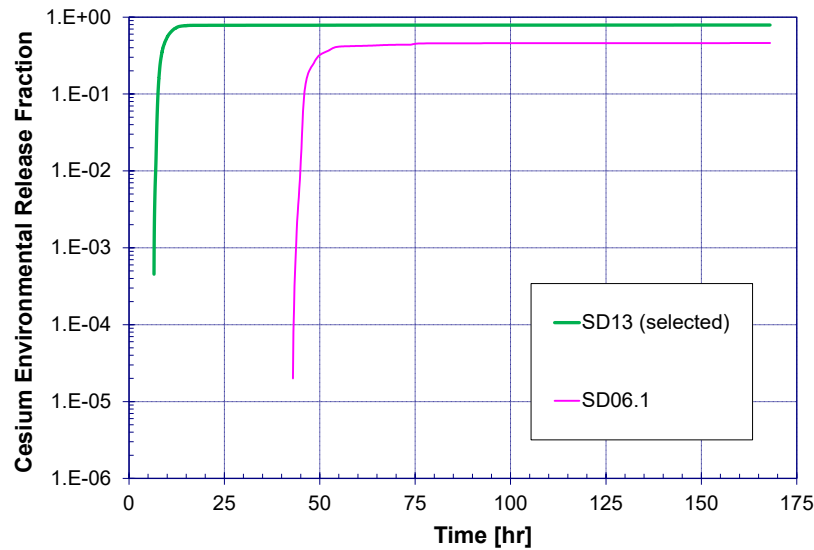


Figure 2-14 Cesium Releases for RC 1-REL-SD-COPEN-RCSVENT

Selection of the Intermediate Containment Failure (ICF-BURN) Representative Source Term

Six MELCOR calculations could potentially describe the typical release for a scenario with intermediate-term containment rupture due to combustion: SD01.3, SD01.4, SD01.6, SD01.7, SD01.8, and SD01.9. All of these are transient scenarios that are assumed to occur in POS 6 (over-drain during mid-loop operation), which the Level 2 PRA model results indicate is the highest contributing POS with 48 percent of the release category's total frequency. The other features which distinguish these calculations are as follows:

- SD01.3: Containment sprays unavailable, containment failure 4 hours after vessel breach, containment failure location near basemat-wall junction.
- SD01.4: Containment spray with 600,000 gallon inventory, containment failure 4 hours after vessel breach, containment failure location near basemat-wall junction.
- SD01.6: Containment spray with 900,000 gallon inventory, containment failure 35 hours after vessel breach, containment failure location near equipment hatch.
- SD01.7: Containment spray with 900,000 gallon inventory, containment failure 35 hours after vessel breach, containment failure location near basemat-wall junction.
- SD01.8: Containment spray with 600,000 gallon inventory, containment failure 35 hours after vessel breach, containment failure location near basemat-wall junction.
- SD01.9: Containment spray with 850,000 gallon inventory, containment failure 35 hours after vessel breach, containment failure location near basemat-wall junction.

The iodine and cesium release fractions for the candidate source terms are shown in Figure 2-15 and Figure 2-16.

The Level 2 PRA frequency results show that more than 99 percent of the frequency of release category 1-REL-SD-ICF-BURN involves sequences with late containment cooling (primarily in the form of EDMG-based spray operation) and resulting in late hydrogen detonation failing the containment. EDMG-based spray operation in all cases (except SD01.3) is assumed to begin two hours after vessel breach and to continue until all water available for spray injection has been exhausted. The amount of water assumed to be available is varied based on possible sources that could be available for spray. The 600,000 gallon case was considered to be the most likely capability based on the reference plant's EDMG guidance.

A probabilistic combustion analysis code called ERPRA-BURN was used to arrive at the relevant combustion-related branch probabilities in the Level 2 PRA logic model. The ERPRA-BURN results suggest that the mean time of containment-failing combustion for scenarios with late containment heat removal is approximately 35 hours after vessel breach, hence the particular timing assumed for SD01.6 through SD01.9. The ERPRA-BURN analysis and associated uncertainties are discussed further as part of the combustible gas model uncertainties in Section 2.6.3.7.

The containment failure location was varied to assess the impact of an alternate release pathway on the release magnitudes. Both containment failure locations are considered to be equally likely failure locations. However, the failure location did not influence the selection of the representative source term. The source term for case SD01.8 was selected because the scenario was deemed the most realistic representation of the dominant accident sequences. The other cases are used to evaluate the impacts of modeling assumptions and uncertainties. The model uncertainties associated with the combustion modeling are discussed in Section 2.6.3.7, and the model uncertainties associated with the release pathway modeling are discussed in Section 2.6.3.13.

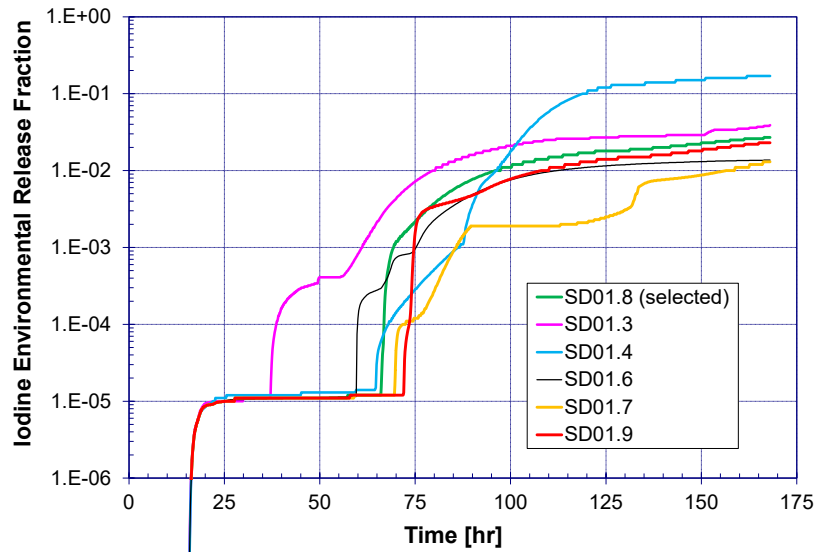


Figure 2-15 Iodine Releases for RC 1-REL-SD- ICF-BURN

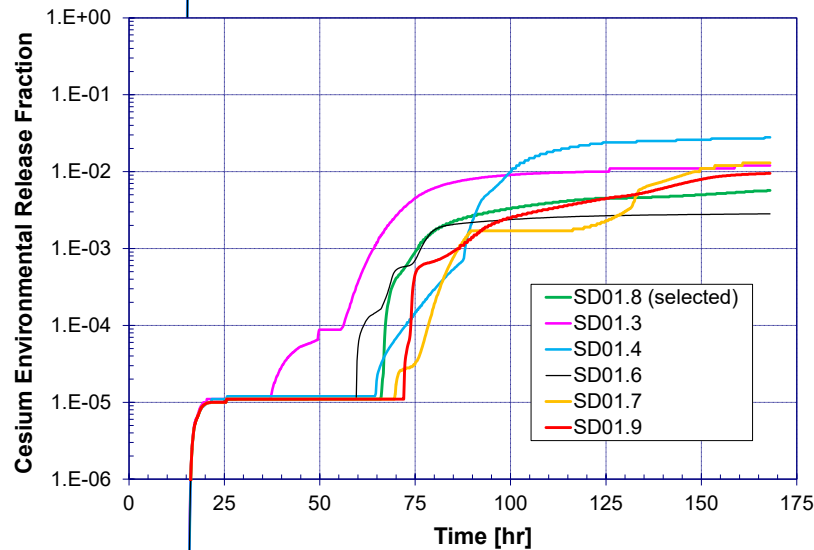


Figure 2-16 Cesium Releases for RC 1-REL-SD- ICF-BURN

Selection of the Late Containment Failure (LCF) Representative Source Term

SD01 (mid-loop overdrain in POS 6, with late containment failure at low elevation), SD01.5 (mid-loop overdrain in POS 6, with late containment failure at high elevation), and SD11 (loss-of-inventory in POS 5A with forced feed) are all reasonable candidates for this source term. The releases for SD11 are somewhat higher in magnitude. However, the two variants of SD01 are better characterizations of about 25 percent of the cutsets in the release category by frequency, whereas SD11 only fairly describes about 4 percent. From the source term magnitudes evident in Figure 2-17 and Figure 2-18, there appears to be little to differentiate between SD01 or

SD01.5, since the lines on the graph nearly overlap. Therefore, the base case calculation SD01, rather than the sensitivity case SD01.5, was chosen as the representative source term. Note that the SD01.5 calculation was extended to 182 hours to evaluate uncertainties on accident termination timing (see Section 2.6.3.15 for additional discussion), but the representative source terms are compared using the nominal 168-hour calculation time for both cases.

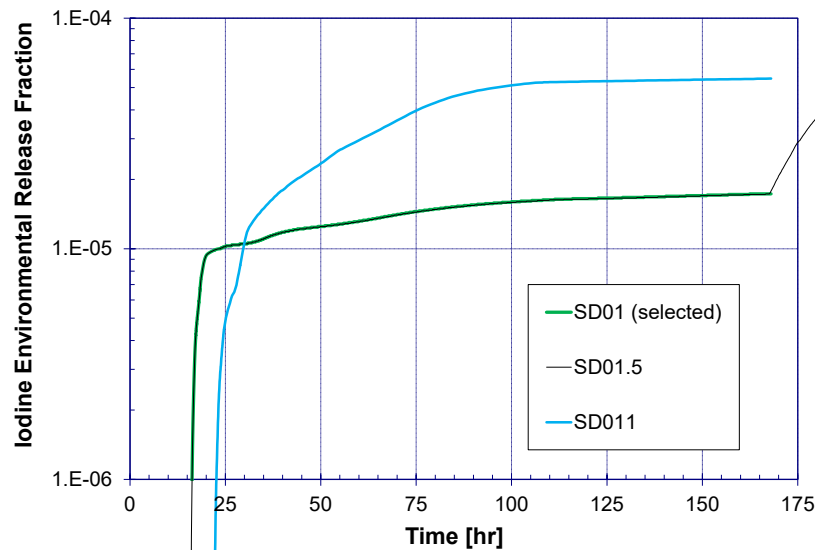


Figure 2-17 Iodine Releases for RC 1-REL-SD-LCF

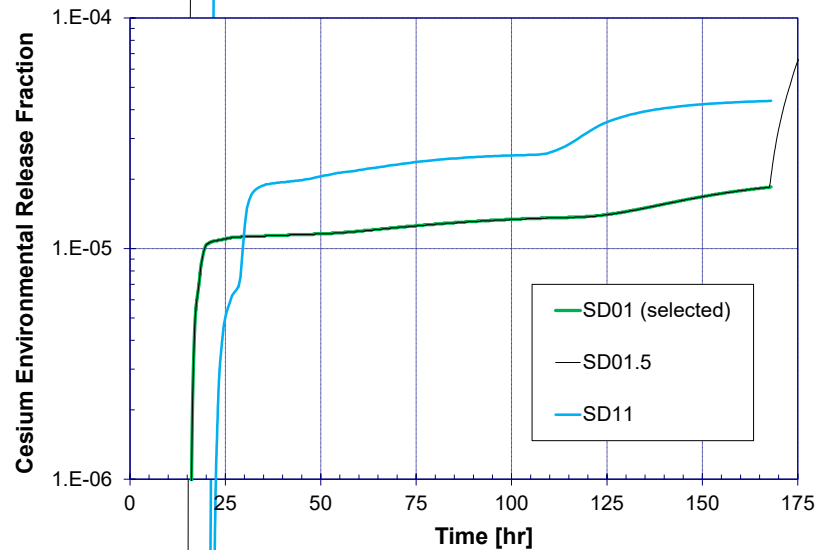


Figure 2-18 Cesium Releases for RC 1-REL-SD-LCF

Selection of the BMT Representative Source Term

SD08 and SD08.2 are both candidate source terms for BMT, and both are variants of a transient in POS 3 with containment spray operation, with the only difference in boundary conditions being that SD08.2 assumes two RCPs continue operating until about 24 hours (at which time they are assumed to trip due to cavitation). The predicted source terms from these two calculations are nearly identical, with SD08 being very slightly higher. Note that there is no reason to believe that the operation of the RCPs is of direct consequence to the progression of core-concrete interactions and BMT, and therefore the small difference in predicted source terms is likely incidental. Nevertheless, SD08.2 was chosen to represent the BMT source term due to the fact that the modeled RCP behavior is believed to more faithfully represent the actual plant response.

The iodine and cesium release fractions for the candidate source terms are shown in Figure 2-19 and Figure 2-20. Recall that this source term only includes airborne releases and does not specifically treat containment failure at the time of BMT.

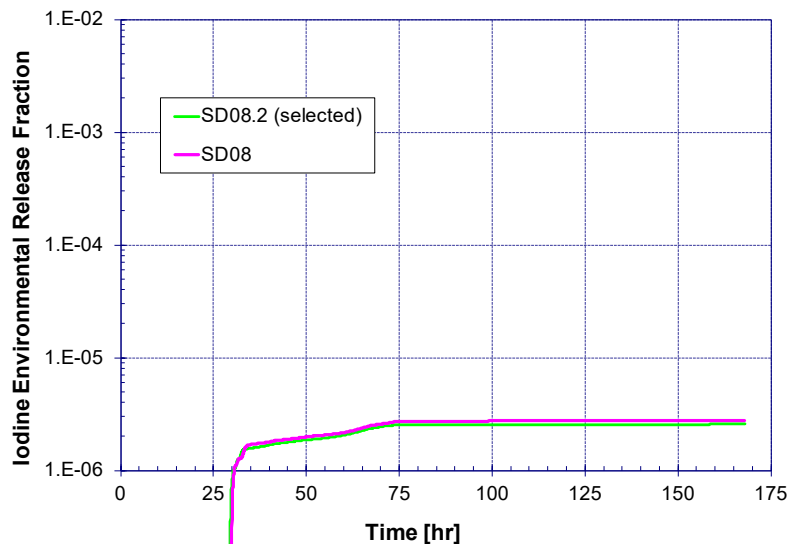


Figure 2-19 Iodine Releases for RC 1-REL-SD-BMT

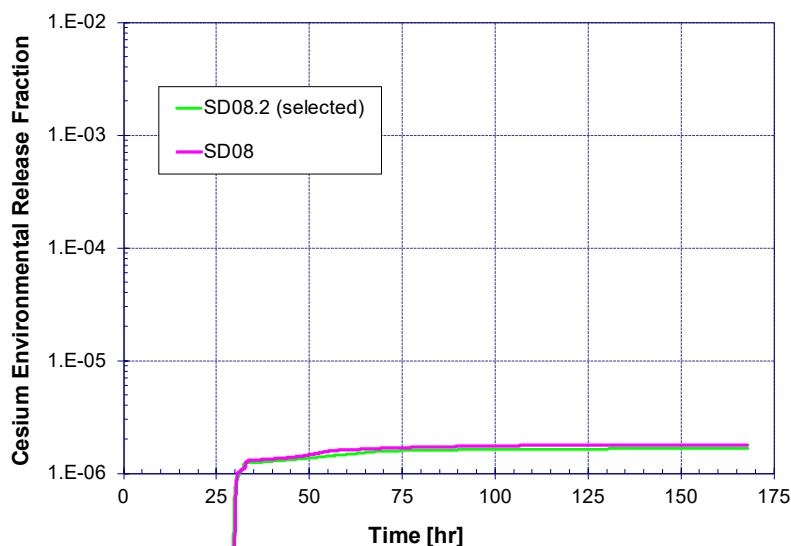


Figure 2-20 Cesium Releases for RC 1-REL-SD-BMT

Selection of the No Containment Failure (NOCF) Representative Source Term

Five of the scenarios analyzed by MELCOR for reactor at-shutdown could potentially describe the 1-REL-SD-NOCF source term: SD01.1, SD10, SD10.1, SD12.1, and SD12. Of these, SD01.1 and SD10.1 were both performed for purposes of obtaining input to the separate hydrogen combustion analysis with combustion being intentionally suppressed in the model. Therefore, they would not ideally serve as realistic source terms. SD10 is a loss-of-inventory accident occurring in POS 8E, whereas SD12 is a loss-of-inventory accident in POS 10. Release category frequency results reveal that accidents in POS 10 make up about 10 percent of this release category frequency, whereas those in POS 8E contribute much less than 1 percent. Also, most of the release category frequency (about 64 percent) is associated with POSs occurring after refueling, which includes POS 10. Therefore, SD12 is a better representative. Note that SD12.1 has a somewhat higher magnitude of release, and it assumes failure of fan coolers in the intermediate time frame due to combustion inside containment. The likelihood of such equipment failure is not considered in the logic model, and therefore SD12.1 is not selected as a source term for intact containment end states. The iodine and cesium release fractions for the candidate source terms are shown in Figure 2-21 and Figure 2-22.

Representative Source Terms for the Remaining Release Categories

Figure 2-23 and Figure 2-24 show the airborne environmental iodine and cesium releases for those release categories that only had one candidate source term; namely, 1-REL-SD-ISLOCA (SD11.1), 1-REL-SD-COPEN-HEADOFF (SD05), 1-REL-SD-ECF (SD01.2), and 1-REL-SD-CF-Before-CD (SD09).

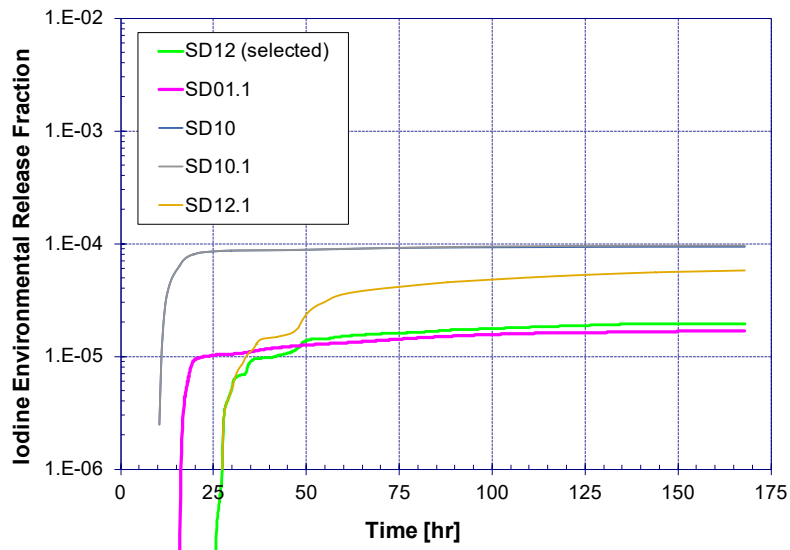


Figure 2-21 Iodine Releases for RC 1-REL-SD-NOCF

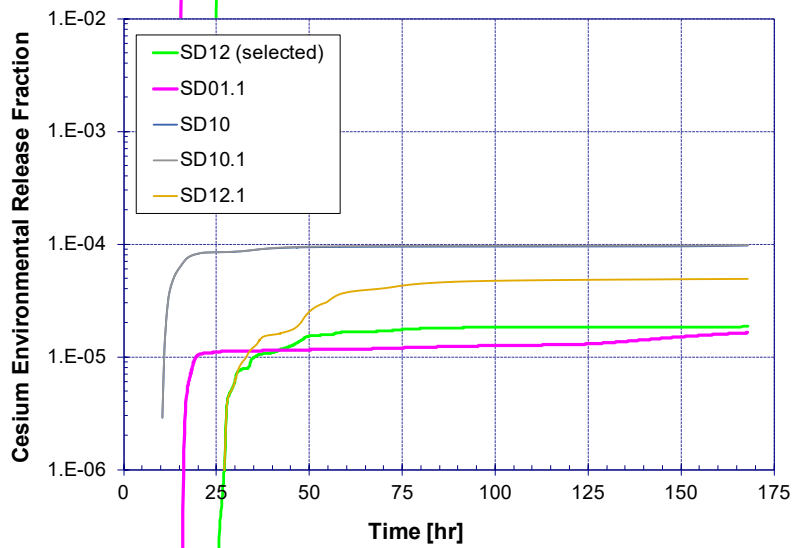


Figure 2-22 Cesium Releases for RC 1-REL-SD-NOCF

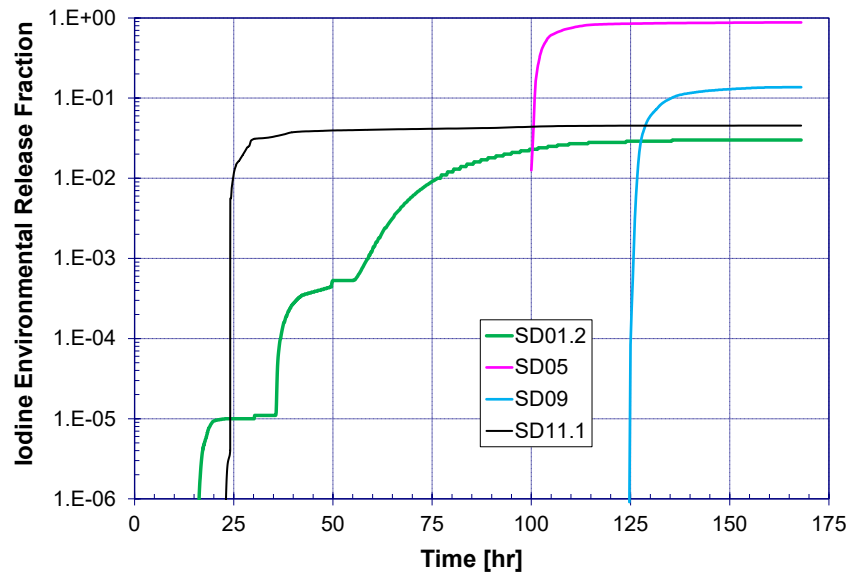


Figure 2-23 Iodine Releases for Other Cases

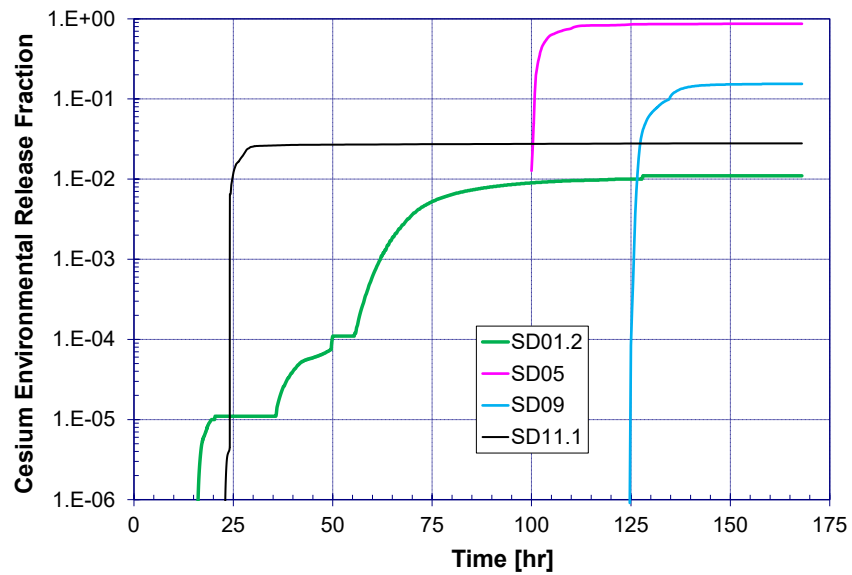


Figure 2-24 Cesium Releases for Other Cases

2.5.3 Consideration of Uncertainties in the Source Term Development

A comprehensive review of modeling uncertainties that can affect the source term development and other aspects of the Level 2 PRA modeling is presented in Section 2.6.3. Refer to that section for additional discussion of modeling uncertainties in the source term development. Several sensitivity studies were performed to assess the impacts of modeling uncertainties on the source term release magnitudes. In particular, the uncertainties related to radiological release pathway modeling can have potentially large impacts on release magnitudes. This is demonstrated by the sensitivity analyses on containment isolation failure release pathway documented in Section 2.6.3.12 and containment failure pathway for long-term overpressurization and combustion-induced failures documented in Section 2.6.3.13.

2.6 Evaluation and Presentation of Results

The results of the reactor at-shutdown Level 2 PRA are presented here. The presentation of results consolidates the deterministic severe accident and source term analyses and the probabilistic frequency results. The severe accident modeling and representative source terms using MELCOR are described in Sections 2.3.3 and 2.5.2. The release category frequency results from the probabilistic model are presented in Section 2.4.6. This section provides a summary of release category and surrogate risk metric results (Section 2.6.1), comparison of the results to previous studies (Section 2.6.2), and a summary of model uncertainties and sensitivity analyses (Section 2.6.3).

2.6.1 Summary of Release Category and Surrogate Risk Metric Results

Three surrogate risk metrics are defined for the reactor at-shutdown Level 2 PRA. The surrogate risk metrics are consistent with the metrics reported in the reactor at-power Level 2 PRA (NRC, 2022a). The following surrogate risk metric definitions are used to report the results:

- **LERF:** Large early release frequency includes release categories if their representative source term has a warning time (based on iodine release exceeding 1 percent) less than 20 hours and the cumulative iodine release fraction is greater than 4 percent.¹¹
- **LRF:** Large release frequency includes 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, or smaller than, the remainder of the intact containment source terms.
- **CCFP:** conditional containment failure probability is defined as the ratio of the release category frequencies involving a failed or bypassed containment to the total release frequency.

The surrogate risk metric results are shown in Table 2-26. The LERF contribution is approximately four percent of the total frequency. Three release categories are identified as contributors to LERF. Two of the LERF release categories (1-REL-SD-COPEN-RCSINTACT and 1-REL-SD-COPEN-RCSVENT) involve large containment openings (i.e., equipment hatch

¹¹ The reactor at-power Level 2 PRA (NRC, 2022a) includes the same definition of LERF based on potential for early injuries and a second definition of LERF based on potential for early fatalities with a warning time criterion of less than 3.5 hours. The more inclusive definition based on potential for early injuries is used in the reactor-at-shutdown Level 2 PRA.

open) as initial conditions. The third LERF contributor is the loss of inventory outside of containment scenario, that is, shutdown ISLOCA (1-REL-SD-ISLOCA). The LRF categories contribute about 49 percent of the overall frequency. Note that the basemat melt-through (1-REL-SD-BMT) and late containment failure (1-REL-SD-LCF) release categories involve containment failures but are excluded from LRF due to the magnitude of the airborne environmental releases being comparable to the intact containment release category (1-REL-SD-NOCF). The CCFP is just over 50 percent of the total frequency with the intact containment category accounting for just over 49 percent of the frequency.

Table 2-26 Surrogate Risk Metric Results

Release Category	Frequency (per year)	% Contribution	LERF (per year)	% Contribution	LRF (per year)	% Contribution	CCFP
1-REL-SD-BMT	1.2E-07	1.0%	n/a	n/a	n/a	n/a	1.0%
1-REL-SD-CF-BEFORE-CD	2.1E-06	16.7%	n/a	n/a	2.1E-06	16.7%	16.7%
1-REL-SD-CIF	1.7E-06	13.8%	n/a	n/a	1.7E-06	13.8%	13.8%
1-REL-SD-COPEN-HEADOFF	7.8E-08	0.6%	n/a	n/a	7.8E-08	0.6%	0.6%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.2%	2.7E-07	2.2%	2.7E-07	2.2%	2.2%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.1%	1.3E-07	1.1%	1.3E-07	1.1%	1.1%
1-REL-SD-ECF	5.1E-10	<0.1%	n/a	n/a	5.1E-10	<0.1%	<0.1%
1-REL-SD-ICF-BURN	1.7E-06	13.8%	n/a	n/a	1.7E-06	13.8%	13.8%
1-REL-SD-ISLOCA	1.3E-07	1.1%	1.3E-07	1.1%	1.3E-07	1.1%	1.1%
1-REL-SD-LCF	5.5E-08	0.4%	n/a	n/a	n/a	n/a	0.4%
1-REL-SD-NOCF	6.1E-06	49.4%	n/a	n/a	n/a	n/a	n/a
Total	1.2E-05	100.0%	5.3E-07	4.3%	6.1E-06	49.2%	50.6%

The surrogate risk metrics are defined in terms of the timing and magnitude of releases. The definitions used in the shutdown PRA are consistent with the at-power PRA. However, the shutdown PRA uses a different initial inventory of fission products based on an end-of-cycle core with a coastdown to shutdown. Whereas the at-power PRA is based on a scenario using a trip from 100 percent power during middle-of-cycle. The shutdown PRA also models accident scenarios that occur during plant operating states at times long after shutdown, and radioactive decay over this time is not explicitly addressed in the surrogate metric definitions. Using the same LERF criteria from the at-power PRA for the shutdown PRA can lead to a potentially conservative assessment of which categories should be included as LERF. However, the impacts of how the definition is applied to the results of this study do not alter the insights. Refer to Section 2.6.3.14 for further discussion on the uncertainties associated with translating the surrogate metric definitions for use in shutdown PRA. Use of a consistent LERF definition for the at-power and shutdown PRAs was determined to be preferable for clarity in interpreting the meaning of the LERF results.

A summary of the scenario timing and release magnitudes is presented in Table 2-27 for each of the release category representative accident scenarios. These results inform the designation of the surrogate release metric results. The representative accident scenarios are simulated with a termination time of 168 hours (7 days) after initiating event occurrence. Table 2-27 shows that some of the shutdown scenarios are slow to develop. In cases with the refueling cavity flooded (during POS 8E), core damage does not occur until after 100 hours into the scenario, and several other scenarios show containment failure occurring late in the accident progression.

Given the slow-developing attributes of some shutdown scenarios, it was deemed worthwhile to consider the potential impacts of recovery and/or mitigating actions that could terminate the accidents at an earlier time and either prevent or limit releases. Actions directed by SAMGs could potentially improve the scenario outcomes. However, SAMG-guided actions are not credited because the reference plant's SAMG guidance explicitly excluded actions for events initiating with the reactor shutdown.¹² Still, the potential for earlier termination of the accidents is considered. A similar analysis was performed for the at-power Level 2 PRA (see Section 2.6.1 of (NRC, 2022a)). For the at-power analysis, two alternate accident termination times were considered, based on 36 hours and 60 hours after the time of transition to the SAMGs. As the SAMGs are not applicable to the shutdown analysis, different criteria are developed for accident termination times, that is, 36 hours and 60 hours after indication of core damage. The indication of core damage is defined as indication of core exit temperature exceeding 1200 °F.

Table 2-28 provides the results of the risk surrogates for the different accident termination times. Similar to the at-power PRA results, LERF is insensitive to truncating the accident analysis at earlier times. LRF and CCFP both show potential reductions if accident termination can be successful within 36 hours after indication of core damage. These results are discussed further as part of the modeling uncertainty analysis in Section 2.6.3.15 (see MU-15.1). A similar modeling uncertainty analysis related to potential credit for severe accident management actions is discussed in Section 2.6.3.3 (see MU-3.2). Also, the implementation of FLEX mitigating strategies is expected to have a positive impact on the release frequency results; however, the FLEX strategies are not included within the scope of this model.¹³

A set of significant release categories is defined to identify the contributions that are important to the surrogate risk metrics LERF and LRF. 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% 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.

¹² The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available and thus not considered at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

¹³ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the L3PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

This assessment yields the following set of significant release categories for the reactor at-shutdown Level 2 PRA:

- 1-REL-SD-CF-BEFORE-CD
- 1-REL-SD-CIF
- 1-REL-SD-COPEN-HEADOFF
- 1-REL-SD-COPEN-RCSINTACT
- 1-REL-SD-COPEN-RCSVENT
- 1-REL-SD-ICF-BURN
- 1-REL-SD-ISLOCA

The risk significant basic events that contribute to the significant release categories are shown in Table 2-29. The table includes all the basic events with a Fussell-Vesely importance greater than 0.1 when combining the cutsets for all the significant release categories. (Additional importance measure results are included in Appendix B.) The following types of events are represented:

- Events representing the fraction of average annual exposure to plant outage and plant operating states. These events indicate that accident scenarios during POS 5A, POS 6, and POS 8E are significant contributors to the significant release categories.
- Events representing shutdown initiating event frequencies, including loss of offsite power and overdraining when entering mid-loop.
- Events related to restoration of AC power during loss of offsite power and station blackout sequences. These include events representing unavailability of an alternate AC power source, failures to recover onsite emergency diesel generators (EDGs), and failure to restore offsite power.
- Failure of operator actions credited during the pre-core damage sequence progression. These include operator failure to established gravity-driven injection from the RWST, failure to isolate containment penetrations, and failure to restore RHR cooling after successful injection.
- Numerous events related to modeling the overall likelihood of energetic combustion events that can result in containment failure. These include the probability of a detonation occurring late in the accident progression, the likelihood of the presence of an ignition source during different phases of the accident progression, and the probability of combustion occurring during different phases of the accident progression.
- A post-core-damage operator action event representing failure to recover injection to the reactor vessel.
- Events representing equipment failures from the Level 1 PRA sequences, including EDG failure to run and reserve auxiliary transformer (RAT) input breaker failure to open.

Table 2-27 Summary of Accident Scenario Timing for Determining Surrogate Risk Metrics

Release Category	MELCOR Rep. Case	GE ¹ (hr)	Time of CD Indication ² (hr)	Warning Time ³ (hr)	Time to LRF Thresh. ⁴	Time of Cont. Failure (hr)	36 hr after CD Indication (hr)	60 hr after CD Indication (hr)
1-REL-SD-BMT	SD08.2 (POS 3)	~28	27	>140	Never	165	63	87
1-REL-SD-CF-BEFORE-CD	SD09 (POS 8E)	~24	126	103	126	118	162	>168
1-REL-SD-CIF	SD03 (POS 8E)	~33	125	95	~127	0 (cont. unisol.)	161	>168
1-REL-SD-COPEN-HEADOFF	SD05 (POS 8E)	~48	100	52	100	0 (cont. hatch open)	136	160
1-REL-SD-COPEN-RCSINTACT	SD07 (POS 5A)	~15	16	2	16	0 (cont. hatch open)	52	76
1-REL-SD-COPEN-RCSVENT	SD13 (POS 5B)	~5	~5	2	7	0 (cont. hatch open)	41	65
1-REL-SD-ECF	SD01.2 (POS 6)	11.5	15	~64	~65	24	51	75
1-REL-SD-ICF-BURN	SD01.8 (POS 6)	11.5	15	~85	69	60	51	75
1-REL-SD-ISLOCA	SD11.1 (POS 5A)	18	22	7	~25	0 (cont. bypass)	58	82
1-REL-SD-LCF	SD01 (POS 6)	11.5	15	>156	Never	168	51	75
1-REL-SD-NOCF	SD12 (POS 10)	30	24	>138	Never	Never	60	84

¹ The timing of General Emergency (GE) declaration depends on the scenario-specific conditions and is indicated in Table 2-23.

² Core damage is indicated when core exit temperature exceeds 1200 °F. Thirty minutes are added to the time indicated in the MELCOR simulation to account for time to assess plant conditions.

³ Warning time is determined by the time at which cumulative environmental iodine release fraction exceeds 1 percent minus the time that GE is declared.

⁴ The time when cumulative environmental release fractions of either iodine or cesium exceed 0.001 is used to indicate a "large" release, which is considered to be larger than (not comparable to) releases from an intact containment (i.e., releases associated with release category 1-REL-SD-NOCF).

Table 2-28 Surrogate Risk Metric Results with Truncation of Accident Termination

Release Category	Total Release		36 hr after CD Indication			60 hr after CD Indication			7 Days after Event Occurs		
	Frequency (per year)	% of Total	LERF	LRF	CCFP	LERF	LRF	CCFP	LERF	LRF	CCFP
1-REL-SD-BMT	1.2E-07	1.0%									1.0%
1-REL-SD-CF-BEFORE-CD	2.1E-06	16.7%		2.1E-06	16.7%		2.1E-06	16.7%		2.1E-06	16.7%
1-REL-SD-CIF	1.7E-06	13.8%		1.7E-06	13.8%		1.7E-06	13.8%		1.7E-06	13.8%
1-REL-SD-COPEN-HEADOFF	7.8E-08	0.6%		7.8E-08	0.6%		7.8E-08	0.6%		7.8E-08	0.6%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.2%	2.7E-07	2.7E-07	2.2%	2.7E-07	2.7E-07	2.2%	2.7E-07	2.7E-07	2.2%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%
1-REL-SD-ECF	5.1E-10	0.0%			0.0%		5.1E-10	0.0%		5.1E-10	0.0%
1-REL-SD-ICF-BURN	1.7E-06	13.8%					1.7E-06	13.8%		1.7E-06	13.8%
1-REL-SD-ISLOCA	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%	1.3E-07	1.3E-07	1.1%
1-REL-SD-LCF	5.5E-08	0.4%									0.4%
1-REL-SD-NOCF	6.1E-06	49.4%									
Total	1.2E-05	100.0%	5.3E-07	4.4E-06	35.4%	5.3E-07	6.1E-06	49.2%	5.3E-07	6.1E-06	50.6%

Table 2-29 Basic Events with Highest Ranking Fussell-Vesely Measures for Significant Release Categories

Basic Event Name	Prob.	FV	Description
1-POT-4	5.53E-02	8.79E-01	PLANT OUTAGE TYPE - 4 (REFUELING)
1-IE-SD-LOOP	1.63E-01	6.57E-01	LOSS OF OFF-SITE POWER - SHUTDOWN
1-POS-8E-POT-4	9.37E-02	5.31E-01	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
1-EPS-XHE-XL-NR04H	6.98E-01	5.07E-01	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
1-OEP-XHE-XL-NR02H	3.26E-01	4.93E-01	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
OEP-VCF-LP-GRWR	4.10E-01	3.98E-01	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
1-LPI-XHE-XM-GFEED-LOOP	5.50E-01	2.79E-01	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
1-L2-BE-H2DET-L-CHR-SD	7.90E-01	2.68E-01	Late Detonation with containment heat removal (Shutdown)
1-CIS-XHE-XL-ISOLPEN-HD	5.02E-01	2.63E-01	Operators fail to isolate containment small containment penetrations (high dependency)
1-L2-OP-EDMGB-SD4	7.41E-01	2.49E-01	Operator fails to recover injection to vessel (LPSD case 4)
1-L2-BE-H2IGSRC-L-AC-SD	9.56E-01	2.37E-01	Ignition Source in Containment Late with AC Power
1-L2-BE-H2IGSRC-E-AC-SD	9.56E-01	2.27E-01	Ignition Source in Containment near time of vessel breach with AC Power
1-L2-BE-H2IGNSRC-VE-AC	9.56E-01	2.26E-01	Ignition Source in Containment before vessel breach, with AC Power
1-EPS-DGN-FR-G4002__	3.30E-02	2.12E-01	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-OEP-XHE-XX-NR02H1	2.73E-01	1.97E-01	Convolution correction factor for failure to recover offsite power in 2 hours
1-POS-5A-POT-4	4.13E-02	1.79E-01	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
1-ACP-CRB-CC-BA0301__	5.35E-03	1.21E-01	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-L2-BE-H2IGN-VE-CHR-SD	5.40E-01	1.20E-01	Burn in Containment before vessel breach (with containment heat removal)
1-IE-SD-ODML-EARLY	5.88E-01	1.20E-01	OVER DRAINING IN MID-LOOP - Before Refueling
1-POS-6-POT-4-MLDRL	4.07E-03	1.20E-01	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)- (demand-based)
1-RHR-XHE-XL-LTREC	1.55E-02	1.04E-01	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection

2.6.2 Comparison to Past Studies

Comparison of the results for the reactor-at-shutdown Level 2 PRA is limited by the scarcity of citable studies that have been performed for other pressurized-water reactors. One such possible basis for comparison is the analysis of LPSD mid-loop accidents at Surry Unit 1 performed by Brookhaven National Laboratories and NRC in 1995, NUREG/CR-6144 (NRC, 1995a). Differences in the assumptions and scope of that study lead to the following differences in results in comparison to the present PRA:

- NUREG/CR-6144 focuses specifically on analysis of reactor-at-mid-loop (including in non-refueling shutdowns), rather than the current study of all shutdown plant operating states for a refueling outage. That is, it quantifies risk only for POSs 6 and 10, and with a different spectrum of plant configurations than is examined here.
- NUREG/CR-6144 assumes that the equipment hatch is open at the start of all accidents based on discussions with operators at the time of that study. The present PRA finds sufficient documentation showing that the hatch would be closed at mid-loop. Although NUREG/CR-6144 assigns HEPs to containment closure for these accidents, the failure probabilities are large and assign unconditional failure in sequences with a prior diagnosis error. The result of these factors is that about 65 percent of core damage frequency in NUREG/CR-6144 occurs with an open containment equipment hatch, as compared with only 4 percent in the present study. This also reflects the changes in outage management that were implemented in response to generic communications and industry guidance on outage management, which were implemented around the mid-1990s (see further discussion on this in the shutdown Level 1 PRA comparison to other studies (NRC, 2025)).
- NUREG/CR-6144 assumes that, even if the containment were to be successfully closed, its effective pressure capacity would be severely degraded, such that it would start to leak at only 45 psig.
- Late AC power recovery is considered very likely in NUREG/CR-6144, whereas late AC recovery after battery depletion is not credited at all in the present study. Therefore, progression of SBOs is significantly different.
- As is typical for combustion analyses at the time, NUREG/CR-6144 assumes that hydrogen would ignite deterministically at a specific threshold concentration (i.e., 10 percent), precluding the possibility of any larger burns. In result, only about 2 percent of core damage frequency consists of containment failure due to hydrogen burning. The large fraction of the frequency with containment unisolated is also a contributing factor to this result. In contrast, for the present study, where combustion is treated stochastically, intermediate containment failure due to energetic combustion events accounts for about 14 percent of the total release frequency.
- The state-of-the-art of the 1990s considered DCH and steam explosions to result in higher conditional probabilities of containment failure than is now believed likely. Accordingly, NUREG/CR-6144 assigns a CCFP of about 1 percent for in-vessel steam explosion (IVSE), and what appears to be on the order of a few percent for DCH. Also, due to the much different spectrum of accidents within the scope of NUREG/CR-6144, there is a far higher relative frequency of accidents with high RCS pressure susceptible

to DCH. In result, NUREG/CR-6144 finds an early containment failure probability of 0.5 percent, which is almost 100 times higher than in the present PRA.

- Flooded cavity conditions are considered likely in the Surry study, in contrast with the dry cavity design for the current reference plant.

In summary, the results of NUREG/CR-6144 and the present shutdown PRA are significantly different. The two studies are not directly comparable due to numerous differences in scope, the state-of-the-art at the time the studies were performed, and plant differences. Nevertheless, the two studies were found to have considered largely the same set of phenomena (in terms of event progression top events and cases), containment failure modes, and equivalent plant damage state attributes, even if they were often quantified very differently.

The magnitudes of the source term predicted using MELCOR and SRSOR in NUREG/CR-6144 are broadly comparable to those arrived at in the present study. This is primarily due to both studies analyzing scenarios with large-sized containment openings during shutdown states that have potential for large releases. The following observations are noted in comparing releases for similar categories from the two studies:

- In NUREG/CR-6144, the release bins with the largest source terms (large-sized containment opening before vessel breach) included roughly 12 to 50 percent release to the environment of volatile radionuclides. This is comparable to (though somewhat smaller than) the containment open releases in the present study (about 25 to 80 percent of volatile radionuclides released to the environment in the case of 1-REL-SD-COPEN-RCSVENT and 1-REL-SD-COPEN-RCSINTACT; note that the Surry study did not include any contribution from sequences resembling 1-REL-SD-COPEN-HEADOFF).
- In NUREG/CR-6144, containment isolation failures appear to lead to about a 10 to 38 percent release of cesium and iodine, which is significantly larger than the 5 to 7 percent arrived at in the present study. However, the minimum size assumed for such failures was 0.1 ft² (about 4-inch equivalent diameter), which is larger than the 2-inch equivalent diameter assumed for the present study.
- In NUREG/CR-6144, releases of volatile radionuclides from intermediate or late containment rupture events appear to be in the range of about 3 to 9 percent, which is roughly consistent with the results of the current study, ranging from less than 1 percent to 15 percent.

Another possible basis for comparison is with NRC's Level 2 reactor at-power PRA (NRC, 2022a). Although the reactor at-shutdown model includes some shutdown-specific elements (e.g., equipment hatch status) and has an entirely different spectrum of accident initiators and plant states at the time of core damage, it is possible to compare the results on a broad scale and arrive at some conclusions. It appears that the two studies are reasonably consistent except for the following differences due to known reasons:

- Equipment hatch closure is an item unique to the shutdown PRA, since the hatch is closed during full-power operation.

- The probability of other forms of containment isolation failure is much higher for reactor at-shutdown, since there is a large swath of accidents in POSs that are initially unisolated and require operator action to isolate.
- A “containment rupture before core damage” release category is present for the reactor at-shutdown study which was not needed in the at-power PRA, since this was determined to occur only for a subset of accidents with the reactor vessel head removed and the refueling cavity flooded (i.e., a much larger amount of water available to boil than for at-power accidents). However, it does represent a type of late containment failure due to overpressurization during a long-term station blackout, which is also a significant contributor to release in the at-power study.
- Early containment failure (due to rupture) probabilities (i.e., from soon after core damage to at or around the time of vessel breach) are somewhat lower for the reactor at-shutdown study, since that study has a much smaller fraction of accidents with high RCS pressure, and therefore certain phenomena leading to this failure mode (e.g., DCH or vessel rocketing) are rendered almost insignificant.
- Intermediate-term containment failure (i.e., rupture) due to hydrogen and carbon-monoxide combustion has roughly comparable conditional probabilities. The reactor at-shutdown PRA entails somewhat smaller quantities of combustible gases in the late time frame, combined with somewhat lower base pressures, but this appears to be balanced out by the higher overall conditional probability of more combustible conditions (due to containment coolers and/or EDMG-based spray operation). Also contributing to the latter factor is the fact that many accident sequences in the reactor at-shutdown PRA occur with a large amount of water in the containment due to refueling operations, or due to losses of inventory, in which this water does not heat up significantly before core damage. In some cases, these conditions were predicted by MELCOR calculations to passively mitigate rising steam concentration in the containment and thereby lead to more combustible conditions in the intermediate time frame.
- BMT and late containment failure release categories have much smaller conditional probabilities for the reactor at-shutdown study due to the lower decay heat, which drives progression to core damage, core-concrete interactions, and containment pressurization.

2.6.3 Model Uncertainty and Sensitivity Analyses

The potential sources of model uncertainty are separated into 16 categories related to the technical elements of the Level 2 PRA development. One or more individual issues subject to uncertainty are identified and discussed within each category. Qualitative analyses along with sensitivity analyses are presented to demonstrate the potential impacts uncertainties may have on the results. This model uncertainty analysis follows a similar approach to the reactor at-power, internal events and floods Level 2 PRA (NRC, 2022a). The current study for reactor at-shutdown builds upon that previous work since much of the Level 2 PRA logic model and the underlying phenomena of importance are largely the same.

The following subsections summarize the individual model uncertainty issues by category, along with a brief discussion of their applicability to the shutdown PRA. Selected alternate modeling assumptions are evaluated to assess the impacts of modeling uncertainties on the LPSD PRA results. The following model uncertainty categories are presented:

- PDS binning model uncertainties (Section 2.6.3.1)
- Equipment performance model uncertainties (Section 2.6.3.2)
- Human reliability analysis model uncertainties (Section 2.6.3.3)
- In-vessel accident progression model uncertainties (Section 2.6.3.4)
- Induced RCS component failure model uncertainties (Section 2.6.3.5)
- Vessel breach and associated energetic event model uncertainties (Section 2.6.3.6)
- Combustible gas model uncertainties (Section 2.6.3.7)
- Long-term containment pressurization model uncertainties (Section 2.6.3.8)
- Ex-vessel coolability and MCCI model uncertainties (Section 2.6.3.9)
- ISLOCA model uncertainties (Section 2.6.3.10)
- SGTR and induced SGTR model uncertainties (Section 2.6.3.11)
- Containment isolation failure model uncertainties (Section 2.6.3.12)
- Release pathway model uncertainties (Section 2.6.3.13)
- Other fission product and emergency preparedness model uncertainties (Section 2.6.3.14)
- Accident termination model uncertainties (Section 2.6.3.15)
- MELCOR solution robustness model uncertainties (Section 2.6.3.16)

2.6.3.1 PDS Binning Model Uncertainties

Table 2-30 identifies issues contributing to model uncertainty in the PDS binning.

Table 2-30 Identified Uncertainties Related to PDS 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 (e.g., failure probability for CCUs based on 24 hours of operation), as well as in screening out some modeling (e.g., lack of modeling of room cooling for containment spray pumps due to heatup taking longer than 24 hours).	These systems may be credited in the Level 2 PRA for longer time periods (i.e., up to 7 days), particularly CCUs.
Number 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 PRA 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, 2008a) 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, RCS status, SG cooling, RWST availability, and ECCS availability and timing) was based on key underlying assumptions.	As an example, gravity injection in SD09 quickly dead-heads due to rising containment pressure. Earlier or later injection timing may lead to different accident progression. The linkage rules are tailored to facilitate investigating the effects of most of these assumptions.
Partial/degraded performance not credited in the Level 1 PRA	Effects are missed for systems that come into play but do not fulfill their Level 1 PRA mission.	This uncertainty may be effectively duplicative to other uncertainties identified above.

(Continued on next page)

Table 2-30 Identified Uncertainties Related to Plant Damage State Binning (cont.)

Item	Description	Other comments
Habitability limits for containment closure	Containment closure HEPs have been evaluated based on the time available to perform the closure for the POS-specific conditions. For many of the modeled HFES, the available time is based on having to complete the action prior to RCS reaching saturation temperature. However, for POS 5B, additional time after RCS saturation is credited for containment closure based on assessment of the habitability conditions (as described in Appendix D). There is no well-established basis for the maximum containment air temperatures that would prevent personnel from performing containment closure actions, but the analysis is informed by a series of plant-specific containment temperature calculations with MELCOR. It is possible that containment closure actions can be successfully performed after the RCS reaches saturation temperature, particularly if protective equipment were used. It is also possible that performance could be negatively impacted or prevented at earlier times in the event progression as the onset of boiling will introduce harsh conditions in addition to the temperature (e.g., noise, humidity, visibility, and fission products in containment) that can impact the ability of operators to perform this action. Note that consideration of containment closure after the time of RCS saturation for the purposes of the PRA should not be viewed as superseding any established regulatory positions and industry guidance to maintain the capability to close containment before steaming inside containment.	Since open containment release categories are significant contributors to LERF, there is reason to expect this could have a significant impact on shutdown risk. There are also several uncertainties associated with the incorporation of habitability limits. The time available prior to reaching the limit of habitability is estimated for different POS conditions, but not all conditions are assessed in detail and some approximations are made. While habitable conditions may continue to exist for some time as the RCS transitions from localized boiling to bulk boiling and the containment gradually heats up, the exact timing of reaching inhabitable conditions is uncertain. Also, other adverse conditions (e.g., humidity, noise, visibility, and radiation) associated with RCS boiling, beyond temperature, could prevent successful containment closure. Sensitivity cases MU-1.2A and B assess the impacts of more pessimistic and more optimistic timing assumptions than the base case for containment closure HFES. This issue is also suggested as a candidate for future work to improve the basis for habitability limits for containment closure and better characterize the potential conditions that can influence this action.

(Continued on next page)

Table 2-30 Identified Uncertainties Related to Plant Damage State Binning (cont.)

Item	Description	Other comments
Containment spray system recovery	Normal containment sprays are unavailable in POSs 4–12, and the at-shutdown bridge tree gives no credit for operator action to potentially return them to service. It is possible that actions could be taken to restore the containment spray system to service.	Under some slow evolving accidents, there would likely be sufficient time to perform restoration actions. Additional analysis is needed to determine if there are adequate cues, procedural guidance, and other conditions that would justify credit for restoration. Note that EDMG-based actions to initiate containment sprays are credited and model uncertainties associated with crediting post-core-damage operator actions are discussed in Section 2.6.3.3.

*Alternate treatment(s)***MU-1.1 – Specific PDS binning assumptions**

Modeling assumptions are applied in defining the PDS binning linking rules based on the attributes of the Level 1 PRA CDF sequences. In some cases, assumptions are made about the availability and timing of systems and functions, such as SG cooling, ECCS injection, and RWST inventory. The availability of these systems and functions is clearly defined for most of the CDF sequences. However, in some cases the status is indeterminate, and assumptions are applied. The impacts of these assumptions can be explored by reformulating the assumptions in a more pessimistic or more optimistic way. A similar sensitivity case was performed for the reactor at-power Level 2 PRA. (See Appendix C, Section 4.1.3 of (NRC, 2022a).) The changes in the PDS logic assumptions did not have a notable effect on the contribution of any release categories for the at-power model. Similarly, these assumptions are not expected to have a notable effect on the shutdown model. The dominant CDF contribution from station blackout scenarios influences the unavailability of systems, and this does not change with a reevaluation of indeterminant linkage rules. Due to the expected minimal impact, no sensitivity cases were performed for the reactor at-shutdown Level 2 PRA.

MU-1.2A/B – Containment closure human error probabilities

HFEs related to closing containment are significant contributors to release category frequencies. The estimation of HEPs is influenced by many uncertainties related to human reliability modeling, most notably for the shutdown Level 2 PRA are the assumptions about habitability in a high temperature environment. Refer to Appendix D on modeling of containment related human actions for details on the HFE assumptions. The sensitivity to the HEP values is assessed by considering a more pessimistic (in case MU-1.2A) and a more optimistic (in case MU-1.2B) time available to execute the containment closure action. Appendix D describes the possible ranges of times that could be available to support habitable conditions for performing the containment closure depending on containment conditions and modeling assumptions (e.g., containment fan coolers in operation, initial containment temperature, and modeling of hatch opening to the environment). The times available for the modeled HFEs and the resulting HEP values for the base and sensitivity cases are summarized in Table 2-31. The available time

excludes the delay time from event initiation through operators diagnosing the event and performing the procedure-directed steps, which prompt the containment closure. The delay time ranges from 15 to 30 minutes for different POS conditions. For some of the modeled actions, assuming additional time has no influence on the HEP value. This is due to the insensitivity of the HRA quantification method to some of the time changes. Also, when HFE dependency is present for the containment closure HFEs, the dependent HEP values are adjusted in the sensitivity cases.

One of the key sources of uncertainty is associated with the timing assumptions for HFEs modeled for POS 5B conditions. POS 5B has a time to reach RCS saturation temperature of about 30 minutes (NRC, 2022a). Under these conditions, successful containment closure prior to RCS reaching saturation temperature is unlikely since the time required to perform the containment closure actions is assumed to be at least 35 minutes. Still, operators are expected to attempt to perform the procedure-directed actions to close containment if habitable conditions to support the action exist. One hour after reaching RCS saturation is credited with habitable conditions to perform containment closure for POS 5B actions. This assumption impacts HFEs 1-CNT-XHE-XM-H1 and 1-CNT-XHE-XM-SBO1. HFE 1-CNT-XHE-XM-SBO1 is applied to SBO sequences and is a dominant contributor to the release frequency for category 1-REL-SD-COPEN-RCSVENT. If more pessimistic assumptions about containment habitability are applied (i.e., actions are completed prior to RCS saturation temperature), then these HFEs are assumed to fail during POS 5B. The same HFEs are also applied to POS 9 conditions, but more time is available before reaching RCS saturation temperature in POS 9 (about 80 minutes) due to the lower decay heat level associated with this POS. Even under pessimistic timing assumptions, there is a possibility of successfully closing containment prior to boiling for POS 9 conditions. The alternate timing assumptions and HEP values in Table 2-31 reflect the time-to-boil differences for POS 5B and POS 9, even though the same HFEs were used for both POSs in the base case.

For POSs with the possibility of SG cooling, HFEs 1-CNT-XHE-XM-H2 and 1-CNT-XHE-XM-SBO2 could potentially benefit from additional time if operators are able to establish and maintain the cooling. The optimistic case considers additional time available for these two actions based on potential SG cooling.

The results of the containment closure HEP sensitivity cases are shown in Table 2-32. The containment closure HEP values have a significant influence on the release category frequency results for categories with the containment open. For the pessimistic case (MU-1.2A), release category frequencies for 1-REL-SD-COPEN-RCSINTACT and 1-REL-SD-COPEN-RCSVENT increase by 362 percent and 346 percent, respectively. Both release category frequencies show significant increases depending on the HEP values used for containment closure. Both release categories are associated with large magnitude releases (due to the open containment hatch), and they contribute to increases in LERF and LRF. The HEP values also influence the frequencies of other release categories because of the contributions from the containment closure HFE success terms. For the pessimistic containment closure case, the largest decreases in frequency are seen in the basemat melt-through (1-REL-SD-BMT) and late containment failure (1-REL-SD-LCF) release categories, which show 27 percent and 34 percent reduction in frequency, respectively. For the optimistic case (MU-1.2B), release category frequencies for 1-REL-SD-COPEN-HEADOFF and 1-REL-SD-COPEN-RCSVENT are significantly reduced.

The LERF results are also significantly influenced by the containment closure HEPs. LERF includes release categories 1-REL-SD-COPEN-RCSINTACT, 1-REL-SD-COPEN-RCSVENT, and 1-REL-SD-ISLOCA. LERF is increased by about 267 percent in the pessimistic case and reduced by about 19 percent in the optimistic case. This sensitivity shows that the assessment of HFEs for closing an open containment, particularly assumptions about habitability and timing, can significantly influence release category frequency and LERF results.

Table 2-31 Containment Closure HEP Sensitivity Cases

HFE Basic Event Name and Condition Description	Base Case		MU-1.2A Sensitivity		MU-1.2B Sensitivity	
	Time Available (hr.)	HEP	Time Available (hr.)	HEP	Time Available (hr.)	HEP
1-CNT-XHE-XM-H1 POS 5B, RCS vented	1.25	1.0E-2	~0.25	1.0	~4	1.0E-3
1-CNT-XHE-XM-H1 POS 9, RCS vented	1.25	1.0E-2	~0.75	9.1E-2	~4	1.0E-3
1-CNT-XHE-XM-H1F RCS vented with forced or gravity feed	6.75	1.0E-3	~2	1.0E-2	~15	1.0E-3
1-CNT-XHE-XM-H2 RCS intact	7.5	2.0E-3	~2	2.0E-2	~30	2.0E-4
1-CNT-XHE-XM-H3 RPV head off, flooded	30.5	4.0E-5	~10	4.0E-4	~48	4.0E-5
1-CNT-XHE-XM-SBO1 POS 5B, RCS vented, SBO	1.25	2.3E-1	~0.25	1.0	~4	2.9E-2
1-CNT-XHE-XM-SBO1 POS 9, RCS vented, SBO	1.25	2.3E-1	~0.75	7.5E-1	~4	2.9E-2
1-CNT-XHE-XM-SBO1F RCS vented with gravity feed, SBO	6.1	2.9E-2	~2	2.3E-1	~15	2.9E-2
1-CNT-XHE-XM-SBO2 RCS intact, SBO	7.5	1.3E-1	~2	6.0E-1	~20	1.3E-1
1-CNT-XHE-XM-SBO3 RPV head off, flooded, SBO	30.5	2.9E-2	~10	2.9E-2	~48	3.0E-3

Table 2-32 Containment Closure HEP Sensitivity Release Frequencies

Release Category	Base Case	MU-1.2A Sensitivity		MU-1.2B Sensitivity	
	Frequency (per year)	Frequency (per year)	% Change	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	9.0E-08	-26.7%	1.3E-07	3.4%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.1E-06	0.0%	2.1E-06	1.9%
1-REL-SD-CIF	1.7E-06	1.4E-06	-16.7%	1.7E-06	1.8%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.2%	7.9E-09	-89.8%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	1.2E-06	361.6%	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	5.8E-07	346.1%	3.2E-08	-75.3%
1-REL-SD-ECF	5.1E-10	4.3E-10	-15.3%	5.2E-10	1.6%
1-REL-SD-ICF-BURN	1.7E-06	1.6E-06	-7.7%	1.7E-06	0.8%
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	-1.7%	1.3E-07	0.2%
1-REL-SD-LCF	5.5E-08	3.6E-08	-34.3%	5.9E-08	7.4%
1-REL-SD-NOCF	6.1E-06	5.2E-06	-14.2%	6.2E-06	1.2%
Release Surrogate Measures					
LERF	5.3E-07	1.9E-06	267.1%	4.3E-07	-18.5%
LRF	6.1E-06	7.1E-06	16.5%	6.0E-06	-1.4%
Overall release	6.3E-06	7.2E-06	15.2%	6.2E-06	-1.2%

2.6.3.2 Equipment Performance Model Uncertainties

Table 2-33 identifies issues contributing to model uncertainty in the equipment performance modeling.

Table 2-33 Identified Uncertainties Related to Equipment Performance Modeling

Item	Description	Other comments
RCP performance	It is assumed that RCPs have been manually tripped or are unavailable during the modeled shutdown POSs. However, it could be argued that the RCPs continue to run during the accident progression until a consequential trip causes them to shut down. The RCP operation will improve the effectiveness of core cooling.	RCPs are in operation for only a portion of the modeled shutdown POSs. RCPs are operating during POSs 3–5A and 13. The RCPs are not in operation during POSs 5B–12.
Equipment/instrument survivability for EDMG implementation	A simplified approach is used to qualitatively assess equipment survivability concerns in the context of modeled post-core-damage operator actions. This affects the context for the HRA and the response of equipment that is credited.	The evaluation of equipment survivability and lack of detailed information is discussed in Section 2.3.6. As an example, insufficient cable routing information is available to address situations where the instrument is not damaged, but it's cabling is damaged elsewhere in the plant (or within containment). ¹⁴ However, qualitative assessments of the credited equipment and general characterizations of severe accident conditions are performed to inform the modeling of post-core-damage operator actions. Additional uncertainties related to the post-core-damage HRA modeling are discussed in Section 2.6.3.3 and sensitivity cases MU-3.1A and B.
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 aerosol radiological material.	This particular system is highlighted because its continued operation may have a strong effect on the results, not only by resulting in a higher frequency of basemat melt-through (by preventing long-term gradual overpressure), but also in leading to a higher contribution to intermediate-term containment failure as the result of conditions more conducive to combustion.

¹⁴ Note that EPRI Report No. 3002005385, "Severe Nuclear Accidents: Lessons Learned for Instrumentation, Control and Human Factors," observes on page 4-1 (based on a review of all major past nuclear accidents), 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 2-33 Identified Uncertainties Related to Equipment Performance Modeling (cont.)

Item	Description	Other comments
Reliability/availability of equipment beyond the Level 1 PRA treatment	All equipment in the Level 1 PRA is subject to a 24-hour mission time; meanwhile, some equipment (e.g., the 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 EDMG-based mitigation equipment has been justified based on the very high HEPs associated with its use (see Section 2.4.1 of (NRC, 2022a)).
Clogging/damage issues when alternate water sources are used	Relates to sump clogging or pump damage when firewater (or another alternate source) is used. This is a larger concern for high-pressure pumps due to their tighter clearances and multi-stage design. Uses of firewater or an alternate water source are infrequent in the HRA and probabilistic model.	Note that for most of the modeled shutdown POSs, the containment sump screens are blocked by protective covers, and the sump is not credited. The uncertainty still applies for those POSs where the sump screens are uncovered.
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.	Analysts contributing to this study pointed out during a visit to the reference plant site that there is a hydrogen supply line routed through the auxiliary building.

Alternative treatment(s)

MU-2.1 – Sensitivity analysis to address uncertainty associated with containment fan cooler survivability.

In the reactor at-shutdown PRA, the availability of the containment fan coolers was assigned based on plant conditions at the start of core damage, and it is assumed that the fan coolers continue to operate effectively for at least 7 days or until any predicted containment failure. However, the survivability of the fan coolers due to energetic events (e.g., hydrogen combustion) or aerosol accumulation has not been assessed in any detail, so this is viewed as a potential source of uncertainty in the PRA results.

The deterministic analysis of accidents at shutdown (described in Section 2.3.3) examined this issue by means of a sensitivity study to scenario SD12 (a loss-of-inventory accident in POS 10 with forced feed prior to core damage and with fan coolers available). In this base case scenario, several hydrogen deflagrations were predicted over the course of the late time frame, with the first significant burn occurring at 29.8 hours and consuming about 200 kg of hydrogen. A sensitivity MELCOR calculation, SD12.1, was performed in which it was assumed that this burn at 29.8 hours had the effect of damaging the fan coolers and halting effective containment heat removal. Note that in neither case, with or without extended containment heat removal, did

the containment fail by overpressure, due to the relatively low decay heat. Table 2-34 compares the predicted source terms with and without assumed fan cooler failure. The differences are relatively small (i.e., a two- to three-fold increase in cumulative release by fission product class), which are in line with the higher containment pressure that results following loss of containment heat removal.

It should also be noted that this sensitivity study considers only the effects of failure of fan cooler operation for a scenario in which the containment remains intact throughout, regardless of fan cooler operation. A larger impact might be expected indirectly, from the effect of fan cooler operation on containment conditions leading to potential containment failure from a large deflagration or detonation. A sensitivity analysis was not performed using SAPHIRE to model failure of the fan coolers in this sense. However, as a general observation from the base case reactor at-shutdown Level 2 PRA results, scenarios with late containment heat removal are more vulnerable to containment failure from combustion, due to low steam concentration and the possibility of detonable conditions. Therefore, modeling fan cooler failure for some portion of sequences with initial fan cooler availability would likely lead to a lower containment failure probability. Also, a large portion of sequences with containment heat removal can be attributed to operator actions resulting in late containment sprays, rather than fan cooler operation, which would dilute the impact of assumed fan cooler failure.

Based on this analysis of fan cooler survivability, it is concluded that:

- The impact of uncertainty in fan cooler survivability after core damage on the late source term, assuming no changes associated with potential containment failure by combustion, is relatively small. Though there is a small increase in the environmental releases, the releases remain small.
- The main impact of uncertainty in fan cooler survivability after core damage is expected to be seen in a reduction in the probability of intermediate-term containment failure by detonation or large deflagration. However, the specific probability of fan cooler failure due to a combustion event is not known, and a sensitivity analysis on the logic model results to assess the impact on combustion-related release frequencies has not been performed.

Table 2-34 Source Terms to Environment with Sensitivity on Fan Cooler Survivability

Fission Product Class	SD12 (CCUs Do Not Fail)	SD12.1 (CCUs Fail at 29.8 h)
Xe	3.2E-3	6.5E-3
Cs	1.8E-5	4.9E-5
I	1.9E-5	5.8E-5
Te	1.8E-5	4.4E-5
Ba	5.3E-7	1.1E-6
Ru	3.1E-7	1.1E-6
Mo	4.3E-6	1.1E-5
Ce	5.3E-9	1.2E-8
La	1.1E-9	2.9E-9

MU-2.2 – Sensitivity analysis to address uncertainty associated with partial sump clogging or damage when alternate injection is used.

This issue is of potential importance both to reactor at-power and reactor at-shutdown accidents. Although no specific sensitivity analysis was performed for an accident scenario at shutdown, a MELCOR sensitivity case (Case 6R1) was performed and compared in the reactor at-power uncertainty analysis document (MU-2.2 in Appendix C of (NRC, 2022a)). The scenario studied consisted of an accident where one train of RHR was recovered after the start of core damage, and the effect of clogging or damage was modeled by reducing injection flow rate from 1500 gpm to 750 gpm. Due to the reduced flow, core support structure failure was predicted by MELCOR to occur only 14 minutes sooner (i.e., at 19.9 hours versus 20.1 hours), and therefore the success of in-vessel core retention in such scenarios was evaluated as relatively insensitive to this issue. The relevant scenario is similar in the shutdown PRA, except that typical decay heat levels are much lower, so the in-vessel progression is slower. Therefore, if anything, the relevance of as much as a 14-minute sooner core support structure failure would be reduced. Another factor to consider is that even the reduced flow rate might be sufficient to cool the core at shutdown decay heat levels, in cases where a reactor at-power accident might require a full and unimpeded train of injection. Based on the above, the reactor at-shutdown PRA is judged to have very low to negligible sensitivity to this issue.

MU-2.3 – Sensitivity analysis on the impact of RCP performance on accident progression

The base case set of MELCOR calculations performed for accidents at shutdown included one scenario taking place in POS 3, RCS cooldown (SD08). In this plant operating state, the RCPs are expected to be running at the time when the accident occurs. The assumption simplistically used in the MELCOR deck is to not credit RCP operation, by assuming that the pumps trip at the instant when the accident begins. However, this is a debatable position, and it could be argued that the pumps would be allowed to continue running until a consequential trip forces them to shut down. In order to analyze this case, a MELCOR sensitivity calculation (SD08.2) was performed in which the RCPs were allowed to keep operating until imminent cavitation (void fraction of 0.1 or higher predicted at the pump suction of any RCP). The results of this calculation showed relatively little difference from the base case, except for slightly earlier in-vessel event progression: the time when core uncover begins was reduced from 25.44 hours to 24.94 hours, and the time of vessel breach from 32.66 hours to 32.09 hours. Meanwhile, as Table 2-35 shows, the radiological source term is also negligibly impacted. The amount of hydrogen generated by oxidation in-vessel was increased from 459 kg to 541 kg; however, this too is on the order of the difference one might expect from normal variation between similar calculations and is not necessarily the consequence of assumptions regarding RCP operation. (The in-vessel oxidation cumulative probability function used in the PRA for such purposes as quantifying hydrogen combustion adequately covers both values.) Therefore, it is concluded that the reactor at-shutdown Level 2 PRA is not particularly sensitive to assumptions regarding operation of the RCPs after the start of core damage.

Table 2-35 Source Terms to the Environment for RCP Sensitivity Case

Fission Product Class	SD08 (RCPs Trip at 0 h)	SD08.2 (RCPs Trip at 24.07 h)
Xe	9.1E-3	8.2E-3
Cs	1.8E-6	1.7E-6
I	2.8E-6	2.6E-6
Te	3.6E-6	3.6E-6
Ba	2.6E-7	2.0E-7
Ru	1.4E-8	1.6E-8
Mo	3.5E-6	2.9E-6
Ce	1.5E-8	1.5E-8
La	4.9E-10	4.8E-10

2.6.3.3 Post-Core-Damage Human Reliability Analysis Model Uncertainties

Table 2-36 identifies issues contributing to model uncertainty in the post-core-damage human reliability analysis modeling.

Alternate treatment(s)

MU-3.1A/B – Sensitivity analysis on post-core-damage human failure events

Sensitivity analyses are performed to assess the impacts of varying the HEP values for post-core-damage modeled actions to consider more pessimistic and more optimistic conditions for operator action. For sensitivity case MU-3.1A, all HEPs for post-core-damage actions modeled in the shutdown PRA are assigned a value of 0.9. For sensitivity case MU-3.1B, the base case HEP values are reduced by a factor of 10. The modeled post-core-damage actions and the HEP sensitivity cases are summarized in Table 2-37.

The results of the post-core-damage sensitivity cases are shown in Table 2-38. Sensitivity case MU-3.1A, with more pessimistic HEP values than the base case, shows a significant increase to the 1-REL-SD-LCF release frequency. The frequency increase is primarily due to the increased failure probabilities for failure to recover injection and failure to initiate containment sprays. This suggests that these actions can be effective for reducing the likelihood of late containment failure. Sensitivity case MU-3.1B, with more optimistic HEP values than the base case, shows reduced release frequencies for 1-REL-SD-BMT, 1-REL-SD-ECF, 1-REL-SD-ICF-BURN, and 1-REL-SD-LCF, with frequency reductions ranging from 90–99 percent. The reduced HEPs for failure to recover injection are the primary contributors to the release frequency reductions. The LERF results are insensitive to the post-core-damage HEP values. The LRF and overall release frequency show only modest increases or decreases with the changes in HEP values.

Table 2-36 Identified Uncertainties Related to Human Reliability Analysis

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.	While specific accident management actions are not addressed, the uncertainties are mainly captured by other related uncertainty items (e.g., see Section 2.6.3.15 on accident termination modeling). Note, at the time of analysis the reference plant SAMGs explicitly excluded shutdown operations, so the SAMGs were not considered within the shutdown Level 2 PRA HRA. The EDMG-related activities are modeled in the shutdown Level 2 PRA HRA.
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 is 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 and in Appendix E. External peer review for the at-power Level 2 PRA also raised this area of uncertainty as related to modeling of HFEs.
Limitations on the number of actions considered	In reality, there would likely be a more complex series of recovery actions attempted that may be more effective at limiting releases than those currently modeled.	The human actions that are considered and modeled are summarized in Sections 2.4.4 and 2.4.5 above.
Limited credit for post-core-damage actions during SBO and late accident progression	The Level 2 PRA HRA approach for this study limits credit for post-core-damage actions for some SBO conditions and long-term actions, due to the lack of instrumentation availability, habitability concerns, and limited resources due to prior failures or depletions.	A similar approach was used for the at-power Level 2 PRA HRA (NRC, 2022a). This was a particular point of concern for both the project's Technical Advisory Group (TAG) and the peer reviewers.
Automatic actuation of equipment not operating at the time of core damage	Unlike the at-power PRA, almost all use of mitigating systems is based on manual action to return equipment to operation. For most plant configurations of interest to the shutdown Level 1 HRA, plant equipment will have been placed in pull-to-lock as part of plant cooldown and refueling outage entrance. Some of the ambiguities on the state of the equipment prior to core damage have not been explicitly addressed. These assumptions impact the potential availability of equipment that can be credited as part of post-core-damage actions.	Refer to Table 2-15 for a summary of the equipment impacted by pull-to-lock assumptions.

Table 2-36 Identified Uncertainties Related to Human Reliability Analysis (cont.)

Item	Description	Other comments
Modeling assumptions associated with habitability, including MCR habitability	This issue is discussed in detail in the at-power Level 2 PRA report Appendix D entry entitled, "Habitability in Accident Management" (NRC, 2022a)—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 containment 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 (NRC, 1995b) alternative source term (depending on the plant's licensing basis). Note that uncertainty due to containment habitability during equipment hatch closure scenarios is addressed in Section 2.6.3.1, above.
Modeling of human actions to isolate and cool containment	Proceduralized actions to isolate containment and to initiate containment cooling units are modeled and discussed in Appendix D. Success or failure of these actions can have an important influence on the accident progression. Assumptions about timing and sequence conditions can influence the estimation of HEPs.	The HFEs are evaluated using a consistent HRA approach with other pre-core damage HFEs modeled in the Level 3 PRA project. The assumptions and results are documented in Appendix D. Alternate modeling assumptions about the HFE context could influence the HEP estimates and are addressed in Section 2.6.3.1.
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 and, therefore, the use of hydrogen recombiners is not considered. The hydrogen recombiners have been removed from the reference plant's Technical Specifications by license amendment. They are considered retired-in-place (i.e., installed but not maintained). Based on discussions with plant staff, there is no indication that they would be used.
Modeling of offsite resources for accident management support	Affects credit given for resources in flooding the reactor cavity.	See the at-power Level 2 PRA report Appendix D entry entitled, "Termination of Radiological Releases" (NRC, 2022a).

Table 2-37 Post-Core-Damage Action HEP Sensitivity Cases

HFE Name	HFE Description	Base Case HEP	MU-3.1A Sensitivity HEP	MU-3.1B Sensitivity HEP
1-L2-OP-EDMGB-SD1	Failure to recover injection to vessel (post-refueling loss of inventory accidents)	2.2E-01 ⁽¹⁾	9.0E-01	2.2E-02
1-L2-OP-EDMGB-SD2	Failure to recover injection to vessel (post-refueling transient accidents)	5.4E-01 ⁽¹⁾	9.0E-01	5.4E-02
1-L2-OP-EDMGB-SD3	Failure to recover injection to vessel (pre-refueling loss of inventory accident)	5.6E-01 ⁽¹⁾	9.0E-01	5.6E-02
1-L2-OP-EDMGB-SD4	Failure to recover injection to vessel (pre-refueling transient accidents)	7.4E-01 ⁽¹⁾	9.0E-01	7.4E-02
1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (RCS intact or vented)	2.4E-01	9.0E-01	2.4E-02
1-L2-OP-EDMGC-SD2	Operator fails to initiate containment sprays (reactor vessel head removed)	5.8E-01	9.0E-01	5.8E-02
1-L2-OP-EDMGH-SD1	Operator fails to inject to SGs	2.4E-01	9.0E-01	2.4E-02
1-L2-OP-EDMGP-SD1	Operator fails to open reactor vessel head vents	1.7E-01	9.0E-01	1.7E-02
Note 1: The nominal HEP estimates for failure to recover injection are 0.14 for loss of inventory events and 0.52 for transient events (including ISLOCAs). The distinction between the HEP values is primarily based on radiological habitability concerns in the vicinity of the auxiliary building and RWST. The HEP estimates are described in Appendix E. The stated probabilities combine the nominal HEP values with the probability of failure of having a coolable core. Success of these events implies both success of the operator action to restore injection and successful conditions for core cooling (i.e., the core is still in a coolable geometry at the time injection is restored). The mean failure probabilities for failure of coolable conditions are 0.5 for pre-refueling POSs and 0.1 for post-refueling POSs. The approach for combining the failure probabilities and establishing the uncertainty ranges is described in Appendix C.				

Table 2-38 Post-Core-Damage Action HEP Sensitivity Release Frequencies

	Base Case	MU-3.1A Sensitivity		MU-3.1B Sensitivity	
Release Category	Frequency (per year)	Frequency (per year)	% Change	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	1.5E-07	23.4%	1.2E-08	-90.5%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.1E-06	0.0%	2.1E-06	0.0%
1-REL-SD-CIF	1.7E-06	1.7E-06	0.2%	1.7E-06	-0.9%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.0%	7.8E-08	0.0%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.7E-07	0.0%	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.3E-07	0.0%	1.3E-07	0.0%
1-REL-SD-ECF	5.1E-10	6.7E-10	31.0%	2.1E-11	-95.9%
1-REL-SD-ICF-BURN	1.7E-06	2.2E-06	28.2%	1.7E-07	-90.0%
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	0.0%	1.3E-07	0.0%
1-REL-SD-LCF	5.5E-08	2.5E-07	355.4%	3.8E-10	-99.3%
1-REL-SD-NOCF	6.1E-06	5.3E-06	-12.9%	7.8E-06	28.4%
Release Surrogate Measures					
LERF	5.3E-07	5.3E-07	0.0%	5.3E-07	0.0%
LRF	6.1E-06	6.6E-06	8.0%	4.5E-06	-25.5%
Overall release	6.3E-06	7.0E-06	11.3%	4.5E-06	-27.4%

MU-3.2 – Sensitivity analysis for operator actions during long-term accident progression

The shutdown HRA approach for post-core-damage operator actions limits the credit for certain station blackout conditions and for actions during long-term accident progression (meaning roughly 6 hours or more after vessel breach) for all scenarios. It is understood that operators would continue to take actions during these longer time frames. However, concerns about instrumentation availability, habitability, and limited resources to support implementation of actions suggest a high likelihood of failure. Also, SAMG-guided actions are not credited because the reference plant's SAMG guidance explicitly excluded actions for events initiating with the reactor shutdown.¹⁵ Long-term recovery actions could include those that were successfully or unsuccessfully carried out at the various Fukushima Daini and Daiichi units in March 2011 (restoration of direct current [DC] power, restoration of AC power, containment venting, etc.). Nevertheless, credit for such actions is beyond the scope of the Level 2 PRA HRA approach used in this study and generally beyond the state-of-practice in Level 2 PRA.

The sensitivity analysis performed here considers three potential types of long-term recovery actions: actions to prevent significant combustion events in containment, actions to control containment pressure, and actions to prevent basemat melt-through. While SAMGs are not credited for shutdown accidents in the base case results, these postulated actions can provide

¹⁵ The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available and thus not considered at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

indications of potential benefits of SAMG-guided actions. These actions and their impacts are discussed further below:

- Actions that prevent significant combustion events in the long term (RF_{combust}) – Examples of these actions include venting combustible gases or igniting at lower flammability levels. Because sequences with combustion-induced containment failure almost exclusively involve late containment heat removal or otherwise mitigated containment pressure, late overpressure is generally not the secondary result if combustion were to be controlled. The sequences affected by the contemplated actions would mainly be driven to releases resulting from basemat melt-through (i.e., from 1-REL-SD-ICF-BURN to 1-REL-SD-BMT).
- Actions that successfully control containment pressure through restoration of containment heat removal or containment venting (RF_{pressure}) – These actions would tend to drive sequences to the intact containment end state (i.e., from 1-REL-SD-CF-BEFORE-CD and 1-REL-SD-LCF to 1-REL-SD-NOCF).
- Actions that flood the cavity with timing and flow rates sufficient to arrest basemat ablation prior to basemat melt-through (RF_{BMT}) – These actions drive frequency from the 1-REL-SD-BMT release category to the 1-REL-SD-NOCF release category.

This sensitivity analysis relies on simple manipulations of the baseline release frequency results and considers these actions to apply across all release category sequences. Postulated recovery factors of 0.1 are applied for each type of action. These represent potential failure probabilities for implementing the recovery actions. The recovery factors are applied in the above order, to the extent that they have overlapping effects. The sensitivity analysis assumes that continuing actions will have an overall positive effect (i.e., the potential for actions to exacerbate the accident are not considered).

The impacts on the risk surrogate measures LERF, LRF, and CCFP are shown in Table 2-39. The results show that LERF is insensitive to the postulated recovery actions because none of the affected release categories contribute to LERF. The largest impact is related to recovery actions intended to control containment pressure, which results in a 31 percent reduction in LRF and CCFP is reduced from about 50 percent to 31 percent. Actions intended to control combustion can also reduce the risk measure results. In this case, LRF is reduced by about 25 percent, and the 1-REL-SD-ICF-BURN release category frequency contribution to LRF is reduced from about 30 percent to about 4 percent. The action to control basemat melt-through shows potential for small reductions in CCFP. Larger reductions to LRF and CCFP are realized when the successful recovery actions are combined. The sensitivity analysis demonstrates the potential benefits of long-term recovery actions, in particular actions related to controlling containment pressure and preventing large combustion events in containment. The discussion of accident termination assumptions in Section 2.6.1 demonstrates similar results if long-term scenarios can be terminated prior to containment failure.

Table 2-39 Risk Surrogate Measures for Postulated Long-Term Recovery Actions

Postulated Recovery Factors			Resulting Risk Surrogate Measures		
RF _{combust}	RF _{pressure}	RF _{BMT}	LERF	LRF	CCFP
1	1	1	5.3E-07	6.1E-06	50.6%
0.1	1	1	5.3E-07	4.6E-06	50.6%
1	0.1	1	5.3E-07	4.2E-06	31.4%
1	1	0.1	5.3E-07	6.1E-06	49.7%
0.1	0.1	1	5.3E-07	2.7E-06	31.4%
0.1	1	0.1	5.3E-07	4.6E-06	43.4%
1	0.1	0.1	5.3E-07	4.2E-06	30.6%
0.1	0.1	0.1	5.3E-07	2.7E-06	22.9%

MU-3.3A/B – Containment isolation and cooling human error probabilities

HFEs related to isolating containment and initiating containment cooling units can have an important influence on accident progression. These actions are prompted by abnormal operating procedure guidance prior to initiation of core damage. The modeling assumptions and HEP estimates are documented in Appendix D. The HEP values can be influenced by the POS-specific conditions and assumptions about the timing of event progression. The modeled accident sequences include a wide range of initiating events and POS conditions, while the analyzed HFEs focus on the conditions related to dominant sequences and may not be specifically tailored to cover the entire range of sequences. The sensitivity analyses performed here consider a more pessimistic case (MU-3.3A) and a more optimistic case (MU-3.3B) for the containment isolation and containment cooling HEPs. The HEP values are increased or decreased by a factor of 10 for the sensitivity cases. Note that actions to close large containment openings (i.e., equipment hatch and personnel airlock) are excluded from this sensitivity. The impacts of alternate HEP values for those actions are considered in Section 2.6.3.1 with sensitivity cases on alternate habitability assumptions. The alternate HEP values for containment isolation and containment cooling are summarized in Table 2-40. When HFE dependency is present for the containment closure HFEs, the dependent HEP values are adjusted in the sensitivity cases.

The results of the containment isolation and containment cooling HEP sensitivity cases are shown in Table 2-41. The primary impact is seen on the containment isolation failure release category frequency, 1-REL-SD-CIF. The frequency is increased by 11 percent for the more pessimistic HEP values. The small decreases in other release category frequencies are primarily due to the inclusion of successful containment isolation events. The HEP values related to initiating CCUs have minimal impacts on the release frequencies. In principle, the operation of CCUs can influence the likelihood of combustion events (e.g., by reducing containment steam concentration resulting in more flammable conditions); however, the EDMG-guided actions to establish containment sprays have a much greater influence on the late term combustion events that can result in containment failure. The more optimistic HEP values (case MU-3.3B) have minimal impact on the release frequencies. Overall, the release frequency results show only small sensitivity to these HEP values.

Table 2-40 Containment Action HEP Sensitivity Cases

HFE Name and Description	Base HEP	MU-3.3A HEP	MU-3.3B HEP
1-CCU-XHE2-SD Failure to restore CCUs during shutdown	3.2E-04	3.2E-03	3.2E-05
1-CCU-XHE2-SD-POS6 Failure to restore CCUs during POS 6	1.4E-03	1.4E-02	1.4E-04
1-CIS-XHE-XL-ISOLPEN Failure to isolate containment penetrations	3.0E-03	3.0E-02	3.0E-04
1-CIS-XHE-XL-PURGE Failure to isolate containment penetrations with purge system operating	2.0E-04	2.0E-03	2.0E-05

Table 2-41 Containment Action HEP Sensitivity Release Frequencies

Release Category	Base Case	MU-3.3A Sensitivity		MU-3.3B Sensitivity	
	Frequency (per year)	Frequency (per year)	% Change	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	1.2E-07	-2.6%	1.2E-07	0.2%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.0E-06	-2.7%	2.1E-06	0.1%
1-REL-SD-CIF	1.7E-06	1.9E-06	11.0%	1.7E-06	-0.8%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.0%	7.8E-08	0.0%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.7E-07	0.0%	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.3E-07	0.0%	1.3E-07	0.0%
1-REL-SD-ECF	5.1E-10	4.9E-10	-2.7%	5.1E-10	0.2%
1-REL-SD-ICF-BURN	1.7E-06	1.7E-06	-2.3%	1.7E-06	0.2%
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	-0.2%	1.3E-07	0.0%
1-REL-SD-LCF	5.5E-08	5.4E-08	-0.8%	5.5E-08	0.1%
1-REL-SD-NOCF	6.1E-06	6.0E-06	-1.4%	6.1E-06	0.1%
Release Surrogate Measures					
LERF	5.3E-07	5.3E-07	0.0%	5.3E-07	0.0%
LRF	6.1E-06	6.2E-06	1.5%	6.1E-06	-0.1%
Overall release	6.3E-06	6.4E-06	1.4%	6.3E-06	-0.1%

2.6.3.4 In-Vessel Accident Progression Model Uncertainties

Table 2-42 identifies issues contributing to model uncertainty in the in-vessel accident progression modeling.

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression

Item	Description	Other comments
Axial power profile	The axial power profile used in the MELCOR model was taken from the reference plant MAAP model. It is fairly flat and bottom-peaked.	Axial power profiles for EOC (end-of-cycle) and BOC (beginning-of-cycle) were provided by the reference plant and support the shape of the EOC profile used in the shutdown MELCOR analysis. The draft Surry (SNL, 2016) and Sequoyah (Barr, 2016) SOARCA uncertainty studies specifically treat the broader issue of using BOC vs. middle-of-cycle (MOC) vs. EOC. As this study focuses on an EOC refueling outage, the use of the EOC profile is appropriate.
Recriticality	This could have an important effect on accident recovery, but only for a small subset of scenarios (i.e., 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 in Section 1.1. It is also discussed generally in Appendix BB 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), the Peach Bottom SOARCA Uncertainty Analysis (NRC, 2016), and Fukushima-related analysis documented in (Denman, 2015) and (SNL 2016a). 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 significant influence on the magnitude of in-vessel hydrogen production, timing of key accident progression events (e.g., RPV lower head failure), and magnitude of source terms.

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression (cont.)

Item	Description	Other comments
In-vessel hydrogen production	This can affect downstream combustion event effects.	It is also worth noting that MELCOR can predict higher in-vessel hydrogen production than MAAP, as explained in (EPRI, 2014). The draft Surry (SNL, 2016) and draft Sequoyah (Barr, 2016) uncertainty analysis results analyze in-vessel hydrogen production as an output metric and show a widespread 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 temperature of in-vessel structures (and thus the timing of core plate failure and lower head failure).
Recovery of a degraded core	Uncertainty in capturing the complex materials and thermal-hydraulics issues is acknowledged. Note that recovery of a degraded core is already treated broadly as a parameter uncertainty through the development of basic events that combine post-core-damage human failure probabilities and the probability of the core being in a non-coolable state. See Sections C.2 and C.3 in Appendix C for more information on these events.	There is a significant degree of uncertainty in this phenomenon, as well as in the human actions and system behaviors required to restore cooling. Recovery of a degraded core is credited for sequences occurring after refueling and with RWST inventory available for injection or already having the refueling cavity flooded as an initial condition (see discussion of core damage recovery in Section 2.4.2). This modeling assumption is supported by evaluated MELCOR scenarios (e.g., cases SD02 and SD04) where the calculation did not reach core damage within the 7-day analysis time. Recovery of a degraded core may be possible for a wider range of sequence types, if RWST refill and extended injection is credited, but this was not credited due to lack of procedural basis for the action. The timing of when the core is reflooded has a significant impact, with recovery being significantly more likely the earlier

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression (cont.)

Item	Description	Other comments
		in core degradation that reflooding occurs. Other studies have varied widely in terms of the credit given. Some recovery calculations in the at-power Level 2 PRA, Appendix B (NRC, 2022a) 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).
Core coolability	The model assumes probabilities of core coolability at the time of alternate injection, based on qualitative arguments and with wide parametric uncertainty bounds.	This uncertainty is related to the recovery of a damaged core, discussed above.
Effect of current CET modeling of slow accident progression sequences	Several cases were identified by shutdown-specific MELCOR analysis where the progression of the accident would be so slow that core damage would require more than the 7-day time frame considered to form the Level 2 PRA duration. Note that, in recognition of the fact that these sequences would still lead to very late core damage in the absence of recovery actions, they are assigned to the “no containment failure” release category instead of a hypothetical “no release” end state.	Further analysis of this uncertainty could show that some portion of the frequency for these slow developing sequences could belong in other, more consequential, release categories. However, given the very long time frame involved (7 days or longer until core damage), other recovery or restoration actions could be considered that could prevent a significant release. See MU-3.2 for additional discussion of long-term recovery actions.
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 dry out of the PRT, including: (i) back-side convection, and (ii) back-side conduction when the tank is partially submerged from flooding of lower containment.	As noted in the discussion of MU-4.2 below, many shutdown accident scenarios are modeled under conditions where the RCS has a large open vent, or the reactor vessel head is removed. In these scenarios, the modeling of the PRT release pathway is not relevant.
Flooding in pressurizer surge line	During loss of RHR transients, there is potential for water hold-up in the pressurizer when the reactor coolant begins to boil. It is possible that the steam velocity in the surge line would be high enough that liquid entrainment prevents water from draining back from the pressurizer. Surge line flooding is a concern prior to core offload, and in situations	An assessment of the potential effect of surge line flooding can be done in any of several ways: (i) hand calculation of the approximate quasi-static steaming flow rate through the surge line, which is then used in the Kutataladze flooding correlation to estimate whether counter-current two-phase flow is likely; (ii) MELCOR calculation to estimate the steaming flow rate, that then either

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression (cont.)

Item	Description	Other comments
	where the RCS is vented via openings in the pressurizer, without accompanying openings elsewhere in the RCS. It can potentially decrease the time to core heatup, if significant water is held up in the pressurizer, or can lead to mis-leading indications of RCS level. It can also affect the viability of RWST gravity drain. It has been a recognized phenomenon since at least the early 1990s.	feeds in to hand calculations or MELCOR-based sensitivity calculations on whether counter-current two-phase flow is likely; (iii) an assessment based on existing literature such as (Fletcher, 1991), (Takeuchi, 1999), (Gonzalez, 2007), and (Carlos, 2012). This issue was considered in the MELCOR analysis performed to support the shutdown Level 1 PRA for accident mitigation timing.
Core radiative heat transfer	Core heatup progression in the PRA is informed by the results of MELCOR calculations, where such heatup is governed by modeling factors that affect radiative heat transfer, including core boundary structure conditions, view factors, and core nodalization. While this issue was not considered to be significant for the reactor-at-power PRA, it was identified as of potentially increased influence in the shutdown study due to slower core heatup and the potential for development of hot and cold regions within the core due to circulation pattern.	The potential impact of this issue could be faster or slower timing of core damage, which could impact numerous downstream phenomena, as well as the radioactive source term.
Air ingress flow effects	This issue refers to the possibility of parallel flow instabilities developing due to localized regions of upwards and downwards flow given air ingress, which might affect heat transfer and the extent of oxidation.	This issue is discussed further in MU-4.4 below.

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression (cont.)

Item	Description	Other comments
ECCS termination assumptions	Some MELCOR scenarios assume ECCS injection ends at the normal at-power indication for switchover to recirculation (RWST low-low level). However, it is uncertain if injection would cease at this point during shutdown when recirculation is not applicable. If injection occurred over a longer period, then the in-vessel progression and timing might be delayed.	The RWST low-low level is reached, there may still be useable volume available in the RWST. Refer to Section 5.3.5 of the LPSD Level 1 PRA (NRC, 2022a) for additional discussion of the ECCS injection modeling assumptions. Extending the injection time to credit additional usable inventory from the RWST could delay the onset of core damage by 2–3 hours and influence the accident progression. Yet, the additional time is not expected to prevent core damage or radiological releases. Actions that could terminate the accident are expected to rely on additional on-site or offsite resources as outlined in EDMGs and SAMGs. Consideration of potential actions that could terminate accidents are discussed in Sections 2.6.3.3 and 2.6.3.15.
Depressurization rate from opening reactor head vents	The model qualitatively assumes that success of operators in opening a reactor head vent could reduce high-pressure sequences to intermediate pressure, but that it could not reduce any to low pressure because of the small leak area.	This modeling assumption is not expected to have a significant impact on the shutdown release frequency results due to the small number of sequences where the depressurization action is credited, which is limited to intact RCS conditions after refueling. The release frequency results are dominated by POS conditions with the RCS vented, vessel head removed, or RCS intact prior to refueling (where limited time available prevents credit for the reactor head vent action).

Table 2-42 Identified Uncertainties Related to In-Vessel Accident Progression (cont.)

Item	Description	Other comments
Loop flow during cooldown, startup, and shutdown conditions	The MELCOR model has not been specifically calibrated to plant data with respect to loop flow (e.g., flow versus heat transfer for a given loop or loop-to-loop variability) during normal operation and accident conditions. This could impact the projected accident sequence timings.	There may be relevant test facility data that could provide calibration input. Note that the calibration issues impact the Level 1 PRA (pre-core damage) modeling assumptions. However, given the model's focus and that the majority of the CDF contributions are associated with POSs where RCS loops are unavailable for cooling, the impact of improving this calibration would be limited. ISP-38 (NEA, 1998), involving the BETHSY facility, may provide useful information for benchmarking mid-loop response to loss of decay heat removal. Other work in this area also exists (e.g., (Liang, 2009)).

Alternative treatment(s)**MU-4.1 – Sensitivity analyses on assumptions in fuel failure modeling**

Two sensitivity cases were performed for the at-power Level 2 PRA (NRC, 2022a) to explore the fuel failure progression. Within the MELCOR code, the parameters related to the eutectic temperature of the fuel and the breakaway temperature (representing the maximum ZrO₂ temperature permitted to hold up molten zirconium in the fuel cladding) were varied in alternative analyses of MELCOR case 1A2. The ranges of these parameters were informed by the uncertainty analysis for the Surry plant as part of the SOARCA study (SNL, 2016). Refer to Section 4.4.5 (sensitivity cases MU 4.3 A and B) in Appendix C of (NRC, 2022a) for a discussion of the analyses. The sensitivity studies show the change in eutectic melt temperature has a noticeable impact on the amount of hydrogen that is formed with the RCS and burns in containment. While these two sensitivities show that the selected parameters have a noticeable effect on in-vessel fuel failure behavior, the overall effect on the cumulative environmental releases is modest. Given these results for the at-power model, a shutdown-specific re-analysis is deemed unwarranted.

MU-4.2 – Sensitivity analysis on pressurizer relief tank modeling

A sensitivity case was performed for the at-power Level 2 PRA to consider alternate modeling of the heat transfer between the pressurizer relief tank (PRT) and the containment atmosphere, as described in Appendix C, Section 4.4.4 (sensitivity MU-4.2) in the at-power Level 2 PRA report (NRC, 2022a). The sensitivity analysis considers convective heat transfer from the PRT to the containment atmosphere and pool scrubbing for the flow paths entering the PRT. The sensitivity analysis shows potential reduction to the environmental releases due to increased retention of fission products in the tank. However, the impact of this modeling uncertainty is limited for the reactor at-shutdown scenarios due to many accident scenarios being modeled under conditions

where the RCS has a large open vent, or the reactor vessel head is removed. In these scenarios, the PRT is not relevant in considering the pathway from RCS to the containment atmosphere. The release category where the PRT modeling could be potentially impactful is 1-REL-SD-COPEN-RCSINTACT, which has a representative scenario modeled during POS 5A with the RCS intact (i.e., MELCOR case SD07). The release pathway from the RCS to containment is through lifting of RHR relief valves to the PRT. A similar reduction in environmental releases as seen in the at-power sensitivity case could be realized for the RCS intact scenario. A specific sensitivity analysis for the RCS intact scenario during shutdown was not performed because the insights are expected to be similar to the at-power sensitivity case. However, a specific sensitivity analysis of the shutdown scenario is identified as a candidate for future work.

MU-4.3 – Sensitivity analysis on pressurizer surge line modeling

A series of MELCOR sensitivities were performed for exploring the issue of surge line flooding and hold-up of water in the pressurizer for a loss of cooling scenario with the pressurizer vented. The analyses were performed to inform the LPSD Level 1 PRA success criteria for gravity injection. The calculations consider alternate assumptions on the momentum exchange length for the surge line flow paths, in addition to sensitivities on other parameters related to gravity-driven injection from the RWST. The calculations show very little dependency on the timing of core uncover related to the surge line modeling assumptions. The variations on gravity injection show more significant impacts on the timing of core damage. While the potential for surge line flooding does not significantly influence the accident progression, the insights from the sensitivity studies did inform the modeling of Level 1 PRA HFEs related to RCS injection and considerations about the uncertainties in pressurizer level indications. Further, the sensitivity analyses that were performed were focused on one type of accident scenario. There may be other impacts on the accident progression that have not been explored in other scenario types. The modeling of surge line flooding is considered an uncertainty that can influence the timing of core damage progression, but based on this limited study, it is not expected to have a significant impact on the Level 2 PRA.

MU-4.4 – Discussion of impact of air ingress on a degraded core

Uncertainties related to the possibility of air ingress into the fuel assemblies can affect the modeling of ruthenium release. There is the potential for enhanced release of ruthenium under oxidizing conditions representative of air ingress into the reactor or SFP assemblies as demonstrated in sensitivity studies in (NRC, 2014) and (NRC, 2001). While those previous studies focused on releases from SFP accidents, similar conditions may be of concern for reactor accident scenarios in which there are large openings in the RCS (e.g., when the RPV head is removed during shutdown) allowing for potential air ingress into the RCS.

The MELCOR calculations for this study predict no significant air ingress. However, MELCOR modeling of the complex multi-dimensional nature of air ingress into the reactor vessel and its impact on oxidation and fuel failure is subject to large uncertainties. While the default parameters of the MELCOR model do not account for the potentially more volatile forms of ruthenium under oxidizing conditions (Gauntt, 2010), the accident scenarios that could lead to such oxidizing conditions are expected to be unlikely. Nevertheless, the potential effects of enhanced ruthenium release have been studied. An approach described in (Gauntt, 2010) demonstrates the impact of MELCOR modeling adjustments to account for a more volatile form of ruthenium under air-oxidation conditions. This enhanced ruthenium release model is

considered as a sensitivity study in the SFP study (NRC, 2014) and applied to accident scenarios involving rapid drain down of the SFP. The study shows a significant increase in the ruthenium release fraction. Also, assessments of data from experimental programs have demonstrated large magnitudes of ruthenium release under controlled conditions [see references (Kärkelä, 2016) and (Ohnet, 2018)], but the modeling of ruthenium chemistry during realistic severe accident conditions is not fully understood at present.¹⁶

Given the current state-of-the art, and the general understanding of ruthenium chemistry compounded by the complex multi-dimensional flow of air inside the reactor fuel assemblies, performance of additional MELCOR sensitivity studies is not expected to provide any new insights into the expected magnitudes of ruthenium release. Rather, a qualitative discussion of the potential impacts of enhanced ruthenium releases is presented here. Informed by the impacts shown in the referenced SFP studies, the enhanced ruthenium release fractions under air-oxidizing conditions could be as much as one to three orders of magnitude higher than currently obtained from the MELCOR calculations performed for the L3PRA project. The most likely impacted release categories for the reactor at-shutdown PRA would be those representing plant conditions with the reactor vessel head removed: 1-REL-SD-COPEN-HEADOFF, 1-REL-SD-CF-BEFORE-CD, and 1-REL-SD-CIF. Significantly increased ruthenium releases for these release categories, could significantly impact the estimated early and latent health consequences from the releases.

While the potential enhanced ruthenium releases would be expected to have significant impacts on the release magnitudes and consequences of some of the shutdown accident scenarios, these should be considered as bounding impacts and not necessarily representative of the actual modeled scenarios. The representative accident scenarios with the reactor vessel head removed are all variations of station blackout events with eventual loss of all reactor decay heat removal. The scenarios show heatup of the refueling cavity water with gradual evaporation and boiloff of the water. These scenarios are much less likely to have potential air ingress than the rapid drain down scenario assessed in (NRC, 2014). Drain down scenarios (i.e., losses of inventory) are modeled in this study, but they do not contribute significantly to any of the release category frequencies. Also, the drain down scenarios that are modeled assume a self-limiting location for the source of inventory loss, which would not result in the fuel being immediately uncovered and exposed to air (i.e., likely inventory loss locations at refueling cavity seals or hot leg and cold leg connected piping are at elevations above the top of fuel). Even if a rapid drain down event were to occur, the timing of fuel heatup, collapse, and relocation is expected to limit the fuel exposure to air, making it much less likely to have the large ruthenium releases that have been shown in other bounding analyses.

While the impacts of an enhanced ruthenium release due to an air oxidizing environment can be significant, the conditions that would lead to such a release are not expected to occur in the representative accident scenarios. However, recognizing that the extent of ruthenium release is a source of modeling uncertainty that could potentially have a significant impact on the health consequences for some severe accidents occurring during reactor shutdown, this issue is identified as a candidate for future work.

¹⁶ Analyses of the expectations and actual releases of ruthenium related to the Fukushima accident were on-going at the time of this analysis. Future insights may become available based on these studies to inform the potential for ruthenium releases from reactor severe accidents.

2.6.3.5 *Induced RCS Component Failure Model Uncertainties*

Table 2-43 identifies issues contributing to model uncertainty in the induced RCS component failure modeling.

Table 2-43 Identified Uncertainties Related to Induced RCS Component Failure

Item	Description	Other comments
Stochastic and high-temperature seizure sticking of power-operated relief valves (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 uncertainty analysis (SNL, 2016) considers this issue from the perspective of passing water, cycling, and high temperature seizure (the latter two issues are assessed in the treatment of the issue in the Peach Bottom SOARCA uncertainty analysis (NRC, 2016)).
Stochastic failure of RHR relief valves	In many of the shutdown scenarios, the RHR relief valves, rather than the PORVs, are the primary means of relieving RCS pressure during a transient. Similar to the PORV uncertainty above, RHR relief valve failure can have a significant effect on accident progression.	RHR relief valve failure can significantly influence the accident progression and release frequency results. In particular, RHR relief valve failure is a significant contributor to shutdown ISLOCA sequences. The modeling of RHR relief valve failures is proposed as an area for future work in Appendix A.
Various modeling uncertainties associated with induced or consequential SGTR	Several modeling uncertainty aspects have been identified in the at-power Level 2 uncertainty analysis (NRC, 2022a) related to induced or consequential SGTR. These include: • Timing of induced SGTR • Fission product retention • Failure of secondary-side relief valves • Modeling of counter-current hot leg/SG flow • Instrument tube failure (as a means of relieving primary pressure and fission product transport to containment) • Modeling of other induced RCS failures, including C-SGTR	Induced or consequential SGTRs are of much lower importance for shutdown compared to at-power. There are few shutdown states where the SGs are in use for RCS cooling, and these states have very small contributions to the overall release frequencies.

(Continued on next page)

Table 2-43 Identified Uncertainties Related to Induced RCS Component Failure (cont.)

Item	Description	Other comments
SG nozzle dam failures	Nozzle dams are installed during mid-loop operation to isolate SGs for surveillance and maintenance. Information on nozzle dam failure mechanisms and operating experience are not widely available, but the impacts of nozzle dam failures are potentially significant.	A sensitivity study was performed for the shutdown Level 1 PRA (see Sections 5.3.5 and 10.3.9 of (NRC, 2022a)). The potential for nozzle dam failure due to pressurization from ECCS injection was explored in a sensitivity case. Potential nozzle dam failure can significantly increase the shutdown CDF. The dominant contributors to the CDF increase are sequences during POS 6 with successful pumped injection. For the shutdown Level 2 PRA, the largest potential impact of nozzle dam failure is expected to be the corresponding release frequency increase. Based on the dominant Level 1 PRA sequences, the largest impact is expected for the 1-REL-SD-ICF-BURN frequency. The nozzle dam failure can also influence the buildup of combustible gases in containment and the transport of fission products from the RCS to containment. Though, the latter issue is not expected to significantly impact releases to the environment because there are other large RCS openings that contribute to fission product releases during the POSs when nozzle dams are installed.

No alternative treatments are proposed.

2.6.3.6 Vessel Breach and Associated Energetic Event Model Uncertainties

Table 2-44 identifies issues contributing to model uncertainty in the reactor vessel breach and associated energetic event modeling.

Table 2-44 Identified Uncertainties Related to 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 the at-power Level 2 PRA, Appendix D entry entitled, "Reactor Cavity and Containment Response at the Time of Vessel Breach" (NRC, 2022a). The same issues are relevant to the shutdown PRA.
RPV lower head failure mechanisms	Will affect the accident progression downstream of vessel failure.	Different possible failure modes exist, as discussed in (EPRI, 2014).
Vessel flange leak-by	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 the at-power Level 2 PRA, Appendix D entry entitled, "Reactor Cavity and Containment Response at the Time of Vessel Breach" (NRC, 2022a), and in terms of potential effects on accident termination in Appendix D entry on "Termination of Radiological Releases" (NRC, 2022a).
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 the at-power Level 2 PRA, Appendix D entry entitled, "Reactor Cavity and Containment Response at the Time of Vessel Breach" (NRC, 2022a).
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. The uncertainties in the modeled basic events are represented by parameter uncertainty distributions. Modeling uncertainties related to these containment failure mechanisms (e.g., uncertainty in modeling the size and location of the containment failure) also exist and are not captured by the parameter uncertainty 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 the discussion of phenomenological issues in Appendix E of the reactor at-power study (NRC, 2022a).
Ex-vessel fuel-coolant interactions (steam explosions)	This could be a low-probability energetic failure event leading to containment failure.	Note that containment failure due to ex-vessel steam explosion is currently disabled in the probabilistic model due to the extremely low estimated likelihood.

Alternative treatment(s)

MU-6.1 – Discussion of lower vessel head failure modeling

The shutdown PRA assumes that lower head failure will be the result of vessel wall melt-through rather than failure of one or more lower head penetrations. Although no shutdown-specific sensitivity analysis was performed for this issue, the at-power, internal event and flood PRA examined it via a MELCOR sensitivity calculation (MU-6.1 in (NRC,2022b)). The results of that calculation showed that the time of first penetration failure was only about half an hour earlier than in the base case with vessel wall melt-through. No significant impact of this earlier vessel failure was found in the predicted radiological source term to the environment. The release magnitude was slightly larger for some fission product classes and slightly smaller for others, but all within the normally expected range of source term uncertainty. Therefore, it was concluded in the reactor at-power study that the PRA is relatively insensitive to uncertainty in the mode of lower head failure. The only essential difference expected for conditions at shutdown is that the decay heat is much lower and the accident progression correspondingly slower. The temperatures of various components in the lower head are expected to be more evenly distributed in this case, and therefore, if anything, the difference in time of vessel wall melt-through and penetration failure would be smaller. Therefore, the shutdown PRA is likewise expected to be insensitive to this uncertainty issue, and no shutdown-specific sensitivity study was performed.

MU-6.2 – Discussion of containment failure size due to high-energy event

In the model uncertainty analysis for the at-power PRA (MU-6.2 in (NRC, 2022a)), two MELCOR sensitivity calculations were performed to explore the impact of a small (2-inch equivalent diameter) or large (8-inch diameter) failure of the containment developing at the time of vessel breach as the result of a hypothetical missile from a high-energy event (steam explosion, high-pressure melt ejection, DCH, or vessel rocketing). Because most of these phenomena are prevalent at high RCS pressure, a high-pressure base case scenario was chosen for analysis. Conditions at shutdown are similar in most essential respects, except that low or medium RCS pressure is the norm (with in-vessel steam explosion the most significant energetic event). The results of the at-power sensitivity calculations, however, did not appear to depend on a particular RCS pressure.

The sensitivity study for reactor at-power discovered that the two cases with containment failure due to a missile showed the expected trends in containment event progression, and that the environmental release of fission products was somewhat larger for the larger containment failure size. However, the environmental release was much higher for both early containment failure cases (i.e., 2-inch and 8-inch failure sizes) than in the base case without early containment failure but later failure due to overpressurization. These qualitative observations would be expected to apply to accidents at shutdown as well. Considering these results, the shutdown PRA includes the potential for early containment failure due to high-energy events as represented by the 1-REL-SD-ECF release category. The size of the containment failure is assumed to be a 4-inch diameter equivalent.

MU-6.3 – Sensitivity analysis on the timing of early containment failure

The shutdown Level 2 PRA model release category for early containment failure (1-REL-SD-ECF) represents all sequences with induced containment rupture any time between the moment

of vessel breach and a time several hours after vessel breach (i.e., enough time for the transient conditions accompanying vessel failure and blowdown to subside and transition to quasi-steady conditions). The main potential causes of early containment failure identified in the shutdown Level 2 PRA are in-vessel steam explosion (i.e., at the time of vessel breach) and hydrogen combustion (which could conceivably occur at any time during the vessel blowdown period). A MELCOR calculation with containment failure at the moment of vessel breach (case SD01.2) is selected as the reference source term for 1-REL-SD-ECF, because in-vessel steam explosions were determined to dominate early containment rupture. However, since the time of containment failure is subject to some uncertainty and may also vary from sequence to sequence, it was thought desirable to analyze the effect of early containment failure timing. For this purpose, the reference source term SD01.2 was compared to the results of a similar calculation, SD01.3, which had otherwise identical boundary conditions except that containment rupture was assumed to occur 4 hours after vessel breach.

Table 2-45 compares the radiological source terms released to the environment for the cases with containment rupture at vessel breach and 4 hours after vessel breach. The release magnitudes are very comparable, although, they are slightly larger for the case with slightly delayed containment failure. This is the result of a threshold effect in which the calculation predicted that tendon gallery water level dropped far enough very late in the calculation (150 hours after the initiating event) to permit freer flow of gases from the containment sump region to the tendon gallery. This is viewed as an idiosyncrasy of the calculation and not directly attributable to the slight delay in containment failure time. Therefore, it is concluded that the reactor at-shutdown Level 2 PRA results are relatively insensitive to differences in the assumed time of early containment rupture.

Table 2-45 Sensitivity Analysis of Timing of Early Containment Failure

Fission Product Class	Case SD01 (no ECF)	Case SD01.2 (ECF at Vessel Breach)	Case SD01.3 (ECF at Vessel Breach + 4 hours)
Xe	1.4E-2	0.95	0.98
Cs	1.8E-5	1.1E-2	1.2E-2
I	1.7E-5	3.0E-2	3.9E-2
Te	2.4E-5	3.4E-3	3.7E-3
Ba	1.6E-6	1.8E-4	2.9E-4
Ru	3.1E-7	8.7E-7	2.5E-6
Mo	1.8E-4	6.0E-2	5.9E-2
Ce	1.0E-7	1.5E-6	3.0E-6
La	1.7E-8	8.7E-7	1.6E-6

MU-6.4 – Sensitivity analysis on probability of IVSE

The current state-of-knowledge regarding IVSE that could fail containment is summarized in (Basu, 1996). Based on the discussion in that document, the probability of a containment failure resulting from an in-vessel steam explosion is very low but also uncertain. A point estimate probability for IVSE occurring of 1×10^{-4} is assumed to be a reasonable estimate based on the expert estimates in (Basu, 1996). A broad uncertainty distribution with an upper bound around

1×10^{-2} is assumed (see Appendix C for description of the uncertainty distribution). The probability is considered more likely at low RCS pressure, so a point estimate of 2×10^{-4} is assumed for low pressure scenarios. A sensitivity analysis was performed where the probability of IVSE was increased by a factor of 100, which corresponds to an upper bound estimate.

Table 2-46 shows the basic events that are used to model the probability of IVSE in the shutdown PRA. Table 2-47 shows the release category frequency results for the upper bound probability values (MU-6.4). The frequency results predictably show that the early energetic containment failure release category, 1-REL-SD-ECF, is significantly increased. The other release category frequencies are not significantly impacted. The LRF and CCFP results show only small increases. While the 1-REL-SD-ECF frequency is significantly increased with the upper bound estimate, the frequency remains a relatively small contributor to the risk metric and overall frequency results. Even with the upper bound estimate, 1-REL-SD-ECF is not a significant release category.

2.6.3.7 Combustible Gas Model Uncertainties

Table 2-48 identifies issues contributing to model uncertainty in the combustible gas modeling.

Table 2-46 Sensitivity Analysis on In-Vessel Steam Explosion Containment Failure Probability

Basic Event Name	Description	Base Case Prob.	MU-6.4 Sensitivity Prob.
1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)	2.0E-04	2.0E-02
1-L2-BE-IVSE-NLP-SD	IVSE Fails Containment (High or Inter RCS Pressure) (Shutdown)	1.0E-04	1.0E-02

Table 2-47 Release Category Frequency Results for In-Vessel Steam Explosion Sensitivity Analysis

	Base Case	MU-6.4 Sensitivity	
Release Category	Frequency (per year)	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	1.2E-07	0.0%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.1E-06	0.0%
1-REL-SD-CIF	1.7E-06	1.7E-06	0.0%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.0%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.3E-07	0.0%
1-REL-SD-ECF	5.1E-10	4.0E-08	7,740%
1-REL-SD-ICF-BURN	1.7E-06	1.7E-06	0.0%
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	0.0%
1-REL-SD-LCF	5.5E-08	5.5E-08	0.0%
1-REL-SD-NOCF	6.1E-06	6.1E-06	0.0%
Release Surrogate Measures			
LERF	5.3E-07	5.3E-07	0.0%
LRF	6.1E-06	6.1E-06	0.6%
Overall release	6.3E-06	6.3E-06	0.6%

Table 2-48 Identified Uncertainties Related to Combustible Gas Modeling

Item	Description	Other comments
In-vessel hydrogen production	This can affect later combustion events and their effects.	This item has been subsumed in Section 2.6.3.4 and repeated here for the sake of cross-referencing.
Ex-vessel combustible gas production	This can affect the likelihood and severity of late combustion events with potential to induce containment failure.	The comparison of the current plant's station blackout results to the draft Surry SOARCA uncertainty analysis base case (SNL, 2016) 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 and 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). The general combustible gas modeling approach is discussed in Appendix E of the at-power Level 2 PRA (NRC, 2022a).
Impact of containment conditions on combustion	Although a probabilistic analysis of combustion was conducted using ERPRA-BURN, this contained point-estimates of MCCI rates, containment atmosphere temperatures and relative humidities obtained from the relevant MELCOR calculations, and therefore uncertainty in containment conditions is not reflected in the combustion modeling results.	
Termination of MCCI	MELCOR simulation results indicate that the level of water in the containment in scenarios with late spray operation may rise to the loop cutouts and spill over into the normally dry reactor cavity. It is uncertain whether there would be sufficient spray inventory to spill over or whether the spill over could quench the debris and end molten core-concrete interactions which produce combustible gases.	

Table 2-48 Identified Uncertainties Related to Combustible Gas Modeling (cont.)

Item	Description	Other comments
Likelihood of ignition source in containment	The presence of ignition sources that could initiate combustion (given combustible conditions) in containment is considered to be likely. However, probability of an ignition source can be influenced by the scenario conditions (e.g., SBO conditions with no energized AC power sources). The likelihood of ignition source presence is assessed qualitatively, and corresponding probabilities are assigned based on judgement to support the stochastic combustion modeling. The assumptions about ignition source probabilities are modeling uncertainties that can influence the estimation of containment failure due to combustion.	

*Alternate treatment(s)***MU-7.1 – Containment conditions affecting intermediate-term containment failure due to combustion events**

The containment event tree issues involving hydrogen combustion were quantified using results from the ERPRA-BURN code. For additional discussion of the ERPRA-BURN approach, refer to Appendix E of at-power Level 2 PRA (NRC, 2022a) and supplemental report on combustion modeling Appendix F of this report. In the model uncertainty analysis for the reactor at-power PRA (MU-7.1 in Appendix C of NRC, 2022a), a sensitivity analysis was performed that amounted to a qualitative validation of the ERPRA-BURN combustion analysis against MELCOR predictions, for scenarios both with and without late containment heat removal. The ERPRA-BURN code generates cumulative probability curves for maximum post-burn containment pressure over a set of Monte Carlo histories. The reactor at-power study performed several MELCOR sensitivity calculations with the BUR package activated at specific times in the simulation in order to force deflagrations at those times (flammable conditions permitting), and the range of MELCOR-predicted containment loads were compared against the ERPRA-BURN cumulative distribution functions to assess the reasonableness of predicted containment failure probabilities.

The overall result of the reactor at-power uncertainty analysis for this issue was that ERPRA-BURN results are reasonable in comparison with MELCOR for the case with containment heat removal (low base pressure in containment), but that ERPRA-BURN could over-estimate late containment loads for scenarios without containment heat removal (high steam concentration and base pressure). It was concluded, therefore, that uncertainty in assumed containment conditions might have a significant impact on the PRA results in terms of conditional probability of containment failure due to combustion.

ERPRA-BURN simulates stochastic combustion histories in the containment, subject to uncertainties in the extent of in-vessel oxidation, the containment fragility, and random timing of individual ignition events. However, for some other input parameters, such as the containment temperature, gas composition, and relative humidity, as well as the rate of MCCI, the code uses point values obtained from a relevant MELCOR calculation. Uncertainty in these latter parameters is not directly addressed in the combustion analysis. Sensitivity cases on the MCCI rate, containment temperature, and relative humidity were developed to assess the impacts of these uncertainties.

Table 2-49 summarizes the results of several ERPRA-BURN sensitivity cases using different boundary conditions. The resulting probability of late containment failure due to combustion is used as a figure-of-merit. The late containment failure probability factors in the probabilities of existence of an ignition source as well as the probabilities of actual ignition. Two ERPRA-BURN cases are evaluated: an SBO case (referred to as case LPSD2 in Appendix F) and a transient with containment heat removal (CHR) available (referred to as case LPSD4 in Appendix F). The SBO case results show the net containment failure probability including both deflagrations and detonations weighted by the relative likelihood of those scenarios. The CHR available case is dominated by containment failures due to detonations, and the containment failure probability only includes failures due to detonation events.

For the SBO case (with no containment heat removal), a few of the sensitivity calculations showed significant reductions on the late containment failure probability. These included the following alternate assumptions:

- 50% Slower MCCI: Under this assumption, it turns out to be difficult to generate enough combustible gases in the late time frame and before steam-inerting to result in any consequential deflagrations (although there is a chance with the amount of in-vessel hydrogen generated from the early time frame).
- 20 to 50°C Hotter Containment: A significantly hotter containment could have enough additional steam content to become inerted to combustion much earlier.
- Higher (60%) Relative Humidity: Similarly, a higher relative humidity could significantly shorten the window of time before the containment becomes steam-inerted and therefore preclude consequential deflagrations.

For the case with containment heat removal available, fewer uncertainties are envisioned, since temperature and relative humidity would be low in any event. Assumptions regarding MCCI rate and higher base pressure in containment (e.g., from MCCI gases) have relatively small impact on the results.

From this analysis, it is concluded that the results for the SBO scenarios could be reduced by considering the alternate assumptions discussed above. None of the SBO scenario alternate assumptions showed significant increases in containment failure probability.

The combustion results for scenarios with containment heat removal available are relatively insensitive to uncertainties in containment boundary conditions.

In the case of the reactor at-shutdown Level 2 PRA, sequences leading to intermediate-term containment failure by combustion (i.e., 1-REL-SD-ICF-BURN) are dominated by containment

heat removal (fan coolers and/or manually operated containment sprays), and therefore, in view of the points above, the overall PRA results are not expected to be impacted significantly by any of these modeling assumptions.

Table 2-49 Sensitivity Cases on Containment Conditions Influencing Containment Failure due to Combustion

ERPRA-BURN Case	Alternate Assumption	Probability of Late Containment Failure by Combustion
SBO Case	Base case	0.14
	50% slower MCCI	0.017
	50% faster MCCI	0.15
	50% slower to 50% faster MCCI (randomly sampled)	0.11
	20 to 50°C cooler containment	0.10
	20 to 50°C hotter containment	0.037
	Humidity decreased from 45% to 30%	0.14
	Humidity increased from 45% to 60%	0.061
CHR Available Case	Base case	0.85
	50% faster MCCI	0.91
	Higher base pressure (+0.5 bar)	0.84

MU-7.2 – MCCI termination

Several deterministic MELCOR calculations were performed (as described in Section 2.3.3) to aid the understanding of event progression, severe accident issue quantification, and radiological source terms for various scenarios. The ERPRA-BURN quantification for scenarios with containment heat removal, for example, was based on conditions derived from the MELCOR results from scenario SD08 from this report.

However, subsequent sensitivity calculations were performed using variations of MELCOR case SD01 (SD01.4, SD01.6, SD01.7, SD01.8, and SD01.9) which underscored that there may be uncertainty with regard to flooding of the reactor cavity by means of spillover of water injected by sprays into the lower containment and through the RCS loop cutouts:

- The amount of water injected by the sprays is based on the available water sources specified in EDMGs. The available inventory for injection is assumed to be 600,000 gallons, but other sources could be available. MELCOR results indicate that 600,000 gallons of water would not quite accumulate to the spillover elevation, but that 850,000 to 900,000 gallons would be sufficient to spill over and fill up the reactor cavity.
- Based on ERPRA-BURN results for the CHR scenario, the average time of containment-failing late combustion in containment was calculated to be about 35 hours following vessel breach. This figure formed the basis for the assumed time of containment failure in scenarios SD01.6 through SD01.8. However, the time of containment failure in ERPRA-BURN is distributed over a wide range, sometimes occurring earlier and

sometimes as late as 100 or more hours after vessel breach. In scenario SD01.6, 900,000 gallons of water are injected via spray operation, which begins flooding the reactor cavity a mere 4.2 hours before the assumed time of containment failure. Considering the uncertainties in the timing of containment failure, flooding of the cavity may not arrest or slow MCCI in time to mitigate or prevent combustion in some scenarios where the containment-failing detonation occurs significantly later than 35 hours after vessel breach.

- The location at which the containment fails following a detonation is uncertain. As indicated for example by scenario SD01.7, failure at a sufficiently low elevation could prevent or reduce the extent of cavity flooding (e.g., water might initially flow into the cavity, but then drainage from the lower containment would result in inability to replenish the cavity water so that it will soon boil dry).

The reference plant reactor cavity design is dry and a relatively confined space, and therefore the molten core materials ejected from the vessel are expected to end up as a molten pool topped by a relatively non-porous solid crust. The expected inventory available for containment spray and the MELCOR sensitivity cases do not find any scenario by which a significant quantity of water could enter the reactor cavity, although, as seen above, some of the sensitivity calculations for spray operation with high water inventories resulted in temporary or sustained cavity flooding some time following vessel breach. The coolability of ex-vessel debris following flooding with limited water in the reactor cavity is uncertain. It is possible that quenching by the water could open cracks or fissures in the top crust that could permit ingress of water and more thorough cooling of the entire pool of debris. However, current understanding based on experimental evidence is that this cooling effect would be reduced if the flooding is delayed long enough after vessel breach for incorporation of a significant amount of concrete and for the crust to become thicker.

Nevertheless, an ERPRA-BURN sensitivity calculation was formulated to examine the potential effects of assuming that overflow of water into the reactor cavity might terminate MCCI at a relatively early point in time. Results from MELCOR scenario SD01.6 indicate that, by the time spillover of water into the cavity begins (at 59.6 hours), the cumulative generation of MCCI gases has reached a value roughly one-quarter of the amount it would reach after 7 full days. Therefore, the ERPRA-BURN sensitivity case uses the CHR scenario and reduces the available amounts of MCCI gases by 75 percent. The results are summarized in Table 2-50. The probabilities that a detonation occurs are presented for the base case and sensitivity case. The probabilities are estimated for scenarios with and without prior combustion events. Note that the dominant sequence contribution to the 1-REL-SD-ICF-BURN release category is associated with containment failures due to detonation with prior combustion events. The detonation probabilities are derived in a manner consistent with the base case probabilities, as described in the supplemental combustion modeling report in Appendix F.

Table 2-50 ERPRA-BURN Sensitivity Case for Assumed MCCI Termination

Derived Failure Probability	Base Case with CHR	CHR Available with 75% Less MCCI Gases
Probability of detonation (with prior combustion)	0.79	0.26
Probability of detonation (without prior combustion)	0.72	0.44

These results indicate that the conditional probability of detonation given one or more prior burns would be reduced approximately by two-thirds (from 0.79 to 0.26) given early termination of MCCI, while the conditional probability of detonation given no prior burns would be reduced by about 40 percent (from 0.72 to 0.44). These results show significant reductions in the likelihood of containment failure due to detonation if MCCI can be terminated. However, as discussed above, it is unlikely that there would be sufficient inventory available to flood the reactor cavity and terminate MCCI. Also, note that ERPRA-BURN models the ex-vessel gas generation at a uniform rate throughout the late time period. A more realistic scenario could involve the gas generation front-loaded and potentially held up in the containment for a long period of time at high concentrations, as opposed to continuous but very slow generation. These conditions are likely more conducive to combustion, and the reduction in detonation probability predicted here is presumed to be an optimistic upper bound.

MU-7.3 – Detonable threshold

The ERPRA-BURN code is used to quantify basic event probabilities related to combustion. The code was designed primarily to model stochastic ignitions and global deflagrations. Deflagration-to-detonation transitions and containment response to detonations are not modeled in detail. However, in recognition of the fact that combustible gas concentrations in the containment can build up significantly in shutdown accidents (e.g., given MCCI sustained over periods of up to seven days), a simplistic condition was added to ERPRA-BURN whereby it was assumed that any ignition in containment under otherwise flammable conditions would transition to detonation if the effective molar concentration of combustible gases is 15 percent or higher.

In actuality, detonable threshold also depends on the steam concentration, the geometry of the containment compartments with respect to where ignition takes place, and other factors. Therefore, the simplistic 15 percent threshold for detonation used in ERPRA-BURN is a source of modeling uncertainty and may lead to conservative results (i.e., detonations could likely require higher concentrations of combustible gases than this). Accordingly, a sensitivity analysis was performed to vary the detonable threshold in ERPRA-BURN. Since the dominant scenario for detonation in this PRA is the case with late containment heat removal via sprays (referred to as case LPSD4 in Appendix F), two ERPRA-BURN sensitivity cases were run with variations on the dominant scenario, in which the minimum detonable concentration threshold was increased to 20 and 25 percent, respectively. Table 2-51 summarizes the late time frame results of these sensitivity cases, alongside a base case with the nominal 15 percent detonable threshold and the case with no detonation modeled (i.e., detonable threshold input as 100 percent or greater). The detonation probabilities are derived in a manner consistent with the base case probabilities, as described in the supplemental combustion modeling report Appendix F.

Table 2-51 ERPRA-BURN Sensitivity Analysis for Minimum Detonable Concentration Threshold

Derived Failure Probability	Base Case 15% Min. Detonable Concentration	No Detonation	20% Min. Detonable Concentration	25% Min. Detonable Concentration
Probability of detonation (with prior combustion)	0.79	0	0.54	0.32
Probability of detonation (without prior combustion)	0.72	0	0.51	0.27

The results for the case with prior combustion show that the probability of detonation drops by about 30 percent in relative terms (i.e., from 0.79 to 0.54) if detonation were assumed to require 20 percent concentration of combustible gases, and by about 60 percent (i.e., from 0.79 to 0.32) if the detonable threshold were raised to 25 percent. Note, since the assumed probability of an ignition source in earlier time frames for this scenario (with AC power available) is so high (0.99 in each time frame), sequences with prior combustion are frequency dominant. Nevertheless, the results for the case without prior combustion are also shown and are similar.

Given the importance of detonations in this PRA to late combustion and the significant relative frequency of the 1-REL-SD-ICF-BURN release category (i.e., intermediate-term containment failure resulting from combustion), this modeling assumption has a sizeable impact. However, the probability of detonation remains significant even under more restrictive assumptions regarding the detonable threshold.

MU-7.4 – Late ignition source frequency

The ERPRA-BURN code models potential ignition events in the containment in a stochastic manner over a set of Monte Carlo simulations of combustion history. Within a particular time frame (e.g., the late time frame following vessel breach), the code assumes random ignition sources are distributed with a uniform probability with respect to time. The functional behavior is similar to the decay of radioisotopes; therefore, one can speak in terms of the frequency of potential ignition events or the half-life or characteristic time in between such events.

The input parameter dictating ignition frequency in ERPRA-BURN is the ignition probability. This is interpreted as the cumulative probability of having at least one potential ignition event over the entire time frame under consideration. That is,

$$P = 1 - e^{-T/\tau},$$

where

- P = the (cumulative) probability of ignition
- T = the duration of the time frame
- τ = the characteristic time in between potential ignition events

and the mean number of such ignition events over the entire time frame can be derived as T/τ .

For the scenario with AC power and late containment heat removal available (referred to as case LPSP4 in Appendix F), the assumed cumulative probability of an ignition source is 0.99.

From the above, and assuming a late time frame duration (from the reference MELCOR calculation results for vessel breach time) of $168 - 32.7 = 135.3$ hours, this is equivalent to a mean of about 4.6 ignition events, or an average of about 29.4 hours between ignition events.

The 0.99 assumption for cumulative ignition source probability is set based on a qualitative view that an ignition source is “highly likely” to be present under such conditions. However, this particular assumption leads to an ignition source frequency that is not very large over the long time frame after vessel breach. It could be argued that the frequency would be significantly higher. The assumption about the ignition source probability is a modeling uncertainty that can influence the estimation of containment failure due to combustion.

In order to examine this issue further, additional ERPRA-BURN sensitivity cases were run with respect to the base case CHR scenario and assuming ignition source probabilities of 0.9999 and 0.999999 in the late time frame, respectively. This is equivalent to doubling or tripling the assumed ignition source frequency, to once every 14.7 hours or once every 9.8 hours. The results are summarized in Table 2-52. The detonation probabilities are derived in a manner consistent with the base case probabilities, as described in the supplemental combustion modeling report in Appendix F.

Note that sequence frequency results are dominated by sequences with prior combustion, since the assumed probability of an ignition source in earlier time frames is very likely. The sensitivity analysis shows that doubling the frequency of ignition sources lowers the probability of detonation by about 19 percent in relative terms (i.e., from 0.79 to 0.63), whereas tripling the frequency of ignition sources lowers the detonation probability by about 43 percent (i.e., from 0.79 to 0.45). The probability of containment failure due to large a deflagration event is also reduced with increasing ignition frequency. Therefore, the containment failure results are sensitive to reasonable variations in late ignition source frequency, although the probability of detonation remains rather high in all cases.

MU-7.5 – Containment failure frequency results considering detonation uncertainties

The SAPHIRE model frequency results leading to intermediate-term combustion-induced containment failure (release category REL-SD-ICF-BURN) are dominated by late detonations in the containment (approximately 99 percent of the release frequency). Items MU-7.2 through MU-7.4 considered several ERPRA-BURN modeling parameter sensitivities that govern the probabilities used in the SAPHIRE model related to combustion:

- Assuming that MCCI could be terminated or significantly slowed through the addition of water to the reactor cavity after vessel breach (e.g., through spillover of injected spray water from the lower containment), such that cumulative MCCI gas production is reduced by 75 percent, reduces the detonation probability for CHR scenarios in relative terms by between 40 and 67 percent. However, the modeling assumptions made in these sensitivity analyses suggest these estimates may overstate the actual expected impacts (see MU-7.2 for additional discussion).

Table 2-52 ERPRA-BURN Sensitivity Analysis for Late Time Frame Ignition Source Frequency

Derived Failure Probability	Base Case	Ignition Source Frequency × 2	Ignition Source Frequency × 3
Cumulative ignition source probability	0.99	0.9999	0.999999
Mean time between ignition source events (hour)	29.4	14.7	9.8
Mean total number of ignition source events	4.6	6.9	13.8
Probability of containment failure given deflagration (with prior combustion)	0.066	0.018	0.004
Probability of containment failure given deflagration (without prior combustion)	0.20	0.055	0.015
Probability of detonation (with prior combustion)	0.79	0.63	0.45
Probability of detonation (without prior combustion)	0.72	0.71	0.56

- Raising the minimum threshold of combustible gas concentration for detonations to take place from 15 percent to 20 or 25 percent resulted in the detonation probability dropping from 0.79 to 0.54 or 0.32, respectively.
- Doubling or tripling the assumed frequency of late ignition source events in the containment resulted in a reduction of the detonation probability from about 0.79 to 0.63 or 0.45, respectively.

In each of these sensitivity cases, the results showed that the detonation probability significantly reduced, ranging from about 20 percent to 67 percent of the base case modeled detonation probability. However, some of these sensitivity cases may not necessarily represent a realistic reduction in the detonation probability. For example, as stated above, the impacts of MCCI termination may be overstated. Also, there is no strong basis for justifying one particular sensitivity value over another. Therefore, a sensitivity case is developed to assess the impact of alternate probabilities for detonation events where the detonation probabilities are reduced by 50 percent. To assess these sensitivity cases, the SAPHIRE model was re-solved with the basic events for the detonation probabilities set to half of their nominal values. The specific basic event adjustments were:

- The probability for basic event 1-L2-BE-H2DET-L-CHR-SD was reduced from 0.79 to 0.395. This represents the probability of detonation for loss of decay heat removal transients with containment heat removal available.
- The probability for basic event 1-L2-BE-H2DET-L-LOIPB-SD was reduced from 0.61 to 0.305. This represents the probability of detonation for loss of inventory event sequences.

The results of the detonation probability sensitivity case are shown in Table 2-53. This sensitivity case shows a significant reduction in the release category frequency for combustion-induced containment failure (1-REL-SD-ICF-BURN). The sensitivity case also results in increased release category frequencies for basemat melt-through (1-REL-SD-BMT) and intact containment (1-REL-SD-NOCF). The changes in release category frequencies also result in reductions in LRF and the overall release frequency, which excludes the intact containment release category. The estimation of LERF is unchanged by this sensitivity case.

Table 2-53 Detonation Probability Sensitivity Release Frequencies

	Base Case	MU-7.5 Sensitivity	
Release Category	Frequency (per year)	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	1.9E-07	52.8%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.1E-06	0.0%
1-REL-SD-CIF	1.7E-06	1.7E-06	0.0%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.0%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.3E-07	0.0%
1-REL-SD-ECF	5.1E-10	5.1E-10	0.0%
1-REL-SD-ICF-BURN	1.7E-06	9.8E-07	-42.3%
1-REL-SD-ISLOCA	1.3E-07	1.3E-07	0.0%
1-REL-SD-LCF	5.5E-08	5.5E-08	0.0%
1-REL-SD-NOCF	6.1E-06	6.7E-06	10.3%
Release Surrogate Measures			
LERF	5.3E-07	5.3E-07	0.0%
LRF	6.1E-06	5.4E-06	-11.9%
Overall release	6.3E-06	5.6E-06	-10.5%

2.6.3.8 Long-term Containment Pressurization Model Uncertainties

Table 2-54 identifies issues contributing to model uncertainty in the long-term containment pressurization modeling.

Table 2-54 Identified Uncertainties Related to Long-term Containment Pressurization and Failure

Item	Description	Other comments
Primary containment structural vulnerabilities, failure modes, and location	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 and temperature-induced seal degradation).	The containment failure modes were identified and characterized as described in Section 2.2. A probable containment failure mode and location were selected and modeled in the accident progression calculations. Alternate containment failure pathway modeling is assessed in MU-13.1 below.
“Normal” containment leakage	Normal (i.e., design-basis) containment leakage can vary between zero and 0.2wt%/day (larger leakage would constitute an isolation failure). The maximum allowable design-basis leakage is used in the base MELCOR model (0.2wt%/day), whereas actual values will vary (e.g., a 2010 integrated leak rate test at the reference plant estimated a leakage rate of 0.07wt%/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 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 modeling uncertainty in MU-13.1 and also discussed in the reactor at-power Level 2 PRA Appendix D entry entitled, “Tendon Gallery Release Pathway” (NRC, 2022a).	The intent of the identified uncertainty is that a variety of approaches can be used, and the outcomes are important. The draft Surry SOARCA uncertainty analysis (SNL, 2016) identifies fragility curve, “design” leakage (before break) path length, and containment convection heat transfer multiplier as uncertain inputs (Section 4.1.4 of SNL, 2016).
Quasi-static failure threshold methods and correlation between failure pressure and leak rate	The state-of-practice is (debatably) not mature for defining leak-before-break containment response for concrete containments.	This is discussed further in the at-power Level 2 PRA Appendix D entry entitled, “Containment Leakage, Effective Sizes Under Normal and Accident Conditions” (NRC, 2022a). The reader is also referred to the analyses done in the draft Surry SOARCA uncertainty analysis (SNL, 2016).

Table 2-54 Identified Uncertainties Related to Long-term Containment Pressurization and Failure (cont.)

Item	Description	Other comments
Heatup of containment structure and equipment	The modeling of heatup of the containment structure, equipment located within containment, and passive heat transfer to the environment or structures outside containment can influence the timing and likelihood of containment pressurization leading to failure.	The impacts of heatup of containment structures and heat losses across the containment boundary are considered for conditions with large amounts of water available to boil in containment (e.g., refueling cavity flooded). The MELCOR analysis in case SD09 assesses these conditions. The uncertainties with modeling heatup of containment structures are discussed further in MU-8.2 below.
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.	This is discussed in Appendix CC 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 ($m = K \cdot d^3$). A 2009 NEA state-of-the-art report on nuclear aerosols (NEA, 2009) states that experimental studies have demonstrated decontamination factors in the range of 10 to 100, and also 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 IRSN and CEA.
Containment leakage after emergency closure	Emergency closure of containment with four bolts is treated the same as a fully intact containment in terms of the leakage response as a function of pressure (both prior to and after overpressure failure), whereas the closure situation could lead to larger leakage. An analogous concern exists for emergency closure of containment penetrations.	Some sites have reported that using only the minimum number of bolts was not sufficient for hatch sealing (see Sections 6.9.2 and 6.9.3 of (NRC, 1993)). For the reference plant, once the hatch is in place, the number of bolts used is not viewed to be important because the pressure-sealing hatch design would tend to improve the seal as pressure increased.

Alternate treatment(s)

MU-8.1 – Sensitivity analysis on containment leakage rate

The MELCOR model used for deterministic severe accident analysis assumes a nominal leak rate for an otherwise intact containment of 0.2 percent air mass per day. In the sensitivity analysis for this issue in the reactor-at-power, internal event and flood PRA (MU-8.2 in Appendix C of (NRC, 2022a)), the impact of uncertainty in the actual containment leakage rate was investigated by means of a MELCOR sensitivity study in which this rate was adjusted to

0.07 percent air mass per day. The effects on accident progression were found to be very small and were attributed mainly to a slightly different predicted timing of accumulator injection and vessel breach rather than a significant long-term impact on containment pressure. The effects of reduced leakage on the radiological source term to the environment were likewise relatively small and within the expected range of uncertainty for MELCOR source terms.

For accidents at shutdown and with an initially intact containment, conditions are not different in any essential respect from the case analyzed for at-power. In fact, since accumulators are isolated for the majority of accidents at shutdown, effects resulting from slight changes in accumulator injection timing would be absent, and the impact would be expected to be even smaller. Therefore, it is qualitatively concluded that the results of the reactor at-shutdown PRA would be relatively insensitive to uncertainty in the assumed containment leakage rate.

MU-8.2 – Sensitivity analysis on heatup of containment structures and equipment

The modeling of heatup of the containment structure, equipment located within containment, and passive heat transfer to the environment or structures outside containment is addressed in MELCOR analyses that result in long-term pressurization and challenges to the containment failure limits. The MELCOR case SD09 demonstrates this issue and is used as the representative case for the release category resulting in containment failure prior to core damage, 1-REL-SD-CF-BEFORE-CD. The case SD09 scenario is a station blackout sequence during POS 8E with the refueling cavity flooded. All potentially opened containment penetrations are assumed to be successfully isolated during the scenario. The containment pressure gradually increases due to evaporation and/or boiling of the refueling cavity water. Containment failure is predicted at approximately 118 hours after event initiation. Core damage is predicted at approximately 125 hours after event initiation.

The MELCOR model accounts for the containment initial conditions, transient heatup of containment heat sinks, condensation, and drain-off from structure surfaces. As with all representative MELCOR cases, the model and results have been reviewed to verify predicted containment failure. The results were also cross-checked by means of hand calculations to verify aspects of the heat losses across the containment boundary and water condensation. Up until the predicted time of containment overpressure, decay heat in the core is still significant (7.9 MW). Around the time prior to containment failure (with containment pressure at 8.0 bar-abs and containment temperature at about 162 °C [435K]), steady-state heat losses across the containment boundary to the environment would be in the range of 0.5 to 0.8 MW, accounting for natural free convection at inner and outer surfaces. A simple estimate of heat conduction through the containment wall can be used to estimate an upper bound of the steady-state heat losses. Assuming a containment surface area of 8500 m², containment wall thickness of 1.14 m, and 0.93 W/mK, the heat loss is limited to no more than about 0.9 MW by steady-state conduction across the thick concrete wall. Heat transfer to the ground and auxiliary building would account for smaller amounts. Therefore, steady-state passive heat loss cannot remove more than a small fraction of decay heat from the system, and the containment will continue to heat up and pressurize.

Furthermore, hand calculations were used to show that heatup of the thicker containment concrete structures to 162 °C would take up a bit more than half of the cumulative core decay heat through the first 100 hours. The smaller structures and containment steel would account for some additional amount. Therefore, by this late point in the accident progression, everything in containment would be in approximate quasi-static equilibrium, and continued heatup and

pressurization would be mainly dictated by steady-state heat losses, as bounded above. This suggests that the overall time scale for containment failure predicted by MELCOR is accurate.

As a point of reference, overpressure for containments of this type is typically expected within roughly 24 to 48 hours for accidents beginning at full reactor power, with most of the pressurization occurring late with largely dry MCCI gas production. By comparison, the level of decay heat in the POS 8E scenario is approximately half of that for a full power scenario 24 to 48 hours after reactor shutdown. The MELCOR predicted containment overpressure failure at 118 hours seems reasonable, especially in view of the very large amount of water available to boil (the main factor contributing to the containment failure). The predicted containment pressure is shown in Figure 2-25. The scenario models water draining from containment heat structures returning to the lower containment rather than to the refueling cavity; however, where the water drains in this scenario does not really affect the dynamics of the quasi-static containment pressurization. If more water were credited to drain back to the refueling cavity, it might delay core uncover, but only slightly, since the recirculated water would be quite warm. Pushing core uncover slightly later would only reinforce the result that containment overpressure occurs prior to core damage. All these considerations have some uncertainty, but the results have been reviewed and verified by considering alternatives and confirmatory calculations.

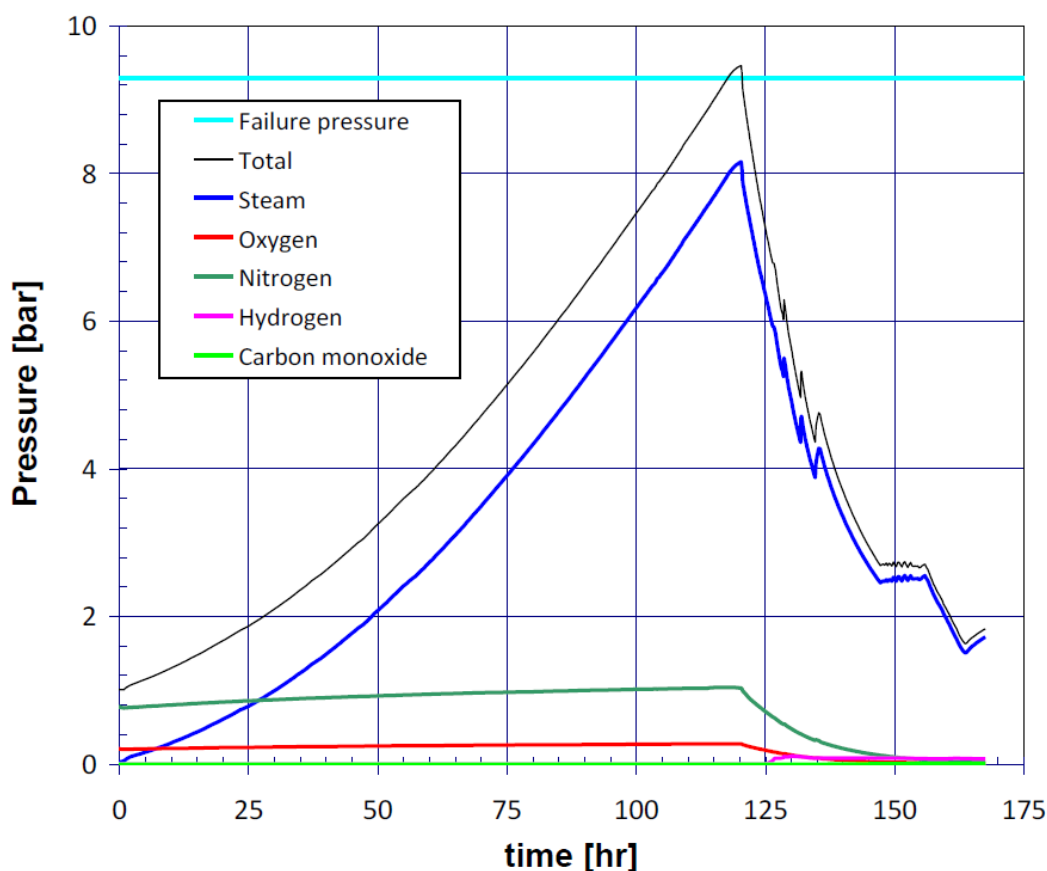


Figure 2-25 MELCOR Predicted Containment Pressure for Long-Term SBO with Refueling Cavity Flooded

MU-8.3 – Sensitivity analysis on effect of decontamination factors for the containment overpressurization failure pathway

This sensitivity considers the impacts of late containment failure due to overpressurization. In the sensitivity analysis for this issue in the reactor-at-power, internal event and flood PRA (MU-8.3 in Appendix C of (NRC, 2022a)), the impact of uncertainty in the decontamination factor due to turbulent deposition is explored. As the fission products pass through the tendon gallery, the larger particles are removed and deposit on the floor. The majority 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. This sensitivity demonstrates that the decontamination factor brought on by turbulent deposition in the containment overpressurization leakage path to the tendon gallery is not significant. Deposition is therefore dominated by the gravitational settling in the tendon gallery which is already modeled in the calculations. A similar impact is expected for the shutdown model. However, the impact on the overall results is expected to have a smaller impact due to the late containment failure category having a lower relative frequency for the shutdown model. Therefore, it is qualitatively concluded that the results of the shutdown PRA would be relatively insensitive to uncertainty in the decontamination factors related to containment failure effects.

2.6.3.9 *Ex-Vessel Coolability and MCCI Model Uncertainties*

Table 2-55 identifies issues contributing to model uncertainty in the ex-vessel debris coolability and MCCI modeling.

No alternative treatments are proposed.

2.6.3.10 *ISLOCA Model Uncertainties*

Table 2-56 identifies issues contributing to model uncertainty in the ISLOCA modeling.

Alternative treatment(s)

MU-10.1A/B – Sensitivity analysis on likelihood of ISLOCA containment bypass

The estimate of the frequency and likelihood of an ISLOCA occurring during shutdown is much different than during at-power operations. For the at-power PRA, an expert elicitation process was performed to inform aspects of the ISLOCA modeling. One of the key uncertainties is the likelihood of failure of containment isolation valves. However, many of the normally closed isolation valves are opened and in operation during shutdown. The estimates for the frequencies and probabilities related to ISLOCA modeling are documented in the shutdown Level 1 PRA (NRC, 2025). The different categories of loss of inventory events are considered for possible losses that could occur outside of the containment boundary. Two probabilities are developed to estimate the likelihood that the loss of inventory occurs outside containment, one for conditions with the RCS pressurized, and one for all other POS conditions. The probability of a loss outside containment with the RCS pressurized is based on a review of operating experience and incorporates the observed number of inventory losses occurring outside containment for shutdown operations with RCS pressurized. For the other POS conditions, there is very limited operating experience and no recent events, so the probability is based on

Table 2-55 Identified Uncertainties Related to Ex-Vessel Coolability and MCCI

Item	Description	Other comments
Ex-vessel debris bed coolability	This issue can affect when releases trail off and whether the accident can be terminated prior to basemat melt-through.	This issue is discussed in the reactor at-power analysis of severe accident management actions in Appendix D of (NRC, 2022a). In the SOARCA Peach Bottom uncertainty analysis (NRC, 2016), the ex-vessel uncertain input parameters include debris overflow head as a function of debris temperature for solid and liquid (DHEADSOL and DHEADLIQ, respectively).
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.	The reactor at-power Level 2 PRA contains discussions on the treatment of this issue. This includes an entry on the ex-vessel molybdenum behavior, which is discussed in Appendix D entry “Cesium Molybdate Treatment (Ex-vessel)” (NRC, 2022a). The possibility that concrete erosion undermines RPV support is discussed briefly in the reactor at-power Level 2 PRA Appendix D entry entitled, “Reactor Cavity Rebar and Under-fill” (NRC, 2022a). The draft Surry SOARCA uncertainty analysis conducted separate sensitivity analyses for concrete aggregate material, and the amount of rebar in the concrete (Sections 6.1.8.2 and 6.1.8.3 of (SNL, 2016)).

the estimated fraction of RHR system piping located outside containment. The approaches for both probability estimates introduce modeling uncertainty. Sensitivity analyses were developed to consider upper-bound and lower-bound probabilities based on the 95th and 5th percentiles of the probability distributions. The probability values, which are documented in the shutdown Level 1 PRA report (NRC, 2025), are shown in Table 2-57.¹⁷

The release category frequency results for the sensitivity analyses are shown in Table 2-58. As expected, the primary impact of the sensitivity analysis is the increase or decrease of the frequency for release category 1-REL-SD-ISLOCA. The frequency is doubled for the upper-bound case and reduced by about 85 percent for the lower-bound case. The other release category frequencies are only marginally impacted with the largest changes to 1-REL-SD-ICF-BURN and 1-REL-SD-ECF being only a few percent. LERF is also impacted by the sensitivity

¹⁷ Note that the probability values in Table 2-57 are combined with the loss of inventory initiating event frequencies to give the frequency of ISLOCA occurrence. The loss of inventory initiating event frequency for RCS pressurized POSs has the highest frequency (i.e., 6.1×10^{-1} per shutdown-year in (NRC, 2025)) compared to all other loss of inventory categories modeled in the shutdown Level 1 PRA, which have frequencies ranging from 1×10^{-5} to 3×10^{-3} per shutdown-year. Accordingly, the RCS pressurized condition has the largest contribution to the shutdown ISLOCA frequency and is the dominant contributor to the 1-REL-SD-ISLOCA release frequency.

Table 2-56 Identified Uncertainties Related to 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).	The representative ISLOCA sequence considers an RHR system failure with a 3-inch diameter failure size resulting in containment bypass to the auxiliary building. Other break sizes and locations can influence the event progression and fission product retentions and releases. The at-power Level 2 PRA considers variations of RHR bypasses to the auxiliary building with cases examining 2-inch, 6-inch, and 8-inch break sizes (NRC, 2022a). The assumptions about break size were found to have significant impacts on the auxiliary building failure and fission product releases. Also, an expert elicitation under the L3PRA project was conducted which provides some related information on ISLOCA break location (NRC, 2022b).
Turbulent deposition in the connected piping	In the Surry SOARCA study (NRC, 2013b), turbulent deposition in RHR piping was predicted to provide significant scrubbing, largely due to back-side flooding of the RHR pipe preventing re-vaporization.	This issue is discussed in the ISLOCA analysis for the at-power Level 2 PRA in Appendix B, Section 5 of (NRC, 2022a). The shutdown Level 2 PRA uses the same approach as the at-power Level 2 PRA with respect to turbulent deposition in connected piping.
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 analyzed in a sensitivity study for the at-power Level 2 PRA in Appendix B, Section 5 of (NRC, 2022a). The shutdown ISLOCA scenario models auxiliary building failure due to deflagration, which limits the impact of the modeling of building filters.
Auxiliary building failure during ISLOCAs	The representative ISLOCA scenario assumes bypass to the auxiliary building via the RHR system. Hydrogen generated in the core is transported to the auxiliary building where a deflagration occurs causing building failure. The modeling of the building response and robustness can significantly influence the environmental releases associated with ISLOCAs.	The at-power Level 2 PRA considers ISLOCA cases for intact and failed auxiliary building in Appendix B, Section 5 of (NRC, 2022a). Building failure in the at-power model is caused by blowdown of RCS pressure and depends on the ISLOCA break size. The at-power analysis also predicts hydrogen combustion events that would be capable of failing the auxiliary building.

Table 2-56 Identified Uncertainties Related to ISLOCA Modeling (cont.)

Item	Description	Other comments
Likelihood of ISLOCA occurrence	The likelihood and frequency of ISLOCAs or losses of inventory occurring outside containment are sources of model uncertainty that influence the ISLOCA release category frequency.	The estimates for likelihood of shutdown ISLOCA are documented in the shutdown Level 1 PRA (NRC, 2025). The estimates are informed by a review of shutdown loss of inventory operating experience and estimates of the likelihood of losses occurring outside (and bypassing) containment.

Table 2-57 Probability of Inventory Loss Occurring Outside Containment

Basic Event Name and Description	Base Case Probability	MU-10.1A Upper-Bound Probability	MU-10.1B Lower-Bound Probability
1-L2-BE-CBYP-GEN Probability of loss of inventory outside containment – general case	2.9E-01	6.2E-01	1.1E-01
1-L2-BE-CBYP-OP Probability of loss of inventory outside containment – RCS pressurized	1.7E-01	3.4E-01	2.3E-02

Table 2-58 ISLOCA Probability Sensitivity Release Frequencies

Release Category	Base Case	MU-10.1A Sensitivity		MU-10.1B Sensitivity	
	Frequency (per year)	Frequency (per year)	% Change	Frequency (per year)	% Change
1-REL-SD-BMT	1.2E-07	1.2E-07	-1.4%	1.2E-07	1.1%
1-REL-SD-CF-BEFORE-CD	2.1E-06	2.1E-06	0.0%	2.1E-06	0.0%
1-REL-SD-CIF	1.7E-06	1.7E-06	-0.1%	1.7E-06	0.1%
1-REL-SD-COPEN-HEADOFF	7.8E-08	7.8E-08	0.0%	7.8E-08	0.0%
1-REL-SD-COPEN-RCSINTACT	2.7E-07	2.7E-07	0.0%	2.7E-07	0.0%
1-REL-SD-COPEN-RCSVENT	1.3E-07	1.3E-07	0.0%	1.3E-07	0.0%
1-REL-SD-ECF	5.1E-10	5.0E-10	-2.1%	5.2E-10	1.7%
1-REL-SD-ICF-BURN	1.7E-06	1.6E-06	-3.4%	1.8E-06	2.8%
1-REL-SD-ISLOCA	1.3E-07	2.7E-07	103.9%	1.9E-08	-85.5%
1-REL-SD-LCF	5.5E-08	5.5E-08	0.0%	5.5E-08	0.0%
1-REL-SD-NOCF	6.1E-06	6.0E-06	-1.2%	6.2E-06	1.0%
Release Surrogate Measures					
LERF	5.3E-07	6.7E-07	26.0%	4.2E-07	-21.3%
LRF	6.1E-06	6.2E-06	1.3%	6.0E-06	-1.0%
Overall release	6.3E-06	6.3E-06	1.2%	6.2E-06	-1.0%

cases, since the ISLOCA frequency is a significant contributor to LERF. LERF is increased by 26 percent for the upper-bound case and reduced by 21 percent for the lower-bound case. These sensitivity cases demonstrate the significant sensitivity of the ISLOCA frequency to the modeling assumptions related to occurrences of inventory losses outside of containment. Several other modeling uncertainties, as identified in Table 2-56 above, can influence the ISLOCA frequency and magnitude of releases. As discussed above, sensitivity analyses for the at-power Level 2 PRA have demonstrated that assumptions about the size and location of the ISLOCA break can significantly influence the environmental releases. As such, aspects of the ISLOCA modeling for shutdown are proposed as candidates for future work and identified in Appendix A.

2.6.3.11 SGTR and induced SGTR Model Uncertainties

Table 2-59 identifies issues contributing to model uncertainty in the SGTR modeling.

No alternative treatments are proposed.

For the reactor at-power Level 2 PRA, sensitivity calculations related to SGTR modeling are documented in Appendix C of (NRC, 2022a). A MELCOR sensitivity calculation was performed assuming earlier tube rupture (MU-11.1 in Appendix C, Section 4.11.3 of (NRC, 2022a)). Another MELCOR sensitivity study was performed in which model changes were made in an attempt to capture the secondary-side decontamination factor (MU-11.2 in Appendix C, Section 4.11.4 of (NRC, 2022a)). Although the effects of these sensitivity studies were significant in the at-power study, induced SGTR is of such low importance in the shutdown study (zero frequency at the current truncation cutoff) that additional evaluation of the uncertainty is not warranted.

Table 2-59 Identified Uncertainties Related to SGTR Modeling

Item	Description	Other comments
Binning of consequential SGTR into a single failure probability / break size	Consequential 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 consequential SGTR calculations performed for this project using the approach described in (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 uncertainty analysis (SNL, 2016) included a separate assessment on SGTR and included the effects of multiple tubes rupturing along with other influential uncertain inputs for SGTR (Sections 6.1.5 and 6.1.8 of (SNL, 2016)).
Time between consequential SGTR and subsequent hot leg creep rupture	This timing is uncertain and has a fairly direct impact on predicted source term. For the at-power model, the case 3A3, described in Appendix B of (NRC, 2022a), assumes a 50-minute time difference, a value which may overestimate the time delay until hot leg creep rupture.	Case 3A3 in the accident analysis for reactor-at-power is believed to provide a somewhat exaggerated response, while Case 3A2 provides the logical extreme of no subsequent hot leg failure. The results of the draft Surry SOARCA uncertainty analysis (SNL, 2016) showed the calculated time difference between induced SGTR, and hot leg rupture ranged from 0.12–144 minutes, with a mean of 28.1 minutes, and the time of hot leg rupture had a large impact on total releases (Sections 6.1.5.6 and 6.1.5.7 of (SNL, 2016)). While this uncertainty can impact the induced SGTR modeling, induced SGTR is of very low importance for the shutdown PRA. Therefore, no sensitivity analyses were performed to address this issue for the shutdown PRA.
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 uncertainty analysis (Section 4.1.3.2 in (SNL, 2016)).

2.6.3.12 Containment Isolation Failure Model Uncertainties

Table 2-60 identifies issues contributing to model uncertainty in the containment isolation failure modeling.

Table 2-60 Identified Uncertainties Related to Containment Isolation Failure

Item	Description	Other comments
Characterization of the containment isolation failure location and size	The containment isolation failure is assumed to be a 2-inch-equivalent diameter break leading into the environment. A smaller or larger leakage area, a different location (e.g., leading to the auxiliary building), or filtering the release pathway would all influence the source term.	During shutdown POSs, many possible containment penetrations are likely to be open. Possible containment isolation failure pathways are assessed in the uncertainty analysis below.

Alternate treatment(s)

MU-12.1 – Sensitivity analysis on characterization of containment isolation failure location and size

The at-power Level 2 PRA (NRC, 2022a) assumes a containment isolation failure equivalent to a 2-inch diameter opening and attributed to a pre-existing containment failure or maintenance error. A sensitivity calculation was performed increasing the failure size to 4-inch diameter (MU-12.1 in Appendix C of (NRC, 2022a)). The larger failure opening size results in larger environmental releases. The impact of containment isolation failure size is potentially greater for the shutdown PRA due to the higher relative frequency contribution from containment isolation failure and the potential for many possible containment penetrations that could be open during shutdown. Sensitivity analyses were performed for containment isolation failure for the reactor at-shutdown considering alternate failure locations and sizes.

For the at-power model, the containment isolation failure models a pre-existing containment failure and failures of several normally closed containment isolation valves. The pre-existing containment failure accounts for over 90 percent of the containment failure probability. For the shutdown model, four different variations of the containment isolation fault tree are developed to represent different POS conditions: (1) conditions similar to at-power when the containment isolation system is active, (2) mid-loop operations when time available for closing penetrations is short, (3) POSs when the containment purge system is operating, and (4) POSs when preparing to move the fuel and automatic containment ventilation isolation is active. These conditions, along with containment isolation valves that are open during shutdown, make for a different mix of failures that contribute to the overall containment isolation failure probability when comparing the at-power results to the latter three fault tree variations.

For shutdown conditions, the largest contributor to containment isolation failure is operator failure to close the many small penetrations (typically 1-inch or less) that are identified in the reference plant operating procedures and containment penetration verification check list. Operator failure to close small penetrations accounts for over 50 percent of the failure probability in each of the fault tree variations. Failures of isolation valves to close collectively account for over 20 percent of the failure probability. Pre-existing containment failures also

account for about 20 percent of the failure probability. Failure to isolate the containment purge system accounts for less than 1 percent up to 4 percent depending on the POS conditions and initial purge system operating status.

These results informed the selection of representative and sensitivity cases for containment isolation failure. The base case (SD03) assumes a 2-inch diameter isolation failure at about 1 foot above the containment floor. This is consistent with the at-power containment isolation size. This was selected as the base case because it was thought to bound significant contributors to isolation failure (i.e., operator failure to close small penetration(s) and pre-existing failures). The purge system isolation failures, while not dominant contributors, could result in much larger sized failure openings. Therefore, the purge system isolation failure was considered for a sensitivity case. It should be noted that the sensitivity cases were developed for a different event scenario (i.e., a loss of RHR transient during POS 6). While this is a different scenario than the base case, the variations in size and location of the isolation failure still provide insights to the potential impacts on releases. The sensitivity cases are summarized as follows:

- Base Case, Case SD03 – 2-inch failure near containment floor, POS 8E SBO scenario
- MU-12.1a, Case SD01.10 – 14-inch filtered failure of purge system piping, POS 6 loss of RHR
- MU-12.1b, Case SD01.11 – 14-inch unfiltered failure of purge system piping, POS 6 loss of RHR
- MU-12.1c, Case SD01.12 – 2-inch failure near containment floor, POS 6 loss of RHR
- MU-12.1d, Case SD01.13 – 2-inch failure near equipment hatch, POS 6 loss of RHR

Early revisions of the containment isolation fault tree modeling suggested that the loss of RHR during POS 6 sequences could significantly contribute to the containment isolation failure release category. This motivated several sensitivity cases with variations of those sequences. Ultimately, the release category frequency was dominated by SBO during POS 8E (contributing over 60 percent of the frequency). Loss of RHR during POS 6 sequences contribute about 5 percent to the release category frequency in the final model. Case SD03 was selected as the base case representative scenario for the containment isolation failure release category (1-REL-SD-CIF) based on these frequency results. While the sensitivity cases are based on a different type of scenario, they still provide insights about the impacts of different containment isolation failure sizes and locations.

Case SD01.10 considers the potential releases related to a failure to isolate the containment purge system. The purge system includes 14-inch and 24-inch diameter piping. Based on the reference plant procedures, the 14-inch piping is expected to be in operation for portions of an outage. Therefore, a 14-inch diameter containment isolation failure is modeled for this sensitivity case. Case SD01.10 also considers the effects of the filtration unit on the purge system exhaust. The results, shown in Table 2-61, show potential for the filters to significantly reduce releases to the environment. However, the analysis does not model failure mechanisms for the filters, which make the predicted releases overly optimistic and unrealistic. The filters would be challenged by deflagration events in containment resulting in pressure spikes that could fail the filters, and the predicted mass deposited in the filters would likely exceed the capacity and

cause clogging. While case SD01.10 provides an optimistic representation of filters, case SD01.11 considers the same scenario without filters. The large differences in the results for these two cases demonstrate that assumptions about filtering releases, and effectiveness and survivability of filters, can significantly influence the predicted releases to the environment.

Additional sensitivity cases were developed for small isolation failures (i.e., 2-inch diameter) because the fault tree modeling shows the small penetration failures dominate the containment isolation failure probability. Case SD01.12 models a 2-inch diameter failure at low elevation near the containment wall and basemat junction. The same failure size and location is modeled in the base case SD03. There are many differences in the scenario progression (e.g., timing of core damage is significantly different) between cases SD01.12 and SD03, which makes it difficult to draw conclusions from a direct comparison of the results. Case SD01.13 models a 2-inch diameter failure at a higher elevation near the equipment hatch. One notable difference between case SD01.13 and SD01.12 is the slower rise of containment pressure for SD01.13, which is attributed to the isolation failure being at a higher elevation and never blocked by water in the containment. The slower rise of containment pressure in case SD01.13 is interpreted as a contributing factor to the lower fractional releases to the environment for most of the fission product classes compared to case SD01.12. Though, some releases are higher for case SD01.12 (e.g., Tellurium and Ruthenium). Table 2-61 summarizes the results of the containment isolation failure sensitivity cases.

Table 2-61 Sensitivity Analysis of Containment Isolation Failure Release Size and Location

Fission Product Class	Case SD03 Base Case	Case SD01.10 14-inch Filtered	Case SD01.11 14-inch Unfiltered	Case SD01.12 2-inch Low Elevation	Case SD01.13 2-inch High Elevation
Xe	6.7E-1	1.0	1.0	9.8E-1	9.9E-1
Cs	6.7E-2	3.6E-4	8.9E-2	1.6E-2	8.9E-3
I	5.9E-2	6.0E-5	3.3E-1	7.9E-2	1.0E-2
Te	5.3E-2	3.1E-4	8.7E-2	7.3E-3	1.1E-2
Ba	1.2E-3	1.5E-5	9.9E-3	6.5E-4	4.0E-4
Ru	1.3E-3	7.2E-6	2.4E-3	1.3E-5	1.4E-4
Mo	1.4E-2	2.6E-4	1.9E-1	5.0E-2	4.8E-2
Ce	3.1E-6	8.8E-7	1.5E-3	2.5E-5	3.5E-6
La	3.2E-7	1.9E-8	1.1E-3	1.7E-6	1.5E-6

2.6.3.13 Release Pathway Model Uncertainties

Table 2-62 identifies issues contributing to model uncertainty in the modeling of radiological release pathways to the environment.

Table 2-62 Identified Uncertainties Related to Release Pathway Modeling

Item	Description	Other comments
Containment failure release pathway modeling	The containment failure mode, location, connected structures, and other accident progression characteristics can significantly influence the releases to the environment.	The containment failure pathway for containment failures due to long-term overpressurization and energetic combustion events is broadly consistent with the modeling approach used in the at-power Level 2 PRA (NRC, 2022a). Several shutdown-specific sensitivity cases were performed to assess the impacts of aspects of the release pathway modeling. See discussion under MU-13.1 below.
Failure to close containment pathway	The shutdown model assumes the release pathway given failure of emergency containment closure is through the equipment hatch. Other containment opening pathways are possible.	The failure to close the open personnel airlock is an alternative release pathway to consider.
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. Related to this, the at-power Level 2 PRA Appendix D entry, "Aqueous Releases, Use of MELCOR in predicting," may also be of interest (NRC, 2022a).
Completeness of release pathway modeling	The MELCOR model treats most potential pathways that can lead to environmental radiological release. Two exceptions that are not modeled are: • <i>Steam jet air ejectors</i> – If a SGTR or induced SGTR leads to significant contamination of the steam lines, the main steam isolation valve on the ruptured SG is open during and following core damage, the condenser continues to be in operation, and the radiological material is not effectively scrubbed by the water and steam in the SG / steam line / condenser, then radiological material could enter the environment through the steam jet air ejectors.	Since the release frequency for induced SGTRs is very low, and since the additional factors, as described in the unmodeled release pathways, would have to occur, this is not viewed as a significant modeling limitation. Also, the use of TDAFW is much less relevant for shutdown conditions than for at-power conditions.

Table 2 62 Identified Uncertainties Related to Release Pathway Modeling (cont.)

Item	Description	Other comments
	• <i>TDAFW turbine discharge</i> – If SGTR or induced SGTR leads to significant contamination of the SGs, and TDAFW supply from the ruptured SG is placed in operation during or following core damage, then radiological material could be released to the environment through the turbine exhaust discharge point.	
Source term attenuation in structures outside the primary containment	This issue involves source term characterization for leakage pathways through adjoining structures, including: • penetration leakage into the auxiliary building • leakage through the tendon gallery • ISLOCA into the auxiliary building • modeling of auxiliary building filtration, where applicable This issue affects the amount of radiological material retained by adjoining structures (versus how much is released to the environment).	For the reference plant, the current understanding of likely failure locations includes (1) the tendon gallery near the containment wall and basemat junction and (2) near the equipment hatch. For plant designs where penetration failures are more prominent, then failure locations involving the control building, and possibly the fuel handling building, would also likely need to be considered. The tendon gallery is discussed in more detail in the at-power Level 2 PRA Appendix D entry entitled, “Tendon Gallery Release Pathway” (NRC, 2022a). The ISLOCA issues are discussed in Section 2.6.3.10 of this report, in the at-power Level 2 PRA Appendix D entry entitled, “Interfacing Systems LOCA, Modeling of Piping Effects” (NRC, 2022a).
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-SD-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-SD-ECF and SGTR have very small contributions to overall release frequency, while ISLOCA sequences are still subject to large releases even if IVR occurs (depending on precisely when during the in-vessel melt progression it occurs).

Alternate treatment(s)

MU-13.1 – Sensitivity analyses on containment failure release pathway modeling

The shutdown Level 2 PRA assumes a containment failure pathway to the tendon gallery, which in turn connects to both the environment and the auxiliary building. The same containment pathway location is applied to long-term overpressurization failures and combustion-induced failures, but with different accident progression attributes. Several modeling choices and accident conditions can influence the release pathway modeling and ultimate environmental releases. Some of the key attributes include:

- location and elevation of containment failure
- size of the containment failure opening
- presence of water at the failure location and timing of boiloff of water
- relative pressures and flows of liquid and gases between connected structures
- timing and inventory available for containment spray

A series of sensitivity studies with associated MELCOR calculations were developed to explore these release pathway modeling attributes and are summarized below.

The reactor at-power Level 2 PRA developed a MELCOR sensitivity case (MU-8.1 in (NRC, 2022a)) to evaluate the impact of containment failure location. The base case assumes the containment failure pathway to be in the tendon gallery, which in turn connects to both the environment and the auxiliary building. The base case MELCOR analysis shows flow may preferentially go directly to the environment. A sensitivity case was performed where the overpressure failure mode/location is modeled to go through the tendon gallery and then completely to the auxiliary building. The impact of the auxiliary building in minimizing the environmental release was significant. The connection to the auxiliary building may have a similar impact for the shutdown model, but other differences in the accident progression make a direct comparison difficult.

The base case for the shutdown model long-term overpressurization (case SD01) models the containment failure release pathway to the tendon gallery with connections to both the auxiliary building and environment, similar to the at-power scenario. However, the environmental releases are much lower for the shutdown case. This is primarily attributed to the very late containment failure at 167.6 hours, just prior to the accident scenario termination time of 168 hours, and the containment failure location remaining submerged at the time of rupture.

An alternate failure location was evaluated to assess the impacts of the break being submerged in water and the connection with the auxiliary building. The alternate failure location was chosen to be near the containment equipment hatch, which the containment capacity analysis identified as another likely failure location. This failure location was at a higher elevation, so it was not submerged, and the pathway connected the containment directly to the environment. Table 2-63 shows the chemical class release fractions to the environment for the base case (SD01) and the alternate failure location case (SD01.5). Without submergence and connection to the auxiliary building, the alternate failure location could intuitively seem to lead to larger environmental releases. However, as can be seen in the table, the environmental releases are not significantly different between the two cases. The principal difference between these cases that limits the releases in case SD01.5 is related to the smaller containment failure size opening. Both cases use the same modeling approach for the size of the containment failure based on containment

pressure. The failure occurs when containment reaches 120 psig with a failure opening area of 0.1 square-foot. The opening size becomes progressively larger with increasing containment pressure until a maximum value of 1 square foot is reached at 130 psig. (This is described in Appendix D of the at-power Level 2 PRA (NRC, 2022a).) The failure in case SD01.5 is not submerged and effectively relieves containment pressure, which rises only slightly above the 120 psig threshold. The opening size never reaches the maximum size realized in case SD01. In both cases the releases are limited by the containment failure occurring so late in the accident progression.

Table 2-63 Sensitivity Analysis of Containment Overpressure Failure Release Location

Fission Product Class	Case SD01 Base Case Containment Overpressure	Case SD01.5 Alternate Failure Location at Equip. Hatch
Xe	1.4E-2	1.7E-2
Cs	1.8E-5	2.1E-5
I	1.7E-5	1.8E-5
Te	2.4E-5	2.4E-5
Ba	1.6E-6	2.2E-6
Ru	3.1E-7	3.1E-7
Mo	1.8E-4	1.9E-4
Ce	1.0E-7	1.1E-7
La	1.7E-8	1.8E-8

The release pathway modeling also influences scenarios with combustion-induced containment failure, and for the shutdown PRA, the combustion-induced containment failure is a higher frequency contributor than the late-term overpressurization failure. Therefore, several sensitivity cases were performed on the combustion-induced containment failure scenario that included variations of the failure pathway modeling. The previously mentioned effect of the auxiliary building connection to the tendon gallery was confirmed to have a similar impact for the shutdown combustion scenario. The potential for significantly reducing the environmental releases was considered to be unduly influenced by the particular MELCOR modeling choices, and this could be a potentially non-conservative result. Therefore, additional sensitivity cases were developed with the connection between the tendon gallery and the auxiliary building closed. Other parameters important to the combustion scenario were also varied. These included the timing of containment failure and availability of inventory for containment sprays. The assessed sensitivity cases are described below:

- Case SD01.4 – Combustion induced containment failure 4 hours after vessel breach, containment failure at tendon gallery (closed connection to auxiliary building), 600,000 gallon inventory for containment spray.
- Case SD01.6 – Combustion induced containment failure 35 hours after vessel breach, containment failure at equipment hatch, 900,000 gallon inventory for containment spray.
- Case SD01.7– Combustion induced containment failure 35 hours after vessel breach, containment failure at tendon gallery (closed connection to auxiliary building), 900,000 gallon inventory for containment spray.

- Case SD01.8 – Combustion induced containment failure 35 hours after vessel breach, containment failure at tendon gallery (closed connection to auxiliary building), 600,000 gallon inventory for containment spray.

Case SD01.4 was developed as a potential representative scenario for the combustion-induced containment failure release category (1-REL-SD-ICF-BURN), but the timing of containment failure was informed by prior analysis of at-power scenarios. After considering the combustion analysis with the ERPRA-BURN code (see Section 2.6.3.7 and Appendix F for additional discussion), the timing of combustion-induced containment failure was adjusted to occur 35 hours after vessel breach in case SD01.6 to better represent the conditions of the shutdown scenario. When adjusted for the specific shutdown scenario conditions (e.g., initial containment conditions, rate of combustible gas generation, frequency of prior combustion events), the ERPRA-BURN analysis showed a slower buildup to combustible gas concentrations that could result in containment failure. Also, this case considered the alternate containment failure location at the equipment hatch and the possibility of additional inventory for containment spray. Similar to the discussion of case SD01.5 above, the higher elevation failure location with direct connection to the environment was anticipated to result in larger environmental releases. However, case SD01.6 has significantly lower releases than case SD01.4, as shown in Table 2-64. The reasons for the lower releases are different than those for case SD01.5 and depend on specific attributes of the accident progression. The later containment failure time contributes to the lower releases. Also, the larger inventory for containment sprays showed unanticipated effects. The containment water level rises after the sprays begin (which is two hours after vessel breach occurs). With the larger inventory, the level is sufficient to reach the hot and cold leg wall cut-outs and spill over into the reactor cavity. While the spill over of water into the reactor cavity does not arrest the MCCI progression, it does inhibit the transport of radionuclides. The boiloff and steaming of water in the cavity also assists in sweeping airborne aerosols from other parts of the containment to deposit on surfaces, which also limits releases to the environment.

Table 2-64 Sensitivity Analysis of Combustion-Induced Containment Failure Location and Other Factors

Fission Product Class	Case SD01.4 (Vessel Breach + 4 hrs, Tendon Gallery, 600k gal. Spray)	Case SD01.6 (Vessel Breach + 35 hrs, Equip. Hatch, 900k gal. Spray)	Case SD01.7 (Vessel Breach + 35 hrs, Tendon Gallery, 900k gal. Spray)	Case SD01.8 (Vessel Breach + 35 hrs, Tendon Gallery, 600k gal. Spray)
Xe	~1.0	~1.0	~1.0	~1.0
Cs	2.8E-2	2.8E-3	1.3E-2	5.7E-3
I	1.7E-1	1.4E-2	1.3E-2	2.7E-2
Te	1.3E-2	6.6E-4	5.5E-3	7.3E-3
Ba	1.0E-3	2.3E-5	5.4E-4	1.0E-3
Ru	5.2E-7	5.8E-7	5.4E-7	8.1E-7
Mo	1.1E-1	1.2E-3	1.0E-1	1.0E-1
Ce	1.9E-6	1.4E-6	2.0E-6	2.7E-6
La	1.8E-6	1.4E-7	1.2E-6	2.4E-6

After identifying the impacts of the larger spray inventory, case SD01.7 was developed with the same conditions as case SD01.6 but with the failure location moved to the tendon gallery. The environmental releases through the tendon gallery are slightly larger than case SD01.6. The

differences are attributed to the extent of flooding in the reactor cavity. With the containment failure occurring at a lower elevation in containment, some of the spray inventory is drained through the failure opening. The level in containment still rises to the point of spilling over into the reactor cavity, but the depth of water in the cavity is less and the duration the water is present is shorter than in case SD01.6.

Another sensitivity case, SD01.8, was analyzed with the inventory available for containment spray reduced to 600,000 gallons. The size of the containment failure opening was also modified to be consistent with the combustion-induced containment failure modeled in the at-power Level 2 PRA, as documented in (NRC, 2022a) in the Appendix D entry on “Containment Leakage, Effective Sizes Under Normal and Accident Conditions.” Cases SD01.6 and SD01.7 used a larger failure size equivalent to the maximum opening under long-term overpressure failures (i.e., 1 square-foot). In case SD01.8 the spray inventory is not sufficient to reach the level to spill over to the reactor cavity, and the cavity remains dry throughout the scenario. Ultimately, case SD01.8 was selected as the representative sequence for the combustion-induced containment failure release category (1-REL-SD-ICF-BURN). However, the selection was not necessarily informed by the environmental release results. The key attributes were selected to represent the most realistic conditions of the scenario being analyzed. These included the timing of the combustion-induced containment failure (informed by the ERPRA-BURN analysis discussed in MU-7.2 above), the inventory available for containment spray (based on the referenced tank volumes in the EDMG-directed action), and the size of the containment failure opening (selected to be consistent with the at-power PRA combustion-induced failure scenario).

The sensitivity cases consider different variations of the containment failure release pathway, but other contributing factors of the accident progression can have significant influences on the environmental releases. The following observations are noted:

- The modeling of the containment failure release through the tendon gallery can be highly influenced by modeling assumptions related to the connected structures. Submergence of the failure location, heatup, boiloff, and flows between connected structures influence the airborne releases and can vary significantly for different accident conditions.
- Containment failures where flow is directed to the auxiliary building have potential for significant retentions in the auxiliary building and reductions in the releases to the environment.
- Two potential containment overpressure failure locations are identified by the containment capacity analysis: at the tendon gallery near the containment wall-basemat junction and near the containment equipment hatch. The higher elevation failure at the equipment hatch connects directly to the environment and avoids issues related to submergence. Intuitively, this could be thought to lead to larger environmental release; however, the sensitivity cases demonstrate that this is not true.
- The availability of additional containment spray inventory, such that the level spills over to the reactor cavity, can inhibit the transport of radionuclides to the environment. If actions can be taken to align additional sources of inventory or refill tanks, then environmental releases can be reduced.

- Other factors can influence the releases to the environment, including the timing of containment failure and the containment failure opening size.

MU-13.2 – Sensitivity on open containment release pathway

The shutdown PRA considers two potential release pathways which would be open in many POSs and would require action by the operators to close in the event of an accident—the equipment hatch and the personnel airlock. The equipment hatch is assumed to be the release pathway of primary interest, since it has a larger opening area and is placed so as to offer few obstructions to the release (e.g., releases from the core in an open-vessel scenario, such as POS 8, can rise from the vessel directly into the dome and then out to the environment via the hatch). However, it is possible that there are conditions when the equipment hatch is closed and the personnel airlock is open, or in responding to emergency conditions, operators successfully close the equipment hatch and fail to close the personnel airlock. Therefore, a sensitivity MELCOR calculation (SD07.1) was performed in order to compare the impact of failure location.

Table 2-65 compares the radiological source terms to the environment for the two cases with different pathways for failure to close the open containment. The source terms for both cases are large, although the case where the release pathway is via the equipment hatch (case SD07) has release magnitudes that for some fission product classes are almost twice as large as for the case where the release pathway transits the personnel airlock. Note that although the open containment release categories collectively make up only about 4 percent of total release frequency, they have some of the highest release magnitudes. Therefore, it is concluded that the shutdown Level 2 PRA results do show a modest degree of sensitivity to the assumed pathway of releases through an open containment.

Table 2-65 Sensitivity on Releases to Environment via Equipment Hatch versus Personnel Airlock

Fission Product Class	Case SD07 (Equipment Hatch Open)	Case SD07.1 (Personnel Airlock Open)
Xe	~1.00	~1.00
Cs	0.25	0.14
I	0.38	0.20
Te	0.34	0.20
Ba	1.8E-2	1.4E-2
Ru	1.1E-3	1.1E-3
Mo	0.64	0.38
Ce	3.7E-4	3.1E-4
La	2.1E-5	1.6E-5

2.6.3.14 Other Fission Product and Emergency Preparedness Model Uncertainties

Table 2-66 identifies issues contributing to model uncertainty in the modeling of fission product releases and emergency preparedness.

Table 2-66 Identified Uncertainties Related to Fission Product Releases and Emergency Preparedness

Item	Description	Other comments
Surrogate risk metric definitions	This study uses surrogate release metrics, such as LERF and LRF, to characterize the release frequencies with respect to potential consequences. For the shutdown study, relying on the same surrogate risk metric criteria in terms of chemical class release fractions may give overly conservative results, if neglecting radionuclide decay for accidents initiated at times long after reactor shutdown.	The specific criteria for determining these surrogate metrics can vary between studies. The at-power Level 2 PRA describes the risk metric definitions in Appendix D of (NRC, 2022a). The extension of the surrogate risk metric criteria for the shutdown model is discussed in MU-14.1, below.
Fission product retention in connected structures	Retentions in containment-connected structures (e.g., auxiliary building) are predicted to be large in some MELCOR calculations.	This issue is discussed below in MU-14.2 and in sensitivity case MU-13.1 on release pathway modeling.
Modeling of ruthenium releases	Uncertainties in the modeling of ruthenium behavior in an air oxidizing environment can affect the current modeling of ruthenium release (using a traditional release model with a modified ruthenium vapor pressure), which could result in either higher or lower releases. This is of particular relevance to the source terms for scenarios with the vessel head removed.	This issue is discussed in the context of modeling air ingress to the core in MU-4.4 above. The issue is also discussed in (Gauntt, 2010).
SCALE analysis use of static masses	To simplify the modeling, static chemical class masses (i.e., inventories) are used, although time-dependent masses could be used. This can have minor effects on fission product transport behaviors (either higher or lower, depending on the evaluation time and chemical class in question).	This issue is discussed further in MU-14.3 below.
SCALE analysis uncertainties	SCALE analysis uncertainties can influence the decay heat and fission product inventories. These uncertainties could have small, systematic effects on the MELCOR analyses.	Examples of SCALE analysis model uncertainties and their impacts on the calculated decay heat are discussed in the at-power Level 2 PRA report in Section 2.3.1 (NRC, 2022a).

Table 2-66 Identified Uncertainties Related to Fission Product Releases and Emergency Preparedness (cont.)

Item	Description	Other comments
SCALE analysis uncertainties specific to assumptions about the timing within a fuel cycle	This could have moderate effect systemically on the MELCOR analyses. Using BOC would result in lower decay heat and lower source terms as compared to EOC. The use of EOC is appropriate for the current shutdown model because the model addresses a planned refueling outage. However, if other outage types are considered, the time-at-cycle assumption could be re-considered.	Current MELCOR model for shutdown uses EOC. The draft Surry SOARCA uncertainty analysis (SNL, 2016) sampled time-at-cycle as BOC (7 days), MOC (200 days), and EOC (505 days) in the integrated uncertainty analysis. In addition, for each of these relative cycle times, families of decay heat curves were sampled to represent deviations from the nominal decay heat curve (see Section 4.1.2.5 of that study).
Timing of emergency action level declarations	Ambiguities in the timing of declaring EALs can influence the emergency response actions and consequences of a release. Any impacts from delayed EAL escalation are not captured; this includes delays with activating the TSC since this is assumed to happen when declaring an ALERT (based on the plant's emergency plan).	For the at-power PRA, the basis for this modeling assumption is described in (NRC, 2022a) in an Appendix D entry entitled, "Emergency Action Level Monitoring Assumptions." Although the shutdown scenarios would have different procedural cues for initiating the emergency plan, the same assumptions regarding possible delays are applied in the shutdown PRA. The timing could potentially affect the consequence modeling within the scope of the Level 3 PRA.
Source term characteristics	This is a broad category of uncertainty in (EPRI, 2012b) that covers release model, fission product transport and deposition, chemistry model, MCCI effects, uncertainties, re-volatilization, and other aspects.	These uncertainties are largely addressed by the numerous related items previously identified herein. Note that the draft Surry SOARCA uncertainty analysis project (SNL, 2016) 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 a percentage of that remaining cesium is introduced to the fuel gap as CsOH, the draft Surry SOARCA uncertainty analysis varied the fraction of remaining cesium as cesium molybdate between 0 to 1 with mode of 0.8 (Section 4.1.5.2 of [SNL, 2016]).
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 SOARCA uncertainty analysis uses a surrogate release fraction based on the fission gas release to the gap. Experimental results (Pontillon, 2005) estimated that the release fraction could vary from 0.2% to 7%. The draft Surry SOARCA uncertainty analysis varied the gaseous iodine percentage in the range of 0 to 3% (Section 4.1.5.1 of [SNL, 2016]).

Table 2-66 Identified Uncertainties Related to Fission Product Releases and Emergency Preparedness (cont.)

Item	Description	Other comments
Aerosol shape factor set to assume spherical shape	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 set to one to indicate a spherical shape. The draft Surry SOARCA uncertainty analysis varies the dynamic shape factor between 1 and 5 (Section 4.1.6.1 of [SNL, 2016]).
Aerosol size distribution	Aerosol dynamics in MELCOR are based on MAEROS, which is a multi-sectional, multicomponent code 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. 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. The treatment of aerosols in MELCOR influences the decontamination factors of release pathway surfaces.	MELCOR's default treatment of aerosols is used. Note that the MELCOR defaults are 0.1 μm and 50 μm for minimum and maximum aerosol diameters, 10 size bins, and two components—one for water vapor and the other for all other radionuclide classes. The potential impacts of the aerosol size modeling treatment and alternative models are discussed in the at-power Level 2 PRA in the context of an induced SGTR scenario (see sensitivity case 3A2 in Appendix B of NRC [2022a]).

Table 2-66 Identified Uncertainties Related to Fission Product Releases and Emergency Preparedness (cont.)

Item	Description	Other comments
Fire suppression system impacts on auxiliary building source term attenuation	This item 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 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.

Alternate treatment(s)

MU 14-1 – Sensitivity analysis on surrogate risk metric definitions and extension to shutdown accidents

Surrogate risk measures are defined in the Level 2 PRA to provide a grouping of release categories that are anticipated to have similar consequences. The consequences are ultimately quantified in the consequence analysis part of the Level 3 PRA. Still, surrogate risk metrics can be useful because they provide indications of the significance and potential consequences of frequency results of the Level 2 PRA. This analysis provides clarification on how the surrogate risk metrics, particularly LERF, are defined in the shutdown Level 2 PRA and maintain consistency with the definitions used in the at-power Level 2 PRA.

The surrogate risk measures reported in the at-power Level 2 PRA are defined in Section 2.6.1 and further discussed in the Appendix D entry, “Risk Metric Surrogate Definitions (LERF, LRF, CCFP),” in (NRC, 2022a). The same definitions are used in the shutdown Level 2 PRA. However, use of consistent criteria for the surrogate risk metrics does not necessarily translate to the same timing and magnitude of radiological releases due to differences between the assumptions about the time during cycle (i.e., middle-of-cycle [MOC] versus end-of-cycle [EOC]) used to estimate initial inventories and radioactive decay between reactor shutdown and the time of the initiating event. While the at-power analysis assumed negligible impact from radioactive decay, the activity can be significantly altered for some shutdown scenarios occurring days into the outage. Also, for accidents initiated after refueling, a portion of the core is replaced with new fuel, which significantly reduces initial radionuclide inventories.

The surrogate risk metrics used in the study include LERF, LRF, and CCFP. The LERF definition depends on the timing and magnitude of iodine releases, and the impacts of this definition for the shutdown analysis is discussed further below. First, the other two surrogate risk metric definitions are reviewed in terms of their applicability to the shutdown Level 2 PRA.

LRF for the at-power Level 2 PRA includes 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, or smaller than, the remainder of the (intact containment)

source terms. The same definition is considered to be appropriate and applicable to the shutdown Level 2 PRA. The specific criteria for inclusion to LRF is adjusted to be defined in terms of the shutdown-specific intact containment source term, that is, the representative source term for release category 1-REL-SD-NOCF. In considering applicability to LRF, the representative source term is characterized in terms of the total release fraction of the cesium class. This was selected due to the importance of cesium releases to the long-term radiological consequences and the relatively long half-lives that make impacts of decay negligible with respect to the time of the accident during the outage. The cesium release fraction for 1-REL-SD-NOCF is 1.8×10^{-5} . A criterion of total cesium release fraction greater than 1×10^{-3} was used for selecting source terms that are grouped as LRF. Such releases are considered significantly greater than, and thus not “comparable to,” the intact containment source term for shutdown.

The CCFP definition for the at-power Level 2 PRA is given as the ratio of the release categories involving a failed or bypassed containment to overall release frequency. As this definition does not depend on the timing or magnitude of releases, the same definition is considered to be appropriate and applicable to the shutdown Level 2 PRA.

As discussed above, additional analysis is required to translate the at-power LERF definition to the shutdown analysis. The at-power Level 2 PRA defined the following two quantitative definitions for LERF:

- Release categories are defined to contribute to LERF (potential early injuries) if their representative source term has a warning time (based on iodine release exceeding 3.2×10^{17} Bq) of less than 20 hours simultaneous with the cumulative iodine release activity being greater than 1.3×10^{18} Bq of iodine and 1.4×10^{17} Bq of iodine-131.
- Release categories are defined to contribute to LERF (potential early fatalities) if their representative source term has a warning time (based on iodine release exceeding 3.2×10^{17} Bq) of less than 3.5 hours simultaneous with the cumulative iodine release activity being greater than 1.3×10^{18} Bq of iodine and 1.4×10^{17} Bq of iodine-131.

The iodine release activity criteria are translated to release fractions for the at-power PRA. The release fraction criteria are based on the initial activity assessed at the time of reactor trip at the midpoint of a representative cycle for the reference plant. The initial core activities for this reference case are 3.2×10^{19} Bq of total iodine and 3.6×10^{18} Bq of iodine-131. The following release fraction criteria are developed corresponding to the initial activities:

- Warning time is indicated by total iodine release exceeding 1 percent (based on iodine release exceeding 3.2×10^{17} Bq). Note that the warning time definition also considers an alternate criterion of noble gas release fraction exceeding 10 percent. The warning time is calculated using the criterion that is exceeded first.
- Cumulative iodine release exceeds 4 percent of total iodine (based on release activity being greater than 1.3×10^{18} Bq).
- Cumulative iodine-131 release exceeds 4 percent (based on release activity being greater than 1.4×10^{17} Bq of iodine-131).

These definitions need to be translated to the reactor at-shutdown situation. The reactor at-shutdown PRA uses a different initial core inventory, which is based on an EOC core with a

controlled shutdown entering a refueling outage. The initial activities are different than the MOC reactor trip that was used for the at-power PRA. Also, the total iodine and iodine-131 activities decay after reactor shutdown. This influences the relationships to the thresholds for accident scenarios that are modeled as initiating at different times throughout the outage. The thresholds will vary by the POS that is modeled for each representative accident scenario.

To address these considerations, information is compiled here to show the impacts of iodine decay on the LERF criteria thresholds. Table 2-67 provides the initial activities of total iodine and iodine-131 in the core for various POSs. These values account for both radioactive decay since reactor shutdown as well as the replacement of roughly half the core with un-irradiated fuel between POS 8E and 8L. Prior to refueling, the decay in the iodine activities could require small adjustments to the LERF thresholds. After refueling, the decreases in iodine activity are significant.

Table 2-68 provides the adjusted LERF release fraction thresholds for the POS-specific activities. The adjusted LERF release fraction criteria show the fraction of the reactor-at-shutdown iodine activity that would need to be released to reach an equivalent activity to the reactor-at-power LERF criteria. The shutdown releases would generally have to be higher to reach an equivalent released activity, and the release fraction increases with accident scenarios occurring at later times after reactor shutdown. The total iodine activity decreases more rapidly than the decrease in iodine-131 activity. The cumulative iodine-131 release fractions for the example POSs prior to refueling vary from approximately 4–7 percent, which is not significantly different than the reactor-at-power release fraction criterion of 4 percent. The post-refueling POSs have significantly reduced iodine activity and corresponding significantly higher release fractions required to meet the LERF criteria. This suggests that post-refueling POSs are unlikely candidates for LERF.

The shutdown release categories were assessed using the reactor-at-power release fraction criteria, as described above. Using these criteria, three release categories were identified as meeting the criteria for potential early injuries: 1-REL-SD-COPEN-RCSINTACT, 1-REL-SD-COPEN-RCSVENT, and 1-REL-SD-ISLOCA. Two of the release categories were identified as meeting the criteria for potential early fatalities: 1-REL-SD-COPEN-RCSINTACT and 1-REL-SD-COPEN-RCSVENT. These release categories were evaluated to consider the impacts of the POS-specific criteria. Table 2-69 shows the adjustments to the warning time criteria if POS-specific thresholds are used. The three release categories that are identified as LERF contributors are shown in the table. The three LERF-designated release categories show small increases in warning time when the criteria are adjusted. However, the warning time changes do not impact their LERF designations. The warning times are still within the defined thresholds for LERF (i.e., 3.5 hours for potential early fatalities and 20 hours for potential early injuries). All other shutdown release categories have warning times in excess of 50 hours. Using POS-specific thresholds would potentially increase the estimated warning times for these categories also, but this would have no impact on identifying categories for LERF.

Table 2-70 shows the impacts of adjusting the cumulative iodine release fraction criteria for POS-specific thresholds. The modeled releases are characterized by their representative source terms, and Table 2-70 shows the total cumulative iodine release fraction for the representative case. The release fraction criteria adjusted for POS-specific decay are generally higher than the 4 percent criterion for at-power releases, though the iodine-131 criteria are only

Table 2-67 Initial Iodine Activities for Various Shutdown Plant Damage States

POS	Activities (Bq)						
		5A	5B	6	8E	8L	10
Time after Shutdown (hr)	t = 0 (Reactor Shutdown)	12 hr	55 hr	82 hr	141 hr	329 hr (Post Refuel)	501 hr (Post Refuel)
I-131	3.38E+18	3.25E+18	2.80E+18	2.57E+18	2.09E+18	6.64E+17	3.56E+17
Total I	1.69E+19	1.09E+19	6.19E+18	4.96E+18	3.36E+18	8.07E+17	3.86E+17

Table 2-68 LERF Iodine Release Fraction Criteria Adjusted for Various Plant Operating States

POS	LERF Release Fraction Criteria						
		5A	5B	6	8E	8L	10
Time after Shutdown (hr)	t = 0 (Reactor Shutdown)	12 hr	55 hr	82 hr	141 hr	329 hr (Post Refuel)	501 hr (Post Refuel)
Warning time Total I activity 3.2×10^{17} Bq (1% of at-power inventory)	1.9%	2.9%	5.2%	6.5%	9.5%	39.7%	82.9%
Cumulative Total I activity 1.3×10^{18} Bq (4% of at-power inventory)	7.7%	11.9%	21.0%	26.2%	38.7%	>100%	>100%
Cumulative I-131 activity 1.4×10^{17} Bq (4% of at-power inventory)	4.1%	4.3%	5.0%	5.4%	6.7%	21.1%	39.3%

Table 2-69 LERF Warning Time Criteria Adjusted for Plant Operating State

Release Category	Rep. POS	At-power Total I Fraction Criterion	Warning Time (hr)	POS-specific Total I Fraction	Adjusted Warning Time
1-REL-SD-COPEN-RCSINTACT	POS 5A	1%	2	2.9%	2
1-REL-SD-COPEN-RCSVENT	POS 5B	1%	2	5.2%	2.5
1-REL-SD-ISLOCA	POS 5A	1%	7	2.9%	12

Table 2-70 LERF Cumulative Iodine Release Criteria Adjusted for Plant Operating State

Release Category	Rep. POS	Rep. Source Term Cumul. I Released ¹	Iodine Radio-nuclide	At-Power Cumul. I Release Fraction Criterion	POS-Specific Cumul. I Release Fraction Criterion	Does Release Exceed POS-Specific Criterion?
1-REL-SD-COPEN-RCSINTACT	POS 5A	38%	Total I	4%	11.9%	Yes
			I-131	4%	4.3%	Yes
1-REL-SD-COPEN-RCSVENT	POS 5B	82%	Total I	4%	21%	Yes
			I-131	4%	5%	Yes
1-REL-SD-ISLOCA	POS 5A	4.5%	Total I	4%	11.9%	No
			I-131	4%	4.3%	Yes
¹ The representative source terms are developed using MELCOR analyses, which track releases by chemical classes, not at the individual radionuclide level. The I-131 release fractions are assumed to be proportional to the total iodine release fractions provided from the MELCOR cases.						

marginally higher. The table shows that even after adjusting for the POS-specific decay, each of the identified LERF release categories would still meet the cumulative iodine release criteria for LERF. The only exception is the total iodine release for 1-REL-SD-ISLOCA. However, this release category is still considered a LERF category based on the iodine-131 release activity.

The following observations are noted from this exercise to adjust LERF criteria for shutdown and POS-specific release activities:

- The assessment of warning times for shutdown accident scenarios can be affected by the radionuclide decay for scenarios that are initiated many hours after reactor shutdown. The warning times will generally be longer due to the longer time to reach an equivalent released activity as the at-power criterion. However, the increase in warning time is typically small. For many of the release categories, for example the categories with containment open, the rate of environmental release is rapid after the release begins. For the release categories assessed for this study, adjusting for POS-specific release activity has no substantial impact on the LERF determination. If considering a scenario with a slow, gradual release, then the assessment of warning times may need to be adjusted to appropriately correct for the timing and magnitude of released activities.
- The cumulative release fraction criteria are not substantially affected by the POS-specific release activities for POSs occurring prior to refueling. While the total iodine activity is reduced due to decay during the pre-refueling period, the activity levels remain significant. In particular, the iodine-131 activity will not be significantly reduced in the several days following reactor shutdown.
- The cumulative release fraction criteria are significantly different for post-refueling POSs. This is due to both the long time after reactor shutdown and the replacement of a fraction of the fuel assemblies with new fuel. The post-refueling POSs would generally require a large magnitude release (e.g., an unmitigated release through a large containment failure or opening) to reach the equivalent release activities as compared to the at-power LERF criteria. Such large releases are possible, especially considering

large containment openings like the equipment hatch; however, the post-refueling accident scenarios tend to develop slowly due to reduced decay heat levels. These aspects make post-refueling accident scenarios unlikely candidates for LERF contributors. For this study, there were no post-refueling POS scenarios that were considered as potential candidates for LERF contributors.

The L3PRA shutdown Level 2 PRA team chose to retain the same release fraction criteria for LERF definitions as used in the at-power PRA. This is acknowledged to be a conservative modeling choice because using the same release fraction criteria for accidents initiated during shutdown could potentially result in capturing representative shutdown scenarios that might not meet the adjusted, POS-specific LERF criteria. However, clarity and consistency in the LERF definitions across all parts of the L3PRA study was deemed to outweigh the benefits of using a tailored definition (or set of definitions) for shutdown POSs.

The release categories identified as contributing to LERF in this study would not have changed if the criteria were adjusted to account for POS-specific iodine-131 release activities. Therefore, the conservative choice for LERF criteria at shutdown has no impact on the Level 2 PRA results. It is important to remember that several important approximations underpin the LERF assignments. Ultimately, the Level 3 PRA offsite consequence analysis for the reactor at-shutdown will determine the early health effects based on the actual release activities, and the results can be used to inform if the LERF designations are too conservative or appropriate.

MU-14.2 – Discussion of MELCOR modeling assumptions and fission product releases

MELCOR simulations have indicated notable fission product retentions in structures connected to the containment. Sensitivity analysis MU-13.1 provides insights into the impacts of different release pathway modeling assumptions. This issue was further investigated after it was observed that the auxiliary building retentions in cases SD01.2 and SD01.3 were seemingly large, even though the flow path between the tendon gallery and the auxiliary building was not open in these simulations. Opening the modeled flow path between the tendon gallery and auxiliary building is normally a part of the MELCOR model's treatment of containment failure. The flow path is opened when the containment pressure reaches the failure pressure set point. Another flow path, connecting the containment with the tendon gallery, also opens on that pressure condition. However, in these two cases of forced containment failure, only the connection from containment to the tendon gallery was opened at the desired time, but not the connection between the tendon gallery and auxiliary building. As a result, no water or gas ever passes from the tendon gallery to the auxiliary building in these calculations.

Upon further investigation, it was recognized that the vast majority of the fission product retentions initially attributed to the auxiliary building are actually retained in the tendon gallery (the tendon gallery is in the same set of MELCOR control volumes used to plot auxiliary building retentions). The simulations were re-run with the flow path between the tendon gallery and the auxiliary building forced open at the time of containment failure. Both sets of simulations taken together illustrate a complex relationship between water level in lower containment relative to the assumed containment failure leak location, containment depressurization, the temperature of the water in the tendon gallery (sub-cooled versus saturated), the role of the connection between the tendon gallery and the auxiliary building, operating of the auxiliary building ventilation and filtration system, and numerical variability caused by slight perturbations in the initial and boundary conditions.

The at-power modeling assumes containment failure manifests in the form of liner shear failure at the basemat junction, but in the original structural investigation, other containment failure locations (the equipment hatch and personnel air lock) were also identified as probable failure locations. The decision to use the liner junction failure location (opening to the tendon gallery) was largely a compromise. Whereas the equipment hatch failure location would result in direct discharge to the environment (and, thus, presumably higher environmental releases) and personnel airlock failure would result in direct discharge to the auxiliary building (and thus potential for greater habitability challenges), the liner junction failure location has aspects of both of these traits since the tendon gallery connects to both the environment and the auxiliary building.

For this reason, the calculations with the connection to the auxiliary building closed generally reflect the situation where containment failure occurs at a location that preferentially discharges fission products to the atmosphere, while the calculations with the auxiliary building connection open generally reflect the situation where containment failure occurs at a location that preferentially discharges fission products to the auxiliary building. Although there was no clear basis for selecting between these modeling options, it was decided to use the calculations with the auxiliary building connection closed as this was thought to be a realistic failure that could potentially lead to larger environmental releases. In this way, the results with the auxiliary building connection open can be thought of as sensitivity studies showing the significant effect of auxiliary building retention for certain containment failure locations coincident with auxiliary building filtering. This investigation prompted additional sensitivity studies (see MU-13.1 above), including consideration of alternate containment failure locations that would not be subject to the same complexities related to the connected structures and water at the failure location.

MU-14.3 – Discussion of use of static masses in estimating source terms

The MELCOR modeling of element chemical class masses (i.e., inventories) is static, not accounting for time-dependence. This was done to simplify the modeling even though time-dependent masses are available. This treatment can have minor effects on fission product transport behaviors (either higher or lower, depending on the evaluation time and chemical class in question).

The reason that the values used by MELCOR are relatively constant over time is because they are the elemental masses which guide chemical behavior, as opposed to the contributing radioisotope masses that guide nuclear behavior (which would behave very differently for all but the very long-lived radioisotopes). Much of this elemental mass is comprised of stable isotopes. For instance, while radioiodine decays relatively quickly (and comprises a higher abundance than it would in nature due to fission yields being skewed to neutron-heavy isotopes), stable iodine is initially abundant (roughly 96 percent of iodine at time zero) and is being generated by the decay of other radioisotopes. As such, elemental mass is much more constant over time than are the activities and decay heat generation of individual radioisotopes. However, some shutdown scenarios are slow developing with core damage a few days after the initiating event. It is possible that accounting time-dependent masses could have non-trivial impacts on fission product transport and releases. Therefore, this topic is considered as a candidate for future work.

2.6.3.15 Accident Termination Model Uncertainties

Table 2-71 identifies issues contributing to model uncertainty in the modeling of accident termination.

Table 2-71 Identified Uncertainties Related to Accident Termination

Item	Description	Other comments
Impact from accident duration truncation of sequence runs	The accident simulation truncation time can affect the estimated amount of radiological material released. For the at-power Level 2 PRA, the termination time had a significant impact on LRF results. For the shutdown model, event progression is much slower. Terminating at earlier times may arbitrarily eliminate slow developing releases. The termination time in the shutdown simulations is seven days after the initiating event. Extending termination times beyond seven days is expected to have limited benefit in most cases because most simulations have reached a stable state.	The results of truncating accident termination for the shutdown representative scenarios are presented in Section 2.6.1 and discussed further in MU-15.1 below. The at-power termination of release is discussed further in Section 2.6.1 of (NRC, 2022a) in the Appendix D entry, "Termination of Radiological Release."

Alternate treatment(s)

MU-15.1 – Sensitivity analysis on accident termination modeling

Table 2-72 demonstrates the impact of alternate accident simulation termination time on the risk metrics LERF, LRF, and CCFP. The LERF metric is unaffected by the choice of termination time. The LRF and CCFP results show modest reduction for the shortest termination time, which is 36 hours after indication of core damage. The sensitivity analysis shows the potential for reducing frequencies of significant releases if post-core-damage actions can be successfully implemented to terminate the accident. A similar analysis was performed for the at-power Level 2 PRA (NRC, 2022a). The results for the at-power scenarios showed a more significant reduction for LRF and CCFP.

Table 2-72 Alternate Accident Termination Time Sensitivity Analysis

	Alternate Accident Termination Time					
	36 hr after CD indication		60 hr after CD indication		7 days after initiator	
Surrogate Risk Metric	Frequency (per year)	% of Total Release	Frequency (per year)	% of Total Release	Frequency (per year)	% of Total Release
LERF	5.3E-07	4.3%	5.3E-07	4.3%	5.3E-07	4.3%
LRF	4.4E-06	35.4%	6.1E-06	49.2%	6.1E-06	49.2%
CCFP	n/a	35.4%	n/a	49.2%	n/a	50.6%

The alternate times for terminating the accident use times occurring after core damage, when the strategies for terminating the accident can shift. Also, there may be potential for offsite resources to become available at these later times in the accident progression. However, for the

shutdown analysis, there are several scenarios that progress slowly, and core damage can occur very late in the progression. For example, the representative accident scenarios during conditions with the refueling cavity flooded (i.e., POS 8) show core damage occurring 100 hours after the initiating event occurs (see Table 2-27). Note that this sensitivity, with focus on post-core-damage termination, does not consider potential actions that could prevent core damage for these slow-developing scenarios. However, it is acknowledged that potential actions could be implemented, such as implementation of FLEX strategies.¹⁸

The slow developing nature of some shutdown scenarios also provides a caution for arbitrarily terminating accident simulations before a stable state is established. While there may be potential for additional resources to address accidents that are developing over several days, this is not assured. The accident simulation approach should consider the proceduralized guidance and give realistic treatment to terminating accidents and releases.

One MELCOR sensitivity analysis extended the simulation time from seven to eight days after the event initiating (see case SD01.5 in Section 2.3.3). The scenario involved late containment failure due to overpressurization, in which the containment failure occurs just prior to the original 7-day simulation end time (specifically, at 167.6 hours). The results, as shown in Figure 2-26, show the releases to the environment continue to increase after seven days, though they remain relatively small. This sensitivity demonstrates potential conditions that could justify extending the accident simulation time.

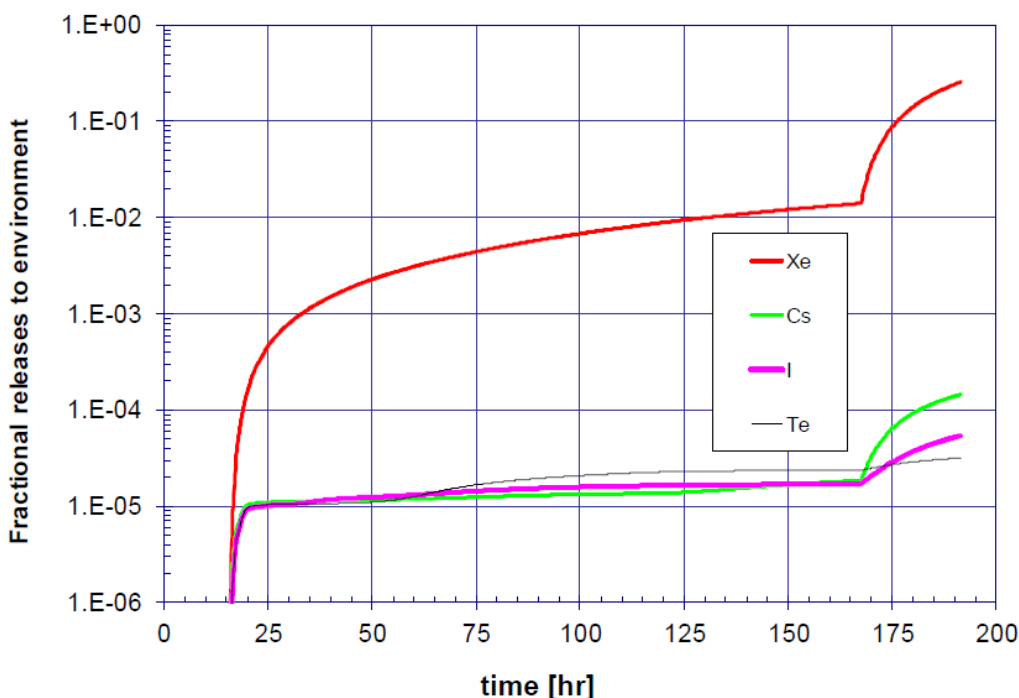


Figure 2-26 Volatile Radionuclide Releases to Environment for Late Containment Failure Scenario Extended to 8-Day (192-Hour) Accident Simulation Termination Time

¹⁸ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the L3PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

2.6.3.16 MELCOR Solution Robustness

Table 2-73 identifies issues contributing to model uncertainty in the MELCOR analyses.

No alternative treatments are proposed.

Table 2-73 Identified Uncertainties Related to MELCOR Solution Robustness

Item	Description	Other comments
Miscellaneous threshold effects in plant response	The existence of thresholds in the plant response modeling mean that small input changes can sometimes result in notable output changes.	These effects are often inevitable in this type of analysis. See MU-8.1 in the at-power Level 2 PRA, Appendix C uncertainty analysis (NRC, 2022a) for an example demonstrating the sensitivity to approaches for the release pathway modeling.
Mass conservation errors introduced by interference between the flashing and hygroscopicity models in MELCOR	These models were 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 was true for the version of the code used in earlier analyses for this project. The present analysis incorporates the current best practice recommendation from the code developers at Sandia National Laboratories (SNL). Thus, the hygroscopic model is active, and the flashing model is not active. This addresses the mass conservation errors in previous model versions. However, the model uncertainty is retained here for potential future sensitivities to explore the impacts of this modeling change.	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 (i.e., pipe break, seal leakage, and relief valve flow paths) in which the flashing model had been active in previous revisions of MELCOR. The hygroscopic/flow path flashing models were corrected in revision 2.1.8611 and later revisions of the code, which is consistent with the MELCOR version used for the shutdown Level 2 PRA (i.e., version 2.2.9607).
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 addressed by developers at SNL.

2.7 Level 2/Level 3 PRA Interface

The Level 2/Level 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 (i.e., 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).

Table 2-74 provides a synopsis of the release category results for use by the Level 3 PRA team. The more complete set of needed information relative to releases is provided in the MELCOR output files designated in the right-most column of this table. 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 2-75 provides a description of the key release paths and their applicability to the shutdown release categories. Table 2-76 describes which release paths (identified by their MACCS identifier number) are modeled for each release category. The dominant release path is identified based on iodine released to the environment. There may be multiple release paths that contribute significantly to the release. There also may be modeled release paths that do not contribute significantly to the environmental release (e.g., normal containment leakage for a failed containment release category). Table 2-77 shows all modeled release paths and includes modeling details (e.g., elevation, location, and relation to MELCOR flow path).

The core inventories are based on an analysis of the EOC with power coastdown to shutdown in preparation for refueling as described in Section 2.3.1. In the at-shutdown analysis, release category frequencies are calculated outage-wide, and then evaluated to obtain the contributions from individual POSs. Meanwhile, the MELCOR source terms are generated assuming that the initiating event occurs at a particular point in the outage. Thus, the consequence analysis will need to consider what accident timing after shutdown (and thus initial core inventory) is most appropriate when utilizing the time-dependent MELCOR release fractions. To facilitate this, Table 2-74 distills information about which POS has the highest contribution to the release category and which POS the MELCOR analysis assumed. The release categories of most concern in this regard are those where the frequency contribution is spread across more than one POS (i.e., is more diffuse): 1-REL-SD-BMT, 1-REL-SD-ICF-BURN, 1-REL-SD-LCF, and 1-REL-SD-NOCF. Refer to Table 2-19 for a complete account of POS contributions to release category frequencies.

In addition to the above considerations, the following are some items for the Level 3 PRA team to be aware of in using the provided results:

- Time zero in MELCOR simulations corresponds to the initial upset condition, which occurs well after the shutdown of the reactor for these source terms. 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 2-74).
- 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.
- Some special handling instructions are provided in Table 2-77. Specifically, some flow paths may require adjustment of the release height, while some flow paths are internal paths defined in the MELCOR model or used for diagnostic purposes and should not be propagated to the offsite consequence analysis because they do not represent releases to the environment.

Table 2-74 Release Category Summary Table for Level 3 PRA

Release Category	Frequency (/year)	Highest-Contributing POS	MELCOR Case Identifier	As-Modeled POS	Warning Time Surrogate ¹ (hr)	Cumul. Iodine Release ³	Cumul. Cesium Release ³	MELCOR File Name
1-REL-SD-ISLOCA (a.k.a., LOI-OC)	1.3×10^{-7}	5A (97%)	SD11.1	5A	7	5%	3%	SD11-1.ptf
1-REL-SD-ISGTR	$<1 \times 10^{-12}$ (2)	-	-	-	-	-	-	-
1-REL-SD-CIF	1.7×10^{-6}	8E (63%)	SD03	8E	95	6%	7%	SD03.ptf
1-REL-SD-COPEN-RCSINTACT	2.7×10^{-7}	5A (62%)	SD07	5A	2	38%	25%	SD07.ptf
1-REL-SD-COPEN-RCSVENT	1.3×10^{-7}	5B (56%)	SD13	5B	2	82%	79%	SD13.ptf
1-REL-SD-COPEN-HEADOFF	7.8×10^{-8}	8E (100%)	SD05	8E	52	88%	87%	SD05.ptf
1-REL-SD-ECF	5.1×10^{-10}	6 (55%)	SD01.2	6	~64	3%	1%	SD01-2.ptf
1-REL-SD-CF-Before-CD	2.1×10^{-6}	8E (100%)	SD09	8E	103	14%	15%	SD09.ptf
1-REL-SD-ICF-BURN	1.7×10^{-6}	6 (48%)	SD01.8	6	~85	3%	6%	SD01-8.ptf
1-REL-SD-LCF	5.5×10^{-8}	5B (52%)	SD01	6	>156	<1%	<1%	SD01.ptf
1-REL-SD-BMT	1.2×10^{-7}	6 (25%)	SD08.2	3	>140	<1%	<1%	SD08-2.ptf
1-REL-SD-NOCF	6.1×10^{-6}	8L (39%)	SD12	10	>138	<1%	<1%	SD12.ptf
¹ This represents the delta between the preliminary GE declaration time and the earlier of the time of noble gas release >10% or iodine release >1%. ² The frequency of this RC is below the model truncation limit of 1×10^{-12} per year. The induced SGTR contribution is insignificant due to the limited reliance on SG cooling during shutdown POSs and during much of the outage SGs are not available at all. ³ These values represent the fractions of mass of the chemical class from the core inventory released to the environment. The released activities, including effects of radionuclide decay during the time after reactor shutdown, are accounted for in the consequence analysis modeling.								

Table 2-75 Key Release Path Descriptions

Containment Failure or Bypass Mechanism	Release Path Description	Applicable Release Categories
Normal containment leakage	The release path is modeled with multiple flow paths distributed through the containment shell above grade with the total flow area adjusted to match the maximum allowable containment leakage flow rate. MACCS release path identifiers: 819, 823, 824 (Two additional flow paths are not included here, which model containment leakage to the auxiliary building.)	1-REL-SD-NOCF 1-REL-SD-BMT¹
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.² MACCS release path identifier: 844	1-REL-SD-ECF 1-REL-SD-ICF-BURN 1-REL-SD-CF-Before-CD
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 the containment pressure, reaching a maximum of 1 square foot, or equivalent to an approximately 13.5-inch diameter opening. The tendon gallery connects to the environment.² MACCS release path identifiers: 820, 821	1-REL-SD-LCF
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 a 2-inch diameter opening. MACCS release path identifier: 842	1-REL-SD-CIF
Containment equipment hatch open	The release path represents the open containment equipment hatch. The flow area is equivalent to a 20-foot (6-meter) diameter opening. MACCS release path identifiers: 9599, 9609	1-REL-SD-COPEN-RCSINTACT 1-REL-SD-COPEN-RCSVENT 1-REL-SD-COPEN-HEADOFF
SGTR	SGTR is not modeled as an initiating event in shutdown POSS. Induced SGTR is considered in the model, but due to limited availability of the SGs for core cooling there are no accident sequences with induced SGTR that contribute above the quantification truncation limit.	Not applicable

Table 2-75 Key Release Path Descriptions (cont.)

Containment Failure or Bypass Mechanism	Release Path Description	Applicable Release Categories
Interfacing systems LOCA (auxiliary building failed)	The release path represents an auxiliary building overpressure failure resulting in unfiltered release to the environment at the building's roof elevation. The flow area is 10.8 square-feet (1 square-meter). MACCS release path identifier: 995	1-REL-SD-ISLOCA
Auxiliary building leakage and exhaust	These release paths represent normal leakage from the auxiliary building to the environment at the building's roof elevation (exhaust vent flow path 27 with flow area of 7.1 square-feet (0.66 square-meters) and at the two above ground level building stories (flow paths 997 and 998 with combined flow area of about 0.75 square-feet (0.07 square-meters)). MACCS release path identifiers: 27, 997, 998. Exhaust vent (27) is not active in SBO scenarios SD03, SD05, SD07, and SD09.	Contributes to many release categories but is not the dominant release path for any release category.
¹ Release category 1-REL-SD-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 consequences analysis. ² See the discussion on "Tendon Gallery Release Pathway" in Appendix D of the at-power Level 2 PRA (NRC, 2022a) for additional details on the release path modeling.		

Table 2-76 Release Category Modeled Release Paths

MELCOR Case Identifier	Release Category	Release Description	Relevant Release Paths
SD01.ptf	1-REL-SD-LCF	Containment gradual overpressure failures through release paths 820 and 821 are the dominant release paths. Other release paths include normal containment leakage (824) and auxiliary building exhaust (27).	27 820 (dominant path) 821 824
SD01-2.ptf	1-REL-SD-ECF	Energetic containment failure with release path through the tendon gallery (844). Other release paths include normal containment leakage (820, 821, 824) and auxiliary building exhaust (27).	27 820 821 824 844 (dominant path)
SD01-8.ptf	1-REL-SD-ICF-BURN	Containment failure due to energetic combustion event with release path through the tendon gallery (844). Other release paths include normal containment leakage (820, 821, 824) and auxiliary building exhaust (27).	27 820 821 824 844 (dominant path)
SD03.ptf	1-REL-SD-CIF	Containment isolation failure through release path 842. Other release paths include normal containment leakage (820, 821, 824) and auxiliary building leakage (998).	820 821 824 842 (dominant path) 998
SD05.ptf	1-REL-SD-COPEN-HEADOFF	Release through open containment equipment hatch modeled by release paths 9599 and 9609. Other release paths include normal containment leakage (820, 821, 824) and auxiliary building leakage (998).	820 821 824 9599 9609 (dominant path) 998
SD07.ptf	1-REL-SD-COPEN-RCSINTACT	Release through open containment equipment hatch modeled by release paths 9599 and 9609. Other release paths include normal containment leakage (820, 821, 824) and auxiliary building leakage (998).	820 821 824 9599 9609 (dominant path) 998
SD08-2.ptf	1-REL-SD-BMT	The modeled release paths for airborne releases are normal containment leakage (820, 821, 824) and auxiliary building exhaust (27).	27 820 (dominant path) 821 824

Table 2-76 Release Category Modeled Release Paths (cont.)

MELCOR Case Identifier	Release Category	Release Description	Relevant Release Paths
SD09.ptf	1-REL-SD-CF-BEFORE-CD	Containment overpressure failure with release path through the tendon gallery (844). Other release paths include normal containment leakage (820, 821, 824) and auxiliary building leakage (997, 998).	820 821 824 844 (dominant path) 997 998
SD11-1.ptf	1-REL-SD-ISLOCA	ISLOCA pathway to the auxiliary building and release to the environment by auxiliary building failure (995). Other release paths include auxiliary building exhaust (27) and leakage (997, 998) and normal containment leakage (820, 821, 824).	27 820 821 824 995 (dominant path) 997 998
SD12.ptf	1-REL-SD-NOCF	The release path is normal containment leakage (820, 821, 824). Auxiliary building exhaust (27) is also an active release path.	27 820 (dominant path) 821 824
SD13.ptf	1-REL-SD-COPEN-RCSVENT	Release through open containment equipment hatch modeled by release paths 9599 and 9609. Other release paths include normal containment leakage (820, 821, 824) and auxiliary building exhaust (27).	27 820 821 824 9599 9609 (dominant path)

Table 2-77 Release Path Characterization

MACCS Identifier ^d	MELCOR Flow Path Name	MELCOR Elevation (m) ^e	Plant Elevation (ft) ^e	Special instructions	Description	Location
Containment (Width: 147.5 feet, Length: 147.5 feet, Height: 402 feet)^a						
820	FL_CONT_LEAK ₁	61.11	359	MACCS identifier 820 is used for MELCOR release path FL_819 for scenarios that do not result in containment overpressure failure.	Normal containment leakage	This leakage is assumed to be distributed throughout the containment shell above grade.
821	FL_CONT_LEAK ₂	43.23	301	MACCS identifier 821 is used for MELCOR release path FL_823 for scenarios that do not result in containment overpressure failure.	Normal containment leakage	
824	FL_CONT_LEAK ₃	34.54	272	None	Normal containment leakage	
820	FL_CONT_FAIL1	4.19	173	This release path connects containment to the environment at the basemat location. The flow path opens when the containment pressure limit is reached.	Containment overpressure failure path	This leakage is located at the junction of the containment wall and containment basemat.
821	FL_CONT_FAIL2	4.19	173	This models containment overpressure failure (along with release path 820) with a larger containment failure opening size at higher pressure.		
842				MACCS identifier 842 uses the same flow path as 821 with the opening area partially opened to model a 2"-diameter isolation failure.	Containment isolation failure	
7824	FL_7824	19.0	221	Alternative containment failure path to FL821 at a higher elevation. This flow path is used for sensitivity studies.	Containment overpressure failure path	This leakage is located on the containment shell at the bottom of the equipment hatch.

Table 2-77 Release Path Characterization (cont.)

MACCS Identifier ^d	MELCOR Flow Path Name	MELCOR Elevation (m) ^e	Plant Elevation (ft) ^e	Special instructions	Description	Location
9599	FL_9599	19.65	223	The flow paths model the release to the environment from containment. The flow paths may be opened, partially opened, or not opened at all depending on the accident scenario.	Equipment hatch	This leakage is located on the containment shell above grade.
9609	FL_9609	22.70	233			
9851	FL_9851	20.18	225	The flow paths model the release to the environment from containment. The flow paths are not opened for all accident scenarios. The flow paths are assumed to be closed prior to release.	Personnel hatch	This leakage is located on the containment shell above grade.
9852	FL_9852	21.19	228			
Auxiliary Building (Width: 220 feet, Length: 125 feet, Height: 288 feet) ^a						
27	FL_027	29.87	257	The elevation of this release path is 402 feet (i.e., top of the containment).	Auxiliary building air ventilation exhaust flow path (through the plant stack)	This leakage is located on the auxiliary building ventilation duct, exhaust side. ^f
995	FL_AUX_995	29.87	257	None	Auxiliary building overpressure failure release pathway	The assumed failure elevation (i.e., roof of the auxiliary building) is reasonable but subject to considerable uncertainty.
997	FL_AUX_997	18	218	None	Normal auxiliary building leakage above grade	This leakage is assumed to be distributed throughout the auxiliary building walls and roof.
998	FL_AUX_998	25	241	None		

Table 2-77 Release Path Characterization (cont.)

MACCS Identifier ^d	MELCOR Flow Path Name	MELCOR Elevation (m) ^e	Plant Elevation (ft) ^e	Special instructions	Description	Location
Tendon gallery access shafts (Height: 220 feet)^b						
810	CONT_FAIL_BE	-	-	This flow path should not be transferred to MACCS. This release path (also referenced as FL 840 in the MELCOR model) is used for modeling releases from containment through the tendon gallery. The releases to the environment from containment through the tendon gallery are accounted for by FL_844.	Containment failure to tendon gallery	This leakage is located at the junction of the containment wall and containment basemat.
844	FL_844	18.6	220	The specified elevation is for the lower of the two associated release points. One tendon gallery access shaft releases at grade, while the other releases at the top of the control building roof. (The third access shaft connects to the auxiliary building.)	Tendon gallery access shafts (2 of 3)	Top of one of the tendon gallery access shafts connects to the environment near the containment structure at grade; another terminates at the control building roof.
Main Steam Valve Room (Height: 262 feet)^b						
372	FL_372	28.74 ^c	253 ^c	These 28 flow paths do not exist in those model versions that omit the SGs.	SG 1 SRV tailpipe	This leakage is located on top of the south main steam valve room, right up against containment
373	FL_373	28.74 ^c	253 ^c		SG 1 SRV tailpipe	
374	FL_374	28.74 ^c	253 ^c		SG 1 SRV tailpipe	
375	FL_375	28.74 ^c	253 ^c		SG 1 SRV tailpipe	
376	FL_376	28.74 ^c	253 ^c	No scenario leads to a ruptured SG tube. Therefore, no radionuclides can ever pass any of these 28 flow paths.	SG 1 lowest setpoint SRV tailpipe	
377	FL_377	28.74 ^c	253 ^c		SG 1 ARV tailpipe	

Table 2-77 Release Path Characterization (cont.)

MACCS Identifier ^d	MELCOR Flow Path Name	MELCOR Elevation (m) ^e	Plant Elevation (ft) ^e	Special instructions	Description	Location
379	FL_379	28.74 ^c	253 ^c		SG 1 secondary-side leakage	This leakage is nominally from top of the south main steam valve room, right up against containment.
472	FL_472	28.74 ^c	253 ^c		SG 2 SRV tailpipe	This leakage is on top of the north main steam valve room, right up against containment.
473	FL_473	28.74 ^c	253 ^c		SG 2 SRV tailpipe	
474	FL_474	28.74 ^c	253 ^c		SG 2 SRV tailpipe	
475	FL_475	28.74 ^c	253 ^c		SG 2 SRV tailpipe	
476	FL_476	28.74 ^c	253 ^c		SG 2 lowest setpoint SRV tailpipe	
477	FL_477	28.74 ^c	253 ^c		SG 2 ARV tailpipe	
479	FL_479	28.74 ^c	253 ^c		SG 2 secondary-side leakage	This leakage is nominally from top of the north main steam valve room, right up against containment.
572	FL_572	28.74 ^c	253 ^c		SG 3 SRV tailpipe	See FL_472
573	FL_573	28.74 ^c	253 ^c		SG 3 SRV tailpipe	See FL_472
574	FL_574	28.74 ^c	253 ^c		SG 3 SRV tailpipe	See FL_472
575	FL_575	28.74 ^c	253 ^c		SG 3 SRV tailpipe	See FL_472
576	FL_576	28.74 ^c	253 ^c		SG 3 lowest setpoint SRV tailpipe	See FL_472
577	FL_577	28.74 ^c	253 ^c		SG 3 ARV tailpipe	See FL_472
579	FL_579	28.74 ^c	253 ^c		SG 3 secondary-side leakage	See FL_479
672	FL_672	28.74 ^c	253 ^c		SG 4 SRV tailpipe	See FL_372
673	FL_673	28.74 ^c	253 ^c		SG 4 SRV tailpipe	See FL_372
674	FL_674	28.74 ^c	253 ^c		SG 4 SRV tailpipe	See FL_372
675	FL_675	28.74 ^c	253 ^c		SG 4 SRV tailpipe	See FL_372

Table 2-77 Release Path Characterization (cont.)

MACCS Identifier ^d	MELCOR Flow Path Name	MELCOR Elevation (m) ^e	Plant Elevation (ft) ^e	Special instructions	Description	Location
676	FL_676	28.74 ^c	253 ^c		SG 4 lowest setpoint SRV tailpipe	See FL_372
677	FL_677	28.74 ^c	253 ^c		SG 4 ARV tailpipe	See FL_372
679	FL_679	28.74 ^c	253 ^c		SG 4 secondary-side leakage	See FL_379
Other						
991	FL_FHB_991	39.37	288	These release paths are not used to model environmental releases in any accident simulations.	Fuel handling building ventilation intakes, exhausts, and leaks	Leakage to the environment from various locations in the fuel handling building.
910	FL_3910	39.37	288			
920	FL_3920	39.37	288			
930	FL_3930	39.37	288			
940	FL_3940	28.52	252			
941	FL_3950	20.42	226		Fuel handling building sliding door	
1310	FL_1310	36.5	279	No radionuclide transport possible due to check valves in RWST piping.	RWST vent	Yard
^a These are the structure dimensions used in the L3PRA Other Hazards analysis as of May 2016. They are provided as a starting place for the Level 3 PRA team in assessing release heights and wake effects. Note that heights are relative to mean sea level and plant grade is 220 feet.						
^b These are the release points assumed in the development of the MELCOR model (see the associated special instructions). They are provided as a starting place for the Level 3 PRA team in assessing release heights and wake effects.						
^c The release point is at elevation 262 feet (top of north main steam valve room at 260 ft plus approximately 2 feet of pipe). The value in MELCOR is more indicative of the top of the valve itself and does not include the relief valve tailpipe.						
^d There are some inconsistencies between the flow path numbering used in the MELCOR model and the numbering of the MACCS output card specifications from the model. The MACCS card numbering is used in this table. However, the same flow paths may have different identifiers in some revisions of the MELCOR model and analysis documents.						
^e MELCOR elevations are referenced to the inside-bottom of the RPV; plant elevations are referenced to mean sea level.						
^f In one deck only, namely the one used for the LOI-OC (a.k.a., ISLOCA) scenario (case 11.1), an additional flow path exists that, similar to FL_027, connects the exhaust side of the AUX ventilation duct to the environment. Its MELCOR name is FL_028. It is identical to FL_027 except for elevation: -10 m (MELCOR). It was not declared as a MACCS flow path. In the results, it never carries a flow to the environment (any flow is from the environment into the duct).						

3 SUMMARY AND CONCLUSIONS

This section summarizes the results and conclusions from the shutdown Level 2 PRA. The Level 2 PRA inherits the same modeling scope as the corresponding shutdown Level 1 PRA (NRC, 2025). The PRA model was developed to assess the risk associated with the shutdown plant operating states of a refueling outage. Six categories of initiating events are assessed: loss of offsite power, loss of residual heat removal (RHR) cooling, loss of inventory, loss of power to a 4,160V AC safety-related bus, overdraining entering mid-loop, and loss of nuclear service cooling water.

The core damage frequency (CDF) accident sequences and results from the Level 1 PRA are carried forward in the Level 2 PRA to characterize radiological releases and their frequencies. The Level 2 PRA covers a broad spectrum of severe accident sequences, and 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.

Eleven release categories are defined to characterize releases in terms of their release timing, energy, location, magnitude, and frequency. A release category for induced steam generator tube rupture was also considered, but the release frequency contribution was found to be negligible due to the limited availability of steam generators during shutdown operating states. The release category model names and brief descriptions that are used to characterize releases are provided below.

- 1-REL-SD-BMT – Containment failure due gradual basemat melt-through.
- 1-REL-SD-CF-BEFORE-CD – Late containment failure due to overpressure from evaporation and boiling of refueling cavity water with failure occurring prior to core damage.
- 1-REL-SD-CIF – Releases to the environment occur via unisolated containment penetration(s), characterized by a 2-inch diameter opening.
- 1-REL-SD-COPEN-HEADOFF – Releases to the environment via open containment equipment hatch where the reactor vessel head has been removed.
- 1-REL-SD-COPEN-RCSINTACT – Releases to the environment via open containment equipment hatch where the RCS pressure boundary is intact.
- 1-REL-SD-COPEN-RCSVENT – Releases to the environment via open containment equipment hatch where the RCS is vented.
- 1-REL-SD-ECF – Early containment failure around the time of vessel breach due to an energetic event.
- 1-REL-SD-ICF-BURN – Intermediate containment failure due to large deflagration or detonation.
- 1-REL-SD-ISLOCA – Releases to the environment via a containment bypass pathway due to a loss of inventory failure location outside of containment.

- 1-REL-SD-LCF – Late containment failure due to long-term quasi-static overpressure.
- 1-REL-SD-NOCF – Intact containment with releases to the environment via design-basis containment leakage.

The release category frequency results are summarized in Section 2.4.6. Notable observations from the release category frequency results include:

- The largest contribution to release frequency (about 49 percent) is associated with the intact containment end state.
- Containment overpressure failure due to boiling of refueling cavity inventory contributes about 17 percent to the total release frequency. This category represents a condition unique to refueling outages where a loss of all AC power and failure to restore cooling results in gradual heatup, evaporation, and boiling of refueling cavity inventory.
- Containment isolation failures contribute about 14 percent to the total release frequency. The containment isolation failures are predominately associated with human failures to isolate open penetrations at the time of the event occurrence. Most of the release category frequency is associated with station blackout (SBO) sequences.
- Combustion-induced containment failure occurring after vessel breach contributes about 14 percent to the total release frequency. An important feature of these accident sequences is the availability of containment heat removal (primarily via containment spray) contributing to combustible conditions. The result is also associated with the use of a stochastic combustion modeling approach, which was also used in the at-power Level 2 PRA portion of this study (NRC, 2022a). Refer to Section 2.6.3.7 for additional discussion of the modeling uncertainties associated with the combustion modeling approach.
- Three release categories associated with large containment openings (i.e., failure to close the open containment equipment hatch) collectively account for about four percent of the total release frequency. While these categories are not among the highest frequency contributors, they lead to the largest release magnitudes and thus are expected to be an important contributor to the reactor at-shutdown risk.
- The initiating events and plant operating states (POSSs) contributing to the significant release categories generally reflect the results of the Level 1 CDF model. Losses of offsite power and SBO sequences are significant contributors for several release categories. Other notable shutdown initiating events include loss of RHR cooling due to overdraining entering mid-loop and loss of inventory events with RCS intact. Significant accident progression sequences involve events initiated during POSSs with the refueling cavity flooded prior to refueling (POS 8E), pressurizer solid operation (POS 5A), and mid-loop operation (POS 6). These attributes reflect the dominant accident progression sequence frequency contributions, and they are also used to inform the representative accident scenarios for developing release category source terms.

Surrogate risk metric results for large early release frequency (LERF), large release frequency (LRF), and conditional containment failure probability (CCFP) are presented in Section 2.6.1. The LERF contribution is approximately 4 percent of the total release frequency. Three release

categories are identified as contributors to LERF, which include ISLOCA and two release categories with large containment openings. The LRF release categories contribute about 49 percent of the total release frequency. Note that the basemat melt-through and late containment failure release categories involve containment failures but are excluded from LRF due to the magnitude of the airborne environmental releases being comparable to the intact containment release category. The CCFP is just over 50 percent of the total release frequency with the intact containment category accounting for just over 49 percent of the frequency.

The representative accident scenarios are simulated with a termination time of 168 hours (7 days) after initiating event occurrence. In some cases (e.g., shutdown states with the refueling cavity flooded), the accident progression scenarios develop slowly. The representative MELCOR analyses show examples where core damage does not occur until after 100 hours into the scenario, and several other scenarios show containment failure occurring late in the accident progression. SBO sequences with failures to restore power are predominant contributors to these scenarios.

The implementation of FLEX mitigating strategies is expected to have a positive impact on the release frequency results; however, the FLEX strategies are not included within the scope of this model.¹⁹ Additionally, severe accident management guideline (SAMG)-directed actions could potentially terminate accidents at earlier times and either prevent or limit releases. While some post-core-damage actions are credited, the reference plant's SAMG guidance was deemed to be not applicable to shutdown conditions at the time of this analysis and is excluded from the scope of the model.²⁰

The potential for recovery, mitigating actions, and earlier termination of accidents are considered as important modeling uncertainties. A modeling uncertainty analysis related to potential credit for severe accident management actions is discussed in Section 2.6.3.3 (see MU-3.2). A related uncertainty analysis showing the impact of reducing the accident termination time is shown in Section 2.6.3.15 (see MU-15.1). These analyses show that LERF is insensitive to additional credit for recovery actions or truncating the accident termination at earlier times. However, LRF and CCFP both show potential for reductions if actions can result in recovery and earlier termination of accidents.

Sources of model uncertainty are identified and assessed in Section 2.6.3. Several model uncertainties were identified that can have significant impacts on the Level 2 PRA results. Sensitivity studies were performed for selected model uncertainties to quantify the impacts on the Level 2 PRA results. The sensitivity studies show the impacts in terms of changes to the release frequencies or the source term release magnitudes. Figure 3-1 shows the impacts of sensitivity studies on the LERF and LRF results. Figure 3-2 shows the impacts of sensitivity studies on release fractions for cesium and iodine.

¹⁹ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. The impacts of FLEX strategies are explored in the L3PRA project overview report on the shutdown Level 1, 2, and 3 PRAs.

²⁰ The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available and thus not considered at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

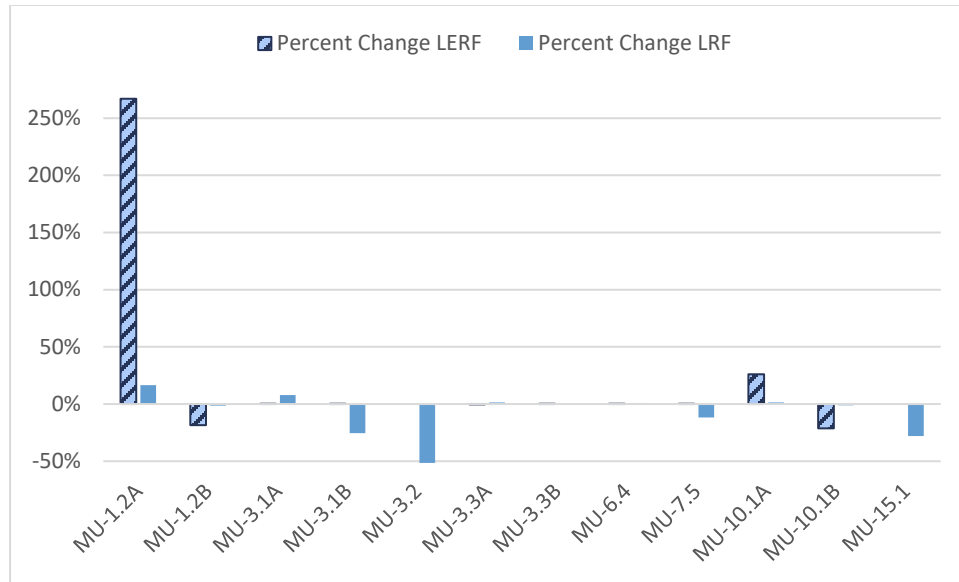


Figure 3-1 Results of Sensitivity Studies Impacting LERF and LRF

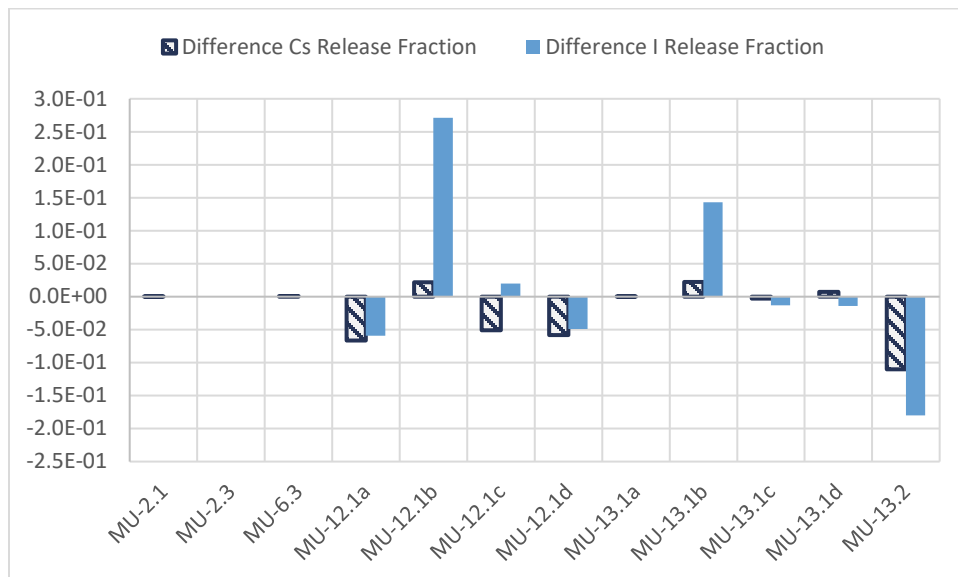


Figure 3-2 Results of Sensitivity Studies Impacting Cesium and Iodine Release Fractions

The largest impacts on LERF are associated with human failure events to close the containment equipment hatch. The LERF results are sensitive to the human error probabilities (HEPs) associated with these events. The HEP values with more restrictive available time assumptions (sensitivity case MU-1.2A) show a 267 percent increase in LERF. This represents a potentially significant increase in LERF, and the LRF results would also increase by about 16 percent. The sensitivity case covers a range of timing assumptions that influence the HEP values for failures to close containment in different POS conditions. One of the key contributors to LERF is the failure to close containment during POS 5B with reduced inventory before mid-loop, with an

open RCS (i.e., vented), and a concurrent SBO. During POS 5B the time to reach RCS saturation temperature is short (i.e., estimated to be about 30 minutes (NRC, 2025)). Assuring completion of the containment closure actions prior to reaching RCS saturation temperature can prevent exposure to harsh environmental conditions in containment that would impact operator actions. However, in assessing the containment closure failure probabilities, there may be additional time to support actions after initially reaching RCS saturation temperature as the RCS transitions from localized to bulk boiling and the containment environment heats. Operators are expected to continue attempts to close containment if there are habitable conditions to support the actions, but there are no well-established criteria for habitability after RCS boiling. The considerations of containment closure after the time of RCS saturation are solely for the purposes of the PRA and should not be viewed as superseding any established regulatory positions and industry guidance to maintain the capability to close containment before steaming inside containment. The results of the sensitivity study on human failure events to close the containment equipment hatch demonstrate that HEPs assumed for these actions can significantly impact the release frequencies.

The LERF results are also sensitive to the probabilities associated with losses of inventory occurring outside of containment resulting in a containment bypass. The upper and lower bounds of these probabilities can result in a 25 percent increase or 21 percent decrease in LERF, as shown in sensitivity cases MU-10.1A and MU-10.1B. The LERF results are insensitive to the other sensitivity studies that were performed.

The largest impact on LRF is associated with potential credit for post-core-damage recovery actions (e.g., actions to control containment pressure or combustion) that could result in a 56 percent reduction in LRF (as shown in sensitivity case MU-3.2). Other potential LRF reductions are associated with alternate HEP values for modeled post-core-damage actions to recovery injection to the reactor vessel (MU-3.1B) and alternate accident termination assumptions (MU-15.1). The largest LRF increase is associated with available time assumptions for containment closure HFEs (MU-1.2A).

The modeling uncertainty areas that show large potential impacts on release magnitudes are related to modeling assumptions about release pathways. Sensitivity case MU-12.1, discussed in Section 2.6.3.12, considers variations of the containment isolation failure pathway. A larger size release pathway and the effects of filters are considered. MU-12.1a shows the impact of effective filters without clogging or failure of the filters, though such effectiveness is acknowledged to be overly optimistic. The filters would be challenged by deflagration events in containment resulting in pressure spikes that could fail the filters, and the predicted mass deposited in the filters would likely exceed the capacity and cause clogging. MU-12.1b shows the large-sized, unfiltered release pathway could significantly increase release magnitudes, particularly for iodine. Two other sensitivity cases (MU-12.1c and MU-12.1d) consider variations of the containment isolation failure pathway occurring in a different type of accident scenario (i.e., loss of RHR cooling during mid-loop). While the different scenario does influence the releases, the base case release scenario (SBO during POS 8E) was selected because it was more representative of the dominant cutset results for the release category.

Sensitivity case MU-13.1 considers variations of the release pathway modeling and location for the late containment overpressure failure (results shown in Table 2-63) and combustion-induced containment failure (results shown in Table 2-64) scenarios. Several variations of the combustion-induced containment failure scenario were evaluated. The largest impact on the release is associated with the timing of containment failure. Case SD01.4 assumes an earlier

time of containment failure (i.e., four hours after vessel breach). The base case scenario (SD01.8) assumes combustion-induced containment failure 35 hours after vessel breach, which is informed by the stochastic modeling of shutdown combustion scenarios discussed further in Section 2.6.3.7.

One more sensitivity case was shown to have a significant impact on the release magnitude. Case MU-13.2 considers an open containment release via the opened containment equipment hatch compared to the opened personnel airlock. The releases to the environment are large for both cases, but the open equipment hatch leads to larger releases. The open equipment hatch release pathway was selected to represent the open containment release categories.

Beyond these sensitivity study results, there are several important model uncertainties that were identified and assessed qualitatively. These are areas with potentially significant impacts on the Level 2 PRA results that could benefit from additional evaluation in future studies and are described below. Appendix A lists these important areas of model uncertainty and additional candidates for future work.

- Modeling related to ISLOCA scenarios can significantly influence the Level 2 PRA results and, in particular, the LERF results. Sensitivity studies on the probabilities related to ISLOCA occurrence are presented in Section 2.6.3.10, which demonstrate the impacts on LERF. Several other modeling assumptions related to ISLOCA, such as the size and location of the release pathway, deposition, filtering, and robustness of the release path building structures, can significantly influence the releases to the environment.
- RHR relief valve failure modeling can influence the likelihood of ISLOCA events and influence environmental releases. Similar to the modeling of relief valve failures in the at-power PRA, this is an area of uncertainty that is proposed for future work.
- Model uncertainties associated with early containment failure from energetic phenomena are discussed in Section 2.6.3.6. Early containment failure caused by in-vessel steam explosion is represented in the model results, but modeling uncertainties could influence the results. While energetic phenomena that could challenge containment are expected to be rare, they could lead to large releases. The characterization of containment failure associated with these phenomena (e.g., size and location of failure) is also uncertain.
- The uncertainties associated with modeling the containment failure release pathway are discussed in Section 2.6.3.13. Submergence of the failure location and flows between the connected structures introduce modeling complications and concerns about impacts on the releases to the environment. Sensitivity studies, including modeling an alternative failure location and pathway, were completed to provide insights into the impacts of modeling alternatives on the releases. However, the MELCOR modeling assumptions for the release pathway could benefit from additional study to ensure all uncertainties are understood in terms of how the results could be influenced under other accident scenarios and conditions (e.g., initial conditions, availability of equipment and mitigation functions, accident timing and progression) beyond those addressed in the sensitivity studies.

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APPENDIX A CANDIDATES FOR FUTURE WORK

The shutdown Level 2 PRA provides a comprehensive assessment of risks associated with radiological releases during shutdown operations for a refueling outage; however, there are several areas that could benefit from additional study. These areas are offered as potential improvements that can be pursued in future revisions or follow-on work to this study. Table A-1 shows a list of specific areas that could be the focus of future work.

Table A-1 Candidates for Future Work

Topic Area	Proposed Area for Future Work
Containment event tree branch logic	The CET logic assumes that containment overpressure before core damage could potentially occur in the case of containment isolation failure, but that most post-core-damage phenomena do not apply. Therefore, for example, some fraction of accidents with a containment isolation failure would end up binned in release category 1-REL-SD-CF-BEFORE-CD, but none would ever end up binned as early, intermediate, or late containment rupture. These assumptions may be somewhat inconsistent with one another, but it was never systematically investigated to what extent a small isolation failure would impact containment pressurization rates and thereby permit or allow such other failure modes. Further investigation with a MELCOR sensitivity calculation could inform how to consistently treat isolation failure sequences.
Containment event tree modeling of RCS status	Sub-classification of open containment release categories currently fails to consider some events which could result in a leak in the RCS, and which might qualify to alter sequences from a “containment open, RCS intact” release to a “containment open, RCS vented/leak” release, since open containment sequences skip most top events in the CET. Further development of those branches (e.g., to not skip the 1-L2-DET-PRESVE-SD and 1-L2-DET-CONTE-SD top events) might help to refine estimates of the probabilities of these release categories.
Core damage arrest modeling assumptions	Treatment of in-vessel core damage arrest currently merges together the operator actions to restore injection with the phenomenological issue of core coolability at the time when injection is restored. Separation of the two issues (e.g., by developing a fault tree for the corresponding top event) would render the uncertainty and importance results more tractable to analysis (i.e., which individual components are more important to the results).
Emergency action level declaration timing	As discussed in Section 2.6.3.14, uncertainty exists with regard to some of the EAL declaration timings. Assumptions about the EAL declarations are made based on the available information, but this topic could benefit from additional study and assessment of the impacts on the consequence analysis modeling.
Equipment survivability	The evaluation of equipment survivability and lack of detailed information is discussed in Section 2.3.6. As an example, insufficient cable routing information is available to address situations where the instrument is not damaged, but it's cabling elsewhere in the plant (or within containment) is. However, qualitative assessments of the credited equipment and general characterizations of severe accident conditions are performed to inform the modeling of post-core-damage operator actions. A more detailed evaluation of equipment survivability could affect the context for modeled post-core-damage operator actions. The study could benefit from enhancement of the

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
	equipment survivability methods beyond the current state-of-practice. This is also identified as a model enhancement in the at-power Level 2 PRA for internal events and floods (NRC, 2022a).
Energetic phenomena causing containment failure	While energetic phenomena occurring around the time vessel breach are expected to be rare, additional study could improve the characterization and impacts of these events. Such events could include steam explosions, high pressure melt ejections, vessel rocketing, or direct containment heating. The induced containment failure characterization (e.g., size and location of failure) could also be improved with additional study.
Extended ECCS injection	The timing of ECCS injection is consistent with the shutdown Level 1 PRA (NRC, 2025) with injection ending when the RWST low-low level indication is reached. Injection could possibly be extended to credit additional usable inventory from the RWST. The additional injection time (no more than 2-3 hours) is not expected to prevent core damage but could influence the accident progression timing and releases. Additional study of capabilities to refill the RWST for extending or recovering injection could also be considered.
Gravity injection	Several MELCOR cases and sensitivities were developed with gravity-driven injection from the RWST. The timing of the injection can potentially influence the accident progression. Some cases show limited effectiveness of gravity injection due to rising RCS or containment pressure that prevents injection. Additional study of variations of conditions and timing could provide insights on the most effective uses of gravity injection.
Habitability limits for containment closure	The habitability limits for POS 5B actions to close containment are based on an evaluation of increasing containment air temperature (see Appendix D). However, other adverse conditions, including those associated with the onset of RCS boiling (e.g., noise, humidity, visibility, and fission products in containment) can impact the performance of containment closure actions. Additional study of the containment conditions in various accident scenarios could improve the basis for limits on the time available for containment closure.
HRA for post-core-damage actions	While the L3PRA project developed an approach for post-core-damage HRA, the model results are shown to be sensitive to the resulting HEP values making them candidates for further study. Also, while the reference plant capabilities circa 2012 limit the scope of actions to be credited, ²¹ future studies may require analyses of different types of actions.
Impact of air ingress on a degraded core	The potential impacts of air ingress on a degraded core are discussed in Section 2.6.3.4. This could lead to enhanced ruthenium release under air-oxidation conditions. While additional MELCOR sensitivity studies are not expected to provide new insights, the uncertainty is acknowledged for the potentially significant impact on health consequences of a release.
ISLOCA modeling	Aspects of the ISLOCA modeling could be further explored to understand the impacts specifically for the shutdown ISLOCA scenario. The modeling issues could include the size and location of the release pathway, deposition, filtering, and robustness of the release path building structures.
Long-term recovery actions and accident termination	Several significant shutdown accident scenarios involve slowly developing accident progression with losses of all AC power and failures to restore power. Given the timeframe (e.g., core damage occurring ~100 hours after

²¹ Note that this study is based on information that reflects the reference plant as it was designed and operated as of 2012. It does not reflect the potential impact of FLEX strategies implemented after 2012. Also, SAMG guidance documents available for plant operation in 2012 did not include shutdown operations within their scope.

Table A-1 Candidates for Future Work (cont.)

Topic Area	Proposed Area for Future Work
	event initiation), potential recovery actions could be considered for terminating or preventing releases. The potential impacts are explored in sensitivity studies (see MU-3.2 in Section 2.6.3.3 and MU-15.1 in Section 2.6.3.15). New or updated information that could support modeling of recovery actions (e.g., implementation of FLEX mitigating strategies or revisions to SAMGs) could provide a basis for additional study of this topic.
MELCOR modeling assumptions for containment failure pathway	A series of sensitivity studies are presented in Section 2.6.3.13 to address modeling uncertainties related to how the containment failure location pathway is modeled in MELCOR. The uncertainties associated with the release pathway are influenced by the specific accident scenarios that are being evaluated. Additional study could improve understanding of the impacts of these modeling choices.
Other outage types	As stated in the assumptions and limitations in Section 1.1, the shutdown Level 2 PRA scope is limited to the analysis of a refueling outage, consistent with the shutdown Level 1 PRA. The inclusion of other outage types could involve other conditions and considerations that are not addressed in the refueling outage POSSs. Specific to the Level 2 PRA, the core radionuclide inventories could be different depending on the time in the cycle when an outage occurs.
Nozzle dam failure	The potential for steam generator nozzle dam failure was considered as a sensitivity study in the Level 1 PRA (NRC, 2025). Nozzle dam failure can significantly increase the shutdown CDF, but the nozzle dam failure scenario was not evaluated in the Level 2 PRA. The predominant effect is expected to be an increase of release frequencies. However, the nozzle dam failure can also influence the buildup of combustible gases in containment, the transport of fission products from the RCS to containment, and possibly other accident progression issues.
Re-criticality of a damaged core	Re-criticality during reflood of a damaged core was not explicitly modeled or considered. Additional effort in this area could lead to additional insights.
RHR relief valve failure modeling	The RHR relief valve failure can significantly influence the accident progression and release frequency for ISLOCA sequences. The RHR relief valve failure is treated in the same way as the power-operated relief valve failure in the at-power Level 2 PRA (NRC, 2022a). Limited operating experience data and uncertainty about the number of cycles experienced by a valve in a given sequence contribute to the difficulty in making failure probability estimates.
Static masses	The use of static chemical class mass inventories, rather than accounting for time dependence, was done to simplify the modeling. The elemental masses guide the chemical behavior and transport as modeled in MELCOR. Much of the elemental mass is comprised of stable isotopes. However, accounting for time-dependent decay of radionuclides could have non-trivial impacts for some shutdown scenarios, particularly those slow-developing scenarios.

APPENDIX B

PRA IMPORTANCE MEASURES AND DOMINANT CUT SETS

PRA importance measures ranked by Fussell-Vesely (FV) importance for the combined Level 1 and Level 2 PRA model are provided in Section B.1 . The top 20 contributing cutsets for each of the release categories are shown in Section B.2 .

B.1 Risk-Significant Basic Events for Integrated Level 1 and Level 2 PRA Model

A list of the most risk-significant basic events for the integrated shutdown Level 1 and Level 2 PRA model, based on FV importance is provided in Table B-1. The FV importance measure provides the percent contribution of the basic event to the overall CDF and is calculated using the following equation:

$$FV_i = F_i(x)/F(x)$$

where,

$F_i(x)$ is the calculated CDF of those cutsets that contain basic event i , and
 $F(x)$ is the overall CDF (i.e., the quantification of all cutsets).

In addition to the FV values, Table B-1 includes the probability, risk achievement worth (RAW) importance measure, and description for each basic event. The RAW importance measure is defined as:

$$RAW = F(1)/F(x)$$

where,

$F(1)$ is the calculated CDF with the basic event failure probability set to 1.0, and
 $F(x)$ is the overall CDF (i.e., the quantification of all cutsets).

Table B-1 includes all Level 1 and Level 2 PRA basic events with FV importance greater than 0.005, for the combined set of significant release categories. 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% 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.

This assessment yields the following set of significant release categories:

- 1-REL-SD-CF-BEFORE-CD
- 1-REL-SD-CIF
- 1-REL-SD-COPEN-HEADOFF
- 1-REL-SD-COPEN-RCSINTACT

- 1-REL-SD-COPEN-RCSVENT
- 1-REL-SD-ICF-BURN
- 1-REL-SD-ISLOCA

Table B-1 Risk-Significant Basic Events for Combined Level 1 and Level 2 PRA for Significant Release Categories

Basic Event Name	Prob.	FV	RAW	Description
1-POT-4	5.53E-02	8.79E-01	1.60E+01	PLANT OPERATING TYPE - 4 (REFUELING)
1-IE-SD-LOOP	1.63E-01	6.57E-01	4.37E+00	LOSS OF OFF-SITE POWER - SHUTDOWN
1-POS-8E-POT-4	9.37E-02	5.31E-01	6.13E+00	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
1-EPS-XHE-XL-NR04H	6.98E-01	5.07E-01	1.22E+00	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
1-OEP-XHE-XL-NR02H	3.26E-01	4.93E-01	2.02E+00	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
OEP-VCF-LP-GRWR	4.10E-01	3.98E-01	1.57E+00	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
1-LPI-XHE-XM-GFEED-LOOP	5.50E-01	2.79E-01	1.23E+00	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
1-L2-BE-H2DET-L-CHR-SD	7.90E-01	2.68E-01	1.06E+00	Late Detonation with containment heat removal (Shutdown)
1-CIS-XHE-XL-ISOLPEN-HD	5.02E-01	2.63E-01	1.07E+00	Operators fail to isolate containment small containment penetrations (high dependency)
1-L2-OP-EDMGB-SD4	7.41E-01	2.49E-01	1.07E+00	Operator fails to recover injection to vessel (LPSD case 4)
1-L2-BE-H2IGSRC-L-AC-SD	9.56E-01	2.37E-01	1.01E+00	Ignition Source in Containment Late with AC Power (Shutdown)
1-L2-BE-H2IGSRC-E-AC-SD	9.56E-01	2.27E-01	1.00E+00	Ignition Source in Containment near time of vessel breach with AC Power (Shutdown)
1-L2-BE-H2IGNSRC-VE-AC	9.56E-01	2.26E-01	1.00E+00	Ignition Source in Containment before vessel breach, with AC Power
1-EPS-DGN-FR-G4002___	3.30E-02	2.12E-01	7.22E+00	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-OEP-XHE-XX-NR02H1	2.73E-01	1.97E-01	1.53E+00	Convolution correction factor for failure to recover offsite power in 2 hours
1-POS-5A-POT-4	4.13E-02	1.79E-01	5.15E+00	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
1-ACP-CRB-CC-BA0301___	5.35E-03	1.21E-01	2.35E+01	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-L2-BE-H2IGN-VE-CHR-SD	5.40E-01	1.20E-01	1.00E+00	Burn in Containment before vessel breach (with containment heat removal) (Shutdown)
1-IE-SD-ODML-EARLY	5.88E-01	1.20E-01	1.08E+00	OVER DRAINING IN MID-LOOP - Before Refueling
1-POS-6-POT-4-MLDRL	4.07E-03	1.20E-01	3.03E+01	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)

Table B-1 Risk-Significant Basic Events for Combined Level 1 and Level 2 PRA for Significant Release Categories (cont.)

Basic Event Name	Prob.	FV	RAW	Description
1-RHR-XHE-XL-LTREC	1.55E-02	1.04E-01	7.59E+00	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-OA-ALIGNPW-G-H	9.20E-02	8.72E-02	1.86E+00	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
1-ACP-CRB-CF-A205301	3.50E-04	8.50E-02	2.43E+02	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
1-L2-BE-RHRVSTUCK-SD	6.69E-02	7.55E-02	1.95E+00	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
1-EPS-SEQ-FO-1821U302	3.33E-03	7.53E-02	2.35E+01	SEQUENCER B FAILS TO OPERATE
1-LPI-MDP-CF-START	4.88E-05	6.96E-02	1.42E+03	CCF OF RHR PUMPS A & B TO START
1-IE-SD-LOINV-OP	6.07E-01	6.82E-02	1.04E+00	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
1-LOI-XHE-XD-POS5A	2.50E-02	6.81E-02	3.66E+00	OPERATOR FAILS TO DIAGNOSE LOI EARLY
1-EPS-DGN-FS-G4002___	2.94E-03	6.55E-02	2.32E+01	DG1B RANDOMLY FAILS TO START
1-L2-OP-EDMGC-SD1	2.40E-01	6.31E-02	9.84E-01	Operator fails to initiate containment sprays (LPSD case 1)
1-RHR-XHE-XL-TRN-MLDRL	1.25E-02	5.88E-02	5.65E+00	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
1-IE-SD-LORHR	3.67E-01	5.78E-02	1.10E+00	LOSS OF RHR - SHUTDOWN
1-IE-SD-LONSCW	1.00E+00	5.61E-02	1.00E+00	LOSS OF NSCW DURING SHUTDOWN
1-POS-5B-POT-4	5.51E-02	4.92E-02	1.84E+00	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
1-RHR-XHE-XM-MLDRL-I	1.02E-02	4.72E-02	5.58E+00	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - mid-loop
1-IE-SWS-MDP-CR-123456	3.03E-05	4.66E-02	1.53E+03	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
1-L2-OP-EDMGB-SD3	5.63E-01	4.56E-02	1.04E+00	Operator fails to recover injection to vessel (LPSD case 3)
1-L2-OP-EDMGB-SD2	5.36E-01	4.55E-02	1.04E+00	Operator fails to recover injection to vessel (LPSD case 2)
1-CNT-XHE-XM-SBO2	1.30E-01	4.40E-02	1.19E+00	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
1-L2-BE-H2IGSRC-L-NAC-SD	3.00E-01	4.35E-02	1.10E+00	Ignition Source in Containment Late without AC Power (Shutdown)
1-RHR-XHE-XL-LTREC-LT5H	1.51E-01	4.20E-02	1.24E+00	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
1-EPS-SEQ-CF-FOAB	2.15E-04	4.05E-02	1.89E+02	CCF OF SEQUENCERS TO OPERATE

Table B-1 Risk-Significant Basic Events for Combined Level 1 and Level 2 PRA for Significant Release Categories (cont.)

Basic Event Name	Prob.	FV	RAW	Description
1-IE-SD-LO4160C	3.40E-01	3.48E-02	1.07E+00	LOSS OF 4160 VAC BUS - SHUTDOWN
1-RHR-XHE-XM-TRIP-MD	1.46E-01	3.00E-02	1.18E+00	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
1-POS-6-POT-4	1.93E-02	2.75E-02	2.40E+00	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
1-HPI-XHE-XM-FFEEED-MLDRL	7.95E-03	2.54E-02	4.18E+00	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
1-L2-BE-H2IGN-L-NACNP-SD	9.90E-01	2.45E-02	1.00E+00	Late burn given ignition source (no AC without prior burn) - LPSD
1-POS-9-POT-4	4.13E-02	2.39E-02	1.56E+00	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
1-L2-BE-CBYP-OP	1.67E-01	2.11E-02	1.06E+00	LOCA Bypasses Containment (LOINV-OP)
1-L2-BE-H2IGN-L-NACPB-SD	8.00E-01	1.90E-02	1.01E+00	Late burn given ignition source (no AC with prior burn) - LPSD
1-POS-10-POT-4	3.44E-02	1.87E-02	1.52E+00	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)
1-L2-BE-H2IGSRC-E-NAC-SD	5.00E-01	1.83E-02	9.93E-01	Ignition Source in Containment near time of vessel breach without AC Power (Shutdown)
1-L2-BE-INDHLF-MP-SD	5.00E-01	1.78E-02	1.02E+00	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
1-CNT-XHE-XM-SBO1	2.30E-01	1.74E-02	1.05E+00	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
1-POS-12-POT-4	7.44E-02	1.58E-02	1.20E+00	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
1-LPI-XHE-XM-GFEED-LD	4.52E-01	1.42E-02	1.02E+00	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED (LOW DEPENDENCY)-LOI
1-POS-4-POT-4	5.51E-03	1.36E-02	3.45E+00	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
1-RHR-XHE-XL-TRN-POS6	1.25E-02	1.35E-02	2.07E+00	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
1-CNT-XHE-XM-SBO3	2.90E-02	1.34E-02	1.02E+00	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
1-LPI-XHE-XM-GFEED-MD	5.05E-01	1.34E-02	1.01E+00	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
1-CIS-XHE-XL-ISOLPEN-LD	5.29E-02	1.22E-02	1.09E+00	Operators fail to isolate containment small containment penetrations (low dependency)

Table B-1 Risk-Significant Basic Events for Combined Level 1 and Level 2 PRA for Significant Release Categories (cont.)

Basic Event Name	Prob.	FV	RAW	Description
1-OEP-XHE-XL-NR01H	5.20E-01	1.10E-02	1.01E+00	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
1-POS-7-POT-4	2.07E-02	1.02E-02	1.48E+00	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
1-NSCWCT-SPRAY	9.04E-01	1.01E-02	1.00E+00	NSCW CTS IN SPRAY MODE (FRACTION OF TIME)
1-OA-OSW-----H	2.00E-02	9.47E-03	1.46E+00	OPERATORS FAIL TO ESTABLISH SINGLE PUMP NSCW PUMP OPERATION
1-EPS-DGN-FR-G4001__	3.30E-02	9.43E-03	1.28E+00	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-POS-11-POT-4	3.31E-02	8.60E-03	1.25E+00	REFILLING REACTOR (POS-11) AFTER REFUELING (POT-4)
1-OEP-VCF-LP-CLOPT	5.30E-03	8.19E-03	2.54E+00	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
1-SWS-MOV-CC-1669A__	3.53E-04	7.41E-03	2.20E+01	NSCW CT B SPRAY VALVE HV1669A FAILS TO OPEN ON DEMAND
1-ACP-XHE-XL-ALTAC-6H	2.00E-03	6.42E-03	4.21E+00	FAIL TO ALIGN ALTERNATE OFFSITE POWER BY PW VIA SAT OR OPPOSITE TRAIN RAT IN 6 HOURS
1-LOI-XHE-XM-ISO2-LT	2.44E-02	6.30E-03	1.25E+00	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
1-LPI-MDP-CF-RUN	3.94E-06	6.25E-03	1.58E+03	CCF OF RHR PUMPS A & B TO RUN
1-LPI-MDP-FS-RHRB__	1.00E-03	5.85E-03	6.84E+00	RHR PUMP B RANDOMLY FAILS TO START
1-L2-BE-H2CF-L-NACNPB-SD	6.22E-01	5.49E-03	1.00E+00	Late CF from Burn (without AC power, without prior burn) (Shutdown)
1-ACP-CRB-CC-AA0205__	5.35E-03	5.07E-03	1.94E+00	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN

B.2 Dominant Cutsets by Release Category

The top 20 contributing cutsets for each release category are shown in the following tables:

Table B-2:	Release Category 1-REL-SD-BMT
Table B-3:	Release Category 1-REL-SD-CF-BEFORE-CD
Table B-4:	Release Category 1-REL-SD-CIF
Table B-5:	Release Category 1-REL-SD-COPEN-HEADOFF
Table B-6:	Release Category 1-REL-SD-COPEN-RCSINTACT
Table B-7:	Release Category 1-REL-SD-COPEN-RCSVENT
Table B-8:	Release Category 1-REL-SD-ECF
Table B-9:	Release Category 1-REL-SD-ICF-BURN
Table B-10:	Release Category 1-REL-SD-ISLOCA
Table B-11:	Release Category 1-REL-SD-LCF
Table B-12:	Release Category 1-REL-SD-NOCF

Each cutset shows the combined cutset frequency and the percent contribution to the release category frequency. Note that a preceding "/" character in the basic event name indicates success of that event. The description column always contains the basic event failure description regardless of whether the event represents success or failure in the cutset.

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT

#	Prob/Freq	% Contr.	Cut Set	Description
1	2.29E-09	1.9		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
2	1.95E-09	1.6		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
3	1.86E-09	1.5		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
4	1.59E-09	1.3		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
5	1.34E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
6	1.34E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
7	1.30E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	1.30E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
9	1.21E-09	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.36E-01		1-L2-OP-EDMGB-SD2	Operator fails to recover injection to vessel (LPSD case 2)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
10	1.21E-09	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.36E-01		1-L2-OP-EDMGB-SD2	Operator fails to recover injection to vessel (LPSP case 2)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
11	1.14E-09	0.9		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
12	1.12E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
13	1.12E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
14	1.11E-09	0.9		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
15	9.72E-10	0.8		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
16	9.43E-10	0.8		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	9.31E-01		/1-L2-BE-H2CF-L-CHRPB-SD	Late CF from Burn (with CHR, with prior burn) (Shutdown)
	2.10E-01		/1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
17	9.32E-10	0.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
18	9.32E-10	0.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	8.22E-10	0.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
20	8.22E-10	0.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown

Table B-2 Top 20 Cut Sets for Release Category 1-REL-SD-BMT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.00E-01		1-L2-BE-BMT-NCHR-SD	BMT Occurs Given No CHR and No Preceding Overpressure Failure (Shutdown)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD

#	Prob/Freq	% Contr.	Cut Set	Description
1	1.89E-07	9.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	1.57E-07	7.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
3	1.56E-07	7.5		
	1.00E+00		1-IE-SD-LONSCW	LOSS OF NSCW DURING SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	3.03E-05		1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	1.55E-07	7.5		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
5	1.28E-07	6.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
6	1.12E-07	5.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
7	9.22E-08	4.5		
	3.67E-01		1-IE-SD-LORHR	LOSS OF RHR - SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	9.19E-08	4.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
9	8.55E-08	4.1		
	3.40E-01		1-IE-SD-LO4160C	LOSS OF 4160 VAC BUS - SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
10	6.99E-08	3.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
11	6.73E-08	3.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	6.17E-08	3.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002__	DG1B RANDOMLY FAILS TO START

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
13	5.72E-08	2.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
14	5.51E-08	2.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
15	5.05E-08	2.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002__	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
16	4.24E-08	2.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
17	4.10E-08	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
18	3.47E-08	1.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
19	2.52E-08	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
20	2.06E-08	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)

Table B-3 Top 20 Cut Sets for Release Category 1-REL-SD-CF-BEFORE-CD (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF

#	Prob/Freq	% Contr.	Cut Set	Description
1	1.90E-07	11.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	1.56E-07	9.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
3	1.13E-07	6.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
4	9.45E-08	5.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
5	9.26E-08	5.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
6	7.05E-08	4.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
7	6.22E-08	3.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
8	5.77E-08	3.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
9	5.62E-08	3.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
10	5.09E-08	3.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
11	4.27E-08	2.5		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	3.50E-08	2.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
13	3.50E-08	2.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
14	3.30E-08	1.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.59E-01		/1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
15	3.09E-08	1.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002___	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
16	2.54E-08	1.5		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301___	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
17	2.45E-08	1.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	5.29E-02		1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
18	2.12E-08	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case

Table B-4 Top 20 Cut Sets for Release Category 1-REL-SD-CIF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	2.08E-08	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
20	2.00E-08	1.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	5.29E-02		1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF

#	Prob/Freq	% Contr.	Cut Set	Description
1	1.13E-08	14.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VPF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	9.27E-09	12		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VPF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
3	6.74E-09	8.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301___	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
4	5.51E-09	7.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
5	4.71E-09	6.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
6	4.19E-09	5.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
7	3.86E-09	5.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
8	3.70E-09	4.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002__	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
9	3.43E-09	4.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
10	3.03E-09	3.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002__	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
11	2.54E-09	3.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	2.08E-09	2.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
13	2.02E-09	2.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
14	1.65E-09	2.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
15	1.51E-09	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
16	1.24E-09	1.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
17	9.41E-10	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
18	8.31E-10	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	7.70E-10	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
20	6.80E-10	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-5 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-HEADOFF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.90E-02		1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002____	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	9.37E-02		1-POS-8E-POT-4	REFUELING OPERATION (POS-8E) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT

#	Prob/Freq	% Contr.	Cut Set	Description
1	4.07E-08	15.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	3.05E-08	11.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
3	2.42E-08	9.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
4	1.69E-08	6.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
5	1.51E-08	5.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
6	1.33E-08	5.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
7	1.31E-08	4.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	9.13E-09	3.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
9	7.46E-09	2.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.31E-02		1-POS-11-POT-4	REFILLING REACTOR (POS-11) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
10	7.26E-09	2.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
11	5.43E-09	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	5.43E-09	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	5.51E-03		1-POS-4-POT-4	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
13	4.58E-09	1.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.31E-02		1-POS-11-POT-4	REFILLING REACTOR (POS-11) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
14	3.38E-09	1.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
15	3.23E-09	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	5.51E-03		1-POS-4-POT-4	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
16	2.98E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.94E-03		1-EPS-DGN-FS-G4002	DG1B RANDOMLY FAILS TO START
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.20E-02		1-OA-ALIGNPW-G-H	OPERATOR FAILS TO ALIGN PLANT WILSON - General case
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
17	2.50E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.31E-02		1-POS-11-POT-4	REFILLING REACTOR (POS-11) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
18	2.26E-09	0.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	5.51E-03		1-POS-4-POT-4	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	2.01E-09	0.8		

Table B-6 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSINTACT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	5.51E-03		1-POS-4-POT-4	COOLDOWN TO AMBIENT TEMP WITH RHR SYSTEM (POS-4) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
20	1.81E-09	0.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	1.30E-01		1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	4.00E-04		1-EPS-TNK-MA-DFOSTKB_	TR B. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-002 IN MAINTENANCE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT

#	Prob/Freq	% Contr.	Cut Set	Description
1	2.20E-08	16.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
2	1.65E-08	12.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	4.13E-02		1-POS-9-POT-4	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
3	1.35E-08	10.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
4	1.01E-08	7.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	4.13E-02		1-POS-9-POT-4	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
5	3.63E-09	2.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4001__	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	1.02E-01		1-OEP-XHE-XX-NR01H2	CONVOLUTION FACTOR FOR 2FTR-OPR (1HR AVAIL)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
6	2.72E-09	2.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4001__	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	1.02E-01		1-OEP-XHE-XX-NR01H2	CONVOLUTION FACTOR FOR 2FTR-OPR (1HR AVAIL)
	4.13E-02		1-POS-9-POT-4	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
7	2.39E-09	1.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	2.26E-01		1-OEP-XHE-XX-NR01H0	CONVOLUTION FACTOR FOR CCF-OPR (1HR Avail)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	2.27E-09	1.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	2.90E-02		1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
9	2.07E-09	1.6		
	2.09E-02		1-IE-SD-LOINV-DS	LOSS OF INVENTORY - SHUTDOWN - inadvertent displacement of RCS inventory
	5.05E-01		1-CNT-XHE-XM-H1-HD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (high dependency)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.20E-03		1-RHR-XHE-XM-M5LOI-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -MODE 5
	5.36E-02		1-RHR-XHE-XM-TRIP-LD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (LOW DEPENDENCY)
10	1.80E-09	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
11	1.79E-09	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	2.26E-01		1-OEP-XHE-XX-NR01H0	CONVOLUTION FACTOR FOR CCF-OPR (1HR Avail)

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.13E-02		1-POS-9-POT-4	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	1.39E-09	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.90E-02		1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
13	1.35E-09	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	4.13E-02		1-POS-9-POT-4	DRAINING REACTOR TO MID-LOOP (POS-9) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
14	1.30E-09	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	2.26E-01		1-OEP-XHE-XX-NR01H1	CONVOLUTION FACTOR FOR 1FTR-OPR (1HR AVAIL)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
15	1.30E-09	1.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4001__	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	2.26E-01		1-OEP-XHE-XX-NR01H1	CONVOLUTION FACTOR FOR 1FTR-OPR (1HR AVAIL)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
16	1.23E-09	0.9		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	1.51E-01		1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	7.75E-03		1-HPI-XHE-XM-FFFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS5A
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.52E-01		1-LPI-XHE-XM-GFEED-LD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
17	1.20E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.68E-05		1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
18	1.15E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.53E-05		1-EPS-MDP-FS-XFERPPS _CC	CCF OF DG FUEL TRANSFER PUMPS TO START
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.20E-01		1-OEP-XHE-XL-NR01H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	1.12E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

Table B-7 Top 20 Cut Sets for Release Category 1-REL-SD-COPEN-RCSVENT (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
20	1.12E-09	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
	2.30E-01		1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF

#	Prob/Freq	% Contr.	Cut Set	Description
1	3.36E-11	6.6		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
2	2.86E-11	5.6		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
3	2.74E-11	5.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
4	2.34E-11	4.6		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
5	1.68E-11	3.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
6	1.61E-11	3.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-04		1-L2-BE-H2CF-E-GEN-SD	Global Deflagration Fails Containment At/Around VB (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
7	1.43E-11	2.8		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
8	1.37E-11	2.7		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-04		1-L2-BE-H2CF-E-GEN-SD	Global Deflagration Fails Containment At/Around VB (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
9	1.31E-11	2.6		

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-04		1-L2-BE-H2CF-E-GEN-SD	Global Deflagration Fails Containment At/Around VB (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
10	1.12E-11	2.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	1.00E-04		1-L2-BE-H2CF-E-GEN-SD	Global Deflagration Fails Containment At/Around VB (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
11	1.01E-11	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
12	9.75E-12	1.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
13	9.12E-12	1.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.36E-01		1-L2-OP-EDMGB-SD2	Operator fails to recover injection to vessel (LPSD case 2)
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
14	8.88E-12	1.8		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
15	8.71E-12	1.7		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	5.05E-01		1-LPI-XHE-XM-GFEED-MD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
16	8.44E-12	1.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
17	8.03E-12	1.6		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	1.00E-04		1-L2-BE-H2CF-E-GEN-SD	Global Deflagration Fails Containment At/Around VB (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSP case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
18	7.57E-12	1.5		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
19	7.42E-12	1.5		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	5.05E-01		1-LPI-XHE-XM-GFEED-MD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
20	7.01E-12	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown

Table B-8 Top 20 Cut Sets for Release Category 1-REL-SD-ECF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	2.00E-04		1-L2-BE-IVSE-LP-SD	IVSE Fails Containment (Low RCS Pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN

#	Prob/Freq	% Contr.	Cut Set	Description
1	9.23E-08	5.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
2	7.86E-08	4.6		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
3	7.53E-08	4.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
4	6.41E-08	3.8		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
5	4.61E-08	2.7		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
6	4.47E-08	2.6		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
7	3.93E-08	2.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
8	3.81E-08	2.2		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	5.63E-01		1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
9	2.91E-08	1.7		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
10	2.48E-08	1.5		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
11	2.44E-08	1.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
12	2.39E-08	1.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.05E-01		1-LPI-XHE-XM-GFEED-MD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
13	2.38E-08	1.4		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
14	2.08E-08	1.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
15	2.04E-08	1.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.05E-01		1-LPI-XHE-XM-GFEED-MD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED- (MODERATE DEPENDENCY)-MLDRL
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN -MLDRL
16	2.03E-08	1.2		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
17	1.75E-08	1.0		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.52E-01		1-LPI-XHE-XM-GFEED-LD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML
18	1.52E-08	0.9		
	1.00E+00		1-IE-SD-LONSCW	LOSS OF NSCW DURING SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	3.03E-05		1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	7.85E-01		/1-SD-NSCW-D-P5B	ALIGN GRAVITY FEED - LONSCW
19	1.51E-08	0.9		
	3.67E-01		1-IE-SD-LORHR	LOSS OF RHR - SHUTDOWN
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-POS6	OPERATORS FAIL TO RECOVER ONE RHR TRAIN
20	1.49E-08	0.9		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	7.95E-03		1-HPI-XHE-XM-FFEED-MLDRL	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-MLDRL
	7.90E-01		1-L2-BE-H2DET-L-CHR-SD	Late Detonation with CHR (Shutdown)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	9.56E-01		1-L2-BE-H2IGSRC-E-AC-SD	Ignition Source in Containment at VB with AC Power (Shutdown)
	9.56E-01		1-L2-BE-H2IGSRC-L-AC-SD	Ignition Source in Containment Late with AC Power (Shutdown)

Table B-9 Top 20 Cut Sets for Release Category 1-REL-SD-ICF-BURN (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	7.60E-01		/1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.52E-01		1-LPI-XHE-XM-GFEED-LD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)
	1.02E-02		1-RHR-XHE-XM-MLDRL-I	OPERATORS FAIL TO DIAGNOSE & ISOLATE DRAINDOWN BEFORE LOSS OF SD COOLING - ML

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA

#	Prob/Freq	% Contr.	Cut Set	Description
1	5.79E-08	43.6		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
2	5.61E-08	42.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseat after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
3	8.94E-09	6.7		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	2.44E-02		1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	1.14E-09	0.9		
	3.00E-03		1-IE-SD-LOINV-SP	LOSS OF INVENTORY - SHUTDOWN - RCS vented, spray event causes small leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	2.44E-02		1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	3.44E-02		1-POS-10-POT-4	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	3.00E-02		1-RHR-XHE-XM-LOISP-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
5	1.09E-09	0.8		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	7.75E-03		1-HPI-XHE-XM-FFeed-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS5A
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.52E-01		1-LPI-XHE-XM-GFEED-LD	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-(LOW DEPENDENCY)-LOI
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
6	7.26E-10	0.6		
	3.00E-03		1-IE-SD-LOINV-SP	LOSS OF INVENTORY - SHUTDOWN - RCS vented, spray event causes small leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	3.44E-02		1-POS-10-POT-4	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.00E-02		1-RHR-XHE-XM-LOISP-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6
7	6.42E-10	0.5		
	3.00E-03		1-IE-SD-LOINV-SP	LOSS OF INVENTORY - SHUTDOWN - RCS vented, spray event causes small leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	2.44E-02		1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	3.00E-02		1-RHR-XHE-XM-LOISP-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6
8	4.99E-10	0.4		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	5.29E-02		1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	2.44E-02		1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
9	4.56E-10	0.3		
	6.40E-05		1-IE-SD-LOINV	LOSS OF INVENTORY - SHUTDOWN - RCS vented, break causes RCS leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	2.44E-02		1-LOI-XHE-XM-ISO2-LT	FAIL TO ISOLATE LOI AFTER SUCCESSFUL INJECTION
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.00E+00		1-RHR-XHE-XM-LOI-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6
10	4.39E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.23E-01		1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A
	1.00E+00		1-OA-NSCW FAN---H	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
11	4.39E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.23E-01		1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A
	1.00E+00		1-OA-NSCWFAN---H	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	4.27E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.75E-03		1-HPI-XHE-XM-FFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS5A
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
13	4.14E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	7.75E-03		1-HPI-XHE-XM-FFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH FORCED RCS MAKEUP-LOI POS5A
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
14	4.08E-10	0.3		

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.00E-03		1-IE-SD-LOINV-SP	LOSS OF INVENTORY - SHUTDOWN - RCS vented, spray event causes small leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	3.00E-02		1-RHR-XHE-XM-LOISP-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6
15	3.58E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	2.72E-03		1-DCP-BAT-MA-AD1B_____	BATTERY 1AD1B IN MAINTENANCE
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.23E-01		1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A
	1.00E+00		1-OA-NSCWFAN---H	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
16	3.58E-10	0.3		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	2.72E-03		1-DCP-BAT-MA-BD1B_____	BATTERY 1BD1B IN MAINTENANCE
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.23E-01		1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A
	1.00E+00		1-OA-NSCWFAN---H	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
17	2.90E-10	0.2		
	6.40E-05		1-IE-SD-LOINV	LOSS OF INVENTORY - SHUTDOWN - RCS vented, break causes RCS leak
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	2.89E-01		1-L2-BE-CBYP-GEN	LOCA Bypasses Containment (all except LOINV-OP)
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.00E+00		1-RHR-XHE-XM-LOI-POS6	OPERATORS FAIL TO DIAGNOSE & ISOLATE LOI BEFORE LOSS OF SD COOLING -POS6
18	1.75E-10	0.1		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	3.00E-03		1-CIS-XHE-XL-ISOLPEN	Operators fail to isolate containment small containment penetrations
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
19	1.69E-10	0.1		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	3.00E-03		1-CIS-XHE-XL-ISOLPEN	Operators fail to isolate containment small containment penetrations
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY

Table B-10 Top 20 Cut Sets for Release Category 1-REL-SD-ISLOCA (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
20	1.16E-10	0.1		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	8.81E-04		1-ACP-INV-MA-AD1I11__	INVERTER 1AD1I11 IN MAINTENANCE
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	8.49E-01		/1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)
	1.67E-01		1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.23E-01		1-LPI-XHE-XM-GFEED-LOI5A	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOI POS 5A
	1.00E+00		1-OA-NSCWFAN---H	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF

#	Prob/Freq	% Contr.	Cut Set	Description
1	2.11E-09	3.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
2	2.11E-09	3.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
3	2.05E-09	3.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	2.05E-09	3.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
5	1.30E-09	2.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
6	1.30E-09	2.4		

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
7	1.26E-09	2.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
8	1.26E-09	2.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	7.70E-01		/1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	5.51E-02		1-POS-5B-POT-4	DRAINING THE REACTOR COOLANT SYSTEM TO MID-LOOP (POS-5B) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
9	9.70E-10	1.8		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSP)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	2.07E-02		1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
10	5.53E-02	1.8	1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	9.70E-10			
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
11	2.07E-02	1.7	1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	9.32E-10			
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	9.32E-10	1.7		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
13	7.94E-10	1.5		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.07E-02		1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
14	7.94E-10	1.5		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	2.07E-02		1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
15	7.62E-10	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
16	7.62E-10	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSP)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSP case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSP case 1)
	1.93E-02		1-POS-6-POT-4	MID-LOOP OPERATION (POS-6) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
17	6.31E-10	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
18	6.31E-10	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	5.00E-01		1-L2-BE-INDHLF-MP-SD	Induced Hot Leg Failure (Intermediate pressure) (Shutdown)
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
19	5.96E-10	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	2.07E-02		1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
20	5.96E-10	1.1		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	5.00E-01		1-L2-BE-CONTOP-GEN-SD	Late containment overpressure, RCS intact or vented (LPSD)
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	5.00E-01		/1-L2-BE-H2IGSRC-E-NAC-SD	Ignition Source in Containment at VB without AC Power (Shutdown)
	7.00E-01		/1-L2-BE-H2IGSRC-L-NAC-SD	Ignition Source in Containment Late without AC Power (Shutdown)

Table B-11 Top 20 Cut Sets for Release Category 1-REL-SD-LCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.41E-01		1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	2.40E-01		1-L2-OP-EDMGC-SD1	Operator fails to initiate containment sprays (LPSD case 1)
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	2.07E-02		1-POS-7-POT-4	FILLING CAVITY (POS-7) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF

#	Prob/Freq	% Contr.	Cut Set	Description
1	1.46E-07	2.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
2	1.44E-07	2.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002__	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
3	1.20E-07	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
4	1.19E-07	2.0		
	1.00E+00		1-IE-SD-LONSCW	LOSS OF NSCW DURING SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	3.03E-05		1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event : 1-IE-SWS-MDP-CF-
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
5	1.19E-07	2.0		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
6	1.18E-07	1.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	3.30E-02		1-EPS-DGN-FR-G4002____	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	2.73E-01		1-OEP-XHE-XX-NR02H1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR AVAIL)
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
7	1.01E-07	1.7		
	5.10E-01		1-IE-SD-ODML-LATE	OVER DRAINING IN MID-LOOP - After Refueling
	9.94E-01		/1-FT-CISOLSD-POS6-F	Operators fail to isolate containment during POS 6
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	4.07E-03		1-POS-10-POT-4-D	MID-LOOP OPERATION (POS-10) AFTER REFUELING (POT-4)- (demand-based)
8	9.81E-08	1.6		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
9	8.65E-08	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
10	8.58E-08	1.4		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
11	7.91E-08	1.3		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
	8.70E-01		/1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	9.00E-01		/1-L2-BE-H2IGNSRC-VE-NAC	Ignition Source in Containment before VB, without AC Power
	9.33E-01		/1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	4.64E-01		/1-L2-OP-EDMGB-SD2	Operator fails to recover injection to vessel (LPSD case 2)
	7.44E-02		1-POS-12-POT-4	REACTOR COOLANT SYSTEM HEATUP/DRAW BUBBLE IN PRESSURIZER (POS-12) AFTER REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
12	7.08E-08	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION
13	7.05E-08	1.2		
	3.67E-01		1-IE-SD-LORHR	LOSS OF RHR - SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
14	7.02E-08	1.2		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.35E-03		1-ACP-CRB-CC-BA0301___	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	4.98E-01		/1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.50E-01		/1-SD-FEED-G-SBO	ALIGN GRAVITY FEED - SHUTDOWN SBO
	4.10E-01		OEP-VCF-LP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
15	6.53E-08	1.1		
	3.40E-01		1-IE-SD-LO4160C	LOSS OF 4160 VAC BUS - SHUTDOWN
	9.94E-01		/1-FT-CISOLSD-POS8-F	Containment penetration isolation during refueling - shutdown
	4.88E-05		1-LPI-MDP-CF-START	CCF OF RHR PUMPS A & B TO START
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
16	6.51E-08	1.1		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	4.37E-01		/1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSP case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
17	6.31E-08	1.0		

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	4.37E-01		/1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSD case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.46E-01		1-RHR-XHE-XM-TRIP-MD	OPERATORS FAIL TO DIAGNOSE & TRIP RHR PUMPS (MODERATE DEPENDENCY)
18	5.87E-08	1.0		
	5.88E-01		1-IE-SD-ODML-EARLY	OVER DRAINING IN MID-LOOP - Before Refueling
	9.47E-01		/1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)
	5.40E-01		1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	2.59E-01		/1-L2-OP-EDMGB-SD4	Operator fails to recover injection to vessel (LPSD case 4)
	4.07E-03		1-POS-6-POT-4-MLDRL	Overdrain RCS entering MID-LOOP OPERATION FOR REFUELING (POT-4)-(demand-based)

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	1.55E-02		1-RHR-XHE-XL-LTREC	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
	1.25E-02		1-RHR-XHE-XL-TRN-MLDRL	OPERATORS FAIL TO RECOVER ONE OPERATING DHR TRAIN - MLDRL
19	5.54E-08	0.9		
	6.07E-01		1-IE-SD-LOINV-OP	LOSS OF INVENTORY - SHUTDOWN with RCS at pressure
	9.98E-01		/1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)
	9.94E-01		/1-FT-CISOLSD-PURGE-OP	Containment penetration isolation with purge system operating - shutdown
	8.33E-01		/1-L2-BE-CBYP-OP	LOCA Bypasses Containment (LOINV-OP)
	4.60E-01		/1-L2-BE-H2IGN-VE-CHR-SD	Burn in Containment before VB (CHR) (Shutdown)
	9.56E-01		1-L2-BE-H2IGNSRC-VE-AC	Ignition Source in Containment before VB, with AC Power
	6.69E-02		1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)
	4.37E-01		/1-L2-OP-EDMGB-SD3	Operator fails to recover injection to vessel (LPSC case 3)
	2.50E-02		1-LOI-XHE-XD-POS5A	OPERATOR FAILS TO DIAGNOSE LOI EARLY
	4.13E-02		1-POS-5A-POT-4	PRESSURIZER WATER SOLID FOR H2 DEGASSING (POS-5A) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	1.51E-01		1-RHR-XHE-XL-LTREC-LT5H	SHUTDOWN -FAIL TO RE-ESTABLISH NORMAL RHR COOLING after successful injection
20	5.39E-08	0.9		
	1.63E-01		1-IE-SD-LOOP	LOSS OF OFF-SITE POWER - SHUTDOWN
	5.02E-01		1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)
	9.71E-01		/1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)

Table B-12 Top 20 Cut Sets for Release Category 1-REL-SD-NOCF (cont.)

#	Prob/Freq	% Contr.	Cut Set	Description
	3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	6.98E-01		1-EPS-XHE-XL-NR04H	OPERATOR FAILS TO RECOVER EMERGENCY DIESEL IN 4 HOURS
	5.50E-01		1-LPI-XHE-XM-GFEED-LOOP	SHUTDOWN - FAILURE TO ESTABLISH LOW PRESSURE GRAVITY FEED-LOOP
	3.26E-01		1-OEP-XHE-XL-NR02H	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS
	7.16E-02		1-POS-8L-POT-4	REFUELING NEW CORE (POS-8L) FOR REFUELING (POT-4)
	5.53E-02		1-POT-4	PLANT OPERATING TYPE - 4 (REFUELING)
	4.10E-01		OEP-VCFLP-GRWR	PLANT WILSON SWITCHYARD IS UNAVAILABLE DURING GR AND WR LOOPS - SPLIT FRACTION

APPENDIX C

LEVEL 2 PRA BASIC EVENT UNCERTAINTY CHARACTERIZATION

This appendix documents the parameter uncertainty distributions that are modeled for shutdown Level 2 PRA basic events. The shutdown Level 2 PRA parameter uncertainty results are presented in Section 2.4.7. The inputs to the parameter uncertainty analysis include basic events developed for the Level 2 PRA model and basic events inherited from the Level 1 PRA model. The shutdown Level 1 PRA uncertainty distributions are discussed in the shutdown Level 1 PRA report (NRC, 2025). Also, many of the Level 1 PRA equipment failure basic events are documented in the at-power Level 1 PRA report (NRC, 2022b).

Section C.1 discusses uncertainty distributions for basic events in the Level 1/Level 2 PRA bridge tree. Section C.2 describes the uncertainty distributions used in the Level 2 PRA containment event tree. Section C.3 provides additional information on the bases for developing uncertainty distributions for certain basic events.

C.1 Uncertainty Distributions for Bridge Tree Modeled Basic Events

C.1.1 SSC Basic Events in Bridge Tree

The structure, system, and component (SSC) events contained in the bridge tree are provided in Table C-1, along with information about their uncertainty characterization. Most of these events were determined in the reactor-at-power PRA to be best described by beta distributions. As discussed in NUREG/CR-7039 (Smith, 2011), Volume 2, SAPHIRE defines the beta distribution, $\text{beta}(\alpha, \beta)$, in terms of the parameters β and μ , where μ , the mean value of the beta distribution, is equal to $\alpha/(\alpha+\beta)$. Similarly, SAPHIRE defines the gamma distribution, $\square(r)$, in terms of the parameters r and λ , where λ is equal to r/μ and μ is the mean value of the distribution. Table C-1 summarizes the uncertainty quantification for reactor-at-shutdown bridge tree SSC-related basic events. Note that since there are no SSC-related events unique to the shutdown bridge tree, this table is just a reproduction of the same table from Appendix C of the reactor at-power, Level 2 PRA for internal events and floods (NRC, 2022a).

C.1.2 Human Failure Events in Bridge Tree

The HRA-related events contained in the bridge tree are provided in Table C-2. Also, “dummy” basic events that exist in the linkage rules only to be substituted eventually by other events in post-processing are not listed. Error factors for these HEPs are assigned following the guidelines that are summarized in Table C-3, which are an expanded version of rules originally found in THERP guidance. The error factor guidelines expand on the THERP guidance by establishing revised error factors for high failure probability events to minimize the fraction of the cumulative distribution that would otherwise fall above a value of 1.0. 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 in accordance with the following rules summarized in Table C-3.

C.1.3 Phenomena Basic Events in Bridge Tree

The phenomena-related events contained in the bridge tree are provided in Table C-4, along with information about correlation class and how uncertainty is estimated. The process used to generate this list uses solved cutsets resulting from the shutdown PRA, 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 4,160V AC bus initiators) are the reason that Train B containment spray nozzle plugging appears in the list while Train A nozzle plugging does not.

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-CIS-AOV-OO-2626_27B-CC	AOV HV-2626B AND AOV HV-2627B FAIL TO OPERATE (HARDWARE)	System	2.93E-05	See related input basic events	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 (CCF multiplier AOV FTOP 2-2 non-staggered T) have beta uncertainty distribution types. The reference plant PRA model is the source of information for these events. <u>1-AVFAL</u> β : 2.105E+4 μ : 6.640E-1 <u>QAOVFTOP2-2NS</u> β : 4.861E+1 μ : 4.410E-2
1-CIS-AOV-OO-HV28_29B-CC	AOV HV-2628B AND AOV HV-2629B FAIL TO OPERATE (HARDWARE)	System	2.93E-05	See related input basic events	
1-CIS-AOV-OO-HV780781-CC	AOV HV-0780 AND AOV HV-0781 FAIL TO OPERATE (HARDWARE)	System	2.93E-05	See related input basic events	
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 is based on the 2010 SPAR model update and plant-specific data. <u>ZT-AOV-OO</u> 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	

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
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 NRC parameter estimate component reliability data sheets. <u>ZT-CKV-CC</u> 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
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. <u>ZA-MDP-FR-02A01</u> β: 4.247E+0 μ: 9.651E-1

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
					<u>ZA-MDP-FR-02A02</u> β : 1.173E+2 μ : 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. <u>ZA-MDP-FS-02A01</u> β: 9.687E+0 μ: 9.750E-1 <u>ZA-MDP-FS-02A02</u> β: 3.774E+2 μ: 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

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

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	SPAR model update and plant-specific data. <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 type is 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)	
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.
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	

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-CSR-MOV-CC-HV9002B	CS SUMP B MOV HV9002B FAILS TO OPEN	System	3.53E-04	MOV-CC	<u>ZT-MOV-CC</u> β : 5.400E+4 μ : 3.530E-4
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	
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. These templates incorporate data from version 4.5.2010.3 of the NRC's CCF database. <u>ZA-MOV-CC-02A01</u> β: 5.280E+1 μ: 1.700E-2 <u>ZA-MOV-CC-02A02</u> β: 1.645E+0 μ: 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	

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
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. These templates incorporate data from version 4.5.2010.3 of the NRC's CCF database. <u>ZA-MOV-OO-02A01</u> β: 8.062E-1 μ: 9.919E-1 <u>ZA-MOV-CC-02A02</u> β: 9.900E+1 μ: 8.080E-3
1-CSR-MOV-OO-HV9017A	CS TRAIN A MOV HV9017A FAILS TO CLOSE	System	3.53E-04	MOV-OO	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.
1-CSR-MOV-OO-HV9017B	CS TRAIN B MOV HV9017B FAILS TO CLOSE	System	3.53E-04	MOV-OO	<u>ZT-MOV-OO</u> β: 5.400E+4 μ: 3.530E-4

Table C-1 Parameter Uncertainty for SSC Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-ESF-SSD-FC-A5161A_B	SAFEGUARDS DRIVER CIRCUIT FAILS	System	5.04E-04	None	This event has a beta distribution with β and μ equal to 9.920E+2 and 5.040E-4, respectively.

Table C-2 Parameter Uncertainty for Human Failure Events Modeled in Bridge Tree

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-CIS-XHE-XL-ISOLPEN	Operators fail to isolate containment small containment penetrations	HRA	3.00E-3	None	Lognormal: mean = 3.00E-3 and error factor (EF) = 10
1-CIS-XHE-XL-ISOLPEN-HD	Operators fail to isolate containment small containment penetrations (high dependency)	HRA	5.02E-1	None	Lognormal: mean = 5.02E-1 and EF = 2
1-CIS-XHE-XL-ISOLPEN-LD	Operators fail to isolate containment small containment penetrations (low dependency)	HRA	5.29E-2	None	Lognormal: mean = 5.29E-2 and EF = 5
1-CIS-XHE-XL-PURGE	Operators fail to isolate containment purge when system operating	HRA	2.00E-4	None	Lognormal: mean = 2.00E-4 and EF = 10
1-CIS-XHE-XL-PURGE-LD	Operators fail to isolate containment purge when system operating (low dependency)	HRA	5.02E-2	None	Lognormal: mean = 5.02E-2 and EF = 5
1-CIS-XHE-XR-PURGE	Operators fail to isolate containment purge after system shutdown (pre-initiator HFE)	HRA	1.00E-4	None	Lognormal: mean = 1.00E-4 and EF = 10
1-CNT-XHE-XM-H1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (AC power available)	HRA	1.00E-2	None	Lognormal: mean = 1.00E-2 and EF = 5
1-CNT-XHE-XM-H1-HD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (high dependency)	HRA	5.05E-1	None	Lognormal: mean = 5.05E-1 and EF = 2
1-CNT-XHE-XM-H1-LD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (low dependency)	HRA	5.95E-2	None	Lognormal: mean = 5.95E-2 and EF = 5
1-CNT-XHE-XM-H1-MD	Operators fail emergency closure of equipment hatch with lowered RCS inventory with AC power (moderate dependency)	HRA	1.51E-1	None	Lognormal: mean = 1.51E-1 and EF = 5
1-CNT-XHE-XM-H1F	Operators fail emergency closure of equipment hatch with successful forced feed	HRA	1.00E-3	None	Lognormal: mean = 1.00E-3 and EF = 10

Table C-2 Parameter Uncertainty for Human Failure Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-CNT-XHE-XM-H1F-LD	Operators fail emergency closure of equipment hatch with successful forced feed (low dependency)	HRA	5.10E-2	None	Lognormal: mean = 5.10E-2 and EF = 5
1-CNT-XHE-XM-H2	Operators fail emergency closure of equipment hatch with RCS intact (AC power available)	HRA	2.00E-3	None	Lognormal: mean = 2.00E-3 and EF = 5
1-CNT-XHE-XM-H2-LD	Operators fail emergency closure of equipment hatch with RCS intact AC power available (low dependency)	HRA	5.19E-2	None	Lognormal: mean = 5.19E-2 and EF = 5
1-CNT-XHE-XM-H3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (AC power available)	HRA	4.00E-5	None	Lognormal: mean = 4.00E-5 and EF = 10
1-CNT-XHE-XM-SBO1	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power)	HRA	2.30E-1	None	Lognormal: mean = 2.30E-1 and EF = 5
1-CNT-XHE-XM-SBO1F	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power)	HRA	2.90E-2	None	Lognormal: mean = 2.90E-2 and EF = 5
1-CNT-XHE-XM-SBO2	Operators fail emergency closure of equipment hatch with RCS intact (no AC power)	HRA	1.30E-1	None	Lognormal: mean = 1.30E-1 and EF = 5
1-CNT-XHE-XM-SBO3	Operators fail emergency closure of equipment hatch with refuel cavity flooded (no AC power)	HRA	2.90E-2	None	Lognormal: mean = 2.90E-2 and EF = 5
1-CCU-XHE2-SD	Operators fail to restore 1 train of CCUs in POS 4-5A and 7-9	HRA	3.20E-4	None	Lognormal: mean = 3.20E-4 and EF = 10
1-CCU-XHE2-SD-POS6	Operators fail to restore 1 train of CCUs in POS 6	HRA	1.40E-3	None	Lognormal: mean = 1.40E-3 and EF = 5
1-CCU-XHE2-SD-POS6-LD	Operators fail to restore 1 train of CCUs in POS 6 (low dependency)	HRA	5.13E-2	None	Lognormal: mean = 5.13E-2 and EF = 5
1-OA-CS-RECIRC	OPERATORS FAILS TO ALIGN CS RECIRCULATION	HRA	9.60E-3	None	Lognormal: mean = 9.60E-3 and EF = 5
1-OA-CS-RECIRC-LD	OPERATORS FAILS TO ALIGN CS RECIRCULATION	HRA	5.90E-2	None	Lognormal: mean = 5.90E-2 and EF = 5

Table C-2 Parameter Uncertainty for Human Failure Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-OA-OP-PHASE-AH	OPERATOR FAILS TO MANUALLY INITIATE PHASE A ISOLATION	HRA	3.00E-3	None	Lognormal: mean = 3.00E-3 and EF = 5

Table C-3 Guidelines for Assigning Error Factors for Human Failure Events

HEP (mean)	EF
HEP < 0.001	10
$0.001 \leq \text{HEP} < 0.3$	5
$0.3 \leq \text{HEP} < 0.4$	3
$0.4 \leq \text{HEP} < 0.45$	2.95
$0.45 \leq \text{HEP} < 0.5$	2.45
$0.5 \leq \text{HEP} < 0.55$	2.10
$0.55 \leq \text{HEP} < 0.6$	1.86
$0.6 \leq \text{HEP} < 0.65$	1.66
$0.65 \leq \text{HEP} < 0.7$	1.51
$0.7 \leq \text{HEP} < 0.75$	1.38
$0.75 \leq \text{HEP} < 0.8$	1.28
$0.8 \leq \text{HEP} < 0.85$	1.20
$0.85 \leq \text{HEP} < 0.9$	1.12
HEP = 0.90	1.05
HEP > 0.90	1.00 (i.e., no uncertainty prescribed)

Table C-4 Parameter Uncertainty for Phenomena Events Modeled in Bridge Tree

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
1-L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS	Phenom	1.11E-3	None	Lognormal distribution with EF = 10 (see Section C.3 for additional information)
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. These templates incorporate data from version 4.5.2010.3 of the NRC's CCF database. <u>ZA-CCF-RATE-02A01</u> β: 4.166E+1 μ: 9.624E-1 <u>ZA-CCF-RATE-02A02</u> β: 1.067E+3 μ: 3.760E-2

Table C-4 Parameter Uncertainty for Phenomena Events Modeled in Bridge Tree (cont.)

Event	Event Description	BE Class	Mean	Corr. Class	Uncertainty Distribution
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. <u>ZT-NZL-PG</u> r: 3.000E-1 λ: 1.000E-6

C.2 Uncertainty Distributions for Containment Event Tree Modeled Basic Events

C.2.1 SSC Basic Events in Containment Event Tree

The SSC-related events contained in the Level 2 PRA model are provided in Table C-5, along with their uncertainty distribution. Note that during the process of quantifying parameter uncertainties, it was decided to substitute the distributions that had been developed for pressurizer power-operated relief valve (PORV) failure in the reactor at-power PRA for RHR relief valve failure in the reactor at-shutdown PRA, despite the fact that the number of cycles and valve failure probability per demand are different. Table C-6 and Table C-7 provide additional context for the relief valve failure modeling in the Level 2 reactor at-power and reactor at-shutdown PRA models. Table C-6 summarizes reference failure probabilities used in the Level 2 PRA and comparisons to values used in the Level 1 PRA or the SOARCA project. Table C-7 includes data on the simulated number of relief valve lifts for different accident scenarios developed for this study.

C.2.2 Human Failure Events in Containment Event Tree

The HRA-related events contained in the Level 2 PRA model are provided in Table C-8. Uncertainty distributions are developed following the same rules as previously outlined for HFEs in Section C.1.2 .

C.2.3 Phenomena Basic Events in Containment Event Tree

The phenomena-related events contained in the Level 2 PRA model are provided in Table C-9. The following different approaches were used to obtain uncertainty distributions:

- For some combustion-related events (i.e., conditional probabilities of ignition, detonation, or containment failure), the computer code ERPRA-BURN that had been used to obtain the point-values also supplies uncertainty distributions in the form of histograms, which were implemented directly into SAPHIRE.
- For some energetic events (in-vessel steam explosion and very early combustion-induced containment failure), past studies and literature review (performed by Energy Research, Inc.) led to a conclusion that the uncertainty distribution would be adequately represented as distributed lognormally around the mean with a 99/50 error factor of two orders of magnitude (which translates to a 95/50 error factor of 76).
- In the case of some phenomenological events, a histogram, uniform distribution, or piecewise-uniform distribution was qualitatively justified based on MELCOR results and/or analyst judgement. This includes a case in which no judgement more precise than complete uncertainty (i.e., uniformly distributed between 0 and 1) could be achieved.
- For any other phenomena events whose uncertainty could not be adequately quantified by any of the above methods, a default lognormal distribution was assigned based on the point estimate magnitude representing the mean, and with an error factor assumed according to the simple lookup table used for HEPs in Sections C.1 . This default scheme, although developed for human reliability purposes, was found to agree well with

the intuition of the team for the spread of uncertainty in otherwise-uninformed phenomena parameters.

There are two fundamental types of phenomena failure probabilities in the model, those that are purely subjective (analyst judgment) and those that are the result of underlying computations (e.g., combustion analysis and in-vessel steam explosion). It is acknowledged that some parameters could be viewed as the uncertainty of an uncertainty (i.e., the split fraction itself is an uncertainty), particularly in the cases where the split fraction originated from a load-capacity interference concept using underlying calculations. Nevertheless, the authors view the assignment of a distribution on the split fraction as appropriate, 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.

In addition, Table C-9 assigns each event to a basic event class and a grouping. The former is not used but is simply a convenience in sorting the events. The latter is not used in the parameter uncertainty propagation but is used in the model uncertainty treatment discussed in Section 2.6.3. In SAPHIRE, correlating two variables infers that (along with being physically correlated) they have identical distributions (both the type and the parameters). Since this is rarely, if ever, the case with the Level 2 PRA phenomena events, sampling correlation is not used.

Table C-5 Parameter Uncertainty for SSC Events Modeled in Containment Event Tree

Event	Event Description	BE Class	Mean	Grouping	Uncertainty Distribution
1-L2-BE-PZRVSTUCK-PORV-SD	Pressurizer PORVs Fail Open During CD (shutdown)	System	0.07	Relief valve	Histogram (see Section C.3)
1-L2-BE-RHRVSTUCK-SD	RHR Relief Valve fails to reseal after lifting during CD sequence (shutdown)	System	0.07	Relief valve	Histogram (see Section C.3)

Table C-6 Relief Valve Failure Modeling Comparison and References

	Event	Used In	Used For	Per Demand Failure Probability	Failure Probability	Failure Probability Per Demand Basis
L3PRA Level 1, at-power (NRC, 2022b)	1-RCS-PRV-OO-RV045*A_	1-PVC and 1-PVC-ATWS	Pressurizer PORV randomly FTC (ATWS or non-ATWS)		1.46E-3 per valve, given a challenge ¹	NUREG/CR-7037, Table 30 (Wierman, 2011)
	1-RCS-SWP-OO-V8010*__	1-PVC	Pressurizer 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
	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 FTO during ATWS		3.51E-3 per valve	Plant-Specific Template. Updated with: 2010 SPAR Update, PORV FTO PPR, Failures: 0, Demands/Hours: 40.3, Entries: 4, Entries with Failures: 0, Build Date: 07/02/2013
	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 from CCF Database Version 4.5.2010.3, Application PWR-RCS-PORV-CC, Build Date: 10/24/2012
	1-IE-SSBI/1-IE-SSBO	1-FPI-SSBI/1-FPI-SSBO	IE for all SSB 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
	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
	1-MSS-ADV-CC-VPV30**	1-SVC-ARV and 1-SVC-SRV	Main steam ARV FTO		5.56E-4	2010 Update to the Parameter Estimation Component Reliability Data Sheets; Section 4.3, Build Date: 07/19/2012

Table C-6 Relief Valve Failure Modeling Comparison and References (cont.)

	Event	Used In	Used For	Per Demand Failure Probability	Failure Probability	Failure Probability Per Demand Basis
L3PRA Level 2, at-power (NRC, 2022a)	1-L2-PZRVSTUCK-PORV	1-L2-DET-PRESVE	Pressurizer PORV FTC during CD	$\sim 4 \cdot 10^{-3}$ (initial), $\sim 3 \cdot 10^{-3}$ (subsequent)	0.07, based on 30 valve cycles and a PORV per lift failure probability	NUREG/CR-7037, Table 18 (Wierman, 2011)
	1-L2-PZRVSTUCK-SRV	1-L2-DET-PRESVE	Pressurizer SRV FTC during CD	$\sim 3 \cdot 10^{-2}$ (initial), $\sim 3 \cdot 10^{-3}$ (subsequent)	0.08, based on 30 valve cycles and a SRV per lift failure probability based on main steam safety valve data	NUREG/CR-7037, Table 20 for MSS Code Safety Valves (Wierman, 2011)
	1-L2-BE-ARVSTUCK-SGTR	1-CET	Probability of stuck-open secondary relief valve during SGTR		Main Steamline ARV	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
				Automatic Demand – Initial	$\sim 3 \cdot 10^{-3}$	
Draft Surry SOARCA Uncert. (SNL, 2016)	SVLAMFTC	Sampled in the individual realizations	Pressurizer SRV stochastic FTC	Beta distribution ($\alpha=17.5$, $\beta=756.5$)	0.64 ⁴	Draft Surry SOARCA uncertainty analysis interpretation of the NUREG/CR-7037 (Wierman, 2011) results
	SVLAMFTO		Pressurizer SRV stochastic FTO	Beta distribution ($\alpha=0.5$, $\beta=773.5$)	0.01 ^{4,5}	
	SVWTR		Pressurizer SRV FTC due to passing water	Beta distribution ($\alpha=0.5$, $\beta=4.5$)	0.04 ⁴	
	SVFAILT		Pressurizer SRV thermal FTC	Temperature-based rather than cycling-based	0 ⁴	
	SVLAMFTC-SG		Main steam SRV stochastic FTC	Beta distribution ($\alpha=17.5$, $\beta=756.5$)	0.95 ⁴	
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 (Wierman, 2011). ² For pressurizer safety valves to be challenged, at least one PORV must fail. ³ For 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 Based on 1,003 realizations, unmitigated short-term station blackout, January 2016 draft (SNL, 2016). Updated analyses for the Surry SOARCA uncertainty analysis are in progress in 2017. ⁵ Individual valves only, none as the system of three valves in parallel.						

Table C-7 Summary of Pressurizer Relief Valve Lifts from At-Power MELCOR Analyses

Source: Appendix B of (NRC, 2022a)

Case ¹	Number of Pressurizer PORV Lifts		Number of Pressurizer SRV Lifts	
	Prior to Core Damage	After Core Damage ²	Prior to Core Damage	After Core Damage ²
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 Numbering of cases in the reactor-at-power, internal event and flood PRA deterministic analysis. ² Understand that once vessel breach has occurred, no further cycles can occur, given the intrinsic RCS depressurization.				

Table C-8 Parameter Uncertainty for Human Failure Events Modeled in Containment Event Tree

Event	Event description	BE Class	Mean	Grouping	Uncertainty Distribution
1-L2-OP-EDMGB-SD1	Operator fails to arrest core damage by recovery of injection (post-refueling, scenario category VI)	HRA + Coolability ¹	0.218	HRA + Coolability	Histogram (see Section C.3)
1-L2-OP-EDMGB-SD2	Operator fails to arrest core damage by recovery of injection (post-refueling, non-scenario category VI)	HRA + Coolability ¹	0.536	HRA + Coolability	Histogram (see Section C.3)
1-L2-OP-EDMGB-SD3	Operator fails to arrest core damage by recovery of injection (pre-refueling, scenario category VI)	HRA + Coolability ¹	0.563	HRA + Coolability	Histogram (see Section C.3)
1-L2-OP-EDMGB-SD4	Operator fails to arrest core damage by recovery of injection (pre-refueling, non-scenario category VI)	HRA + Coolability ¹	0.741	HRA + Coolability	Histogram (see Section C.3)
1-L2-OP-EDMGC-SD1	Operator fails to spray containment late (closed or vented RCS)	HRA	0.24	HRA	Lognormal: Mean = 0.24 and EF = 5
1-L2-OP-EDMGC-SD2	Operator fails to spray containment late (reactor head removed)	HRA	0.58	HRA	Lognormal: Mean = 0.58 and EF = 1.86
1-L2-OP-EDMGH-SD1	Operator fails to inject to steam generators	HRA	0.24	HRA	Lognormal: Mean = 0.24 and EF = 5
1-L2-OP-EDMGP-SD1	Operator fails to open reactor vessel head vents	HRA	0.17	HRA	Lognormal: Mean = 0.17 and EF = 5
¹ Success of this event is the product of a successful operator action (restoring injection to the reactor vessel) and favorable phenomenological conditions (the core is still in a coolable geometry at the time injection is restored). The probability of failure of the operator action (p_{OP}) was determined to be 0.14 (lognormal distribution with mean of 0.14 and error factor of 5) for scenarios of Category VI, and 0.52 (lognormal distribution with mean of 0.52 and error factor of 2.1) for scenarios not of Category VI. The probability of non-coolability (p_{cool}) was, on the basis of MELCOR calculations at shutdown, qualitatively established as 0.5 pre-refueling (uniform distribution between 0 and 1) and 0.1 post-refueling (piecewise-uniform distribution, with 90 percent of the cumulative distribution between 0 and 0.1, and 10 percent between 0.1 and 1, such that the mean value is 0.1). The value of $1 - (1 - p_{OP})(1 - p_{cool})$ was found for each of 5,000 simple Monte Carlo samples performed in an Excel spreadsheet, and the results were discretized to form approximate histograms for these four separate basic events. See Section C.3 for the resulting histogram input data.					

Table C-9 Parameter Uncertainty for Phenomena Events Modeled in Containment Event Tree

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-BMT-NCHR-SD	Basemat melt-through occurs late given MCCI	1-L2-BMT	Adaptation of MELCOR results ²	0.1	None	None
1-L2-BE-CCI-DISP	MCCI occurs given energetically dispersed debris	1-L2-MCCI	User judgement informed by past studies	0.01	None	Lognormal: Mean = 0.01 and EF = 5
1-L2-BE-CONTOP-GEN-SD	Containment fails due to quasi-static over-pressurization late	1-L2-CONTOP-L	Adaptation of MELCOR results	0.5	None	Uniform between 0 and 1
1-L2-BE-EVSE-GEN-SD	Ex-vessel steam explosion occurs given submerged reactor cavity	1-L2-EVSE	User judgement informed by past studies	0.1	Other Energetic	Lognormal: Mean = 0.1 and EF = 5
1-L2-BE-H2CF-E-GEN-SD	Combustion fails containment given very early ignition	1-L2-H2CF-E	ERPRA-BURN results and user judgement	1.0E-4	Combustion	Lognormal: Mean = 1.0E-4 and EF = 76
1-L2-BE-H2CF-L-CHRNPB-SD	Combustion fails containment given late ignition (with containment heat removal; no prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.18	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-CHRPB-SD	Combustion fails containment given late ignition (with containment heat removal; prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.05	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-LOINPB-SD	Combustion fails containment given late ignition (loss-of-inventory scenarios; no prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.20	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-LOIPB-SD	Combustion fails containment given late ignition (loss-of-inventory scenarios; prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.07	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-NACNPB-SD	Combustion fails containment given late ignition (without AC power; no prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.62	Combustion	Histogram (see Section C.3)

Table C-9 Parameter Uncertainty for Phenomena Events Modeled in Containment Event Tree (cont.)

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-H2CF-L-NACPB-SD	Combustion fails containment given late ignition (without AC power; prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.33	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-NCHRN-SD	Combustion fails containment given late ignition (without containment heat removal; no prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.23	Combustion	Histogram (see Section C.3)
1-L2-BE-H2CF-L-NCHRPB-SD	Combustion fails containment given late ignition (without containment heat removal; prior burn occurred)	1-L2-H2CF-L	ERPRA-BURN results	0.13	Combustion	Histogram (see Section C.3)
1-L2-BE-H2DET-L-CHR-SD	Detonation occurs in containment late given ignition (with containment heat removal)	1-L2-H2DET-L	ERPRA-BURN results	0.79	Combustion	Bimodal histogram ¹
1-L2-BE-H2DET-L-LOIPB-SD	Detonation occurs in containment late given ignition (loss-of-inventory scenarios)	1-L2-H2DET-L	ERPRA-BURN results	0.61	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-L-NACNP-SD	Ignition occurs in containment late given an ignition source (without AC power; no prior burn occurred)	1-L2-H2IGN-L	ERPRA-BURN results	0.99	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-L-NACPB-SD	Ignition occurs in containment late given an ignition source (without AC power; prior burn occurred)	1-L2-H2IGN-L	ERPRA-BURN results	0.8	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-L-NCHPB-SD	Ignition occurs in containment late given an ignition source (without containment heat removal; prior burn occurred)	1-L2-H2IGN-L	ERPRA-BURN results	0.98	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGNSRC-VE-AC	Ignition source exists in containment very early (with AC power)	1-L2-H2IGNSRC-VE	User judgement	0.99	Combustion	Histogram (see Section C.3)

Table C-9 Parameter Uncertainty for Phenomena Events Modeled in Containment Event Tree (cont.)

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-H2IGNSRC-VE-NAC	Ignition source exists in containment very early (without AC power)	1-L2-H2IGNSRC-VE	User judgement	0.1	Combustion	Histogram (see Section C.3)
1-L2-BE-H2IGN-VE-CHR-SD	Ignition occurs in containment very early (with containment heat removal)	1-L2-H2IGN-VE	ERPRA-BURN results	0.54	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-VE-LOCA-SD	Ignition occurs in containment very early (loss-of-inventory conditions)	1-L2-H2IGN-VE	ERPRA-BURN results	0.92	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-VE-SBO-SD	Ignition occurs in containment very early (without AC power)	1-L2-H2IGN-VE	ERPRA-BURN results	0.39	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGN-VE-TRAN-SD	Ignition occurs in containment very early (with AC power)	1-L2-H2IGN-VE	ERPRA-BURN results	0.83	Combustion	Bimodal histogram ¹
1-L2-BE-H2IGSRC-E-AC-SD	Ignition source exists in containment early (with AC power)	1-L2-H2IGNSRC-E	User judgement	0.99	Combustion	Histogram (see Section C.3)
1-L2-BE-H2IGSRC-E-NAC-SD	Ignition source exists in containment early (without AC power)	1-L2-H2IGNSRC-E	User judgement	0.5	Combustion	Histogram (see Section C.3)
1-L2-BE-H2IGSRC-L-AC-SD	Ignition source exists in containment late (with AC power)	1-L2-H2IGNSRC-L	User judgement	0.99	Combustion	Histogram (see Section C.3)
1-L2-BE-H2IGSRC-L-NAC-SD	Ignition source exists in containment late (without AC power)	1-L2-H2IGNSRC-L	User judgement	0.3	Combustion	Histogram (see Section C.3)
1-L2-BE-INDHLF-MP-SD	Temperature-induced hot leg failure occurs given intermediate RCS pressure	1-L2-INDHLF	User judgement informed by MELCOR results	0.5	Induced Piping Fail	Lognormal: Mean = 0.5 and EF = 1.9
1-L2-BE-INDSGTR-HDL	Temperature-induced SGTR occurs given high/dry/low conditions	1-L2-INDSGTR	C-SGTR Calculator	0.02	Induced Piping Fail	Lognormal: Mean = 0.02 and EF = 5

Table C-9 Parameter Uncertainty for Phenomena Events Modeled in Containment Event Tree (cont.)

Event	Event Description	Associated Top Event	Underlying Point Estimate Source	Mean	Grouping	Uncertainty Distribution
1-L2-BE-IVSE-LP-SD	In-vessel steam explosion fails containment (low RCS pressure)	1-L2-IVSE	User judgement informed by past studies	2.0E-4	Other Energetic	Lognormal: Mean = 2.0E-4 and EF = 76; see Section C.3 for additional discussion
1-L2-BE-IVSE-NLP-SD	In-vessel steam explosion fails containment (intermediate or high RCS pressure)	1-L2-IVSE	User judgement informed by past studies	1.0E-4	Other Energetic	Lognormal: Mean = 1.0E-4 and EF = 76; see Section C.3 for additional discussion
1-L2-BE-VROCK-HP-SD	Vessel rocketing fails containment (high RCS pressure)	1-L2-VROCK	Adaptation of ERPRA-ROCKET results	1.0E-4	Other Energetic	Lognormal: Mean = 1.0E-4 and EF = 10
¹ For each sample performed in ERPRA-BURN – which is a possible stochastic history of combustion in the containment – either ignition occurs or it does not in a given time frame, or it either results in a detonation or it does not. Therefore, the conditional probability of the event occurring in any given sample is either 0 or 1. The probabilistic output from the code reflects this in the form of a histogram with only two bins with values of $(0, 1-\bar{p})$ and $(1, \bar{p})$, where $\bar{p}$ is the mean probability of the event over all samples. Such a distribution also has the feature of conservatively having the maximum magnitude of uncertainty for a given mean value. This histogram is input directly into SAPHIRE, with the only minor change being that the first bin value is changed from zero to 1.0E-06 to avoid a warning from the code. ² Assigned a point-estimate BMT probability of 0.1 qualitatively based on extrapolated MELCOR results indicating it would require about 190 to 440 hours, as compared with the 168-hour termination time used in the reactor at-shutdown Level 2 PRA. However, with no MELCOR sensitivity cases having been performed specifically to address uncertainties in MCCI rate or boundary conditions, it was impossible to precisely assign a particular uncertainty distribution to this basic event's failure probability.						

C.3 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 upon the mean. In the case of Table C-10, Table C-28, and Table C-29, the standard approach is used, and additional discussion is provided. Note that, in the case of some basic events shared between the at-power and at-shutdown models, this appendix simply recapitulates the original uncertainty quantification developed for the at-power model (NRC, 2022a) rather than any new development.

Table C-10 Parameter Uncertainty for 1-L2TEAR

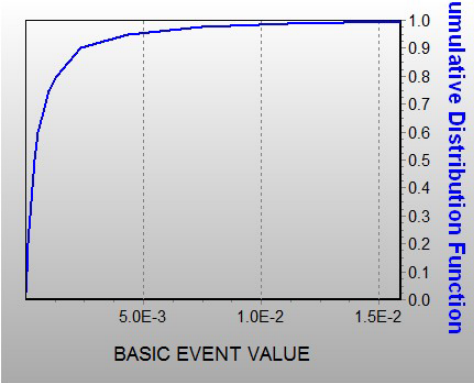
Distribution: Lognormal with Mean of $1.1\text{E-}3$ and EF = 10 See description below and CDF to the right.	
Basis: This basic event represents the failure of containment due to pre-existing maintenance errors, cracks, or tears which allow a direct pathway outside containment. The probability of such 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 $1.1\text{E-}3$ is used in the base model. In addition, the group also looked at additional supporting documents including EPRI report 1018243, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals,” dated October 2008 (EPRI, 2008b). The contributing staff convened to discuss the variability in the expert elicitation data in Table F-6 of EPRI 1018243; selection of the appropriate leakage size to be associated with the basic event; and selection of an uncertainty distribution type. This discussion included a review of Table D-2 from EPRI 1018243 which 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 a number of 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.	
Contributors: Anders Gilbertson (lead), Don Helton, Tina Ghosh, Hossein Esmaili	

Table C-11 Parameter Uncertainty for 1-L2-BE-PZRVSTUCK-PORV-SD and 1-L2-BE-RHRVSTUCK-SD

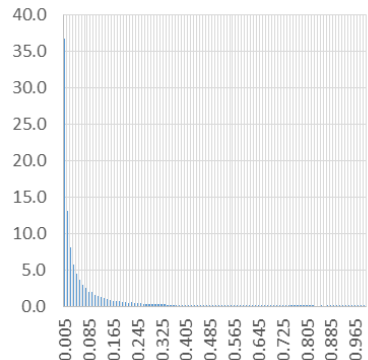
Distribution: Histogram See description below and PDF to the right. Encoded in the model as HIST-L2-PZRVSTUCK-PORV.									
Basis: (Note: This table documents the distribution originally developed for 1-L2-BE-PZRVSTUCK-PORV, the corresponding event for a stuck-open pressurizer PORV in the reactor-at-power PRA. The decision was made by NRC to use the same mean value and uncertainty distribution for the pressurizer PORVs and RHR relief valves in the reactor-at-shutdown model, both in the interest of time and due to the fact that the reactor-at-power quantification of this issue was considered to be well supported) 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 pressurized-water reactors than it is for boiling-water reactors, owing to the much more tortuous path to the valves in question (owing to the surge line and pressurizer). 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). 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.									
<table><tr><th>Type of Valve</th><th>Distribution of Probability of Failure-to-Close on Initial Demand</th><th>Distribution of Probability of Failure-to-Close on Subsequent Demand</th><th>Source of Data from NUREG/CR-7037 (Wierman, 2011), Failure to Close, All Failures</th></tr><tr><td>Pressurizer PORV</td><td>Beta($\alpha = 0.5$, $\beta = 100.5$)</td><td>Beta($\alpha = 0.5$, $\beta = 181.5$)</td><td>NUREG/CR-7037 (Wierman, 2011) Table 18, RCS PORVs, Automatic demands</td></tr></table>	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 (Wierman, 2011), Failure to Close, All Failures	Pressurizer PORV	Beta($\alpha = 0.5$, $\beta = 100.5$)	Beta($\alpha = 0.5$, $\beta = 181.5$)	NUREG/CR-7037 (Wierman, 2011) Table 18, RCS PORVs, Automatic demands	
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 (Wierman, 2011), Failure to Close, All Failures						
Pressurizer PORV	Beta($\alpha = 0.5$, $\beta = 100.5$)	Beta($\alpha = 0.5$, $\beta = 181.5$)	NUREG/CR-7037 (Wierman, 2011) Table 18, RCS PORVs, Automatic demands						
<i>Distribution of # of valve cycles: lognormal distribution with mean = 30, 5th percentile = 3, 95th percentile = 100</i>									
Numerically convolving these distributions results in a distribution with the following attributes: • Mean = 0.07 • Median = 0.02 • 5th percentile = 1·10⁻⁴ • 95th percentile = 0.32 This data was then approximated and input to SAPHIRE using a 100-point “percentage”-type histogram.									
Contributors: Tina Ghosh (lead), Don Helton, Hossein Esmaili									

Table C-12 Parameter Uncertainty for 1-L2-BE-H2IGNSRC-VE-AC, 1-L2-BE-H2IGNSRC-E-AC, and 1-L2-BE-H2IGNSRC-L-AC

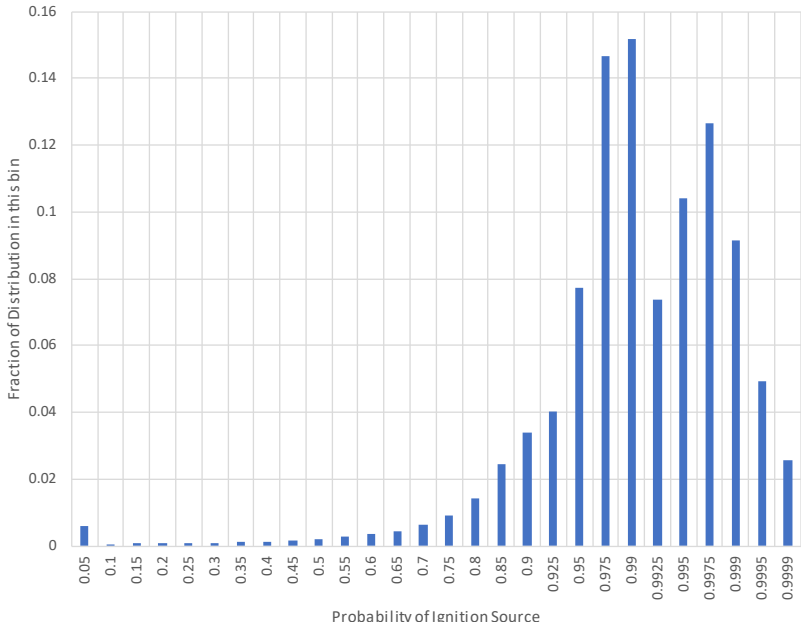
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2IGNSRC-AC (Type: Percent).	 <table border="1"> <caption>Estimated data for the histogram in Table C-12</caption> <thead> <tr> <th>Probability of Ignition Source (Bin Center)</th> <th>Fraction of Distribution in this Bin</th> </tr> </thead> <tbody> <tr><td>0.05</td><td>0.005</td></tr> <tr><td>0.1</td><td>0.002</td></tr> <tr><td>0.15</td><td>0.001</td></tr> <tr><td>0.2</td><td>0.001</td></tr> <tr><td>0.25</td><td>0.001</td></tr> <tr><td>0.3</td><td>0.001</td></tr> <tr><td>0.35</td><td>0.001</td></tr> <tr><td>0.4</td><td>0.001</td></tr> <tr><td>0.45</td><td>0.001</td></tr> <tr><td>0.5</td><td>0.001</td></tr> <tr><td>0.55</td><td>0.002</td></tr> <tr><td>0.6</td><td>0.003</td></tr> <tr><td>0.65</td><td>0.004</td></tr> <tr><td>0.7</td><td>0.006</td></tr> <tr><td>0.75</td><td>0.008</td></tr> <tr><td>0.8</td><td>0.012</td></tr> <tr><td>0.85</td><td>0.025</td></tr> <tr><td>0.9</td><td>0.035</td></tr> <tr><td>0.925</td><td>0.04</td></tr> <tr><td>0.95</td><td>0.075</td></tr> <tr><td>0.975</td><td>0.145</td></tr> <tr><td>0.99</td><td>0.15</td></tr> <tr><td>0.9925</td><td>0.075</td></tr> <tr><td>0.995</td><td>0.105</td></tr> <tr><td>0.9975</td><td>0.125</td></tr> <tr><td>0.999</td><td>0.09</td></tr> <tr><td>0.9995</td><td>0.045</td></tr> <tr><td>0.9999</td><td>0.025</td></tr> </tbody> </table>	Probability of Ignition Source (Bin Center)	Fraction of Distribution in this Bin	0.05	0.005	0.1	0.002	0.15	0.001	0.2	0.001	0.25	0.001	0.3	0.001	0.35	0.001	0.4	0.001	0.45	0.001	0.5	0.001	0.55	0.002	0.6	0.003	0.65	0.004	0.7	0.006	0.75	0.008	0.8	0.012	0.85	0.025	0.9	0.035	0.925	0.04	0.95	0.075	0.975	0.145	0.99	0.15	0.9925	0.075	0.995	0.105	0.9975	0.125	0.999	0.09	0.9995	0.045	0.9999	0.025
Probability of Ignition Source (Bin Center)	Fraction of Distribution in this Bin																																																										
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Basis: These three basic events represent the probabilities of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting) before, around the time of, and in the long time frame after vessel breach, respectively, when AC power is available. In this situation, the availability of ignition sources is considered to be highly likely due to the continuous operation of electrically powered equipment, and a point value probability of 0.99 was subjectively assigned. Ordinarily, the use of such a high “failure” probability would dictate that no uncertainty distribution is necessary, according to the NRC-developed parameter uncertainty quantification guidelines. However, there may well be cases in which a lower failure probability is worse; for example, if ignition is precluded from earlier time frames, it can lead to more oxygen and combustible gases being available for a large, potentially containment-challenging burn in a later time frame. Therefore, an uncertainty distribution was developed on the basis of user judgement. The converse of this event’s failure probability (i.e., the probability of there being no ignition source) was considered to be distributed lognormally with a mean value of 0.01 and an error factor of 10, consistent with the default method described in Section C.2 for other phenomenological issues. Since SAPHIRE does not allow for an uncertainty distribution of type “one minus lognormal”, it was then necessary to discretize the converse of the resulting probability distribution into a histogram for input to the code (pictured above). Note that the unevenness of the bin values in this figure are an artifact of the non-uniform widths of the bins, as it was necessary in the interest of precision to use finer increments as the bin value approaches unity (e.g., reducing the width of each bin from about 0.05 to 0.005 or less). Also, the slightly higher bin percentage at the left end of the distribution reflects the long lower tail of the original lognormal distribution. Note that the same histogram is used for the corresponding three basic events in the very early, early, and late time frames, although they are assumed to be uncorrelated (i.e., a separate independent random variable is used for each of the three events in any given Monte Carlo sample).																																																											
Contributors: Michael Zavisca																																																											

Table C-13 Parameter Uncertainty for 1-L2-BE-H2IGNSRC-E-NAC

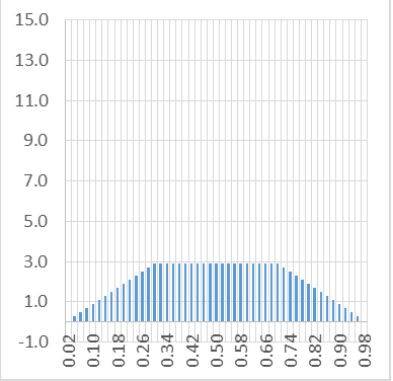
Distribution: Histogram See description below and PDF to the right. Encoded in the model as HIST-L2-H2IGNSRC-E-NAC.	
Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting), around the time 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 PRA 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. Contributors: Don Helton (lead), Hossein Esmaili, Tina Ghosh	

Table C-14 Parameter Uncertainty for 1-L2-BE-H2IGNSRC-E-NAC

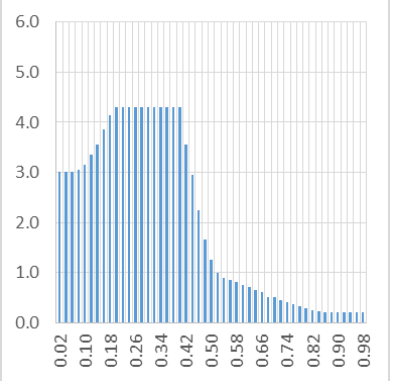
Distribution: Histogram See description below and PDF to the right. Encoded in the model as HIST-L2-H2IGNSRC-L-NAC.	
Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting), in the intermediate and late timeframe (i.e., well after the time of vessel breach but prior to containment quasi-static overpressure 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 PRA 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. Contributors: Don Helton (lead), Hossein Esmaili, Tina Ghosh	

Table C-15 Parameter Uncertainty for 1-L2-BE-H2IGNSRC-VE-NAC

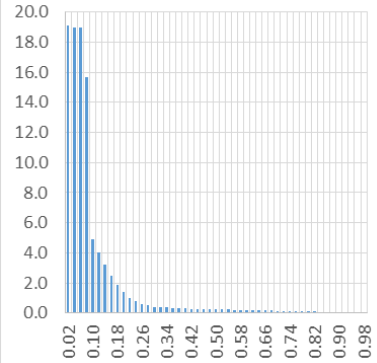
Distribution: Histogram See description below and PDF to the right. Encoded in the model as HIST-L2-H2IGNSRC-VE-NAC.	
Basis: This basic event represents the probability of having an ignition source present (spatially concurrent with where in containment combustible gases are collecting), in the very 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 PRA 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. Contributors: Don Helton (lead), Hossein Esmaili, Tina Ghosh	

Table C-16 Parameter Uncertainty for 1-L2-OP-EDMGB-SD1

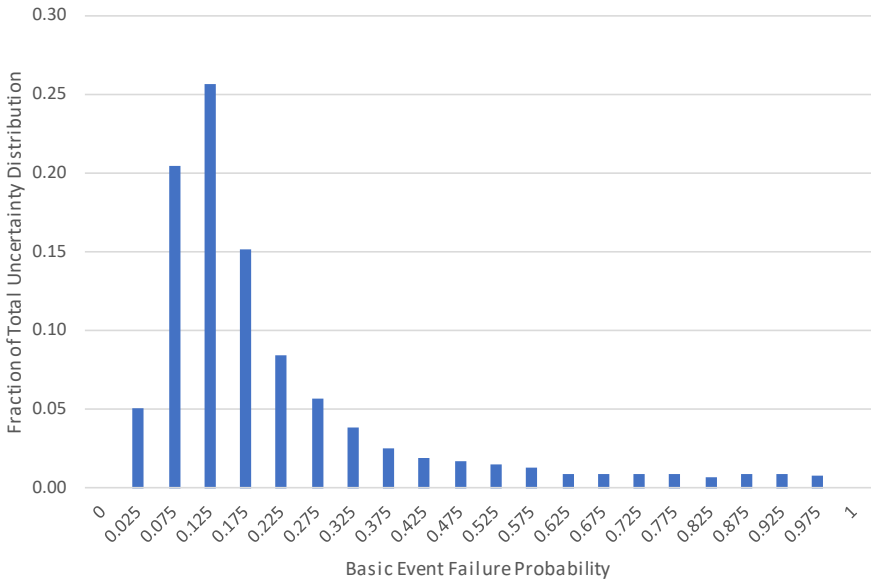
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-EDMGB1 (Type: Percent).	 The histogram displays the frequency of basic event failure probabilities. The x-axis represents the Basic Event Failure Probability from 0 to 1, and the y-axis represents the Fraction of Total Uncertainty Distribution from 0.00 to 0.30. The distribution is unimodal and right-skewed, with the highest frequency occurring in the 0.125 to 0.175 bin. <table border="1"> <thead> <tr> <th>Basic Event Failure Probability Bin</th> <th>Fraction of Total Uncertainty Distribution</th> </tr> </thead> <tbody> <tr><td>0.025 - 0.075</td><td>0.05</td></tr> <tr><td>0.075 - 0.125</td><td>0.20</td></tr> <tr><td>0.125 - 0.175</td><td>0.25</td></tr> <tr><td>0.175 - 0.225</td><td>0.15</td></tr> <tr><td>0.225 - 0.275</td><td>0.08</td></tr> <tr><td>0.275 - 0.325</td><td>0.05</td></tr> <tr><td>0.325 - 0.375</td><td>0.04</td></tr> <tr><td>0.375 - 0.425</td><td>0.03</td></tr> <tr><td>0.425 - 0.475</td><td>0.02</td></tr> <tr><td>0.475 - 0.525</td><td>0.02</td></tr> <tr><td>0.525 - 0.575</td><td>0.01</td></tr> <tr><td>0.575 - 0.625</td><td>0.01</td></tr> <tr><td>0.625 - 0.675</td><td>0.01</td></tr> <tr><td>0.675 - 0.725</td><td>0.01</td></tr> <tr><td>0.725 - 0.775</td><td>0.01</td></tr> <tr><td>0.775 - 0.825</td><td>0.01</td></tr> <tr><td>0.825 - 0.875</td><td>0.01</td></tr> <tr><td>0.875 - 0.925</td><td>0.01</td></tr> <tr><td>0.925 - 0.975</td><td>0.01</td></tr> <tr><td>0.975 - 1.000</td><td>0.01</td></tr> </tbody> </table>	Basic Event Failure Probability Bin	Fraction of Total Uncertainty Distribution	0.025 - 0.075	0.05	0.075 - 0.125	0.20	0.125 - 0.175	0.25	0.175 - 0.225	0.15	0.225 - 0.275	0.08	0.275 - 0.325	0.05	0.325 - 0.375	0.04	0.375 - 0.425	0.03	0.425 - 0.475	0.02	0.475 - 0.525	0.02	0.525 - 0.575	0.01	0.575 - 0.625	0.01	0.625 - 0.675	0.01	0.675 - 0.725	0.01	0.725 - 0.775	0.01	0.775 - 0.825	0.01	0.825 - 0.875	0.01	0.875 - 0.925	0.01	0.925 - 0.975	0.01	0.975 - 1.000	0.01
Basic Event Failure Probability Bin	Fraction of Total Uncertainty Distribution																																										
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Basis: The basic event 1-L2-OP-EDMGB-SD1 represents failure of the operators to arrest core damage and prevent vessel breach by means of alternate injection of water to the RCS, in severe accidents assigned to Category VI which occur post-refueling. It is somewhat unique in that its success is the product of two events: a human action (operators restore injection to the RCS) and a phenomenological question (the core remains in a coolable configuration as of the time when injection is restored). Therefore, the failure probability p_{net} of the basic event is given by $1 - (1 - p_{EDMG})(1 - p_{cool})$, where p_{EDMG} is the HEP of the action to restore injection and p_{cool} is the probability that the core is not coolable. The Level 2 PRA HRA for reactor-at-shutdown arrived at a point value of 0.14 for the HEP, and the method described in Section C.1 is used to assign a lognormal distribution with a mean of 0.14 and an error factor of 5 to this component of the probability. MELCOR results combined with user judgement were employed to arrive at the qualitative judgement that the core would likely be coolable by the time injection could be restored, for scenarios occurring post-refueling, and therefore a point value of 0.1 for failure of coolability was assigned. For this component of probability, a simple piecewise-uniform uncertainty distribution was assumed, with 90% of the cumulative distribution lying uniformly between values of 0 and 0.1, and the remaining 10% of the distribution uniformly between 0.1 and 1. This distribution therefore has the property that its mean value is 0.1. A simple Monte Carlo simulation was performed in Excel with 5000 samples for both p_{EDMG} and p_{cool} in order to arrive at a distribution for p_{net} according to the formula above. This distribution was then discretized to construct a histogram (illustrated above) for input to SAPHIRE.																																											
Contributors: Michael Zavisca																																											

Table C-17 Parameter Uncertainty for 1-L2-OP-EDMGB-SD2

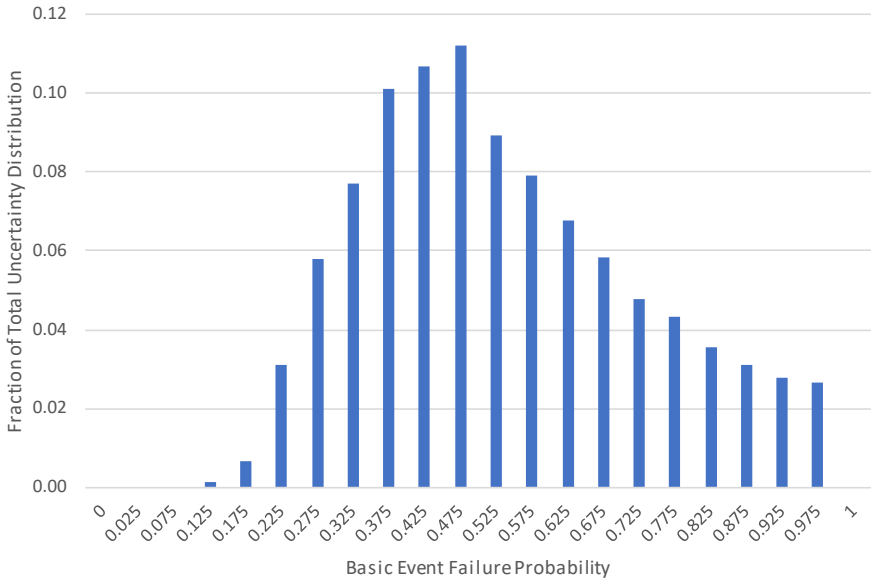
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-EDMGB2 (Type: Percent).	
Basis: The basic event 1-L2-OP-EDMGB-SD2 represents failure of the operators to arrest core damage and prevent vessel breach by means of alternate injection of water to the RCS, in severe accidents assigned to other than Category VI which occur post-refueling. It is somewhat unique in that its success is the product of two events: a human action (operators restore injection to the RCS) and a phenomenological question (the core remains in a coolable configuration as of the time when injection is restored). Therefore, the failure probability p_{net} of the basic event is given by $1 - (1 - p_{EDMG})(1 - p_{cool})$, where p_{EDMG} is the HEP of the action to restore injection and p_{cool} is the probability that the core is not coolable. The Level 2 PRA HRA for reactor-at-shutdown arrived at a point value of 0.52 for the HEP, and the method described in Section C.1 is used to assign a lognormal distribution with a mean of 0.52 and an error factor of 2.1 to this component of the probability. MELCOR results combined with user judgement were employed to arrive at the qualitative judgement that the core would likely be coolable by the time injection could be restored, for scenarios occurring post-refueling, and therefore a point value of 0.1 for failure of coolability was assigned. For this component of probability, a simple piecewise-uniform uncertainty distribution was assumed, with 90% of the cumulative distribution lying uniformly between values of 0 and 0.1, and the remaining 10% of the distribution uniformly between 0.1 and 1. This distribution therefore has the property that its mean value is 0.1. A simple Monte Carlo simulation was performed in Excel with 5000 samples for both p_{EDMG} and p_{cool} in order to arrive at a distribution for p_{net} according to the formula above. This distribution was then discretized to construct a histogram (illustrated above) for input to SAPHIRE.	
Contributors: Michael Zavisca	

Table C-18 Parameter Uncertainty for 1-L2-OP-EDMGB-SD3

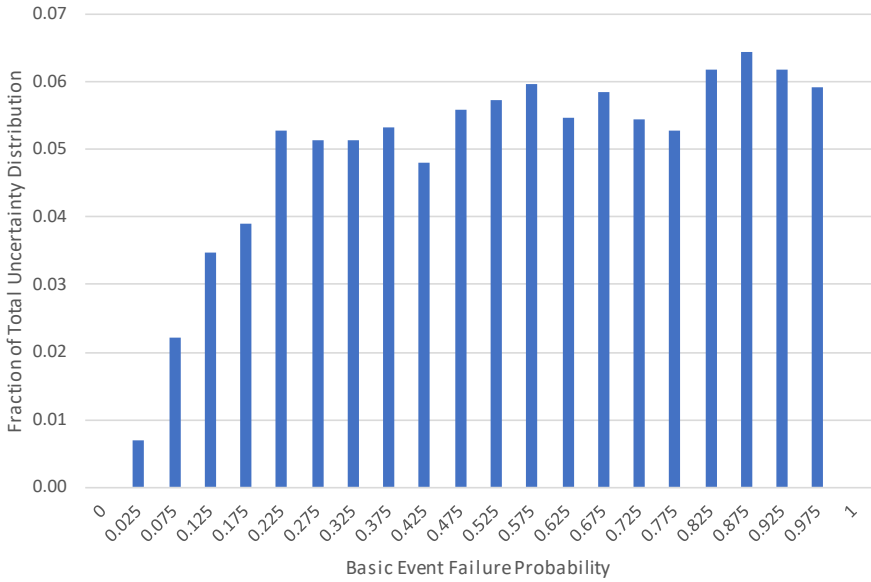
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-EDMGB3 (Type: Percent).	 <table border="1"> <caption>Estimated Data for Histogram</caption> <thead> <tr> <th>Basic Event Failure Probability Interval</th> <th>Fraction of Total Uncertainty Distribution</th> </tr> </thead> <tbody> <tr><td>0.025</td><td>0.007</td></tr> <tr><td>0.075</td><td>0.022</td></tr> <tr><td>0.125</td><td>0.035</td></tr> <tr><td>0.175</td><td>0.039</td></tr> <tr><td>0.225</td><td>0.053</td></tr> <tr><td>0.275</td><td>0.051</td></tr> <tr><td>0.325</td><td>0.051</td></tr> <tr><td>0.375</td><td>0.053</td></tr> <tr><td>0.425</td><td>0.048</td></tr> <tr><td>0.475</td><td>0.056</td></tr> <tr><td>0.525</td><td>0.058</td></tr> <tr><td>0.575</td><td>0.060</td></tr> <tr><td>0.625</td><td>0.055</td></tr> <tr><td>0.675</td><td>0.059</td></tr> <tr><td>0.725</td><td>0.055</td></tr> <tr><td>0.775</td><td>0.053</td></tr> <tr><td>0.825</td><td>0.062</td></tr> <tr><td>0.875</td><td>0.065</td></tr> <tr><td>0.925</td><td>0.062</td></tr> <tr><td>0.975</td><td>0.059</td></tr> </tbody> </table>	Basic Event Failure Probability Interval	Fraction of Total Uncertainty Distribution	0.025	0.007	0.075	0.022	0.125	0.035	0.175	0.039	0.225	0.053	0.275	0.051	0.325	0.051	0.375	0.053	0.425	0.048	0.475	0.056	0.525	0.058	0.575	0.060	0.625	0.055	0.675	0.059	0.725	0.055	0.775	0.053	0.825	0.062	0.875	0.065	0.925	0.062	0.975	0.059
Basic Event Failure Probability Interval	Fraction of Total Uncertainty Distribution																																										
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Basis: The basic event 1-L2-OP-EDMGB-SD3 represents failure of the operators to arrest core damage and prevent vessel breach by means of alternate injection of water to the RCS, in severe accidents assigned to Category VI which occur pre-refueling. It is somewhat unique in that its success is the product of two events: a human action (operators restore injection to the RCS) and a phenomenological question (the core remains in a coolable configuration as of the time when injection is restored). Therefore, the failure probability p_{net} of the basic event is given by $1 - (1 - p_{EDMG})(1 - p_{cool})$, where p_{EDMG} is the HEP of the action to restore injection and p_{cool} is the probability that the core is not coolable. The Level 2 PRA HRA for reactor at-shutdown arrived at a point value of 0.14 for the HEP, and the method described in Section C.1 is used to assign a lognormal distribution with a mean of 0.14 and an error factor of 5 to this component of the probability. MELCOR results combined with user judgement were employed to arrive at the qualitative judgement that the coolability of the core would be highly uncertain, since the amount of time required to align injection was about the same as the amount of time for the core to begin relocating significantly. Therefore, a uniform distribution was assumed for coolability between values of 0 and 1 (with a mean value of 0.5). A simple Monte Carlo simulation was performed in Excel with 5000 samples for both p_{EDMG} and p_{cool} in order to arrive at a distribution for p_{net} according to the formula above. This distribution was then discretized to construct a histogram (illustrated above) for input to SAPHIRE.																																											
Contributors: Michael Zavisca																																											

Table C-19 Parameter Uncertainty for 1-L2-OP-EDMGB-SD3

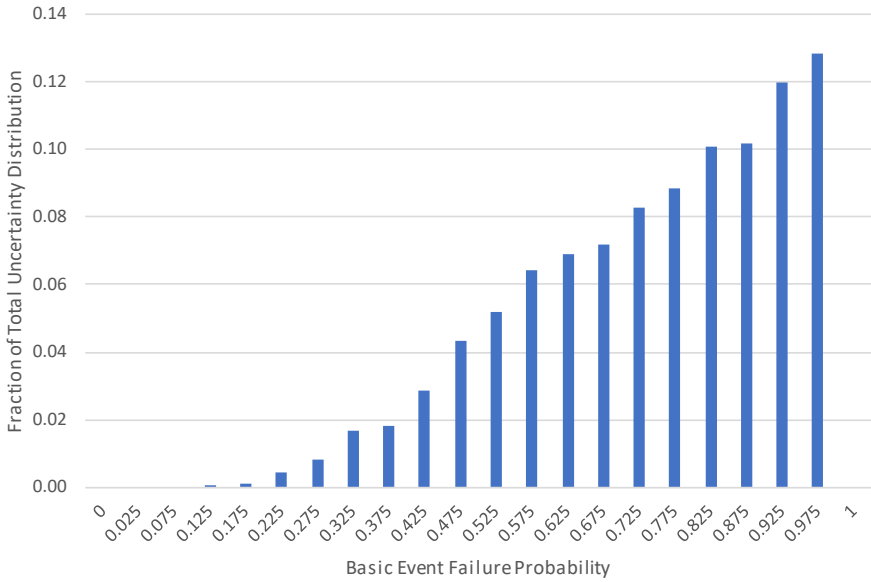
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-EDMGB4 (Type: Percent).	 <table border="1"> <caption>Estimated Data for Basic Event Failure Probability Histogram</caption> <thead> <tr> <th>Basic Event Failure Probability Range</th> <th>Fraction of Total Uncertainty Distribution</th> </tr> </thead> <tbody> <tr><td>0.000 - 0.025</td><td>0.000</td></tr> <tr><td>0.025 - 0.050</td><td>0.000</td></tr> <tr><td>0.050 - 0.075</td><td>0.000</td></tr> <tr><td>0.075 - 0.100</td><td>0.000</td></tr> <tr><td>0.100 - 0.125</td><td>0.000</td></tr> <tr><td>0.125 - 0.150</td><td>0.000</td></tr> <tr><td>0.150 - 0.175</td><td>0.000</td></tr> <tr><td>0.175 - 0.200</td><td>0.000</td></tr> <tr><td>0.200 - 0.225</td><td>0.001</td></tr> <tr><td>0.225 - 0.250</td><td>0.002</td></tr> <tr><td>0.250 - 0.275</td><td>0.005</td></tr> <tr><td>0.275 - 0.300</td><td>0.008</td></tr> <tr><td>0.300 - 0.325</td><td>0.015</td></tr> <tr><td>0.325 - 0.350</td><td>0.018</td></tr> <tr><td>0.350 - 0.375</td><td>0.028</td></tr> <tr><td>0.375 - 0.400</td><td>0.042</td></tr> <tr><td>0.400 - 0.425</td><td>0.052</td></tr> <tr><td>0.425 - 0.450</td><td>0.065</td></tr> <tr><td>0.450 - 0.475</td><td>0.068</td></tr> <tr><td>0.475 - 0.500</td><td>0.072</td></tr> <tr><td>0.500 - 0.525</td><td>0.082</td></tr> <tr><td>0.525 - 0.550</td><td>0.088</td></tr> <tr><td>0.550 - 0.575</td><td>0.100</td></tr> <tr><td>0.575 - 0.600</td><td>0.102</td></tr> <tr><td>0.600 - 0.625</td><td>0.118</td></tr> <tr><td>0.625 - 0.650</td><td>0.120</td></tr> <tr><td>0.650 - 0.675</td><td>0.130</td></tr> </tbody> </table>	Basic Event Failure Probability Range	Fraction of Total Uncertainty Distribution	0.000 - 0.025	0.000	0.025 - 0.050	0.000	0.050 - 0.075	0.000	0.075 - 0.100	0.000	0.100 - 0.125	0.000	0.125 - 0.150	0.000	0.150 - 0.175	0.000	0.175 - 0.200	0.000	0.200 - 0.225	0.001	0.225 - 0.250	0.002	0.250 - 0.275	0.005	0.275 - 0.300	0.008	0.300 - 0.325	0.015	0.325 - 0.350	0.018	0.350 - 0.375	0.028	0.375 - 0.400	0.042	0.400 - 0.425	0.052	0.425 - 0.450	0.065	0.450 - 0.475	0.068	0.475 - 0.500	0.072	0.500 - 0.525	0.082	0.525 - 0.550	0.088	0.550 - 0.575	0.100	0.575 - 0.600	0.102	0.600 - 0.625	0.118	0.625 - 0.650	0.120	0.650 - 0.675	0.130
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Basis: The basic event 1-L2-OP-EDMGB-SD4 represents failure of the operators to arrest core damage and prevent vessel breach by means of alternate injection of water to the RCS, in severe accidents assigned to other than Category VI which occur pre-refueling. It is somewhat unique in that its success is the product of two events: a human action (operators restore injection to the RCS) and a phenomenological question (the core remains in a coolable configuration as of the time when injection is restored). Therefore, the failure probability p_{net} of the basic event is given by $1 - (1 - p_{EDMG})(1 - p_{cool})$, where p_{EDMG} is the HEP of the action to restore injection and p_{cool} is the probability that the core is not coolable. The Level 2 PRA HRA for reactor-at-shutdown arrived at a point value of 0.52 for the HEP, and the method described in Section C.1 is used to assign a lognormal distribution with a mean of 0.52 and an error factor of 2.1 to this component of the probability. MELCOR results combined with user judgement were employed to arrive at the qualitative judgement that the coolability of the core would be highly uncertain, since the amount of time required to align injection was about the same as the amount of time for the core to begin relocating significantly. Therefore, a uniform distribution was assumed for coolability between values of 0 and 1 (with a mean value of 0.5). A simple Monte Carlo simulation was performed in Excel with 5000 samples for both p_{EDMG} and p_{cool} in order to arrive at a distribution for p_{net} according to the formula above. This distribution was then discretized to construct a histogram (illustrated above) for input to SAPHIRE.																																																									
Contributors: Michael Zavisca																																																									

Table C-20 Parameter Uncertainty for 1-L2-BE-H2CF-L-CHRNPB-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-CHRNPB-SD (Type: Percent).	L-NPB Histogram <table border="1"> <caption>Approximate data points for L-NPB Histogram</caption> <thead> <tr> <th>CCFP</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0</td><td>0.42</td></tr> <tr><td>0.05</td><td>0.26</td></tr> <tr><td>0.1</td><td>0.06</td></tr> <tr><td>0.15</td><td>0.01</td></tr> <tr><td>0.2</td><td>0.01</td></tr> <tr><td>0.25</td><td>0.01</td></tr> <tr><td>0.3</td><td>0.01</td></tr> <tr><td>0.35</td><td>0.01</td></tr> <tr><td>0.4</td><td>0.01</td></tr> <tr><td>0.45</td><td>0.01</td></tr> <tr><td>0.5</td><td>0.01</td></tr> <tr><td>0.55</td><td>0.01</td></tr> <tr><td>0.6</td><td>0.01</td></tr> <tr><td>0.65</td><td>0.01</td></tr> <tr><td>0.7</td><td>0.01</td></tr> <tr><td>0.75</td><td>0.04</td></tr> <tr><td>0.8</td><td>0.03</td></tr> <tr><td>0.85</td><td>0.02</td></tr> <tr><td>0.9</td><td>0.01</td></tr> <tr><td>0.95</td><td>0.01</td></tr> <tr><td>0.999</td><td>0.04</td></tr> <tr><td>1</td><td>0.01</td></tr> </tbody> </table>	CCFP	Probability	0	0.42	0.05	0.26	0.1	0.06	0.15	0.01	0.2	0.01	0.25	0.01	0.3	0.01	0.35	0.01	0.4	0.01	0.45	0.01	0.5	0.01	0.55	0.01	0.6	0.01	0.65	0.01	0.7	0.01	0.75	0.04	0.8	0.03	0.85	0.02	0.9	0.01	0.95	0.01	0.999	0.04	1	0.01
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Basis: This basic event represents the conditional probability of containment failure given a global hydrogen/carbon monoxide deflagration in the late time frame of a severe accident scenario in which containment heat removal (i.e., containment sprays and/or fan coolers) is effective, and in which no earlier combustion events occurred in the time frames around or before vessel breach. Its uncertainty distribution is output directly by the ERPRA-BURN code which had also been used to calculate the mean (point) value of this probability, in the form of a Percent-type histogram (see figure above). This histogram is input to SAPHIRE directly. Note that the distribution predicted by ERPRA-BURN is highly bimodal (mainly very low and very high conditional containment failure probabilities), reflecting the high degree of uncertainty in the stochastic process of multiple burns occurring in the late time frame. Contributors: Michael Zavisca																																															

Table C-21 Parameter Uncertainty for 1-L2-BE-H2CF-L-CHRPB-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-CHRPB-SD (Type: Percent).	L-PB Histogram <table border="1"> <caption>Estimated Data for L-PB Histogram</caption> <thead> <tr> <th>CCFP Bin Range</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0 - 0.05</td><td>0.58</td></tr> <tr><td>0.05 - 0.1</td><td>0.22</td></tr> <tr><td>0.1 - 0.15</td><td>0.07</td></tr> <tr><td>0.15 - 0.2</td><td>0.02</td></tr> <tr><td>0.2 - 0.25</td><td>0.02</td></tr> <tr><td>0.25 - 0.3</td><td>0.01</td></tr> <tr><td>0.3 - 0.35</td><td>0.01</td></tr> <tr><td>0.35 - 0.4</td><td>0.01</td></tr> <tr><td>0.4 - 0.45</td><td>0.01</td></tr> <tr><td>0.45 - 0.5</td><td>0.01</td></tr> <tr><td>0.5 - 0.55</td><td>0.01</td></tr> <tr><td>0.55 - 0.6</td><td>0.01</td></tr> <tr><td>0.6 - 0.65</td><td>0.01</td></tr> <tr><td>0.65 - 0.7</td><td>0.01</td></tr> <tr><td>0.7 - 0.75</td><td>0.01</td></tr> <tr><td>0.75 - 0.8</td><td>0.01</td></tr> <tr><td>0.8 - 0.85</td><td>0.01</td></tr> <tr><td>0.85 - 0.9</td><td>0.01</td></tr> <tr><td>0.9 - 0.95</td><td>0.01</td></tr> <tr><td>0.95 - 0.999</td><td>0.01</td></tr> <tr><td>0.999 - 1</td><td>0.01</td></tr> </tbody> </table>	CCFP Bin Range	Probability	0 - 0.05	0.58	0.05 - 0.1	0.22	0.1 - 0.15	0.07	0.15 - 0.2	0.02	0.2 - 0.25	0.02	0.25 - 0.3	0.01	0.3 - 0.35	0.01	0.35 - 0.4	0.01	0.4 - 0.45	0.01	0.45 - 0.5	0.01	0.5 - 0.55	0.01	0.55 - 0.6	0.01	0.6 - 0.65	0.01	0.65 - 0.7	0.01	0.7 - 0.75	0.01	0.75 - 0.8	0.01	0.8 - 0.85	0.01	0.85 - 0.9	0.01	0.9 - 0.95	0.01	0.95 - 0.999	0.01	0.999 - 1	0.01
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Contributors: Michael Zavisca																																													

Table C-22 Parameter Uncertainty for 1-L2-BE-H2CF-L-LOINPB-SD

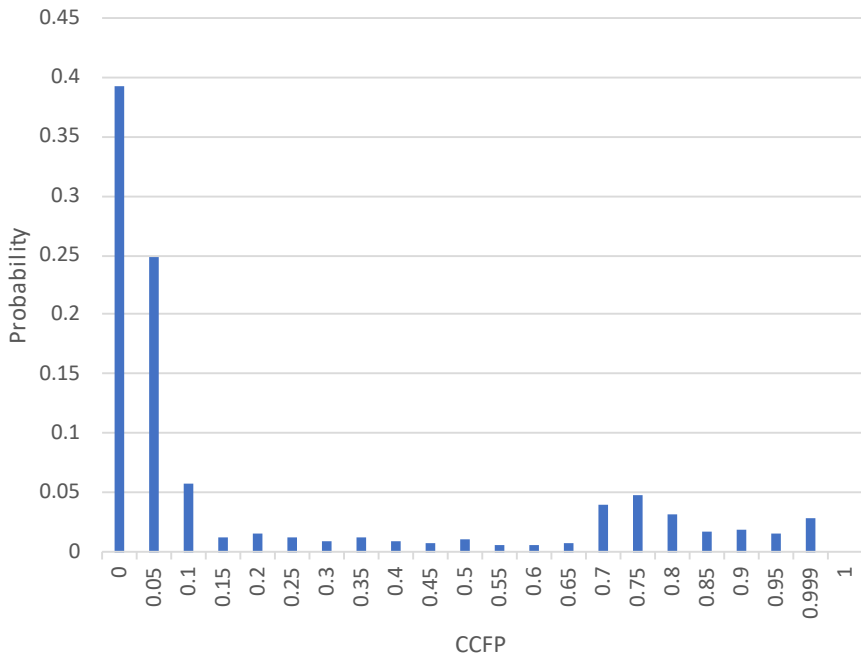
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-LOINPB-SD (Type: Percent).	L-NPB Histogram  <table border="1"> <caption>Approximate data points for L-NPB Histogram</caption> <thead> <tr> <th>CCFP</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0</td><td>0.39</td></tr> <tr><td>0.05</td><td>0.25</td></tr> <tr><td>0.1</td><td>0.06</td></tr> <tr><td>0.15</td><td>0.01</td></tr> <tr><td>0.2</td><td>0.01</td></tr> <tr><td>0.25</td><td>0.01</td></tr> <tr><td>0.3</td><td>0.01</td></tr> <tr><td>0.35</td><td>0.01</td></tr> <tr><td>0.4</td><td>0.01</td></tr> <tr><td>0.45</td><td>0.01</td></tr> <tr><td>0.5</td><td>0.01</td></tr> <tr><td>0.55</td><td>0.01</td></tr> <tr><td>0.6</td><td>0.01</td></tr> <tr><td>0.65</td><td>0.01</td></tr> <tr><td>0.7</td><td>0.04</td></tr> <tr><td>0.75</td><td>0.05</td></tr> <tr><td>0.8</td><td>0.03</td></tr> <tr><td>0.85</td><td>0.02</td></tr> <tr><td>0.9</td><td>0.02</td></tr> <tr><td>0.95</td><td>0.02</td></tr> <tr><td>0.999</td><td>0.03</td></tr> </tbody> </table>	CCFP	Probability	0	0.39	0.05	0.25	0.1	0.06	0.15	0.01	0.2	0.01	0.25	0.01	0.3	0.01	0.35	0.01	0.4	0.01	0.45	0.01	0.5	0.01	0.55	0.01	0.6	0.01	0.65	0.01	0.7	0.04	0.75	0.05	0.8	0.03	0.85	0.02	0.9	0.02	0.95	0.02	0.999	0.03
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Table C-23 Parameter Uncertainty for 1-L2-BE-H2CF-L-LOIPB-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-LOIPB-SD (Type: Percent).	L-PB Histogram <table border="1"> <caption>Estimated Data for L-PB Histogram</caption> <thead> <tr> <th>CCFP Bin Range</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0 - 0.05</td><td>0.53</td></tr> <tr><td>0.05 - 0.1</td><td>0.22</td></tr> <tr><td>0.1 - 0.15</td><td>0.09</td></tr> <tr><td>0.15 - 0.2</td><td>0.03</td></tr> <tr><td>0.2 - 0.25</td><td>0.02</td></tr> <tr><td>0.25 - 0.3</td><td>0.02</td></tr> <tr><td>0.3 - 0.35</td><td>0.02</td></tr> <tr><td>0.35 - 0.4</td><td>0.01</td></tr> <tr><td>0.4 - 0.45</td><td>0.01</td></tr> <tr><td>0.45 - 0.5</td><td>0.01</td></tr> <tr><td>0.5 - 0.55</td><td>0.01</td></tr> <tr><td>0.55 - 0.6</td><td>0.01</td></tr> <tr><td>0.6 - 0.65</td><td>0.01</td></tr> <tr><td>0.65 - 0.7</td><td>0.01</td></tr> <tr><td>0.7 - 0.75</td><td>0.01</td></tr> <tr><td>0.75 - 0.8</td><td>0.01</td></tr> <tr><td>0.8 - 0.85</td><td>0.01</td></tr> <tr><td>0.85 - 0.9</td><td>0.01</td></tr> <tr><td>0.9 - 0.95</td><td>0.01</td></tr> <tr><td>0.95 - 0.999</td><td>0.01</td></tr> <tr><td>0.999 - 1</td><td>0.01</td></tr> </tbody> </table>	CCFP Bin Range	Probability	0 - 0.05	0.53	0.05 - 0.1	0.22	0.1 - 0.15	0.09	0.15 - 0.2	0.03	0.2 - 0.25	0.02	0.25 - 0.3	0.02	0.3 - 0.35	0.02	0.35 - 0.4	0.01	0.4 - 0.45	0.01	0.45 - 0.5	0.01	0.5 - 0.55	0.01	0.55 - 0.6	0.01	0.6 - 0.65	0.01	0.65 - 0.7	0.01	0.7 - 0.75	0.01	0.75 - 0.8	0.01	0.8 - 0.85	0.01	0.85 - 0.9	0.01	0.9 - 0.95	0.01	0.95 - 0.999	0.01	0.999 - 1	0.01
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Table C-24 Parameter Uncertainty for 1-L2-BE-H2CF-L-NACNPB-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-NACNPB-SD (Type: Percent).	L-NPB Histogram <table border="1"> <caption>Approximate data for L-NPB Histogram</caption> <thead> <tr> <th>CCFP</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0</td><td>0.16</td></tr> <tr><td>0.05</td><td>0.10</td></tr> <tr><td>0.1</td><td>0.04</td></tr> <tr><td>0.15</td><td>0.02</td></tr> <tr><td>0.2</td><td>0.01</td></tr> <tr><td>0.25</td><td>0.01</td></tr> <tr><td>0.3</td><td>0.01</td></tr> <tr><td>0.35</td><td>0.01</td></tr> <tr><td>0.4</td><td>0.01</td></tr> <tr><td>0.45</td><td>0.01</td></tr> <tr><td>0.5</td><td>0.01</td></tr> <tr><td>0.55</td><td>0.01</td></tr> <tr><td>0.6</td><td>0.01</td></tr> <tr><td>0.65</td><td>0.01</td></tr> <tr><td>0.7</td><td>0.01</td></tr> <tr><td>0.75</td><td>0.01</td></tr> <tr><td>0.8</td><td>0.01</td></tr> <tr><td>0.85</td><td>0.01</td></tr> <tr><td>0.9</td><td>0.01</td></tr> <tr><td>0.95</td><td>0.01</td></tr> <tr><td>0.999</td><td>0.10</td></tr> <tr><td>1</td><td>0.44</td></tr> </tbody> </table>	CCFP	Probability	0	0.16	0.05	0.10	0.1	0.04	0.15	0.02	0.2	0.01	0.25	0.01	0.3	0.01	0.35	0.01	0.4	0.01	0.45	0.01	0.5	0.01	0.55	0.01	0.6	0.01	0.65	0.01	0.7	0.01	0.75	0.01	0.8	0.01	0.85	0.01	0.9	0.01	0.95	0.01	0.999	0.10	1	0.44
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Table C-25 Parameter Uncertainty for 1-L2-BE-H2CF-L-NACPB-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-NACPB-SD (Type: Percent).	L-PB Histogram <table border="1"> <caption>Estimated Data for L-PB Histogram</caption> <thead> <tr> <th>CCFP</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0</td><td>0.35</td></tr> <tr><td>0.05</td><td>0.12</td></tr> <tr><td>0.1</td><td>0.06</td></tr> <tr><td>0.15</td><td>0.02</td></tr> <tr><td>0.2</td><td>0.02</td></tr> <tr><td>0.25</td><td>0.02</td></tr> <tr><td>0.3</td><td>0.02</td></tr> <tr><td>0.35</td><td>0.02</td></tr> <tr><td>0.4</td><td>0.02</td></tr> <tr><td>0.45</td><td>0.02</td></tr> <tr><td>0.5</td><td>0.02</td></tr> <tr><td>0.55</td><td>0.01</td></tr> <tr><td>0.6</td><td>0.02</td></tr> <tr><td>0.65</td><td>0.02</td></tr> <tr><td>0.7</td><td>0.02</td></tr> <tr><td>0.75</td><td>0.02</td></tr> <tr><td>0.8</td><td>0.02</td></tr> <tr><td>0.85</td><td>0.02</td></tr> <tr><td>0.9</td><td>0.02</td></tr> <tr><td>0.95</td><td>0.02</td></tr> <tr><td>0.999</td><td>0.09</td></tr> <tr><td>1</td><td>0.09</td></tr> </tbody> </table>	CCFP	Probability	0	0.35	0.05	0.12	0.1	0.06	0.15	0.02	0.2	0.02	0.25	0.02	0.3	0.02	0.35	0.02	0.4	0.02	0.45	0.02	0.5	0.02	0.55	0.01	0.6	0.02	0.65	0.02	0.7	0.02	0.75	0.02	0.8	0.02	0.85	0.02	0.9	0.02	0.95	0.02	0.999	0.09	1	0.09
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Table C-26 Parameter Uncertainty for 1-L2-BE-H2CF-L-NCHRN-SD

Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-NCHRN-SD (Type: Percent).	L-NPB Histogram <table border="1"> <caption>Estimated Data for L-NPB Histogram</caption> <thead> <tr> <th>CCFP Bin</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0.00-0.05</td><td>0.38</td></tr> <tr><td>0.05-0.10</td><td>0.22</td></tr> <tr><td>0.10-0.15</td><td>0.07</td></tr> <tr><td>0.15-0.20</td><td>0.02</td></tr> <tr><td>0.20-0.25</td><td>0.02</td></tr> <tr><td>0.25-0.30</td><td>0.01</td></tr> <tr><td>0.30-0.35</td><td>0.01</td></tr> <tr><td>0.35-0.40</td><td>0.01</td></tr> <tr><td>0.40-0.45</td><td>0.01</td></tr> <tr><td>0.45-0.50</td><td>0.01</td></tr> <tr><td>0.50-0.55</td><td>0.01</td></tr> <tr><td>0.55-0.60</td><td>0.01</td></tr> <tr><td>0.60-0.65</td><td>0.01</td></tr> <tr><td>0.65-0.70</td><td>0.01</td></tr> <tr><td>0.70-0.75</td><td>0.01</td></tr> <tr><td>0.75-0.80</td><td>0.01</td></tr> <tr><td>0.80-0.85</td><td>0.01</td></tr> <tr><td>0.85-0.90</td><td>0.01</td></tr> <tr><td>0.90-0.95</td><td>0.01</td></tr> <tr><td>0.95-1.00</td><td>0.06</td></tr> <tr><td>1.00-1.05</td><td>0.08</td></tr> </tbody> </table>	CCFP Bin	Probability	0.00-0.05	0.38	0.05-0.10	0.22	0.10-0.15	0.07	0.15-0.20	0.02	0.20-0.25	0.02	0.25-0.30	0.01	0.30-0.35	0.01	0.35-0.40	0.01	0.40-0.45	0.01	0.45-0.50	0.01	0.50-0.55	0.01	0.55-0.60	0.01	0.60-0.65	0.01	0.65-0.70	0.01	0.70-0.75	0.01	0.75-0.80	0.01	0.80-0.85	0.01	0.85-0.90	0.01	0.90-0.95	0.01	0.95-1.00	0.06	1.00-1.05	0.08
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0.35-0.40	0.01																																												
0.40-0.45	0.01																																												
0.45-0.50	0.01																																												
0.50-0.55	0.01																																												
0.55-0.60	0.01																																												
0.60-0.65	0.01																																												
0.65-0.70	0.01																																												
0.70-0.75	0.01																																												
0.75-0.80	0.01																																												
0.80-0.85	0.01																																												
0.85-0.90	0.01																																												
0.90-0.95	0.01																																												
0.95-1.00	0.06																																												
1.00-1.05	0.08																																												
Basis: This basic event represents the conditional probability of containment failure given a global hydrogen/carbon monoxide deflagration in the late time frame of a severe accident scenario with AC power but no containment heat removal, and in which no earlier combustion events occurred in the time frames around or before vessel breach. Its uncertainty distribution is output directly by the ERPRA-BURN code which had also been used to calculate the mean (point) value of this probability, in the form of a Percent-type histogram (see figure above). This histogram is input to SAPHIRE directly. Note that the distribution predicted by ERPRA-BURN is highly bimodal (mainly very low and very high conditional containment failure probabilities), reflecting the high degree of uncertainty in the stochastic process of multiple burns occurring in the late time frame. Contributors: Michael Zavisca																																													

Table C-27 Parameter Uncertainty for 1-L2-BE-H2CF-L-NCHRPB-SD

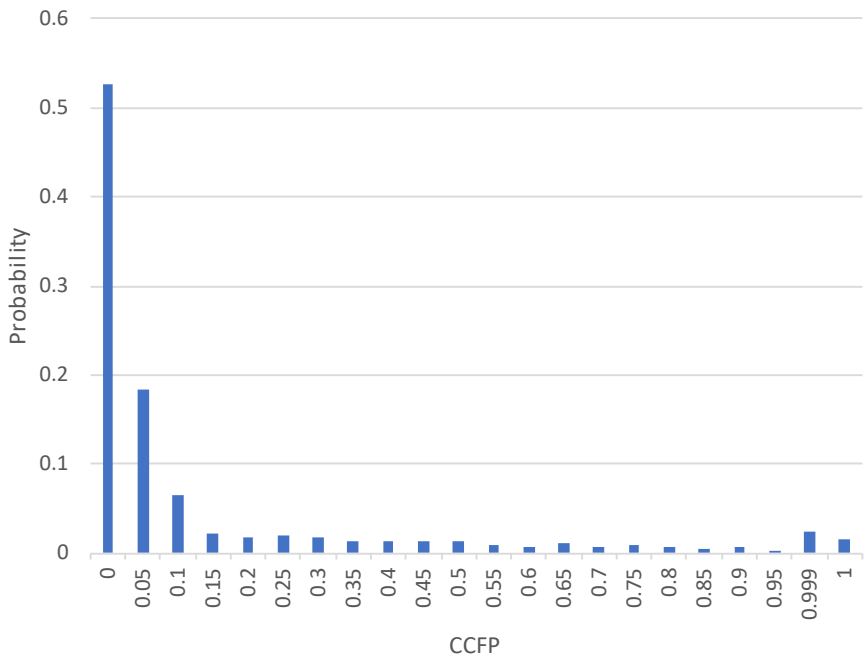
Distribution: Histogram See description below and Percent-type histogram to the right. Encoded in the model as HIST-L2-H2CF-L-NCHRPB-SD (Type: Percent).	L-PB Histogram  <table border="1"> <caption>Approximate data for L-PB Histogram</caption> <thead> <tr> <th>CCFP</th> <th>Probability</th> </tr> </thead> <tbody> <tr><td>0</td><td>0.53</td></tr> <tr><td>0.05</td><td>0.18</td></tr> <tr><td>0.1</td><td>0.06</td></tr> <tr><td>0.15</td><td>0.02</td></tr> <tr><td>0.2</td><td>0.02</td></tr> <tr><td>0.25</td><td>0.02</td></tr> <tr><td>0.3</td><td>0.02</td></tr> <tr><td>0.35</td><td>0.01</td></tr> <tr><td>0.4</td><td>0.01</td></tr> <tr><td>0.45</td><td>0.01</td></tr> <tr><td>0.5</td><td>0.01</td></tr> <tr><td>0.55</td><td>0.01</td></tr> <tr><td>0.6</td><td>0.01</td></tr> <tr><td>0.65</td><td>0.01</td></tr> <tr><td>0.7</td><td>0.01</td></tr> <tr><td>0.75</td><td>0.01</td></tr> <tr><td>0.8</td><td>0.01</td></tr> <tr><td>0.85</td><td>0.01</td></tr> <tr><td>0.9</td><td>0.01</td></tr> <tr><td>0.95</td><td>0.01</td></tr> <tr><td>0.999</td><td>0.02</td></tr> <tr><td>1</td><td>0.02</td></tr> </tbody> </table>	CCFP	Probability	0	0.53	0.05	0.18	0.1	0.06	0.15	0.02	0.2	0.02	0.25	0.02	0.3	0.02	0.35	0.01	0.4	0.01	0.45	0.01	0.5	0.01	0.55	0.01	0.6	0.01	0.65	0.01	0.7	0.01	0.75	0.01	0.8	0.01	0.85	0.01	0.9	0.01	0.95	0.01	0.999	0.02	1	0.02
CCFP	Probability																																														
0	0.53																																														
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0.999	0.02																																														
1	0.02																																														
Basis: This basic event represents the conditional probability of containment failure given a global hydrogen/carbon monoxide deflagration in the late time frame of a severe accident scenario with AC power but no containment heat removal, and in which one or more earlier combustion events occurred in the time frames around or before vessel breach. Its uncertainty distribution is output directly by the ERPRA-BURN code which had also been used to calculate the mean (point) value of this probability, in the form of a Percent-type histogram (see figure above). This histogram is input to SAPHIRE directly. Note that the distribution predicted by ERPRA-BURN is highly bimodal (mainly very low and very high conditional containment failure probabilities), reflecting the high degree of uncertainty in the stochastic process of multiple burns occurring in the late time frame. Contributors: Michael Zavisca																																															

Table C-28 Parameter Uncertainty for 1-L2-BE-IVSE-LP-SD

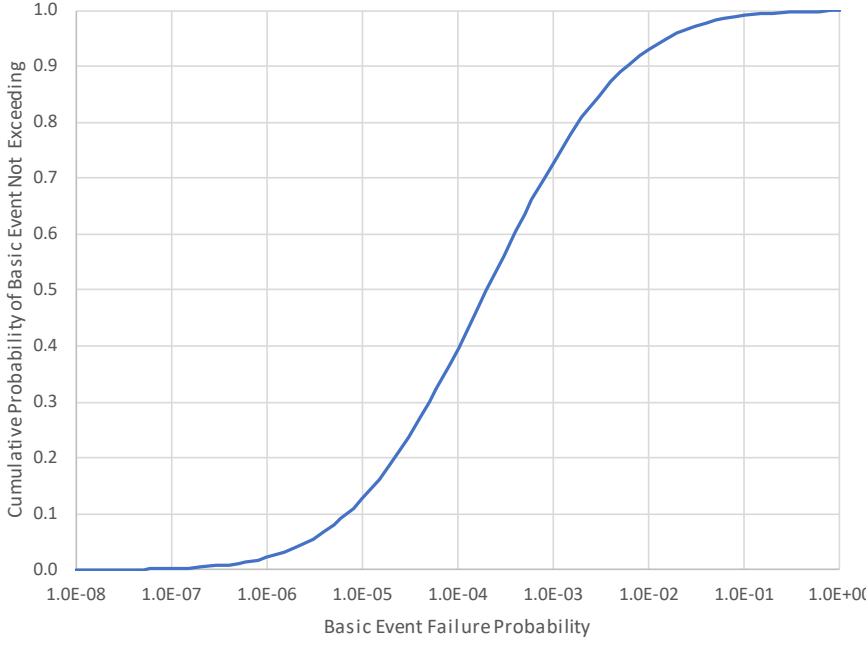
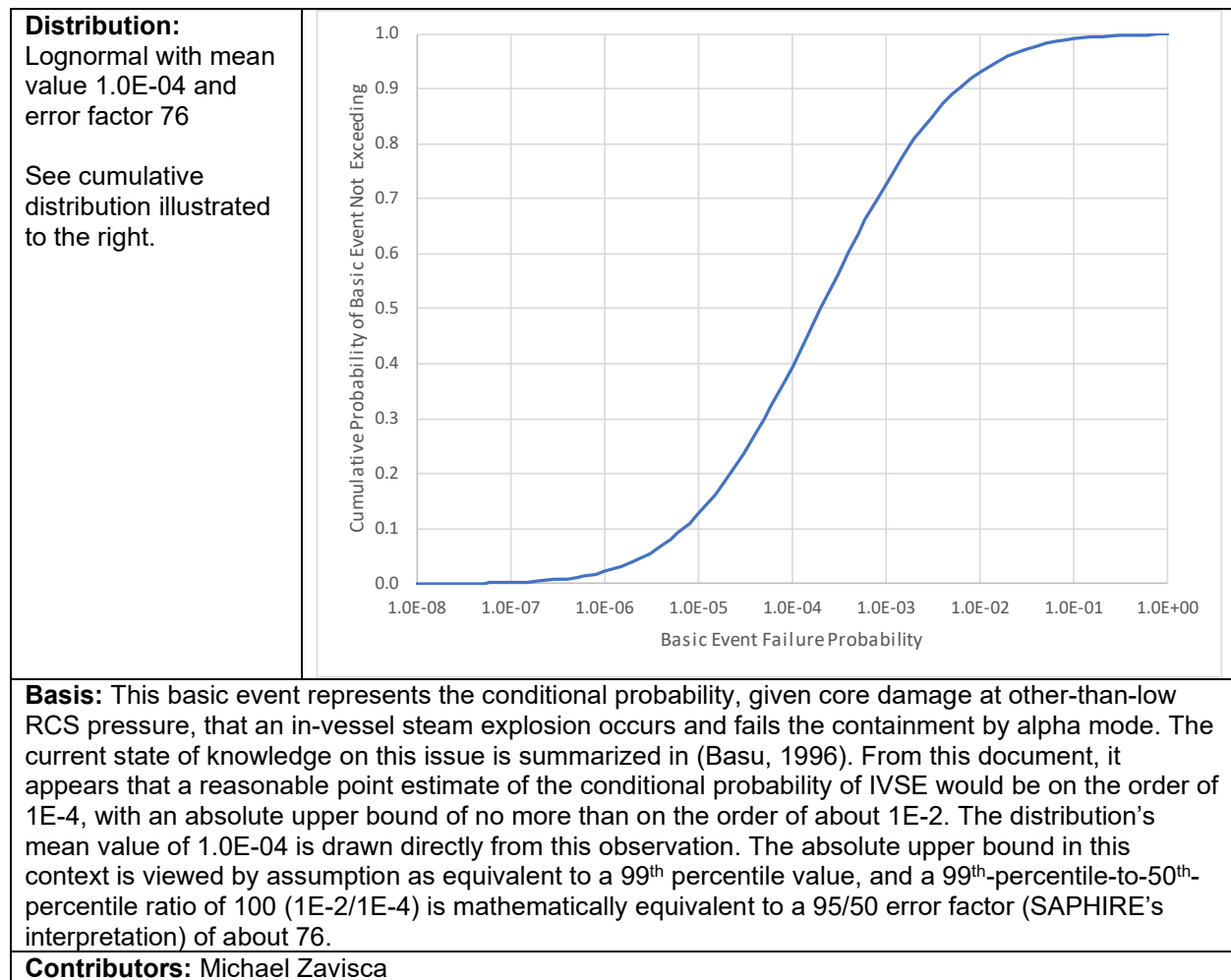
Distribution: Lognormal with mean value 2.0E-04 and error factor 76 See cumulative distribution illustrated to the right.	
Basis: This basic event represents the conditional probability, given core damage at low RCS pressure, that an in-vessel steam explosion occurs and fails the containment by alpha mode. The current state of knowledge on this issue is summarized in (Basu, 1996). From this document, it appears that a reasonable point estimate of the conditional probability of IVSE would be on the order of 1E-4, with an absolute upper bound of no more than on the order of about 1E-2. However, IVSE is also viewed as somewhat more likely under conditions of low RCS pressure, and therefore the mean value for this event in the model was assumed to be 2.0E-04. The absolute upper bound in this context is viewed by assumption as equivalent to a 99th percentile value, and a 99th-percentile-to-50th-percentile ratio of 100 (2E-2/2E-4) is mathematically equivalent to a 95/50 error factor (SAPHIRE's interpretation) of about 76.	
Contributors: Michael Zavisca	

Table C-29 Parameter Uncertainty for 1-L2-BE-IVSE-LP-SD



APPENDIX D

HUMAN RELIABILITY ANALYSIS FOR LEVEL 1/LEVEL 2 BRIDGE TREE CONTAINMENT ACTIONS

D.1 Introduction

This appendix documents the human reliability analysis (HRA) performed to support the Level 1 to Level 2 PRA interface. The human actions associated with containment systems, which are not modeled in the Level 1 PRA, are evaluated here. The bridge tree refers to the modeling element that is used to transfer the Level 1 PRA sequences to the Level 2 PRA and assess the status of containment systems. The bridge tree is described in Section 2.1.1.

The following top events are treated in the bridge tree:

- containment closure
- containment isolation system
- containment spray system
- containment cooling unit (CCU) system

Section D.2 of this report describes the human actions associated with the first two items, containment closure and containment isolation. The next sections describe the detailed human failure event (HFE) evaluations for containment closure and isolation actions. Section D.3 describes the HFEs related to closing the large containment openings (i.e., the equipment hatch and personnel airlock). Section D.4 describes the HFEs for isolating other containment penetrations.

There are no shutdown-specific actions modeled for the containment spray system. For POSs where the containment spray system is active, the modeling is treated the same as the at-power model and using the same human failure events. For most of the modeled shutdown POSs the containment spray system is not in service and would require manual action to restore. In these cases, there are other conditions that make it unlikely that aligning containment spray would be successful. These can include station blackout conditions and exhaustion of RWST inventory due to RCS injection for alternate core cooling. Due to these conditions, there is no credit given for manually restoring the containment spray system.

Although restoring containment spray is not modeled in the bridge tree, it is assessed as part of the post-core-damage accident management modeling. The approach for post-core-damage human reliability analysis for shutdown is described in Appendix E.

Operation of the CCU system can have a significant impact on the likelihood of containment failure. Though, for the majority of the refueling outage the CCUs are not in operation, and they may be unavailable due to maintenance activities on the CCUs or supporting systems. Operator actions are required to place the CCUs in operation. Section D.5 describes the actions related to the CCU system.

The SPAR-H method (NRC, 2005a) along with guidance in (NRC, 2006) and (Whaley, 2011) was used to quantify human error probabilities (HEPs) for each HFE. The results of the HEP quantification are shown in Table C-2.

A summary of the HFE dependency analysis is provided in Section D.6 .

D.2 Containment Isolation and Integrity

During low power and shutdown (LPSD), the major containment openings that may require closure include the main equipment hatch, the personnel air lock, and containment purge system lines. In addition, there are several small (e.g., 0.75 inch-diameter) containment penetrations that could be unisolated during an outage. Actions for establishing containment integrity are initiated the plant abnormal operating procedures for loss of residual heat removal or other applicable accident conditions.

Depending on the plant operating state (POS), the equipment hatch and the personnel air lock may be open or closed, and the purge system may be in service. Table D-1 summarizes the expected conditions for the containment openings in each POS that require appropriate operator actions. The estimated times for the RCS to reach saturation temperature during different POS conditions are based on the plant-specific calculations that were performed as part of the success criteria analysis for the Level 1 shutdown PRA (NRC, 2025). The table includes estimates of the time to heat up the containment air temperature to 140 °F, which is used as a reference upper limit of the habitability conditions for operators to perform actions in containment. These times were based on plant-specific calculations performed with MELCOR and address different boundary conditions (i.e., containment heat up with the hatch fully opened or closed, CCUs operating or not). For most of the human reliability estimates, the timings of the actions are assessed for completion prior to the RCS reaching saturation temperature. The assumptions used in the human reliability analysis of the modeled bridge tree containment actions are solely for the purposes of the PRA and should not be viewed as superseding and established regulatory positions and industry guidance to maintain the capability to close containment before steaming inside containment.

The following summarizes the various actions and the time requirement for the containment closure.

D.2.1 Equipment Hatch Closure

The reference plant's equipment hatch is a 20-foot diameter pressure-seating hatch design with 30 bolts that swing away from the door. The hatch could be closed using a minimum of four swing bolts approximately 90 degrees apart. The reference plant also has an emergency hatch closure procedure, which provides closure instructions for situations with power or loss of power condition and the preparation of supporting equipment. The reference plant procedures indicate that the hatch closure crew should be able to close the equipment hatch within 25 minutes of notification of station blackout conditions. During the mid-loop operation, the hatch closure crew are on standby at the equipment hatch. Under other conditions, the hatch closure crew are "in the vicinity," which is defined as the condition whereby designated crew members can be mobilized at the equipment hatch in a time period not to exceed 10 minutes. Therefore, the hatch closure would need between 25 minutes (during a mid-loop with the RCS water level below 191 feet) and 35 minutes (for other POSs).

Although operating procedures address hatch closure during mid-loop, the model assumes the equipment hatch is closed during these operations. The assumption of the closed hatch during mid-loop is based on a review of past refueling outage histories, which indicate the equipment hatch was closed when entering the mid-loop condition. The basis also considers the reference plant's procedure for monitoring outage risk, which is one of the overall guiding procedures for managing outage operations. The procedure outlines a qualitative assessment of key safety

features and limits entry into high-risk conditions. Plant conditions would normally prevent entering mid-loop prior to refueling (POS 6) according to the outage risk assessment. Maintaining the equipment hatch open during mid-loop after refueling (POS 10) could be possible depending on the estimation of time to reach RCS boiling and the contingency plans for closing containment. Still, recent past outage experience indicates that the equipment hatch is closed during both pre-refueling and post-refueling mid-loop conditions.

The use of only four bolts, as opposed to all 30 bolts, is considered for possible impacts on the containment equipment hatch being closed. Some sites have reported that using only the minimum number of bolts was not sufficient for hatch sealing (see NRC [1993], Sections 6.9.2 and 6.9.3). For the case of the reference plant, once the hatch is in place, the number of bolts used is not viewed to be important because the pressure-sealing hatch design would tend to improve the seal as pressure increased. Also, the pressure conditions for shutdown accident scenarios are expected to be well below containment design pressures for a wide range of configurations and timeframes.

Regarding the possible impacts of containment pressurization on the ability to close the hatch, it is estimated that above a five psig pressure difference, there could be some vibration and tilting as the hatch is closed. Also, there would be significant wind through the opening and whistling that would impact personnel habitability. However, such a pressure difference would not be possible with the hatch open. As the hatch is closed and the opening becomes smaller, the containment will begin to pressurize, but the time spent in this condition is expected to be short (a few minutes). The act of closing the hatch will exacerbate the egressing flow rate, the whistling, any tilting, and the containment pressure rise. However, the differential pressure is not expected to rise quickly enough to see significant impacts during the short closing time. It was judged that these effects would not become severe enough to impact the completion of closing the hatch. This determination included consideration of NUREG/CR-3855 (Shackelford, 1985).

Considering the containment status as outlined in Table D-1, a set of conditions are identified for assessing actions to close the containment equipment hatch. While each POS and accident scenario does present a unique set of conditions, some POSs are grouped to focus on the key attributes that would influence the human actions to close the equipment hatch.

- Hatch closure during early part of outage (POSs 4 and 5A) considering both AC power available and station blackout (SBO) conditions.
- Hatch closure with RCS vented and level reduced near reactor head flange (POS 5B) considering both AC power available and SBO conditions.
- Hatch closure with flooded refueling cavity (POS 7, 8E, and 8L) considering both AC power available and SBO conditions.
- Hatch closure during post-refueling POSs.

D.2.2 Personnel Airlock

Even though the personnel airlock consists of two interlocking air lock doors, the air lock door interlock mechanism may be disabled during core alteration and movement of fuel assemblies within the containment. However, according to the reference plant's safety analysis, the air lock must always be isolable by at least one air lock door with a designated individual available to

close the air lock door, or at least one air lock door must be closed. The reference plant uses proceduralized administrative controls to ensure that the designated person available to close the personnel and/or emergency air lock doors does not have other duties that would preclude the ability to operate the door in a timely manner.

The reference plant's safety analysis states that the air lock doors will be closed within 15 minutes of a fuel handling accident. The PRA analysts for this study identified a time frame of about 20 minutes for securing the personnel airlock would be consistent with other known PRA studies. Also, the loss of power event as described in NUREG-1410 (NRC, 1990) indicates that within 21 minutes after securing the equipment hatch, all personnel were evacuated, and the personnel airlock interface was made functional. Hence, it can be concluded that securing the personnel access airlock will take about 20 minutes.

Therefore, the closure timing for the equipment hatch is bounding, and air locks are assumed to be capable of being closed before the equipment hatch is closed.

D.2.3 Main and Mini Purge Systems

As described in the reference plant's safety analysis, the main purge system (24-in lines) are open only during a cold shutdown condition and are provided with an isolation valve capable of 10 second closure. The mini-purge system (14-in. lines) may be open during normal plant operation as well as during shutdown condition and are provided with isolation valve capable of five second closure against the peak calculated containment pressure following a LOCA.

Indication of the operational status of the containment purge exhaust fans and the fans in the containment is provided in the control room. All fans and air handlers operate from control room; the containment cooling units can also be operated from the remote shutdown panel. The containment isolation valves of the mini-purge supply and exhaust are equipped with air operators inside and outside of containment. The air operated valves are designed to fail closed in the event of loss of offsite power. The containment isolation valves of main purge supply and exhaust are equipped with motor-operated isolation both inside and outside containment. These valves fail as-is in the event of loss of offsite power.

The containment vent effluent air particulate monitor measures radioactivity in the containment purge vent. The containment purge system is automatically isolated following an abnormal release of radioactivity in the containment. During fuel movement with the equipment hatch open, the containment purge system will normally be operating with an inward flow of air into containment. Because of the fan noise, during fuel handling the exhaust purge system may be shut down while the hatch is open to reduce noise and maintain comfort level. If the purge system is shut down and hatch is open, the opening will be monitored by health physics air monitoring station. The purge exhaust system needs to be shut down during hatch closure activities. The operators can close and isolate the containment purge system from the control room. The actions to isolate the containment purge system are assessed in Section D.4 .

Table D-1 Plant Operating State Initial Conditions, Time for Containment Heat Up, and Time for Equipment Hatch Closure

POS (Time after Shutdown/ Duration [hr])	Initial Conditions					Estimated Time for Containment Air Temperature to Reach 140 °F after RCS Saturation When the Containment is Closed/Open (hr) ¹		Time Required for Equip. Hatch Closure, Non-SBO ² (min)
	RCS Water Level	Time for RCS Temp. to Reach Saturation (hr)	Status of Equipment Hatch / Personnel Airlock	Purge	Cont. Cooling Units	With fan coolers	Without fan coolers	
3 (5/3)	80% level in PRZ	Not Available	Closed and bolted / Closed	Mini-purge operated as-needed	All running in high speed	Not estimated	Not estimated	Not Applicable
4 (8/4)	80% level in PRZ	Not Available	Open / Open	Mini-purge operating	Status unknown	0.5/1	0.08/0.5	35
5A (12/30)	Water solid	23.7	Open / Open	Mini-purge operating	Status unknown	1.5/3	0.25/1.5	35
5B (42/40)	192' (below the flange)	0.47	Open / Open	Mini-purge operating	Status unknown	2.5/5	0.83/2.5	35
6 (82/14)	188' ("mid-loop")	0.31	Closed with min. of 4 bolts / One door closed	Off and closed	Status unknown	3.5/7	1.5/3.5	Not Applicable
7 (126/15)	207'	5.7	Open / Open	Off and closed	Status unknown	Not estimated	Not estimated	35
8E (141/68)	218'6"	7.8	Open / Open	Mini-purge operating	Status unknown	23/>48	10/23	35
8L (329/52)	218'6"	18.2	Open / Open	Mini-purge operating	Status unknown	>48/>>48	>24/>48	35
9 (471/30)	192'	1.3	Open / Open	Off and closed	Status unknown	8.5/17 ³	2.8/8.5 ³	35

Table D-1 Plant Operating State Initial Conditions, Time for Containment Heat Up, and Time for Equipment Hatch Closure (cont.)

POS (Time after Shutdown/ Duration [hr])	Initial Conditions					Estimated Time for Containment Air Temperature to Reach 140 °F after RCS Saturation When the Containment is Closed/Open (hr) ¹		Time Required for Equip. Hatch Closure, Non-SBO ² (min)
	RCS Water Level	Time for RCS Temp. to Reach Saturation (hr)	Status of Equipment Hatch / Personnel Airlock	Purge	Cont. Cooling Units	With fan coolers	Without fan coolers	
10 (501/25)	188' ("mid-loop")	1.1	Closed with min. of 4 bolts / One door closed	Off and closed	1-train (4 CCUs) operating in high speed	12/24	5/12	Not Applicable
11 (526/24)	201'	Not Available	Open / Open	Off and closed	1-train (4 CCUs) operating in high speed	Not estimated	Not estimated	35
12 (550/54)	80% level in PRZ	161.6	Open / Open	Mini-purge operating	1-train (4 CCUs) operating in high speed	Not estimated	Not estimated	35
13 (604/20)	80% level in PRZ	Not Available	Closed and bolted / Closed	Mini-purge operated as-needed	1-train (4 CCUs) operating in high speed	3/6	0.5/3	Not Applicable
¹ If containment is initially open, once the decision is made to close the equipment hatch, the containment could be considered partially closed in a short period (about 30–50 minutes depending on the power supply availability). ² For a station blackout (SBO), it is estimated to add another 10 minutes to the time required to perform equipment hatch closure. ³ Estimated using the time in POS 5B and prorated based on the ratio of decay heat in POS 5B (13.55 MWt) over that in POS 9 (3.95 MWt).								

D.3 HRA Characterization for Emergency Containment Closure – Large Access Areas

As indicated above, containment generally has three access points: the main equipment hatch, the personnel airlock, and the emergency airlock. When in Mode 5 or 6, a containment closure plan is implemented as required by the reference plant's procedures. An adequate containment closure plan is defined as all containment penetrations that are open, (air to air path) are tracked, and can be closed.

Consistent current industry practices, containment closure is expected to be staged whenever the level drops below the vessel flange with fuel in vessel. As indicated in Table D-1, only in POS 6 and POS 10, the RCS level is below the reactor vessel head flange (i.e., below reference elevation 191 feet). In these POSs the containment hatch and personnel airlock are expected to be closed, and the containment purge system is off. However, the reference plant operating procedures provide conditions that shall be met if the equipment hatch is opened during mid-loop operations. These conditions provide a reference point for containment closure during other POSs. Based on review of the reference plant procedures, the time required to close the containment equipment hatch under emergency closure conditions is assumed to be 25 minutes plus the time required to assemble the hatch closure crew.

During Mode 5 and 6, when the RCS water level is at or above 192 feet, the hatch closure crew should be assembled first prior to hatch closure activities. According to procedures, the hatch closure crew will be in the vicinity of the equipment hatch performing other duties and can be mobilized at the equipment hatch in a time period not to exceed ten minutes. The time required for hatch closure is expected to be 35 minutes for all POSs that the hatch is open, and the RCS level is above 191 feet elevation.

As indicated in Table D-1, for most POSs both the containment equipment hatch and the personnel airlock are open. The closure of equipment hatch is expected to take a longer time than the closure of the personnel interlock. Generally, the equipment hatch if open, must be closed first; all personnel should leave the containment via the personnel airlock (or emergency airlock if needed), and then the personnel airlock is to be secured. It is assumed that the closure of one personnel access door is sufficient. Successful containment closure assumes securing the personnel access door in parallel with securing the hatch. The time for the containment closure crew to exit through the personnel airlock is expected to be incidental compared with the total time required for closure and is assumed to be included in this time.

The closure process includes the equipment hatch being lowered in place using an installed hoist and being secured with at least four bolts. According to reference plant procedures, prior to each refueling or a forced outage that requires hatch opening, the closure crew will be required to receive training on emergency equipment hatch closure methods with a step-by-step review of the procedure.

If the decay heat removal function is lost when the RCS is vented or the reactor vessel head is removed, then the decay heat will be directly released to containment in the form of steam after RCS boiling. The released steam will heat up the containment and increase both the temperature and humidity inside the containment. The operation of fan coolers and the air flow through the open containment could somewhat slow down the heat up process. The containment environment for personnel habitability conditions must be considered for any of the containment closure actions while RCS is boiling and releasing steam into the containment. The

timing assessment of the habitable conditions is an important input into the time available for establishing containment integrity. The assessment of personnel habitability is discussed in Section D.3.1 .

D.3.1 Assessment of Personnel Habitability for Containment Closure Actions

This section addresses the question of the length of time for containment to become uninhabitable after the RCS begins to boil during LPSD conditions. The information contained in the detailed discussion below is summarized and extrapolated in Table D-2. These inform the sequence timing values for determining the time available to complete containment closure activities for the Level 3 PRA project, when combined with estimates of the time between the initiating event and the RCS reaching saturation temperature. For most of the containment closure actions, the time before RCS reaching saturation temperature is used to determine the time available for the action. However, for POS 5B conditions with approximately 30 minutes available before reaching saturation temperature, successful closure of the containment equipment hatch in this timeframe is unlikely. Additional time with habitable conditions to perform the closure is expected to be available if accounting for the time from initially reaching saturation temperature to bulk boiling in the RCS and heat up of the containment environment. The exact threshold for habitability is not known and is a source of uncertainty in the HEP estimates. The assumptions used in the human reliability analysis of the actions to close containment during POS 5B are solely for the purposes of the PRA and should not be viewed as superseding and established regulatory positions and industry guidance to maintain the capability to close containment before steaming inside containment.

Table D-2 provides reference calculations for heat up of the containment temperature to 140 °F, though habitability limits are expected to be reached before this time. Factors other than timing which may affect human performance (stress, environmental conditions, other time factors contributing to delay of actions, etc.) when closing containment are not considered here, but are considered in the human reliability analysis.

A series of MELCOR accident analyses is used to inform the assumptions about the timing of containment temperature heat up. In the MELCOR model, the containment air space is modeled using approximately 25 control volumes. For this reason, water and air temperatures taken from MELCOR analysis are bulk temperatures (i.e., they represent average temperatures within a large volume). Localized boiling will occur sooner than bulk boiling, and the cited air temperatures are not peak temperatures. Spatial variability is not considered further here and are broadly expected to be comparable to other uncertainties at play in the estimates herein.

Calculations are leveraged that were performed as part of the MELCOR shutdown model development. Table D-3 summarizes several cases that were analyzed. The table gives the time from the initiating event to reach RCS saturation temperature and then reaches 140 °F and 200 °F in the containment environment. Note that, for the most part, the cases in which there are no containment openings or fan coolers available are most limiting in terms of the time from boiling to loss of habitability (which is expected to occur prior to containment reaching 140 °F). The case with the shortest time difference is the cold shutdown case (Case 5) in spite of containment being open and fan coolers running, because it has the highest decay heat. This case is explored in more detail later.

Table D-2 Time Between RCS Saturation and Containment Bulk Temperature of 140 °F

Notional Plant Conditions (with Loss of Decay Heat Removal)	Containment Open		Containment Closed	
	Fan Coolers	No Fan Coolers	Fan Coolers	No Fan Coolers
POS 2-3 (Hot standby and hot shutdown)	Notionally, less time than POS 4			
POS 4 – Cold shutdown – 14 hours after subcriticality	1 hour ²	30 minutes (estimated ¹)	30 minutes ²	5 minutes ²
POS 5A/5B (Water solid and draining)	Notionally, ramps from POS 4 values to POS 6 values			
POS 6 - Hot mid-loop – 84 hours after subcriticality	7 hours (estimated ¹)	3.5 hours (estimated ¹)	3.5 hours	1.5 hours
POS 7 (Filling refueling cavity)	Notionally, ramps from POS 6 values to POS 8 values			
POS 8E - Refueling with RPV head off and refueling cavity flooded – 96 hours after subcriticality	>> 48 hours	23 hours	23 hours (estimated ¹)	10 hours (estimated ¹)
Defueled	Infinite			
POS 8L – Refueling with new core	>> 48 hours (estimated ¹)	> 48 hours (estimated ¹)	> 48 hours (estimated ¹)	> 24 hours (estimated ¹)
POS 9 – Draining to mid-loop	Notionally, ramps from POS 8L values to POS 10 values			
POS 10 – Cold mid-loop	24 hours (estimated ¹)	12 hours (estimated ¹)	12 hours (estimated ¹)	5 hours (estimated ¹)
POS 11/12 – RCS refill and heatup	Notionally ramps from POS 10 values to POS 13 values			
POS 13 – Cold shutdown	6 hours (estimated ¹)	3 hours (estimated ¹)	3 hours (estimated ¹)	30 minutes (estimated ¹)
POS 14 – Reactor startup	Notionally similar to POS 13 values			
¹ These times are simply projected based on the times that have been calculated and associated decay heat levels and are not intended to be very precise.				
² For these cases, saturation is reached at the time that the RCS depressurizes, which is in turn caused by assumed failure of the RHR relief valve upon first passing water.				

A useful reference for assessing containment habitability is NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States* (NRC, 1993). Subsection 6.9.4.1 of NUREG-1449 estimates a maximum of 1 hour for containment air temperature to go from 70 degrees F (°F) (21 degrees C [°C]) and atmospheric pressure to 150 °F (66 °C) and 5.9 psig (40.7 kPa) (see Figure 6.1 of NUREG-1449). The NUREG states that the selection of 150 °F (66 °C) may be questionable, because breathing air at 160 °F (71 °C) is considered fatal (lungs will burn). Furthermore, Subsection 6.9.4.2 of NUREG-1449, “Radiological Considerations,” states, “[b]oiling of coolant within an opened reactor system following a postulated loss of shutdown cooling would release dissolved fission products within the containment atmosphere.” The pressurized-water reactor equivalent dose rate estimates (see Figures 6.2 and 6.3 in NUREG-1449) suggest that conditions could exceed the derived air concentrations for occupational dose limits as defined in 10 CFR Part 20. Though, the estimates do not consider the potential reduction in the soluble iodine and noble gases in the RCS through cleanup that occurs prior to opening the RCS. Under these conditions protective clothing, respirators, or possibly self-contained breathing apparatus (SCBA) may be required. But the humidity may challenge the filtration efficiency of SCBA.

Table D-3 Timings for Onset of Boiling and Un-Inhabitability Calculations

Case	Initiating Event / Time Since Subcriticality (hr)	Containment Opening(s)	Fan Coolers Available?	Time of Reaching Saturation (min)	Time of Reaching 140 °F (60 °C) (min)	Diff-erence (min)	Time of Reaching 200 °F (93 °C) (min)	Diff-erence (min)
1	Station blackout in POS 6 / 84	None	No	21	122	101	413	392
2	Station blackout in POS 6 / 84	None	No	20	127	107	411	391
3	Station blackout in POS 6 / 84	None	No	21	122	101	925	904
4	Station blackout in POS 8E / 96	Fuel transfer tube, equip hatch, personnel airlock	No	161 hrs	184 hrs	23 hrs	186 hrs	25 hrs
5	Loss of decay heat removal + induced loss-of-inventory in POS 5A / 14	Fuel transfer tube, equip hatch, personnel airlock	Yes	483	538	55	N/A	N/A
6	Loss of decay heat removal in POS 8E / 96	Equip hatch, personnel airlock	Yes	70 hrs	86 hrs	16 hrs	101 hrs	31 hrs
7	Loss of decay heat removal in POS 8E / 96	Equip hatch, personnel airlock	Yes	70 hrs	86 hrs	16 hrs	N/A	N/A
8	Loss-of-inventory in POS 6/ 84	None	Yes	22.5	236	213	325	302
9	Loss-of-inventory in POS 8E / 96	Fuel transfer tube, equip hatch, personnel airlock	Yes	327	761	435	N/A	N/A

If 140 °F (60 °C) bulk air temperature is used as the habitability metric, the time of loss of habitability in the station blackout Case 1 is around 580 percent of the time to the onset of boiling. In other words, there is a significant difference between the time of reaching bulk saturation in the pool and the time the containment bulk temperature reaches 140 °F (60 °C). Figure D-1 shows the RCS water level and Figure D-2 shows the RCS and containment temperatures as simulated in the Case 1 scenario. Comparatively, simulations done for the spent fuel pool (SFP) PRA show the fuel handling building (FHB) generally has a much shorter period of time between boiling and un-inhabitability (time of loss of habitability is between 101 and 103 percent of the time to the onset of boiling).

Table D-4 provides the necessary fundamental data to explain this difference. In the SFP situation, the volume of water boiling is about 28 times more than that in reactor Case 1. Meanwhile, the free volume is 3.4 times smaller in the FHB versus the containment. The total

decay heat in the SFP during a full core offload is comparable to the reactor decay heat load around the time of refueling, because the discharged fuel dominates the SFP heat load. So conceptually, the SFP situation is a much larger pot of water in a smaller room. This means that it takes (comparatively) much longer for the SFP to reach bulk boiling, but once it does, the FHB heats up at a much greater rate than the containment.

Table D-4 Conditions in the RCS and Containment for Cases 1 and 5

	SFP	Reactor (Case 1)	Reactor (Case 5)
Initial water temperature (F)	85	112	225
Initial air temperature (F)	80	103	106
Decay heat (MW)	2.8 to 11.6 (both pools combined)	12.1 (initially)	18.3 (initially)
Initial water (gallons)	~1,000,000 (all pools/canals connected with fuel transfer tube closed)	~35,000	~85,000
Free Volume (m ³)	22,723	77,871	77,871

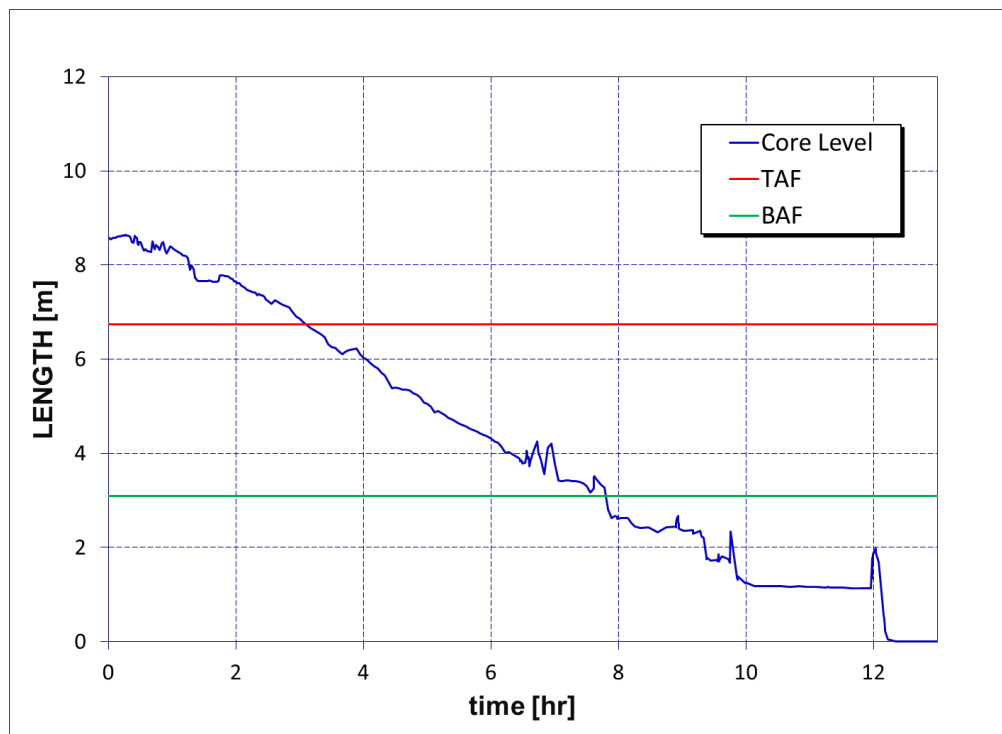


Figure D-1 The Water Level in the RCS as a Function of Time in Case 1

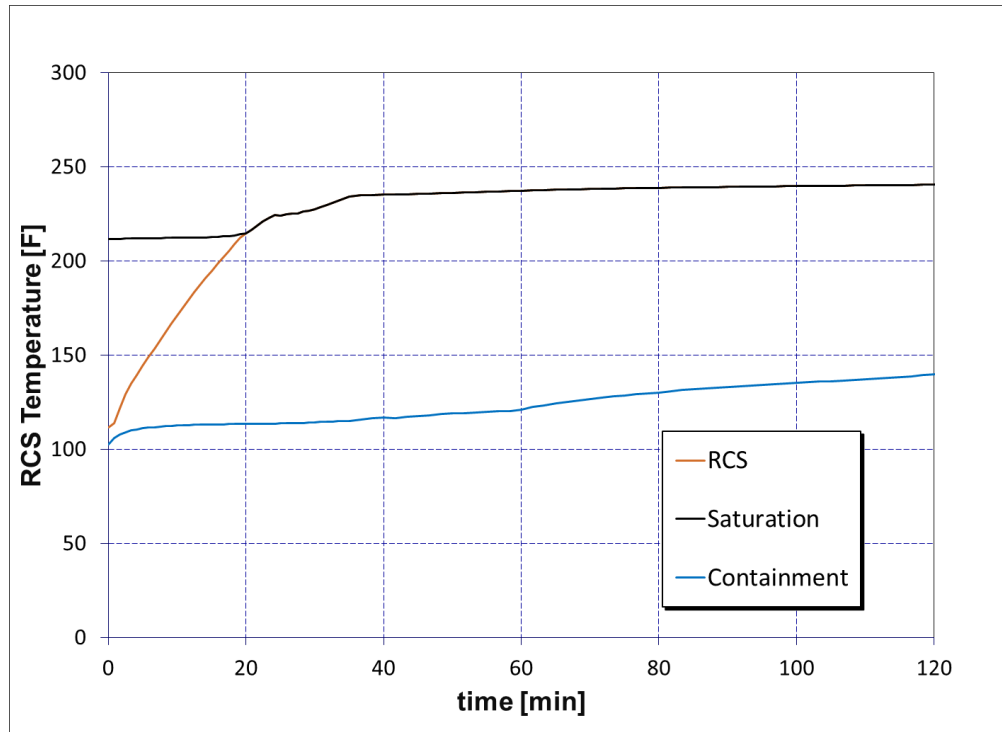


Figure D-2 Temperature in the RCS and Containment in Case 1

Case 5 is considered to be the more limiting case due to the shorter time between reaching saturation and containment heat up to 140 °F. Being the more limiting case, three sensitivities of Case 5 were simulated in MELCOR with the containment isolated/closed. The first (5a) assumes that fan coolers are available the entire duration of the accident, the second (5b) assumes that fan coolers become unavailable when the RCS starts to boil (t=485 minutes), and the third (5c) assumes that fan coolers are unavailable when the initiating event occurs (i.e. at t=0s). Table D-5 and Table D-6 below are respectively analogous to Table D-3 and Table D-4 above. Figure D-3, Figure D-4, and Figure D-5 provide the RCS and containment temperatures for each simulation. In looking at these figures, note that the sharp drop in RCS saturation pressure at eight hours is caused by the failing open of the RHR relief valve and the subsequent failure of the pressurizer relief tank rupture disk, which follows the scenario definition (failure of the relief valve upon first passing water).

Table D-5 Timings for Onset of Boiling and Un-Inhabitability for Sensitivity Cases on Case 5

Case	Initiating Event / Time since Subcriticality (hr)	Containment Opening(s)	Fan Coolers Available?	Time of Reaching Saturation (min)	Time of Reaching 140 °F (min)	Diff-erence (min)	Time of Reaching 200 °F (min)	Diff-erence (min)
5a	Loss of decay heat removal + induced loss-of-inventory / 14	None	Yes	485	512	27	N/A	N/A
5b			Unavailable at saturation	485	497	12	655	170
5c			No	485	490	5	633	148

Table D-6 Initial Conditions in the RCS and Containment for Sensitivity Cases on Case 5

	Reactor (Case 5a)	Reactor (Case 5b)	Reactor (Case 5c)
Initial water temperature (F)	225	225	225
Initial air temperature (F)	108	108	115
Decay heat (MW)	18.3 (initially)	18.3 (initially)	18.3 (initially)
Initial water (gallons)	85,000	85,000	85,000
Free Volume (m ³)	77,871	77,871	77,871

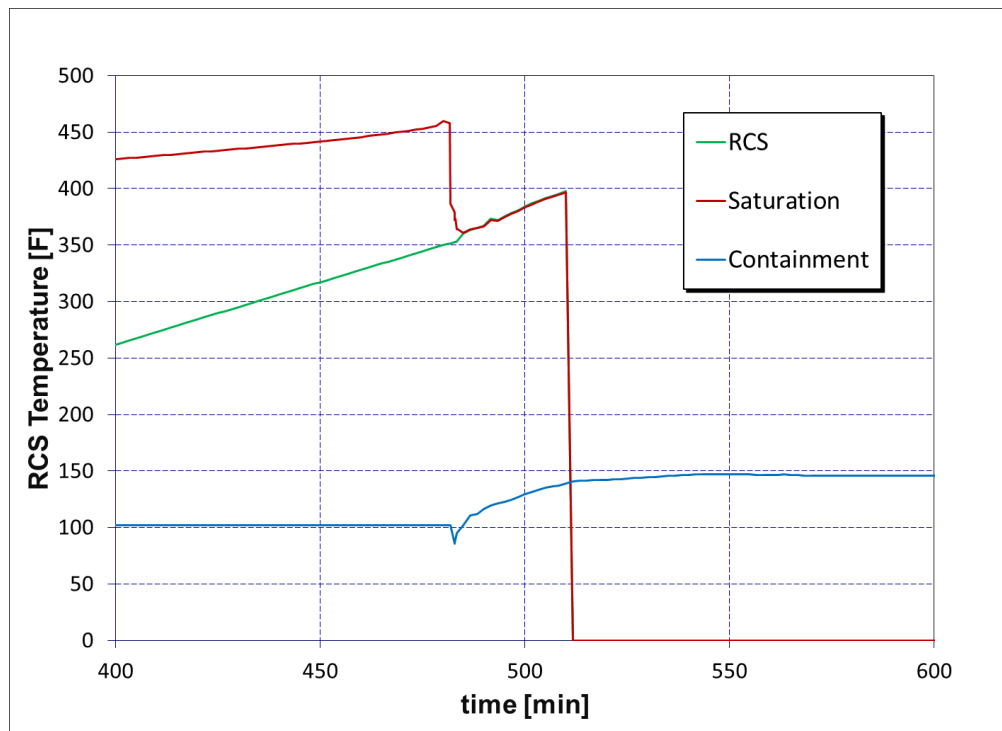


Figure D-3 Temperature in the RCS and Containment in Case 5a Sensitivity with Fan Coolers Running

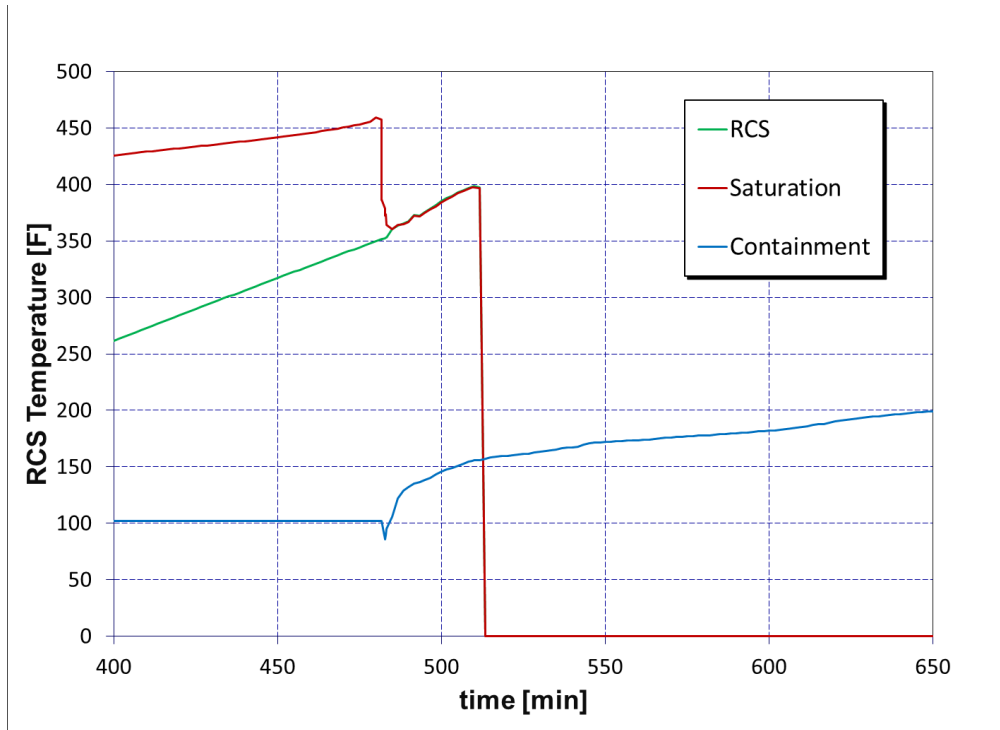


Figure D-4 Temperature in the RCS and Containment in Case 5b Sensitivity with Fan Coolers Running

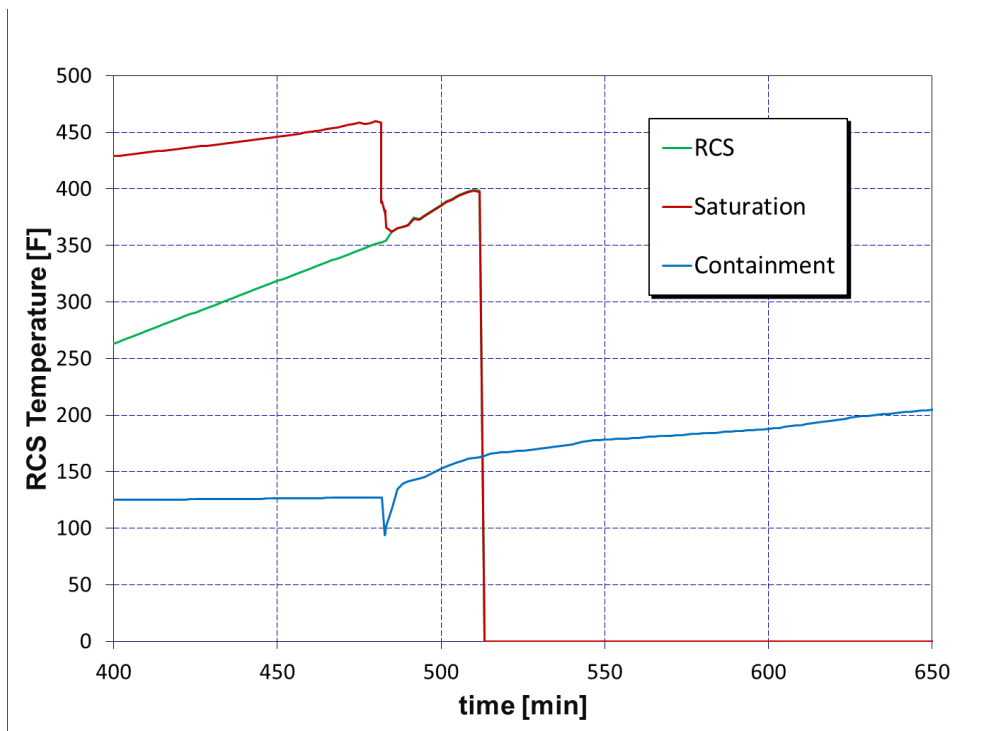


Figure D-5 Temperature in the RCS and Containment in Case 5c Sensitivity with Fan Coolers Assumed Unavailable

NUREG -1449 (NRC, 1993) also addresses the question of habitability timing in Section 6.9.4. That study estimated that if RHR was lost 48 hours after shutdown, it would take 1 hour for containment to heat from 70 °F to 150 °F (or, looking at Figure 6.1 of the NUREG, about 40 minutes from the time of boiling to reach 140 °F) with no fan coolers available and no containment openings. While this study used a smaller containment volume (56,633 m³), the amount of water assumed to evaporate when containment reaches 140 °F was 22,680 kg. By comparison, the amount of water that evaporates is 21,611 kg in Case 1 and 44,033 kg in Case 5c. Table D-7 shows that the NUREG-1449 containment heatup timing falls between the two MELCOR calculations, as would be expected based on the evaluation times.

Table D-7 Comparison between the Initial Conditions and Results of Cases 1 and 5C and Those Found in NUREG-1449

	Case 5C	NUREG-1449	Case 1
Time since shutdown (hrs)	14	48	84
Initial air temperature (°F)	108	70	103
Time from saturation to 140 °F (min)	5	40	101
Water deposited to containment when reach 140 °F (kg)	44,033	22,680	21,611
Containment Free Volume (m ³)	77,871	56,633	77,871
Ratio of water addition to containment free volume (kg/m ³)	0.57	0.4	0.28

D.3.2 Assessment of Equipment Hatch Closure for POS 5B Conditions

The following provides a discussion of the time available and the time required for the hatch closure in POS 5B and POS 9. These two operating states have the minimum available times for the hatch closure. The urgency of the hatch closure is dominated by the available time for the RCS to boil. As indicated in Table D-1, for all POSs where containment closure is needed POS 5B has the shortest time for the RCS to reach the saturation temperature of 0.47 hours, followed by POS 9 of 1.33 hours. Note, in POS 6 and POS 10, the containment hatch, personnel hatch, and purge systems are closed. In these POSs, the time to boil does not limit the time available for ensuring containment integrity. For other POS conditions, the time available will be longer than that available in POS 5B. Consequently, there is expected to be a higher likelihood of success for hatch closure under other POS conditions.

In POS 5B, RCS water level is at or above 192 feet. RCS is at atmospheric pressure, reactor coolant temperature is about 110 °F (~ 43 °C), pressurizer safety valves are removed, and vessel head is intact and venting. The decay heat is about 13 MW thermal. Based on these conditions, the time to start boiling is about 0.47 hours (~30 minutes). This time frame is less than what is required to close the hatch. However, the progression from localized to bulk boiling and gradual heat up of the containment environment may allow for additional time to close the hatch.

The plant specific analyses performed for developing the time required for the bulk temperature in the containment to reach 140 °F (60 °C) are summarized in Section D.3.1 above. These analyses provide estimates for different accident scenarios and conditions. While these estimates have uncertain aspects, they do give a sense of the range of times under different scenarios. Also, although several cases were analyzed, not all possible conditions were

assessed. For many cases, the estimated time for containment heat up requires interpolation between estimates that bound the conditions being analyzed.

The estimates in Table D-2 are referenced for assessing the conditions POS 5B. The time for POS 4 with containment open and no fan coolers (30 minutes) is used as a bounding estimate for the minimum time for heat up. The time for POS 6 with containment open and no fan coolers (3.5 hours) is used as a bounding maximum estimate. The decay heat levels from POS 4 to POS 5B will have reduced from 19.9 MW to 13.6 MW. If the containment heat up time is interpolated based only on the decay heat level, then the time to heat up to 140 °F in POS 5B is expected to be 2.8 hours or about 170 minutes. However, using 140 °F containment temperature threshold for human habitability is likely too optimistic. Inhabitable conditions are expected to be reached before this point. It is noted that the available time depends on variations in the scenario conditions. Availability of fan coolers can significantly increase the available time. Due to the uncertainty in these estimates, the analysis uses 60 minutes beyond the time to saturation temperature for assessing the human reliability. Including the time to reach saturation temperature (approximately 30 minutes) the total time window for closing containment from the time of the initiating event to reaching 140 °F in the containment environment is 90 minutes. The assumptions used in the human reliability analysis of the actions to close containment during POS 5B are solely for the purposes of the PRA and should not be viewed as superseding and established regulatory positions and industry guidance to maintain the capability to close containment before steaming inside containment.

The time window required for closing containment includes the time to perform the closure (35 minutes, as described in Section 2.1) plus the delay time to initiate the action. In POS 5B, if a loss of RHR were to occur, it is estimated that the first 5 minutes would be lost to the operator diagnosing the condition and entering the appropriate abnormal operating procedure. The operators will then evaluate the steps in the procedure in their attempts to restore the heat removal function. Assessing the procedure steps is estimated to take between 5 and 10 minutes before reaching procedural direction to establish containment closure. Therefore, for the purposes of developing the HRA worksheets, a total delay time of 15 minutes is used. The time window from the initiating event to completing the containment closure is 50 minutes.

The timing evaluation is an important input to the human reliability analysis, but other performing shaping factors can also influence the human reliability. The full assessment of the performance shaping factors (PSFs) has been performed for each HFE according to the SPAR-H method. The results of the SPAR-H evaluations are provided in Table D-9. The HFE 1-CNT-XHE-XM-H1 is the event that assesses containment closure in POS 5B. The PSF evaluation provides the basis for the HEP values shown in Table C-2.

The evaluation of PSFs can be influenced by different accident conditions. For most initiating events, the timing evaluation and procedure steps will be the same or similar to the loss of RHR scenario. However, one significant condition that will influence the containment closure is the loss of offsite power leading to a station blackout. When AC power is not available, the normal containment closure process with an electric motor-driven hoist cannot be used. Instead, an air motor with compressed nitrogen bottles is used to control the hatch closure.

The timing evaluation is similar to the situation where AC power is available, but 10 minutes are added to the time required to close containment due to the SBO conditions and the need to use special equipment. The total time required to perform the containment closure is 60 minutes. A delay time of 15 minutes is assumed, and 45 minutes are required to perform the closure. In

addition to the timing evaluation, all other PSFs are evaluated considering the other challenges (e.g., lighting, communications) that could be present during an SBO. See Table D-9 for the SPAR-H PSF evaluation for HFE 1-CNT-XHE-XM-SBO1 for assessment of containment closure in POS 5B without AC power.

One other timing consideration is made for assessing HFEs for POS 5B conditions. If forced or gravity-driven injection from RWST to RCS is established, then this would provide a temporary means of alternate cooling before the usable RWST inventory is depleted. This successful action extends the available time by 6 to 8 hours. An alternate HFE is applied for these conditions, which accounts for the additional time available. See Table D-9 for HFE 1-CNT-XHE-XM-H1F for assessment of containment closure in POS 5B with forced or gravity feed available. For SBO conditions with gravity feed available, HFE 1-CNT-XHE-XM-SBO1F is substituted for 1-CNT-XHE-XM-SBO1.

In reviewing the POS conditions in Table D-1, the next most time-constrained conditions are seen in POS 9. In POS 9, RCS water level is at or above 192 feet. RCS is at atmospheric pressure, pressurizer safety valves are removed, and vessel head is being reinstalled and vented. The decay heat is about 4 MW. Based on these conditions, the time to start boiling is about 1.3 hours (~80 minutes). This is comparable to the estimated time available for containment closure in POS 5B, even without accounting for containment heat up. In this POS, the time for the containment temperature to reach 140 °F would be longer than POS 5B due to the lower decay heat level. Simply based on the ratio of decay heat levels, POS 9 containment heat up would be 3.4 times longer than that for the POS 5B.

However, given the uncertainty in the times for containment heat up, a conservative approach is taken in applying the HFE for POS 9 conditions. This is based on the following considerations. First, the time available for containment closure is generally similar to the time frame for habitable conditions in containment. Yet, it is expected that actions will be taken to limit the crew's exposure to hazardous conditions to as low as reasonably achievable. Also, the procedure cues suggest that containment closure actions would be initiated before RCS boiling, if achievable. The reference plant's loss of RHR procedure includes steps to monitor containment ambient temperature. All personnel working inside containment are to be evacuated if temperature reaches 120 °F. The containment closure crew are considered essential for establishing containment integrity, so they would remain, if necessary, but the closure should be completed before reaching 120 °F, if possible.

Even with the above considerations, the time available in POS 9 will be longer than POS 5B. However, the somewhat coarse timing assessment used in SPAR-H does not significantly distinguish between the two situations. In order to adjust the PSF for timing, the time available must be greater than 5 times the time required. With the uncertainty in the time estimates and the habitability considerations above that would tend to push the performance of the closure earlier in the timeline, the approach for POS 9 is to use the same HFE evaluation that was developed for POS 5B.

D.3.3 Assessment of Equipment Hatch Closure for POS 4/5A Conditions

This section describes the time available and time required for containment hatch closure during POS 4 and POS 5A. These states occur early in the outage when the highest decay heat levels are seen with an open containment status. The time for heating up containment to uninhabitable conditions is relatively short. Table D-2 shows 30 minutes to 1 hour after RCS reaches

saturation temperature with containment open. However, due to the RCS status the time to reach saturation temperature is much longer than the lowered inventory states in POSs 5B and 9. This allows for longer time available for closing the equipment hatch.

The POS 4 conditions have slightly higher decay heat, but the POS 5A conditions are expected to be of higher risk significance due to the longer time spent in the POS and the water solid conditions. Therefore, the HRA is assessed for POS 5A conditions. During POS 5A, RCS temperature is maintained between 190°F and 170°F, and RCS pressure is maintained between 350 psig and 225 psig. Steam generators are available as a heat sink with inventory levels initially at 95 percent. As long as steam generators can be used for heat removal, the time to reach RCS saturation and ultimately the time available for containment closure is significantly increased.

The habitability assessment in Table D-3 shows that for POS 5A conditions (Case 5) the time to RCS saturation temperature is approximately eight hours. However, the analysis assumes an RHR relief valve fails to close and remains open after lifting. If this failure does not occur, then additional time would be available. A similar case was analyzed for POS 5A conditions for the shutdown Level 1 PRA success criteria analysis. See Table 5.3-1 in (NRC, 2025). In this case, the RHR relief valves are cycled open as pressure increases but are not assumed to fail open. Under these conditions, steam generators continue to be used for RCS heat removal until they are boiled dry at around 20 hours after the initiating event (assuming no successful actions to feed the steam generators). So, the range of estimates for RCS to reach saturation temperature is around 8 to over 20 hours. An additional 30 minutes to 1 hour are available until uninhabitable conditions in containment are reached, but this additional time is not credited due to the uncertainty about how long the conditions would remain habitable after RCS saturation temperature has been reached. Considering this range, a conservative estimate of 8 hours after the initiating event is assumed available for closing the equipment hatch.

The time required to perform the equipment hatch closure is dependent on assessment of the plant conditions and procedural prompts. During POS 5A the RCS is intact (i.e., not vented) and steam generators can be used for heat removal. The procedure for loss of RHR cooling during these conditions does not emphasize expedient containment closure to the same extent as during other outage conditions. If RHR cooling is lost, operators will establish steam generator cooling to control RCS temperature and pressure and pursue efforts to restore RHR cooling. The availability of steam generator cooling makes it much more likely that RCS cooling can be maintained and any significant radiological release prevented.

After assessing RHR status and venting the system, if necessary, operators are directed to establish one train of RHR. If this cannot be established, then the procedure directs initiation of emergency classification determination and evacuation of non-essential personnel from containment. Actions to close containment are also expected to occur around this time. The direction will likely depend on the specific conditions of the scenario and whether an emergency action level ALERT is declared. The ALERT declaration will also initiate the Technical Support Center (TSC). The TSC is expected to direct actions to close containment.

The conditions that would prompt an ALERT declaration are outlined in the reference plant's emergency plan procedures. The conditions that are applicable during cold shutdown and refueling operations (Technical Specification Mode 5 and Mode 6) are:

- Loss of RCS inventory – indicated by RPV level, or RCS level cannot be monitored for greater than 15 minutes and possible loss of RCS inventory indications.
- Loss of all offsite power and loss of all onsite AC power to essential busses – indicated by loss of power to or from the relevant electrical systems (as listed) and failure to restore power within 15 minutes.
- Inability to maintain plant in cold shutdown with irradiated fuel in the RPV – indicated by an unplanned event resulting in RCS temperature exceeding 200°F or RCS pressure increasing greater than 10 psig with timing considerations and containment closure and RCS integrity conditions as listed.

The timing of the ALERT declaration depends on the details of the event progression. For SBO conditions, an ALERT could be declared as early as 15 minutes into the event. A loss of RHR cooling would result in rapid increases in RCS temperature and pressure that would meet an ALERT condition. Under other conditions, the progression of the event could drive the ALERT declaration later, possibly hours, into the event timeline.

The time required to close the containment hatch is assumed to be 35 minutes. The delay time from the start of the event is assumed to be 30 minutes to diagnose the event and initial evaluation of the emergency plan threshold values. For SBO conditions, a delay time of 30 minutes is again applied, and 45 minutes are required to perform the closure.

Refer to Table D-9 for the results of the SPAR-H PSF evaluations for HFEs 1-CNT-XHE-XM-H2 for containment closure in POS 5A and 1-CNT-XHE-XM-SBO2 for containment closure in POS 5A without AC power.

D.3.4 Assessment of Equipment Hatch Closure for POS 8E Conditions

This section describes the time available and time required for containment hatch closure during POS 8E. This state occurs when the refueling cavity is flooded and preparations for removing fuel from the reactor vessel are underway. A similar condition is reached after refueling in POS 8L, but the decay heat levels are significantly reduced after refueling the core. Prior to POS 8E is POS 7, which represents the period of flooding the refueling cavity. POS 7 is represented by a midpoint in the POS with the cavity partially flooded. Like POS 8E, POS 7 will have long times available before reaching RCS saturation temperature due to the large inventory of water available. Although POS 8E has even more time to reach saturation temperature, POS 8E conditions are considered to be of greater overall risk due to the maintenance unavailabilities that occur after flooding the cavity and the longer duration of POS 8E compared to POS 7. Therefore, the equipment hatch closure HFE is assessed for POS 8E conditions, and the same HFE is applied to POSs 7, 8E, and 8L.

During POS 8E the time to boil after loss of RHR cooling is estimated as 7.8 hours, as discussed in the shutdown Level 1 PRA (NRC, 2025). Supporting calculations of the refueling cavity heat up show that the refueling cavity water does not reach saturation temperature. The estimated time to boiling represents the time when the refueling cavity water rapidly heats up to approach its maximum temperature. The refueling cavity water level will begin to fall due to evaporation without boiling.

The time to heat up containment before uninhabitable conditions is significantly longer compared to other POS conditions. Table D-2 shows an estimate of 23 hours with containment open and no fan coolers in operation. Additional time would be available if the fan coolers are used, which is consistent with directions in the applicable procedure. Also, note that the several of the calculations for estimating containment heat up times (shown in Table D-3) show very long times to reach saturation temperature.

The total time available to close containment is estimated to be 31 hours (23 hours plus approximately 8 hours), though the time could potentially be much longer.

The time window required for closing containment includes the time to perform the closure (35 minutes, as described in Section D.2.1) plus the delay time to initiate the action. In POS 8E, the reference plant's loss of RHR procedure directs operators to turn on the containment cooling units, which will allow for additional time for closing containment. Establishing containment closure is also directed by the procedure. A delay time of 30 minutes is assumed for assessing the loss of RHR cooling and performing the procedure steps. The time window from the initiating event to completing the containment closure is 1 hour and 5 minutes. For SBO conditions, a delay time of 30 minutes is again applied, and 45 minutes are required to perform the closure.

Refer to Table D-9 for the SPAR-H PSF evaluations for HFEs 1-CNT-XHE-XM-H3 for containment closure in POS 8E and 1-CNT-XHE-XM-SBO3 for containment closure in POS 8E without AC power.

D.3.5 Summary of Equipment Hatch Closure Human Failure Events

The preceding sections outline the assessment of human failure events to model failure to close the containment equipment hatch under emergency conditions. The HFEs represent the likelihood of failure under different accident scenario conditions. There are three sets of POS conditions that are assessed in Sections D.3.2 , D.3.3 , and D.3.4 .

- POS 5B conditions, which are also applied to POS 9
- POS 4/5A conditions, which are also applied to POS 11 and POS 12
- POS 8E conditions, which are also applied to POS 7 and POS 8L

In addition to the POS conditions, there are variations for the HFEs to account for accident sequence specific conditions. This includes SBO conditions and the availability of pumped or gravity-driven injection to the RCS, which can extend the time available for performing the containment closure. Additional adjustments to the HFE modeling are made for loss of inventory events. Loss of inventory event sequences generally have faster progression to core damage and consequently less time available for containment closure actions. For some scenarios, the HFEs for loss of inventory events are substituted to account for the reduced time available. However, for other scenarios the differences in timing do not warrant adjustments to the HFEs.

A summary of the how the different POS and accident sequence conditions are modeled is provided in Table D-8. The table shows which HFEs are used for the different conditions. The SPAR-H PSF evaluations of each HFE can be found in Table D-9.

Table D-8 Summary of Equipment Hatch Closure Human Failure Event Modeling

Plant Operating States	Initiator Type / Condition	Applied HFE
POS 5B, POS 9	SBO, no gravity feed	1-CNT-XHE-XM-SBO1
POS 4, POS 5A POS 11, POS 12	SBO, no gravity feed	1-CNT-XHE-XM-SBO2
POS 7, POS 8E, POS8L	SBO, no gravity feed	1-CNT-XHE-XM-SBO3
POS 5B, POS 9	Loss of RHR, no forced or gravity feed	1-CNT-XHE-XM-H1
POS 4, POS 5A POS 11, POS 12	Loss of RHR, no forced or gravity feed	1-CNT-XHE-XM-H2
POS 7, POS 8E, POS8L	Loss of RHR, no forced or gravity feed	1-CNT-XHE-XM-H3
POS 5B, POS 9	Loss of inventory, no forced or gravity feed	1-CNT-XHE-XM-H1
POS 4, POS 5A POS 11, POS 12	Loss of inventory with stuck open relief valve, no forced or gravity feed	1-CNT-XHE-XM-H1
POS 4, POS 5A POS 11, POS 12	Loss of inventory, no stuck open relief valve, no forced or gravity feed	1-CNT-XHE-XM-H2
POS 7, POS 8E, POS8L	Loss of inventory, no forced or gravity feed	1-CNT-XHE-XM-H2
POS 5B, POS 9	SBO, successful gravity feed	1-CNT-XHE-XM-SBO1F
POS 4, POS 5A POS 11, POS 12	SBO, successful gravity feed	1-CNT-XHE-XM-SBO2
POS 7, POS 8E, POS8L	SBO, successful gravity feed	1-CNT-XHE-XM-SBO3
POS 5B, POS 9	Loss of RHR, successful forced or gravity feed	1-CNT-XHE-XM-H1F
POS 4, POS 5A POS 11, POS 12	Loss of RHR, successful forced or gravity feed	1-CNT-XHE-XM-H2
POS 7, POS 8E, POS8L	Loss of RHR, successful forced or gravity feed	1-CNT-XHE-XM-H3
POS 5B, POS 9	Loss of inventory, successful forced or gravity feed	1-CNT-XHE-XM-H1
POS 4, POS 5A POS 11, POS 12	Loss of inventory, successful forced or gravity feed	1-CNT-XHE-XM-H2
POS 7, POS 8E, POS8L	Loss of inventory, successful forced or gravity feed	1-CNT-XHE-XM-H3

Table D-9 Evaluation of SPAR-H Performance Shaping Factors for Containment Closure Human Failure Events

HFE Basic Event Name	HFE Description and PSF Evaluation							
1-CNT-XHE-XM-H1 (Execution PSFs)	Operators fail emergency closure of equipment hatch with lowered RCS inventory (AC power available) POS applicability: assessed for 5B; also applied to POS 9 HFE type: Execution							
	<i>Time PSF:</i> Nominal	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-SBO1 (Execution PSFs)	Operators fail emergency closure of equipment hatch with lowered RCS inventory (no AC power) POS applicability: assessed for 5B; also applied to POS 9 HFE type: Execution							
	<i>Time PSF:</i> Nominal	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Poor	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-H1F (Execution PSFs)	Operators fail emergency closure of equipment hatch with successful forced feed POS applicability: assessed for 5B; also applied to POS 9 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-SBO1F (Execution PSFs)	Operators fail emergency closure of equipment hatch with successful gravity feed (no AC power) POS applicability: assessed for 5B; also applied to POS 9 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Poor	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal

Table D-9 Evaluation of SPAR-H Performance Shaping Factors for Containment Closure Human Failure Events (cont.)

HFE Basic Event Name	HFE Description and PSF Evaluation							
1-CNT-XHE-XM-H2 (Execution PSFs)	Operators fail emergency closure of equipment hatch with RCS intact (AC power available) POS applicability: assessed for 5A; also applied to POS 4 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Available, but poor	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-SBO2 (Execution PSFs)	Operators fail emergency closure of equipment hatch with RCS intact (no AC power) POS applicability: assessed for 5A; also applied to POS 4 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Available, but poor	<i>Ergonomics PSF:</i> Poor	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-H3 (Execution PSFs)	Operators fail emergency closure of equipment hatch with refueling cavity flooded (AC power available) POS applicability: assessed for 8E; also applied to POS 7, POS 8L HFE type: Execution							
	<i>Time PSF:</i> ≥ 50x time required	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CNT-XHE-XM-SBO3 (Execution PSFs)	Operators fail emergency closure of equipment hatch with refueling cavity flooded (no AC power) POS applicability: assessed for 8E; also applied to POS 7, POS 8L HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Poor	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal

D.4 HRA Characterization for Closing Containment Penetrations

During the refueling outage, it may be necessary to open containment penetrations, which can result in potential containment bypass release pathways if not closed in response to accident conditions.

The containment purge system is one of the primary concerns for containment bypass due to large openings to the containment environment (24-inch and 14-inch diameter lines) and the operation of the system during outages. However, there could be many other smaller penetrations that are potential release pathways.

The assessment of actions to close containment penetrations is divided into three sections. Section D.4.1 describes isolating the purge system when not in operation. Section D.4.2 describes isolating the purge system when the system is operating. Section D.4.3 describes the actions to isolate all other small penetrations that are potentially open during an outage.

D.4.1 Isolating Containment Purge System When Not Operating

The assumptions regarding the containment purge system operation are summarized in Table D-1. The mini-purge is assumed to be in operation during POSs 3 through 5B, during POS 8E and 8L in preparing for fuel movement, and POS 12 through 13. The purge system is off and isolated during POSs 6, 7, 9, 10, and 11. The human action described in this section apply to the POSs when the system is off and presumed to be isolated.

The reference plant's system procedure clearly directs closure of the isolation valves for system shutdown from normal operation. The procedure includes a written checklist with independent verification. A separate check of component status is made later. Also, the containment penetration status is monitored each shift as part of the reference plant's outage risk monitoring procedure to ensure the containment integrity safety function.

An HFE for a pre-initiator failure to restore the supply and exhaust isolation valves during purge system shutdown is used to model isolation failure in the containment isolation fault trees. The approach used for estimating failure probabilities for pre-initiator HFEs is described in NUREG/CR-4772, *Accident Sequence Evaluation Program Human Reliability Analysis Procedure* (Swain, 1987). According to method, the basic HEP of every pre-initiator is 1×10^{-2} and each factor of recovery credit may be 0.1. According to the ASME/ANS PRA standard supporting requirement HR-D4 (ASME/ANS, 2009), the following four factors, with appropriate procedural direction, may be considered for crediting recovery of the error:

- Post-maintenance tests,
- Independent verification component status after maintenance or testing,
- Verification of component status made at a later time,
- Routine (e.g., daily) checks of component status.

At least two recovery factors can be applied considering the procedures and independent verification tasks described above. This results in a pre-initiator human failure probability of 1×10^{-4} for this event. This pre-initiator HFE is represented in the model with basic event name 1-CIS-XHE-XR-PURGE.

D.4.2 Isolating Containment Purge System When Operating

The human action described in this section apply to the POSs when the mini-purge system is operating. The mini-purge is assumed to be in operation during POSs 3 through 5B, during POS 8E and 8L in preparing for fuel movement, and POS 12 through 13. The isolation valves on the 24-inch main purge lines are assumed to be isolated, as only the mini-purge is routinely used during outages.

The automatic containment isolation system is assumed to be in operation during POS 3 and 13, so no HFE is modeled for those POSs. The purge system can also be automatically isolated by a containment ventilation isolation (CVI) signal. The CVI actuation is required to be active for fuel movement during refueling. The purge isolation valves will close if high radiation is detected. Radiation alarms in the control room would also prompt manual closing of the valves. The automatic CVI actuation is assumed to be active during POS 8E and 8L. During these POSs, failure to isolate the purge system requires failing both the manual action and the automatic CVI actuation.

A manual action for isolating the purge system is modeled for the remaining POSs with the purge system in operation (POS 4, 5A, 5B, and 12). The action involves closing the four air-operated isolation valves on the 14-inch diameter supply and exhaust lines. There are two series isolation valves on each line, one inside containment and one outside. The valves are closed remotely from the control room. The time for the valves to close is five seconds. The isolation valves will fail to a closed position on a loss of power.

The time available to perform the action depends on the time to reach core damage conditions because this is when there would first be potential for significant radiation release through the bypass pathway. The specific conditions for the HFE timing are assessed for POS 5B, which are expected to be the most limiting conditions due to the lowered RCS level and potential for reaching core damage conditions early in the accident progression. The timing analysis assumes availability of forced or gravity-driven injection because these are likely to be successful and are the dominant sequence contributions to core damage frequency. Under these conditions in POS 5B, the minimum time available for action prior to core damage is approximately 10 hours from the time of the initiating event. For the other POS conditions, the time available is greater than 10 hours.

The operator response to an initiating event in POS 5B is guided by the reference plant's abnormal operating procedure for loss of RHR. The operators will diagnose the entry conditions for the procedure, and then evaluate the steps in the procedure in their attempts to restore the heat removal function. According to the procedure, the operators will focus on performing continuous actions to control RCS level and monitoring conditions. The procedure directs operators to initiate actions to establish containment closure. A delay time of 15 minutes is assumed to reach this procedure step.

The performance of the action to close the purge system isolation valves is straightforward and is performed from the control room. The estimated time to complete the action is 15 minutes. There is ample time for performing the action. A summary of the SPAR-H assessment of the performance shaping factors influencing this action is provided Table D-10. The basic event used to model this HFE is named 1-CIS-XHE-XL-PURGE.

D.4.3 Isolating Small Containment Penetrations

The opening of containment penetrations can occur throughout an outage to support the various maintenance and testing activities being performed. The potential large containment openings (e.g., equipment hatch, personnel airlock, and purge system) are modeled with other HFEs. This section describes the action to close the possible small that could be open.

The action is assessed for conditions in POS 6. In POS 6, both RHR trains are operating. The RCS temperature is assumed to be 110 °F. The RCS level during this POS is being drained from level near the reactor vessel flange to just above the center line of the hot leg. The level remains near mid-loop for most of the POS while steam generator nozzle dams are installed. Per the Level 1 PRA POS definition, the containment equipment hatch is closed and secured with a minimum of four bolts. The emergency and personnel airlocks are also closed.

The reference plant's operating procedures for this operating state include directions for tracking the status of containment penetrations and having a plan for containment closure. As such, it is anticipated that at the time of the accident, there would be good general operational awareness of the existence of open containment penetrations (and specifically which penetrations), as well as the means for closing them, despite the hatch and airlocks already being closed.

During this mode of operation, the RCS temperature is monitored with two core exit thermocouples, and the integrated plant computer is set to alarm if the thermocouple inputs are 10 °F above T_{avg} temperature. Other temperature indications that are available in the main control room are the hot leg wide range temperature indications on each of the four loops and the RHR heat exchanger inlet temperature indications. Following an initiating event (e.g., loss of RHR), operators will transition to the relevant abnormal operating procedure, which will direct actions to establish containment integrity.

Considering the procedural direction for a loss of RHR event, the steps leading up to isolating containment involve:

- Checking if RHR pumps should be stopped,
- Initiating continuous actions to maintain RCS level and monitor RCS temperature,
- Checking and adjusting valve positions/alignments to control RCS level,
- Initiating emergency classification determination,
- Determining the time to RCS boiling,
- Initiating actions to protect personnel in containment.

The above steps will require time to evaluate. If RHR cooling is lost, then operators would have approximately 20 minutes to diagnose the situation and perform actions to restore cooling before RCS boiling would occur. The procedure directs evacuation of non-essential personnel from containment. It would be unlikely that confirmation of all containment penetrations could be confirmed and closed during this initial response to the event. However, if operators establish RCS injection as an alternate means of core cooling, then the sequence timeline is extended for approximately 8 hours until RWST inventory is exhausted. If injection is successful, then this time would be available to confirming status of containment penetrations. If injection is failed, then the time available would be limited to approximately 3 hours, which is when the onset of core damage would begin. The actions would also be limited to areas outside of containment due to habitability concerns inside containment. The basis for this action assumes successful

RCS injection. If RCS injection is failed, then this human action is adjusted according to HFE dependency analysis.

The 8 hours available for the POS 6 conditions are considered to be the most limiting conditions for the containment isolation action. Conditions during POS 5B are similar. The higher decay heat level would exhaust the RWST inventory in approximately 7 hours, but there is more time available before RCS boiling compared to POS 6. During POSs 7 and 8E, the RWST is displaced to the refueling cavity, which would leave less inventory for injection (duration of three to five hours), but there is also much longer time available before RCS boiling. Also, the post-refueling POSs have much longer time available for isolating containment penetrations. Therefore, the 8 hours for POS 6 conditions are used as a reference point for the containment isolation action, and other POS conditions are expected to have longer times available.

The execution of this action involves reviewing the containment building penetrations procedure, verify the equipment status logs referenced in the procedure, dispatching staff to close any opened containment penetrations, and verifying that those penetrations are closed. Executing the initial steps of the procedure to identify the open penetrations and dispatching field operators is estimated to take 10 minutes. A review of the reference plant's procedures identified 15 potential containment penetrations that could be open along with the equipment hatch, personnel airlock and emergency airlock. It is assumed that several field operators or maintenance staff experienced and trained in closing containment penetrations are available, but not a sufficient number to close all penetrations in parallel. It is not expected that all penetrations would be open at the time of the event. It is assumed that each available personnel member (or team) must serially close two penetrations. It is assumed to take 20 minutes for closing each penetration, given time to transit to the penetration and perform the closure. Therefore, the total time required for the action is assumed to be 1 hour, 10 minutes for the initial procedure steps and logs review and 20 minutes for closing two penetrations. All the identified penetrations are confirmed to have valves or flanges that can be closed from a location outside containment.

The HFE for closing small containment penetrations is modeled with basic event 1-CIS-XHE-XL-ISOLPEN. See Table D-10 for the SPAR-H assessment of PSFs. The timing and other sequence conditions can significantly influence the likelihood of failure for this event. Therefore, the dependency analysis, discussed in section D.6, should also be consulted for the failure probability of closing containment penetrations. This includes dependency on prior failures that can result in SBO sequences. Most penetrations can be closed by manual valves without power available, but the occurrence of an SBO would significantly influence the failure probability. Also, prior failures of actions to establish RCS injection are considered in the dependency analysis, which would decrease the time available to perform the action.

Table D-10 Evaluation of SPAR-H Performance Shaping Factors for Isolating Containment Penetration Human Failure Events

HFE Basic Event Name	HFE Description and PSF Evaluation							
1-CIS-XHE-XR-PURGE	Operators fail to isolate containment purge after system shutdown (pre-initiator HFE) POS applicability: POSs, 6, 7, 9, 10, and 11 HFE type: Pre-initiator with reference HEP from (Swain, 1987) with credit for two recovery factors							
1-CIS-XHE-XL-PURGE (Execution PSFs)	Operators fail to isolate containment purge when system operating POS applicability: assessed for POS 5B; also applied to POSs 4, 5A, and 12 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Nominal	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CIS-XHE-XL-ISOLPEN (Execution PSFs)	Operators fail to isolate containment small containment penetrations POS applicability: assessed for POS 6; also applied to POSs 4 – 5B and 7 – 12 HFE type: Execution							
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> Extreme	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal

D.5 HRA Characterization for Containment Cooling Units

This section describes the actions to establish operation of the containment cooling units during the time before the onset of core damage. The status of the CCUs during the outage is outlined in Table D-1. The CCUs are assumed to be in operation (i.e., either one train or both trains are operating) during POSs 3, 10, 11, 12, and 13. Therefore, the actions described in this section apply to the POSs where the CCUs are not operating, POS 4 through POS 9.

For any reactor at-shutdown initiating event, and if containment is already isolated/closed or has been isolated/closed in response to the accident, there will be a desire to establish (or maintain) containment coolers in order to control containment conditions. If containment is not closed or isolated, containment coolers will have limited effect (due to the containment being open to the environment). If AC power is unavailable, the containment coolers are also assumed unavailable and cannot be restored. The actions described here apply to the accident sequences where AC power is available, conditions and procedural direction indicate the need to close and isolate containment (should it not already be closed and isolated at the time of the accident), and to subsequently restore containment coolers (should they not already be running at the time of the accident).

The containment cooling system involves two safety-related trains, each with four cooling units. Air is drawn from upper containment, passed over large radiator-like cooling units (which are cooled by the nuclear service cooling water system), and discharged into lower containment. There are indications in the main control room of containment atmosphere status, and whether the containment fan coolers are operating (supply and return flow indicators for each pair of coolers on the main control board panel). The reference plant's operating procedures applicable during cooldown, refueling, and mid-loop operations indicate the practices for containment fan cooler operation under normal outage conditions, while the abnormal operating procedures that govern the relevant accident response have more specific cues to restore CCU operation. When the relevant procedure step is reached, the procedure gives clear direction to operate the containment cooling fans.

The time available to restore containment coolers is associated with the time to core damage. This time varies dramatically depending on the particular accident sequence conditions. Table D-11 summarizes the time available prior to core damage for accidents assuming no intervening successful mitigation actions, based on calculations for the Level 1 shutdown PRA success criteria analysis (NRC, 2025). The actual time available can vary for individual accident sequence conditions (e.g., successful RCS injection can extend the time available), but the table provides a reference point for the variation in the available time. As can be seen, there are generally many hours between the accident initiation and the time of core damage, even for the sequences without successful mitigation. The most limiting conditions are associated with POSs 5B and 6 with available time on the order of 3–5 hours, while for other POS/sequence combinations it will be many hours. Based on this discrepancy in the available time, the actions to establish CCU operation is treated separately for POS 5B/6 versus all other relevant POSs.

Table D-11 Summary of Available Time to Core Damage for Shutdown Accident Scenarios

Plant State Analyzed	Time to Core Damage (hr)
POS 5A (pressurizer solid) event leading to LORHR	27
POS 5B event leading to LORHR	5.3
POS 5B LORHR with successful gravity feed	10
POS 6 event leading to LORHR	3.3
POS 7 event leading to LORHR	52
POS 8E event leading to LORHR	90
POS 8L event leading to LORHR	214
POS 9 event leading to LORHR	15

D.5.1 Operation of Containment Cooling Units during POSs 5B and 6

The specific status of the CCUs is not clearly indicated in past outage records. During parts of the outage, one train of CCU support systems (e.g., electrical power, service water) is unavailable due to maintenance. It is assumed one train of the CCUs is also unavailable and cannot be restored. While most maintenance activities occur later in the outage, specifically during POS 8, it is conservatively assumed that only one train of CCUs is available for restoration throughout POS 5B and POS 6.

It is further assumed that the remaining train of CCUs has been “racked out” as part of standard outage work planning activities, but that it is otherwise operational and has necessary and available support systems. In order to restore the CCUs, operators would need to notify outage work planning of their intent to return the containment coolers to service, “rack in” the appropriate breakers located in the control building and actuate the coolers from the main control board.

The time window required for restoring the CCUs is estimated to be 20 minutes to perform the action plus the delay time to initiate the action. In POS 6, if a loss of RHR were to occur, it is estimated that the action must be completed within three hours of the event occurrence. The first minutes of the response would be focused on the operator diagnosing the condition and entering the appropriate abnormal operating procedure. The operators will then evaluate the steps in the procedure to restore the heat removal function, control RCS level, and establish containment closure. The delay time of 30 minutes is assumed before the procedure step for containment cooling is reached and action to restore CCUs begins. It is assumed that the prior actions and continuous actions identified in the procedure are proceeding in parallel. Therefore, the time window from the initiating event to completing the CCU operation is 50 minutes. See Table D-12 for the SPAR-H evaluation of PSFs for HFE 1-CCU-XHE2-SD-POS6 for assessment of containment closure in POS 5B and POS 6 and estimation.

D.5.2 Operation of Containment Cooling Units during POSs 4, 5A, 7, 8E, 8L, and 9

The assessment of the action to restore containment cooling units during the remaining POSs is very similar to the action during POSs 5B and 6, which is assessed in Section D.5.1, above. There are similar procedural prompts and the required actions to be performed are the same. The primary difference from POSs 5B and 6 is the time available to perform the action. As can be seen in Table D-11, the other POS conditions have significantly longer times before core

damage would be reached. The most limiting conditions occur during POS 9 with 15 hours from the time of the initiating event to core damage, assuming no intervening successful mitigating actions. Some of the other POSs have significantly longer times available.

As a modeling simplification, the timing assessment is based on POS 9 conditions and applied to all other POSs. The time window required for restoring the CCUs is estimated to be 20 minutes to perform the action plus the delay time to initiate the action. The delay time of 30 minutes is assumed before the procedure step is reached and action begins to restore CCUs. It is assumed that the prior actions and continuous actions identified in the procedure proceed in parallel. The time available to complete the action is 15 hours from the time of the initiating event. See Table D-12 for the SPAR-H evaluation of PSFs for HFE 1-CCU-XHE2-SD for assessment of containment closure in POSs 4, 5A, and 7–9.

Table D-12 Evaluation of SPAR-H Performance Shaping Factors for Containment Cooling Human Failure Events

HFE Basic Event Name	HFE Description and PSF Evaluation							
1-CCU-XHE2-SD-POS6 (Diagnosis PSFs) (Execution PSFs)	Operators fail to restore 1 train of CCUs in POS 6 POS applicability: assessed for POS 6; also applied to POSs 5B HFE type: Diagnosis and Execution							
	<i>Time PSF:</i> Expansive	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Nominal	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
	<i>Time PSF:</i> ≥ 5x time required	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
1-CCU-XHE2-SD (Diagnosis PSFs) (Execution PSFs)	Operators fail to restore 1 train of CCUs in POS 4-5A and 7-9 POS applicability: assessed for POS 9; also applied to POSs 4 – 5A and 7 – 8L HFE type: Diagnosis and Execution							
	<i>Time PSF:</i> Expansive	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Nominal	<i>Training PSF:</i> Nominal	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal
	<i>Time PSF:</i> ≥ 50x time required	<i>Stress PSF:</i> High	<i>Complex PSF:</i> Moderate Complex	<i>Training PSF:</i> Low	<i>Procedure PSF:</i> Nominal	<i>Ergonomics PSF:</i> Nominal	<i>Fitness PSF:</i> Nominal	<i>Work Processes PSF:</i> Nominal

D.6 Human Failure Event Dependency Analysis

To ensure that all key HFE dependency combinations were identified, the plant damage states (including the Level 1 accident sequences and the bridge tree) in the L3PRA project shutdown model were solved while setting the HEPs for applicable HFEs to 0.9.²² It was determined that a cut set contribution cutoff of 1×10^{-6} was sufficient to prevent any potential HFE combinations that could significantly affect the sequence frequencies from being eliminated from the evaluation prior to the review. This resulted in 5296 cut sets (with elevated HEPs) with frequencies greater than or equal 1×10^{-6} .²³ Using a lower cut set screening frequency threshold could add a significant number of additional HFE combinations; however, it is not believed that the additional combinations will result in many new HFE pairs (i.e., most, if not all, HFE pairs are already included in this evaluation).

The cut sets were evaluated to identify 380 different HFE combinations, each consisting of two or more HFEs. Some of the HFE combinations were eliminated from further dependency evaluations because they were not relevant to the bridge tree dependency analysis. The following types of HFE combinations were removed from further consideration:

- HFE combinations that included only successful bridge tree HFEs and no failed bridge tree HFEs
- HFE combinations that included only Level 1 HFEs and no bridge tree HFEs (these combinations are addressed in the Level 1 PRA dependency analysis)
- HFE combinations that involve pre-initiator HFEs to restore systems or components

From the remaining cutsets, 150 distinct HFE pairs were identified.

The dependency analysis considers the time sequence of each HFE pair. The second HFE in the sequence is evaluated for potential dependency on the first HFE. This dependency analysis evaluates the potential dependency between the modeled Level 1 PRA HFEs and the Level 2 PRA bridge tree HFEs related to containment integrity and isolation. The dependency is only evaluated for pairs where the Level 1 PRA HFE occurs first in the sequence, and the bridge tree HFE occurs second. However, for several of the HFE pairs, the bridge tree HFE occurs first in the timing of the sequence. Consider the following example HFE sequence:

1. HFE₁ is a Level 1 HFE modeling failure to restore shutdown cooling. HFE₁ occurs first in the sequence.
2. HFE₂ is a bridge tree HFE modeling failure to isolate containment penetrations. HFE₂ occurs after HFE₁.
3. HFE₃ is a Level 1 HFE modeling failure to establish forced injection after other actions to restore shutdown cooling have failed. HFE₃ occurs after HFE₂.

In this example, the dependency between the HFE pair HFE₁-HFE₂ is evaluated as part of this dependency analysis. The dependency between HFE pair HFE₂-HFE₃ is not evaluated. The

²² HEPs that were currently set to 1.0 (or TRUE) were left at that value while all others were set to 0.9.

²³ The corresponding frequencies for these cut sets will be at least an order of magnitude lower even if complete dependency existed, because the 1st HFE in the combination is independent (i.e., its HEP would not be elevated).

dependency between HFE pair HFE₁-HFE₃ is evaluated as part of the Level 1 PRA dependency analysis. The intervening action modeled in HFE₂ is not directly evaluated in the Level 1 PRA. However, the intervening action is considered in evaluating the dependency between HFE₁ and HFE₃, and this action can influence several of the PSFs (e.g., timing, procedures, and complexity). No further dependency is considered between HFE₂ and HFE₃. This treatment reflects a fundamental limitation in HRA application, wherein the PRA does not consider all operator actions, and thereby can exclude intervening actions that have some potential effect on HEP dependency.

After the dependent HFEs were identified, the level of dependency was determined using the guidance in NUREG-1792 (NRC, 2005b). NUREG-1792 presents a few specific elements that should be evaluated for determining level of dependency, such as:

- The same crew member(s) are responsible for the actions.
- The actions can be considered to take place relatively close in time such that a common crew mindset may carry over from one action to the next.
- There are similar plant conditions between the actions and they are being directed by identical (or nearly so) procedure and cue.
- The actions are performed in the same location and performed in similar ways.

More generally, NUREG-1792 suggests evaluating if the actions have similar PSFs and if there is reason to believe that the crew's interpretation of the need or decision for an action might influence the crew's decisions for actions later in the scenario.

The LPSD model considers these elements in determining the level of dependency, and used the same approach is used during the L3PRA at-power internal events analysis.²⁴ The approach for determining the dependency level specifically considered the elements outlined in NUREG-1792 of same crew, timing of cues and action, same procedures and cues, and same location. The approach applied for the LPSD HRA also assessed if enough resources are available for the successful completion of both actions. The combination of all these elements starts to address the more general suggestion by NUREG-1792 to evaluate similar PSFs across the actions. In addition, the evaluation of same procedures and cues for the actions is asserted to address an element termed "common cognitive" referring to the crew having a common mindset or belief in approaching the actions. This "common cognitive" is likely to also influence the crew's decisions for actions later in the scenario. Specifically, the following criteria are evaluated through this approach:

- **Same Crew.** If the actions are assumed to be performed by different crew, the HFEs may be considered to be independent. If the difference in time between the cues for

²⁴ Note that a change is made from the method employed for the L3PRA at-power internal events analysis (NRC, 2022b). The impacts of intervening successes were not evaluated for the full power dependency evaluation. For the LPSD dependency evaluation, intervening success was assessed on a case-by-case basis. This assessment is consistent with the approach used in the EPRI HRA Calculator that states an intervening success would break dependency between a pair. In addition, the end points of the dependency evaluation tree often show a lower level of dependency than shown in the dependency evaluation tree published in the at-power document. This change is due to the current application of the tree not evaluating stress as "high" as was done in the L3PRA at-power analysis.

each of the HFEs is greater than the length of the shift, a new crew can be assumed to be responding to the cue of the second HFE.

- **Time.** This element assesses the amount of time that is estimated to have elapsed between the cues for each of the HFEs. The options are either occurring simultaneously or in one of the following intervals:
 - $0 < \text{Time} \leq 15$ minutes
 - $15 < \text{Time} \leq 30$ minutes
 - $30 < \text{Time} \leq 60$ minutes
 - $\text{Time} > 60$ minutes
- **Common Cognitive / Same Cues and Procedures.** If the crew can be assumed to be in a common cognitive mindset while responding to both HFEs, complete dependency is assigned. This element is assessed by evaluating if the cue and/or procedure steps being used are identical for the HFEs being evaluated. NUREG-1921²⁵ defines a cue as “a change in condition or signal that triggers the need for an action” (page A-2); therefore, a cue can be thought of as either an indication, procedure step, or alarm that alerts the crew member to need for a response.
- **Adequate Resources.** This element assesses whether an adequate number of staff is available to support the required actions. For the dependency determination, this assessment is only relevant if the actions are required to be completed during the same time. If staffing is found to be insufficient, complete dependency is assumed.
- **Same Location.** The location refers to the room or general area in which the actions will be executed. If the actions are executed in the same location, a higher level of dependency is typically assessed.

Using this approach, a level of dependency is assigned as shown in Figure D-6²⁶ of either: complete, high, moderate, low, or zero.

²⁵ Electric Power Research Institute and U.S. Nuclear Regulatory Commission, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines,” NUREG-1921, July 2012.

²⁶ This figure differs in some areas from the dependency evaluation tree included in the L3PRA full power internal events analysis. The representation of the dependency evaluation for this LPSD report represents the intended application for the L3PRA project, however, some decisions were made after the L3PRA full power internal events model was completed, and therefore, were not included in the original version of that model. Specific changes include an evaluation of intervening success and the final end points on the evaluation tree reflect those values associated with “low stress” used within the dependency evaluation tree proposed within the EPRI HRA Calculator (Electric Power Research Institute, “Human Reliability Analysis Calculator Version 4.21,” TR 1022814, June 2011.)

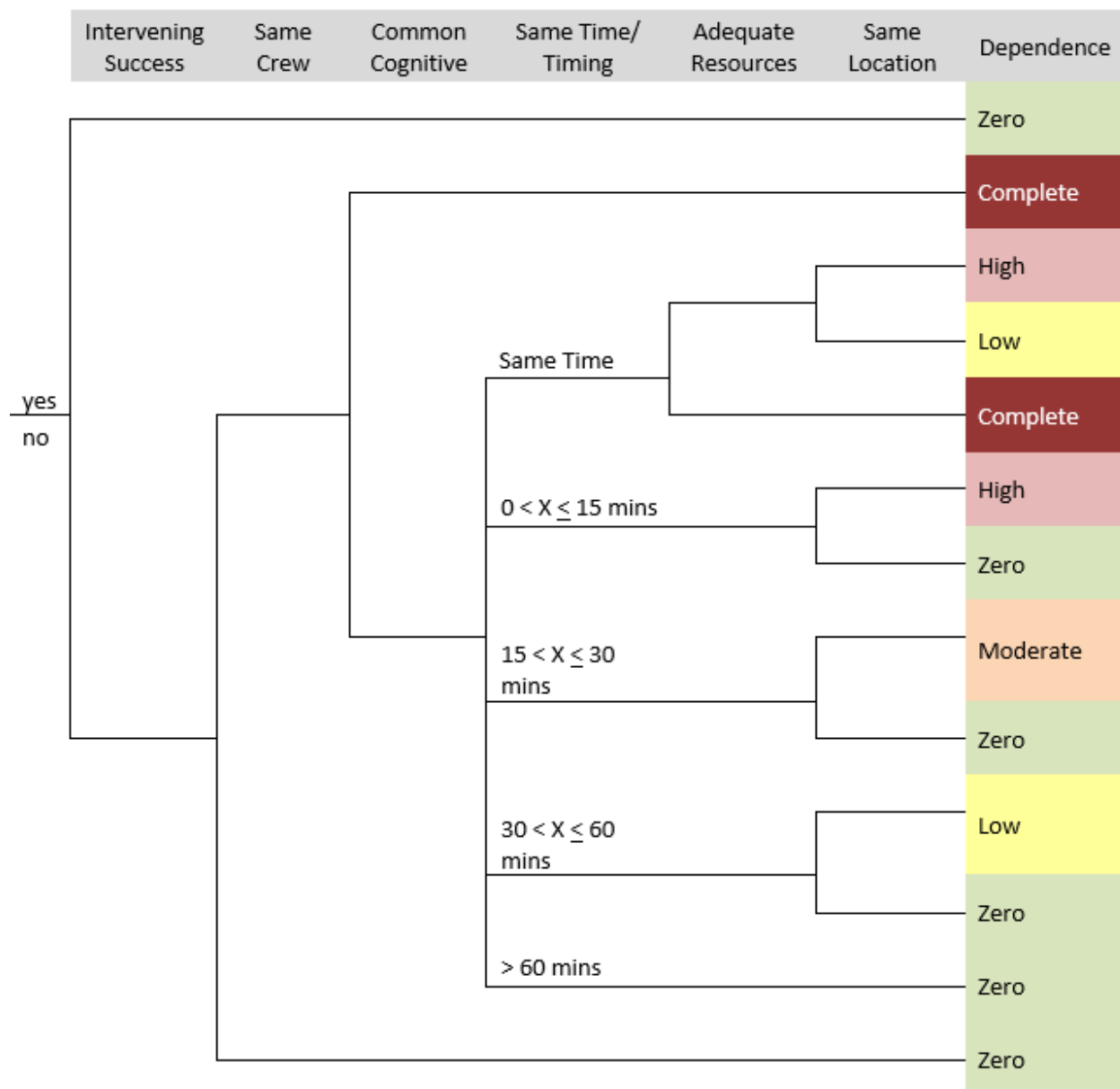


Figure D-6 L3PRA Modified Dependency Decision Tree

Following the assignment of the dependency level, for the dependent HFE, the HEPs were calculated by applying the following dependency formulas given in THERP²⁷ and shown in Table D-13.

²⁷ U.S. Nuclear Regulatory Commission, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, August 1983.

Table D-13 Human Error Probability Assignment for Dependency Levels

Dependence Level	Equation
Zero	HEP
Low	$(1 + 19 \times \text{HEP}) / 20$
Moderate	$(1 + 6 \times \text{HEP}) / 7$
High	$(1 + \text{HEP}) / 2$
Complete	1.0

If more than two HFEs are considered to be dependent in a sequence, the dependency calculation for each successive HFE is calculated based only on the immediately preceding HFE. In other words, if three HFEs termed HFE₁, HFE₂, and HFE₃ were found to be dependent, the calculation of the dependent HEP for HFE₂ would be based on the dependency level determined between HFE₁ and HFE₂. The calculation for the dependency HEP for HFE₃ would be based on the dependency level determined between HFE₂ and HFE₃. This calculation is based on guidance given in THERP.

Table D-14 lists those HFE pairs evaluated for dependency.

Table D-14 Shutdown Bridge Tree HFE Dependency Pairs and Evaluation

HFE 1	HFE 2	Dependency Level
1-OEP-XHE-XL-NR02H	1-CIS-XHE-XL-ISOLPEN	high
1-RHR-XHE-XM-M5LOI-I	1-CNT-XHE-XM-H1	high
1-HPI-XHE-XM-FEED-6H	1-CIS-XHE-XL-ISOLPEN	low
1-HPI-XHE-XM-FEED-6H	1-CNT-XHE-XM-H1	low
1-HPI-XHE-XM-FFEED-6H	1-CCU-XHE2-SD-POS6	low
1-HPI-XHE-XM-FFEED-6H	1-CIS-XHE-XL-ISOLPEN	low
1-HPI-XHE-XM-FFEED-6H	1-CNT-XHE-XM-H1	low
1-HPI-XHE-XM-FFEED-LOI	1-CIS-XHE-XL-ISOLPEN	low
1-HPI-XHE-XM-FFEED-LOI	1-CNT-XHE-XM-H1	low
1-HPI-XHE-XM-FFEED-LOI5A	1-CIS-XHE-XL-ISOLPEN	low
1-HPI-XHE-XM-FFEED-LOI5A	1-CNT-XHE-XM-H1	low
1-HPI-XHE-XM-FFEED-LOI-6	1-CIS-XHE-XL-ISOLPEN	low
1-HPI-XHE-XM-FFEED-MLDRL	1-CCU-XHE2-SD-POS6	low
1-HPI-XHE-XM-FFEED-MLDRL	1-CIS-XHE-XL-ISOLPEN	low
1-LOI-XHE-XM-ISO2-LT	1-CCU-XHE2-SD-POS6	low
1-LOI-XHE-XM-ISO2-LT	1-CIS-XHE-XL-ISOLPEN	low
1-LOI-XHE-XM-ISO2-LT	1-CNT-XHE-XM-H1	low
1-LPI-XHE-XM-GFEED-LD	1-CCU-XHE2-SD-POS6	low
1-LPI-XHE-XM-GFEED-LD	1-CIS-XHE-XL-ISOLPEN	low
1-LPI-XHE-XM-GFEED-LD	1-CNT-XHE-XM-H1	low
1-LPI-XHE-XM-GFEED-LOI	1-CNT-XHE-XM-H1	low
1-LPI-XHE-XM-GFEED-LOI5A	1-CIS-XHE-XL-ISOLPEN	low
1-LPI-XHE-XM-GFEED-LOI5A	1-CNT-XHE-XM-H1	low
1-LPI-XHE-XM-GFEED-MD	1-CCU-XHE2-SD-POS6	low
1-RHR-XHE-XD-3H	1-CNT-XHE-XM-H1	low
1-RHR-XHE-XL-LTREC	1-CIS-XHE-XL-ISOLPEN	low
1-RHR-XHE-XL-LTREC	1-CIS-XHE-XL-PURGE	low
1-RHR-XHE-XL-LTREC	1-CNT-XHE-XM-H1F	low
1-RHR-XHE-XL-LTREC	1-CNT-XHE-XM-H2	low
1-RHR-XHE-XL-TRN-3H	1-CNT-XHE-XM-H1	low
1-RHR-XHE-XL-TRN-M5LOI	1-CNT-XHE-XM-H1	low
1-RHR-XHE-XL-TRN-STBY-2H	1-CNT-XHE-XM-H1	low
1-LOI-XHE-XD-POS5A	1-CNT-XHE-XM-H1	moderate
1-RHR-XHE-XL-TRN-POS9	1-CNT-XHE-XM-H1	moderate
1-ACP-XHE-XL-ALTAC-6H	1-CNT-XHE-XM-H3	zero
1-HPI-XHE-XM-FEED-6H	1-CCU-XHE2-SD	zero
1-HPI-XHE-XM-FEED-6H	1-CNT-XHE-XM-H2	zero
1-HPI-XHE-XM-FFEED-6H	1-CIS-XHE-XL-PURGE	zero
1-HPI-XHE-XM-FFEED-LOI	1-CCU-XHE2-SD	zero

Table D-14 Shutdown Bridge Tree HFE Dependency Pairs and Evaluation (cont.)

HFE 1	HFE 2	Dependency Level
1-HPI-XHE-XM-FFFEED-LOI	1-CIS-XHE-XL-PURGE	zero
1-HPI-XHE-XM-FFFEED-LOI5A	1-CCU-XHE2-SD	zero
1-HPI-XHE-XM-FFFEED-LOI5A	1-CIS-XHE-XL-PURGE	zero
1-LOI-XHE-XD-POS5A	1-CCU-XHE2-SD	zero
1-LOI-XHE-XD-POS5A	1-CIS-XHE-XL-ISOLPEN	zero
1-LOI-XHE-XD-POS5A	1-CIS-XHE-XL-PURGE	zero
1-LOI-XHE-XD-POS5A	1-CNT-XHE-XM-H2	zero
1-LOI-XHE-XM-ISO2-LT	1-CCU-XHE2-SD	zero
1-LPI-XHE-XM-GFEED-LD	1-CCU-XHE2-SD	zero
1-LPI-XHE-XM-GFEED-LD	1-CIS-XHE-XL-PURGE	zero
1-LPI-XHE-XM-GFEED-LD	1-CNT-XHE-XM-H2	zero
1-LPI-XHE-XM-GFEED-LOI	1-CCU-XHE2-SD	zero
1-LPI-XHE-XM-GFEED-LOI	1-CCU-XHE2-SD-POS6	zero
1-LPI-XHE-XM-GFEED-LOI	1-CIS-XHE-XL-ISOLPEN	zero
1-LPI-XHE-XM-GFEED-LOI	1-CIS-XHE-XL-PURGE	zero
1-LPI-XHE-XM-GFEED-LOI5A	1-CCU-XHE2-SD	zero
1-LPI-XHE-XM-GFEED-LOI5A	1-CIS-XHE-XL-PURGE	zero
1-LPI-XHE-XM-GFEED-LOI5A	1-CNT-XHE-XM-H2	zero
1-LPI-XHE-XM-GFEED-LOI-6	1-CIS-XHE-XL-ISOLPEN	zero
1-LPI-XHE-XM-GFEED-LOOP	1-CNT-XHE-XM-SBO3	zero
1-LPI-XHE-XM-GFEED-MD	1-CIS-XHE-XL-PURGE	zero
1-LPI-XHE-XM-GFEED-MLDRL	1-CCU-XHE2-SD-POS6	zero
1-LPI-XHE-XM-GFEED-MLDRL	1-CIS-XHE-XL-ISOLPEN	zero
1-OEP-XHE-XL-NR02H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XD-3H	1-CCU-XHE2-SD	zero
1-RHR-XHE-XD-3H	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XD-3H	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XD-3H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XD-3H	1-CNT-XHE-XM-H1F	zero
1-RHR-XHE-XD-3H	1-CNT-XHE-XM-H2	zero
1-RHR-XHE-XD-ML	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XD-ML	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XD-ML	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XD-POS3	1-CCU-XHE2-SD	zero
1-RHR-XHE-XD-POS3	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XD-POS3	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XD-POS3	1-CNT-XHE-XM-H2	zero
1-RHR-XHE-XD-POS8E	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-LTREC	1-CCU-XHE2-SD	zero

Table D-14 Shutdown Bridge Tree HFE Dependency Pairs and Evaluation (cont.)

HFE 1	HFE 2	Dependency Level
1-RHR-XHE-XL-LTREC	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-LTREC	1-CNT-XHE-XM-H1	zero
1-RHR-XHE-XL-LTREC-LT5H	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-LTREC-LT5H	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-LTREC-LT5H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-LTREC-LT5H	1-CNT-XHE-XM-H2	zero
1-RHR-XHE-XL-LTREC-LT5H	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-STBY-POS7	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-STBY-POS7	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-STBY-POS7	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-TRA-POS8E	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-TRB-POS7	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-TRB-POS7	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRB-POS7	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-TRB-POS8E	1-CNT-XHE-XM-H3	zero
1-RHR-XHE-XL-TRN-3H	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-3H	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-3H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-3H	1-CNT-XHE-XM-H1F	zero
1-RHR-XHE-XL-TRN-LOOP	1-CNT-XHE-XM-SBO3	zero
1-RHR-XHE-XL-TRN-M5LOI	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-TRN-M5LOI	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-M5LOI	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-M5LOI	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-MLDRL	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-MLDRL	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-MLDRL	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-ML-P10	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-ML-P10	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-POS10	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-POS10	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-POS6	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-POS6	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-POS6	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-POS9	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-TRN-POS9	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-POS9	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-POS9	1-CNT-XHE-XM-H1F	zero
1-RHR-XHE-XL-TRN-POS9	1-CNT-XHE-XM-H2	zero

Table D-14 Shutdown Bridge Tree HFE Dependency Pairs and Evaluation (cont.)

HFE 1	HFE 2	Dependency Level
1-RHR-XHE-XL-TRN-STBY-2H	1-CCU-XHE2-SD	zero
1-RHR-XHE-XL-TRN-STBY-2H	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-STBY-2H	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-STBY-2H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XL-TRN-STBY-2H	1-CNT-XHE-XM-H1F	zero
1-RHR-XHE-XL-TRN-STBY-ML	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XL-TRN-STBY-ML	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XL-TRN-STBY-ML	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-LOISP-POS6	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-LOISP-POS6	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-M5LOI-I	1-CCU-XHE2-SD	zero
1-RHR-XHE-XM-M5LOI-I	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XM-M5LOI-I	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-M5LOI-I	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-MLDRL-I	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XM-MLDRL-I	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-MLDRL-I	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-POS10-I	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-POS10-I	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-TRIP-HD	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-TRIP-LD	1-CCU-XHE2-SD	zero
1-RHR-XHE-XM-TRIP-LD	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XM-TRIP-LD	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-TRIP-LD	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-TRIP-LD	1-CNT-XHE-XM-H1	zero
1-RHR-XHE-XM-TRIP-MD	1-CCU-XHE2-SD	zero
1-RHR-XHE-XM-TRIP-MD	1-CCU-XHE2-SD-POS6	zero
1-RHR-XHE-XM-TRIP-MD	1-CIS-XHE-XL-ISOLPEN	zero
1-RHR-XHE-XM-TRIP-MD	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-TRIP-MD	1-CNT-XHE-XM-H2	zero
1-RHR-XHE-XM-TRNA-3H	1-CCU-XHE2-SD	zero
1-RHR-XHE-XM-TRNA-3H	1-CIS-XHE-XL-PURGE	zero
1-RHR-XHE-XM-TRNA-3H	1-CNT-XHE-XM-H2	zero

APPENDIX E

POST-CORE-DAMAGE HUMAN RELIABILITY ANALYSIS TO SUPPORT THE LEVEL 2 REACTOR AT-SHUTDOWN PRA

E.1 Introduction

This appendix documents work done to develop human failure events (HFEs) and supporting human error probabilities (HEPs) associated with mitigative actions credited in the reactor at-shutdown Level 2 PRA and associated human reliability analysis (HRA). It heavily leverages past work on this project for the at-power Level 2 PRA (NRC, 2022a), and in so doing presumes the reader has knowledge of that work.

Feedback received on the at-power Level 2 HRA was considered in developing this report. Some of this feedback could be directly considered (e.g., the reference plant staff feedback on generic timing estimates) while other feedback could not (e.g., the view on the part of some Technical Advisory Group Members and peer reviewers that actions during station blackout events should be credited, the view by some individual Advisory Committee on Reactor Safeguards (ACRS) members that the HRA should model multiple parallel potential actions). Of the two examples of items that could not be addressed, the former is not addressed here due to continued lack of credit for power recovery after station batteries have depleted and a fundamental lack of a state of practice in this area, while the latter is not addressed as a practical resource and computational complexity limitation. Individual ACRS members also expressed the thoughts that the Level 2 PRA HRA work should rely more heavily on operator input (beyond the limited operator feedback realized during the internal events work). On this point, it is particularly important to keep in mind that the as-designed/as-operated plant being studied is the reference plant as operated in 2012, while the present work is ongoing several years later. In particular, in 2012 (and as will be discussed later), plant SAMGs specifically restricted use to situations involving accidents that occurred while at-power.²⁸ In addition, the accessibility of plant staff and the difficulty in reconstructing their perspectives after several years of significant plant changes germane to accident management and its guidance, further limit the usefulness of this exercise. For these reasons, the work in this report is based on NRC staff and contractor understanding and perspectives developed during the at-power Level 2 PRA HRA work.

The following sections explain the scope of the HRA, the approach taken relative to the reactor at-power Level 2 PRA HRA, the context of the considered HFEs, and the estimation of the associated HEPs.

E.2 HRA Scope and Identification of Candidate Actions

The purpose of this discussion is to document the procedural/guidance scope of the reactor at-shutdown Level 2 HRA, specifically regarding how the Severe Accident Management Guidelines (SAMGs) and Extensive Damage Mitigation Guidelines (EDMGs) are, or are not, considered in the post-core-damage accident management modeling. In the timeframe at which this project is modeling the reference plant (circa 2012-2016), the SAMGs, EDMGs, and Emergency Plan all

²⁸ The reference plant provided revisions to their SAMG guidance documents that include shutdown operations within their scope. These updated SAMGs were not available and thus not considered at the time the analysis was performed. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012.

work concurrently, but are not integrated into a single governing framework. It is important to keep in mind that some of the same individuals (e.g., the Emergency Director [ED]) and organizational entities (e.g., the Technical Support Center [TSC]) have roles in the various processes outlined in these document sets.

The SAMGs are specifically focused on providing guidance to onsite accident management staff when core damage is imminent or has occurred, and thus when the emergency operating procedures no longer supply sufficient direction. This approach is based on the version of the reference plant's SAMGs (circa 2012) that were available at the time of the analysis, which follow the contemporary Westinghouse SAMGs in place since 1998. The SAMGs provide guidance to the Main Control Room crew (MCR) and the TSC, and by extension to the Operations Support Center (OSC) and ED. The SAMGs are specifically intended for reactor, at-power operation. They do not provide any guidance for reactor shutdown modes of operation. A subsequent revision of the reference plant's SAMGs were provided after the initial analysis was completed. The updated SAMGs include shutdown operations, but the analysis was not revised to incorporate the updated SAMGs. This is consistent with the overall L3PRA project approach, which reflects the reference plant as it was designed and operated as of 2012. On this basis, the project team does not consider the SAMGs within the reactor at-shutdown Level 2 PRA HRA.

The EDMGs, on the other hand, are guidelines focused on responding to a loss of large area of the plant due to fire(s) or explosion(s), including situations in which normal command and control may be compromised. They focus on providing alternative means for (i) re-establishing command and control, (ii) providing core, containment, and SFP cooling, and (iii) dealing with other emergency-related needs and responsibilities. A flowchart covers the full scope of EDMG-related activities, while a set of supporting guidelines focus on implementation of specific strategies. These strategies can be seen in Table E-1.

The EDMGs rely on a mix of pre-existing, installed equipment and portable equipment that was acquired specifically for this purpose. The EDMGs are not restricted to at-power reactor operation or post-core-damage conditions; however, some of the specific guidance and strategies are only applicable to at-power operations (the governing procedure is silent about their intended scope in this respect). Based on a detailed review of the reference plant's documents and detailed discussions with plant staff, as well as a drill critique, the project team has confidence that this guidance set would be in the forefront of responders' minds during a severe accident. On this basis, the project team is considering the use of EDMGs in the reactor at-shutdown Level 2 PRA HRA. To this end, Table E-1 provides a mapping of the individual EDMG strategies to the conditions that may prompt their use during reactor at-shutdown accident sequences.

Table E-1 Mapping of EDMG Strategies to Potential Shutdown Level 2 HRA Use

EDMG Strategy	Relevance to Reactor At-Shutdown Level 2 HRA
Extensive Damage Mitigation Guideline	Addresses situations where a loss of large area of the plant due to fire or explosion has compromised the main control room
Refueling Water Storage Tank (RWST) Makeup Using External Sources	Situations relying on injection from the RWST, for which regular RWST makeup means are unavailable
Containment Flooding Using Godwin ¹ Pump	Situations with (i) an intact containment and rising containment pressure or (ii) known high radiation levels within containment, and for which the engineered containment spray system is not restorable.
Makeup to the Condensate Storage Tank (CST)	Situations relying on injection from the CST, for which regular CST makeup means are unavailable
Fire System Strategies	General supporting function for other EDMGs strategies
Manual Operation of the Turbine-Driven Auxiliary Feedwater (TDAFW) System	Limited relevance to shutdown HRA (no or insufficient steam source during many modes of shutdown and irrelevant when the steam generator (SG) nozzle dams are installed)
Manually Depressurize SGs	Situations with rising reactor coolant system (RCS) pressure, and for which normal means of opening SG atmospheric relief valves (ARVs) are unavailable
AFW Supply Using Godwin ¹ Pump	Situations relying on secondary-side heat removal, for which regular means of feeding the SGs are unavailable
Spraying Down Release Points	Situations with known, sustained fission product releases from a specific location (e.g., the equipment hatch)
Godwin ¹ Pump Setup and Operation	General supporting function for other EDMGs
Godwin ¹ Pump and Accessories	
SFP Makeup Using Internal Sources	
SFP Makeup or Spray Using External Sources	
SFP Leakage Control Strategies	
SG Indication Measurement	Situations relying on secondary-side heat removal, for which regular means of monitoring SG level are unavailable
Reactor Vessel Head Vent Strategy	Situations with rising RCS pressure, and for which normal means of controlling RCS pressure are unavailable
Note 1: 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," "EMG pump," or "B5b pump."	

In the absence of the SAMG framework, it is reasonable to assume that the plant staff and TSC may continue to try those mitigation actions that failed to prevent core damage, which most typically would involve either forced or gravity feed into the reactor coolant system. The availability of additional time and additional plant staff may or may not warrant credit for these actions post core damage.

Following that, the above EDMGs are expected to be consulted, the EDMGs relevant to the ongoing accident would need to be identified, and the available options would need to be prioritized. Here, it is assumed that the general hierarchy of the SAMGs (i.e., Severe Challenge Guidelines (SCGs) are higher priority than Severe Accident Guidelines (SAGs)) would influence

the TSC's prioritization of the EDMGs. This would result in the following general set of priorities, from highest to lowest:

- address a large ongoing release;
- address containment pressure, if it is well above design pressure (not expected for shutdown accidents, at least for a significant period of time);
- address flammability, if a combustion could challenge containment;
- address containment pressure, if a vacuum exists;
- inject into the SGs if they are dry;
- depressurize the RCS, if it is at high pressure;
- etc.

Using this paradigm, Table E-2 shows the notional priority of upset conditions and the associated candidate actions, categorized in terms of the RCS and containment status (which are expected to be the two high-level plant states with the most influence on identifying and prioritizing actions). Table E-3 then re-casts this information to the relevant PDS and bridge tree top events, so that it can be employed in the PRA model. The PDS attributes are used to define seven categories of accident types (category I through VII), which are discussed further as part of the HFE development in Section E.4 .

Table E-2 Mapping of Candidate Action Priorities by Plant State

Condition	Candidate Action	Candidate Action Priority for Plant State			
		RCS Intact / Cont. Intact	RCS Vented / Cont. Intact	RCS Intact / Cont. Open	RCS Vented / Cont. Open
General	Continued attempts to accomplish pre-core damage injection/recovery	1	1	1	1
Evidence of core damage from insufficient decay heat removal	(If SGs available) Inject into SGs using Godwin pump	2	n/a	2	n/a
Rising RCS pressure	Open reactor head vents	3	n/a	4	n/a
Rising containment pressure	Spray containment (internally) with Godwin pump	4	2	n/a	n/a
Ongoing fission product release	- Spray containment (internally) with Godwin pump - Spray containment (externally) with Godwin pump	5	3	3	2
Secondary (supporting) strategies	- RWST makeup using external sources - Makeup to CST - Manually depressurize SGs - Diverse SG level indication	As needed to support above strategies	As needed to support above strategies	As needed to support above strategies	As needed to support above strategies

Table E-3 Re-Casting of Candidate Actions by Relevant PDS and Bridge Tree Top Events

Cat.	ACCTYPE ¹	RCS VentFlood	Containment	1 st Pri.	2 nd Pri.	3 rd Pri.
I	Transient	Not vented	Closed & Isolated	Continued attempts to accomplish pre-core damage injection / recovery	Inject into SGs	Open head vents
II			Open or unisolated			Spray internally / externally in/on containment
III		Vented or Flooded	Closed & Isolated		Spray in containment	Spray internally / externally in/on containment
IV			Open or unisolated		Spray internally / externally in/on containment	No other candidate actions identified
V	Station Blackout	All	All	No HRA credit is given		
VI	Loss Of Coolant Accidents (LOCAs)	All	Closed & Isolated	Continued attempts to accomplish pre-core damage injection / recovery	Spray in containment	Spray internally / externally in/on containment
VII			Open or unisolated		Spray internally / externally in/on containment	No other candidate actions identified
¹ This is the accident type based on plant conditions at the onset of core damage, as defined in the plant damage state binning; in particular note that LOCA does not necessarily infer a pipe rupture (it may be a diversion of water or a deliberate opening of an RCS vent path) and transient does not necessarily infer a transient initiating event (a transient initiator can lead to a feed and bleed situation while a loss-of-inventory initiator can result in an isolated RCS).						

E.3 HRA Approach

The approach taken to performing the scoping and detailed HRA here heavily leverages the at-power Level 2 PRA HRA approach discussed in (NRC, 2022a). In so doing, there are two fundamental aspects of that approach that were considered.

First, the approach was based on plant-specific observations about conduct of operations and accident response philosophy, including good awareness of the EDMGs as not only a guidance set useful for responding to accidents in general, but also as a collection of guidance excerpts useful for fulfilling various functional needs. Given this, the approach is well-suited to the reactor at-shutdown situation described in the preceding section.

Second, the at-power approach considers that the post-core-damage situation is significantly different than the pre-core damage situation. For reactor at-shutdown, a decision needed to be made as to whether it makes more sense to extend the at-power Level 2 PRA HRA approach to shutdown operations versus extending the reactor at-shutdown Level 1 PRA HRA approach to the post-core-damage situation. A judgment was made that the post-core-damage aspects were more controlling in this regard, and that extension of the at-power Level 2 PRA HRA approach was more appropriate.

Given this decision, there are several aspects of the at-power approach that warrant re-visiting and potential alteration, and this is covered in the subsequent sub-sections.

E.3.1 Timing

The at-power Level 2 PRA HRA used criteria for developing timing estimates, as follows:

Generic timings from the at-power Level 2 PRA HRA

- Guideline entry:
 - 30 minutes after conditions exist for SCG and SAG entry 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 more straight-forward local actions
 - 1 hour for local actions involving harsh environments, set up of equipment, etc.

While it did not wind up impacting the at-power Level 2 PRA HRA (because of sufficient additional time margin), the at-power Level 2 PRA HRA discussed the timing assumptions with

the reference plant staff. 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 they 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 might be an addition of 15–45 minutes to the overall times.

Adjusting for the different diagnosis paradigm in the at-shutdown situation (i.e., SAMGs being out-of-scope, EDMGs playing a larger role, and candidate action selection being more organic), and accounting for the above feedback, the revised set of timing criteria used here will be:

Generic timings for the at-shutdown Level 2 PRA HRA (used here)

- Candidate action selection:
 - 30 minutes after onset of core damage or completion of a preceding post-core-damage action
- Development of the specific approach to be taken:
 - 30 minutes if the action is a specific EDMG action; primarily associated with assessing status of relevant structures, systems and components; coordinating across engineering and operations; and getting ED buy-in
 - 1 hour if the action is not a specific EDMG action (to account for additional time needed to develop the equivalent step-by-step approach that the EDMGs would otherwise provide)
- Start of implementation:
 - 15 minutes for control room actions (to allow relaying of direction and any clarification)
 - 30 minutes for more straight-forward local actions
 - 1 hour for local actions involving harsh environments, setting up equipment, etc.

E.3.2 Screening Identification and Probability Assignment

The screening HRA approach for the at-power Level 2 HRA is discussed in Appendix D of (NRC, 2022a). It includes two initial screening criteria that are founded in the SAMG-oriented nature of the at-power approach, as follows:

- For the identified HFE, the associated SAMG is at any time the highest priority for the next 12 hours and the area where the HFE is performed is habitable.
- OR
- For the identified HFE, the associated SAMG is the second highest priority for at least 2 hours during the initial 12 hours and the area where the HFE is performed is habitable.

These criteria do not directly apply here, since the SAMGs are not in play. The criteria were important to the at-power Level 2 PRA HRA, because there were 13 possible candidate high-level actions which had periodically-changing applicability depending on accident conditions. Here, there is not a need to narrow down to a manageable number of candidate high-level actions, because the approach described previously in this report already narrows the scope to three (or fewer) potential candidate actions per scenario. For this reason, these two criteria from the at-power model are not applied here.

The at-power Level 2 PRA HRA also presents an approach to assigning screening HEPs to candidate actions based on various considerations, such as priority and applicability, DC power availability, upstream HFEs, accident-altering events, relevance to more familiar emergency operating procedure (EOP)-based actions, and EDMG relevance (on top of the inherent SAMG relevance stemming from that approaches structure). There is some analogue to priority in the at-shutdown case, given the three-level priority actions previously established in Table E.3. The criteria related to DC power availability, upstream HFEs, accident-altering events, and EOP relevance are all directly applicable here. The nature of SAMG and EDMG relevance is different here, as has been covered at length previously in this report.

Given this, the at-power screening HEP assignment criteria have been lightly modified to capture the same general intent, but factoring in the differences associated with the at-shutdown situation. This re-tooled set of criteria are presented in Table E-4. One additional modification is the elimination of the criterion related to the existence of upstream Level 2 PRA HFE failures. This criterion was used as an attempt in the at-power model to bring HEP dependency into the screening HRA, but since the intent is to subsequently apply the detailed HRA method and perform an HEP dependency review, it is extraneous and potentially confusing here.

Table E-4 At-Shutdown Screening HEP Assignment Criteria

Screening HEP Value	Screening HEP Criteria
1.0 (100% chance action is NOT taken)	If DC power is unavailable during the period of diagnosis or execution
0.9 (90% chance action is NOT taken)	It is never the first or second highest priority OR An accident-altering event ¹ occurs during the implementation period (i.e., occurs during setup, occurs during continued local action, or occurs in a location that would impact equipment for an action that is on “auto-pilot”).
0.1 (10% chance action is NOT taken)	It is very similar to an EOP action in terms of the action’s function ² AND It is the highest perceived priority, and not concurrent with an accident-altering event (as clarified above) AND During the above time period there is no habitability or survivability concern
0.5	If not covered by one of the categories above
¹ This term is defined here to be hot leg creep rupture, vessel failure, containment failure, a combustion event, or failure of a surrounding structure. ² For instance, injecting in to the SGs is a function the plant staff are familiar with based on training/operation (because feeding the SGs is part of normal operation and a key means of preventing core damage); meanwhile, venting containment or de-energizing potential ignition sources to control hydrogen flammability is a less familiar concept in the context of the EOPs, and this criterion represents the belief that the plant staff will be more hesitant to take (and/or less likely to be successful at implementing) such an action.	

E.3.3 Feasibility and Habitability Screening

These aspects are unchanged from the at-power approach, as the concepts of ensuring that timing is sufficient and areas are accessible is equally applicable here.

E.3.4 Diagnosis Decision Tree

Likewise, the majority of the diagnosis decision tree aspects apply here, with the following minor adjustments:

- Entry condition will also consider the availability of indication from the perspective of shutdown operations, where removal of equipment from operation and changing plant configurations affect what indication is relevant and/or available; see the considerations below for more details on available indications during shutdown.
- The “Procedural Support” top event is tweaked to account for the role of general accident management (SAMG) training vs. the SAMGs themselves.

- For the 2nd and 3rd top events, their invocation is approached somewhat more pessimistically given the lack of the SAMG structure and the 'off-normal' outage situation (equipment out-of-service, etc.).

In the corresponding at-power Level 2 PRA HRA approach (discussed in Appendix D of NRC [2022a]), some over-arching information and viewpoints are presented regarding the availability of information and trends in the context of the at-power post-core-damage accident management. Much of this discussion remains applicable to the at-shutdown situation, despite the differences discussed earlier in this document. But there are a few aspects of that writeup that warrant re-visiting here:

- Site releases – The situation for indication availability and monitoring is generally analogous to the at-power situation (i.e., little trending prior to core damage and relatively sparse indication).
- Containment pressure – Many aspects are analogous to the at-power situation (good indication and routine monitoring). That said, if containment is open at the time of the accident, containment pressure will only become relevant if the containment is successfully re-closed. In addition, there are a much broader set of circumstances in which the RCS can directly dump energy into containment (e.g., RCS vented; vessel head off) relative to the at-power situation.
- Containment hydrogen – Again, there will be a broader set of situations in which hydrogen in the RCS can transit quickly to containment. Meanwhile, the abnormal operating procedures (AOPs) that are relevant to shutdown events will not address the design-basis aspects of hydrogen monitoring addressed in the EOPs, and the SAMGs (by virtue of not being applicable) will not provide default curves in the absence of indication for shutdown conditions, so indication and trending prior to core damage will be generally poor.
- Steam generator level – The situation here is generally analogous to at-power conditions, for plant operating states where the steam generators are available (i.e., good indication and monitoring).
- RCS pressure – The situation here is generally analogous to at-power conditions, for plant operating states prior to RCS venting (i.e., good indication and monitoring).
- RCS temperature – Means of measuring RCS temperature will be changing during the outage (see more below), but otherwise, indication and awareness should be generally good.
- Containment water level – With the sump covered for the majority of the outage, this parameter will have less relevance than the at-power situation.
- RCS level: In general, the reactor vessel level indication system (RVLIS),²⁹ narrow range RCS level, wide range RCS level, and pressurizer level will remain available (given DC power and no instrument failures) for most plant operating states, though the AOPs will

²⁹ Note that like the at-power Level 2 PRA, in interpreting the MELCOR results with regard to RVLIS, the collapsed core level (as opposed to the un-collapsed core level or the collapsed or un-collapsed downcomer level) is used.

have the operators focus on the subset of these instruments that are most relevant (e.g., narrow range RCS level in the case of accidents at mid-loop). A key exception is with the vessel head off (Mode 6), where RVLIS will not be available, and pressurizer level will be via the cold calibration level indication. Reactor cavity and containment sump levels may be monitored, but only to ensure that they are not increasing (that being a sign of unexpected leakage). The RCS sight glass is also a source of level indication when relevant and can be viewed from a video monitor outside of the containment.

- RCS temperature: For some plant operating states (and particularly those associated with refueling), some or all core-exit thermocouples may be out-of-service. In these instances, hot leg wide-range temperature instruments become the primary source of information. RHR pump discharge temperature indication is also a possible source, but not likely to be relevant if core damage is ensuing.

The at-power diagnosis decision tree is re-produced in Figure E-1. The same diagnosis decision tree is used for the reactor at-shutdown Level 2 PRA HRA.

E.3.5 Execution Decision Tree

There was no need to make shutdown-specific alterations to this part of the method, as all the considerations apply equally here.

The at-power execution tree is re-produced in Figure E-2.

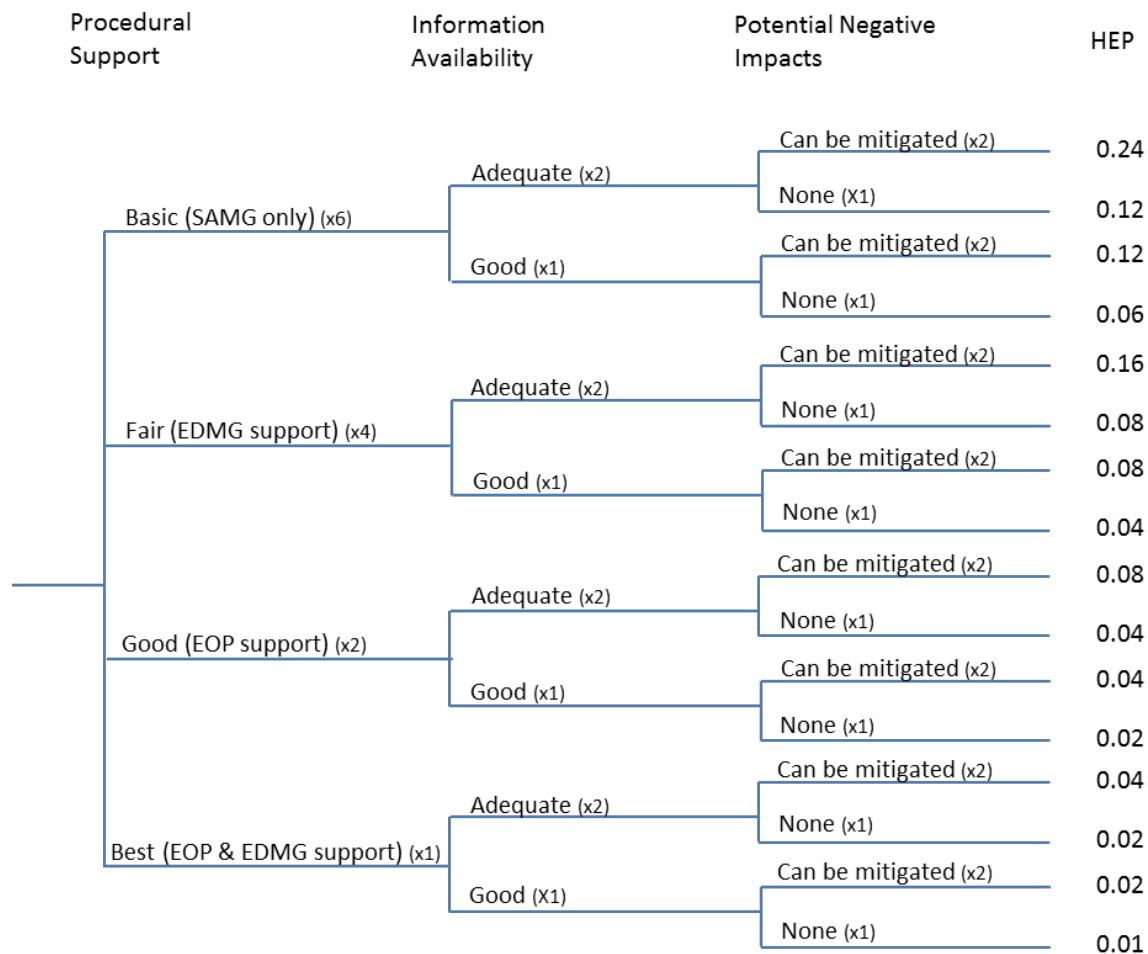


Figure E-1 Decision Tree for Diagnosis

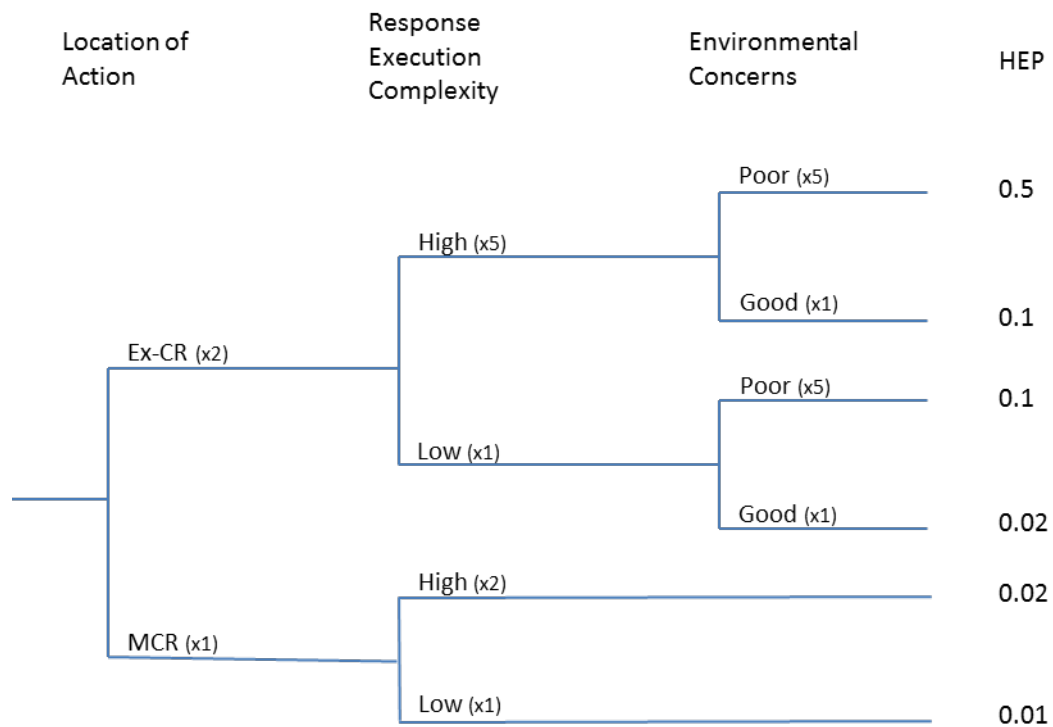


Figure E-2 Decision Tree for Execution

E.4 HFE Development and HEP Quantification

The different accident scenario situations captured in Table E-3 can be mapped to the representative sequences modeled in MELCOR (described in Section 2.1.5 and Section 2.3.3), and this mapping is shown in Table E-4. Of particular note is that four basic categories of scenarios fall out from this mapping. These are:

- Category I - Transient accident type,³⁰ with the RPV not vented, and containment closed and isolated – Sequence SD08;
- Category II - Transient accident type, with the RPV not vented, and containment open – Sequence SD06;
- Category V - Station blackout sequences (where post core damage HRA credit is not considered) – Sequences SD03, SD04, SD05, SD07, SD09, and SD13; and
- Category VI - LOCA accident type with a closed and isolated containment at the time of core damage – Sequences SD01, SD02, SD10, SD11, and SD12.

There are three categories of scenarios envisioned in Table E-3 not captured, and these are:

- Category III - Transient accident type, with the RPV vented or flooded, and containment closed and isolated – the candidate actions and priorities considered by this category are identical to those from Category VI and the HRA findings can be applied (with caution) to this category;
- Category IV – Transient accident type, with the RPV vented or flooded, and containment open or un-isolated – the candidate actions and priorities considered by this category are similar to those from Category VI and the HRA findings can be extended (with caution) to this category;
- Category VII – LOCA accident type with an open or un-isolated containment – the candidate actions and priorities considered by this category are similar to those from Category VI and the HRA findings can be extended (with caution) to this category.

The lack of ideal coverage of the representative sequences for this part of the HRA is a function of the representative sequences being based on plant damage state binning (pre-core damage) versus this part of the HRA focusing on severe accident management (post core damage).

The representative MELCOR sequences are mapped to corresponding accident scenarios for identifying modeled post-core-damage actions. Table E-5 summarizes the accident scenarios and their associated candidate actions. The following subsections describe each of the representative accident scenarios.

³⁰ This is the accident type based on plant conditions at the onset of core damage, as defined in the plant damage state binning; in particular note that LOCA does not necessarily infer a pipe rupture (it may be a diversion of water or a deliberate opening of an RCS vent path) and transient does not necessarily infer a transient initiating event (a transient initiator can lead to a feed and bleed situation while a loss-of-inventory initiator can result in an isolated RCS).

Table E-5 Mapping of MELCOR Representative Sequences to Plant Condition-Based Candidate Action Priority Scheme

Scen. #	Plant/Accident Conditions	1 st Priority	2 nd Priority	3 rd Priority
1 (SD01)	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
2 ¹ (SD02)	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
3 (SD03)	Station blackout, containment un-isolated	No credit for post-core-damage human actions is considered		
4 ¹ (SD04)	Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
5 (SD05)	Station blackout, containment open	No credit for post-core-damage human actions is considered		
6 ^{1, 2} (SD06)	Transient accident type, RPV not vented, containment open	Continued attempts to accomplish pre-core damage injection/recovery	Inject into SGs	Spray int/ext containment
7 (SD07)	Station blackout, containment open	No credit for post-core-damage human actions is considered		
8 (SD08)	Transient accident type, RPV not vented, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Inject into SGs	Open head vents
9 (SD09)	Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
10 (SD10)	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
11 (SD11)	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
12 (SD12)	LOCA accident type, containment closed & isolated	Continued attempts to accomplish pre-core damage injection/recovery	Spray in containment	Spray int/ext containment
13 (SD13)	Station blackout, containment open	No credit for post-core-damage human actions is considered		

Table E-5 Mapping of MELCOR Representative Sequences to Plant Condition-Based Candidate Action Priority Scheme (cont.)

Scen. #	Plant/Accident Conditions	1 st Priority	2 nd Priority	3 rd Priority
¹ These scenarios did not actually proceed to core damage in the representative sequence analysis, for reasons related to either (a) details of their boundary condition specification (e.g., that a valve would not fail despite cycling an inordinate number of times or (b) simplifications in the Level 1 PRA accident sequence modeling (e.g., outage symmetry despite decay heat differences, neglect of SG cooling when SGs are in wet layup). ² The sensitivity calculation shown in Section 2.3.3 was considered as an alternative that could be used to generate HFEs for this scenario, since it proceeds to core damage. However, the reason it proceeds to core damage is the consideration of a relief valve failure which would actually make the simulated scenario a LOCA (rather than a Transient) in the plant damage state ACCTYPE categorization. The transient-oriented HFEs, most notably injecting into the SGs would thus not be the appropriate HFEs for this situation.				

E.4.1 Scenario #1 Scenario Information

Scenario #1 is a LOCA type accident in POS 6 with the containment closed and isolated at the time of core damage. Table E-6 shows the general accident scenario timeline. Table E-7 gives the general survivability and habitability considerations.

Table E-6 Scenario #1 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	1	
Plant Operating State / Tech Spec Mode	6 / 5	"Hot" mid-loop; SG dams are installed; all 3 pressurizer safety valves are held open or removed
Initiating event	LoRHR	(Loss of Residual Heat Removal)
Level 1 successes/failures	→	SI and charging inject from t=15 minutes until RWST depletion at ~8 hours; operators close all containment penetrations but are unable to restore RHR and containment coolers
Containment status	→	Initially closed and un-isolated; isolation occurs prior to core damage
Contain. systems	→	Containment sprays and fan coolers are unavailable
AC power status	Avail.	
DC power status	Avail.	
Invocation of Emergency Classification in AOPs	< 1	Loss of Residual Heat Removal AOP directs initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 5	Conditions will apply starting at ~9 hours; based on the simulation results, RCS temperature conditions briefly apply around t=30 minutes, and temperatures are close to conditions thereafter; it is assumed that an ALERT is declared at the mid-point of this time window, based on ED discretion
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	11	RPV level at top of active fuel with containment closure established
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	11.5	Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig NOTE: The containment level rise is interpreted to be "unexplained" in the context of the first item; separately, containment closure is expected to be with 4 bolts (thus the containment closure cited above, as opposed to the alternate Tech Spec intact state).
Time of onset of core damage ²	15	
Time of vessel breach	24	
Time of combustion event(s)	~24 & 43	These were estimated from the MELCOR plots, so they are approximate numbers
Time of containment overpressure liner tear	168	
Time of basemat melt-through	>168	
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific Emergency Action Level Matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

Table E-7 Scenario #1 General Survivability and Habitability Perspective

Timing (Top of the Hour)	General I&C and Equip. Survivability Concerns	Habitability Concerns	Accident-Altering Event ¹
Onset of core damage			
16	With a vented RCS, and containment closure (as opposed to Tech Spec leakage conditions), and with containment pressure steadily rising, areas adjacent to containment will experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion	For the same reasons as described to the left, habitability near containment penetrations may be impeded; no wide-spread habitability concern is expected, though cumulative noble gas releases to the environment are approaching 0.1% during this timeframe, so there may be a need for recurring dose surveys associated with onsite actions	
17			
18			
19			
20			
21			
22			
23			
24	In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach		Vessel breach, combustion
25			
26			
27			
28			
29			
30			
¹ This refers to a combustion event, hot leg creep rupture, vessel breach, containment failure, or failure of a surrounding structure.			

Scenario description:

The accident is initiated by a loss of residual heat removal while near the beginning of POS 6 (mid-loop operations prior to re-fueling) of a re-fueling outage. The minimum duration from reactor shutdown to the start of this POS is 82 hours after reactor shutdown (NRC, 2025, Table 2.5-1). The accident is assumed to occur early in this POS, and it is assumed that the steam generator nozzle dams have been put in place (and no feedwater to the steam generators is considered). The RCS is completely depressurized by means of having held open or removed all three pressurizer safety relief valves, and coolant temperature is approximately 110 °F. Water level in the RCS is at mid-loop level, which is about 1 foot above the hot leg centerline.

Upon loss of RHR, operators enter the AOP associated with loss of RHR. In this PRA sequence, operators are unable to restore RHR to service (either by addressing the problem that stopped the running train or returning the opposite train to operation). Subsequently in the AOP, operators are directed to isolate containment, restore containment cooling, and begin injection using ECCS. In this PRA sequence, these actions constitute a mixture of successes and failures:

- Operators succeed in manually actuating forced injection via charging and safety injection (SI) with a notional action time of 15 minutes from the start of the accident and succeed in throttling injection to maintain a minimum pressurizer level and a maximum core temperature.
- Operators succeed in isolating open containment penetrations, and this notionally occurs a few hours into the event.

- Operators fail to restore containment cooling to service prior to core damage.
- Operators fail to perform RWST refill and after the RWST is emptied, the sequence proceeds to core uncovering since the sump is unavailable (covered) and there is no source of water left for injection.

The failure to refill the RWST is currently an assumed failure in the Level 1 PRA model. The action is modeled in the Level 1 PRA event trees but is set to fail (TRUE) due to broad concerns about makeup flow rate capacity limitations and lack of proceduralized direction. Separately, containment sprays are unavailable (out-of-service), and the relevant AOPs will not direct their restoration or use. Thus, at the onset of core damage, the systems that may warrant re-consideration for restoration/recovery in the Level 2 PRA HRA are the RWST (for ECCS injection), containment coolers, and containment sprays. Since the context of these failures may be important, the table below provides some additional information on their failure contributions.

Table E-8 Scenario #1 Additional Information on Level 1 Failures/Unavailabilities

	Diagnosis	Execution
RWST refill	Not explicitly modeled; concerns exist about procedural direction	Not explicitly modeled; concerns exist about makeup capacity
Containment sprays	Not explicitly modeled; concerns exist about procedural direction and its shared use of RWST inventory for ECCS injection	Not explicitly modeled; concerns exist about RWST inventory and any ongoing maintenance at the time of the accident
Containment coolers	For POS 6, the pre-core damage diagnosis HEP is more than an order-of-magnitude lower than the execution HEP	For POS 6, the pre-core damage execution HEP dominates (though it's still fairly low), and has negative performance shaping factors (PSFs) for stress, complexity, experience/training, and procedures. The only positive PSF is available time.

As just discussed, there are three capabilities that may be of heightened interest at this point in the accident and may have been the subject of previous restoration attempts; these are RWST refill, containment sprays, and containment coolers. Replenishing the RWST (with subsequent RCS injection) would significantly improve the potential for arresting core damage, restoring containment sprays would promote fission product scrubbing (particularly since the RCS is already vented to the containment atmosphere), and restoring containment coolers represents a long-term benefit to containment integrity. There will be competing interests in this decision-making. For instance, SAMG training (though not directly applicable here) emphasizes a re-focusing on the containment barrier, but slowly increasing containment pressure and low RCS pressure will suggest that this is not an immediate concern. As a second example, fission product releases will be small early in core degradation (because of the intact containment), but operators will know that the containment closure situation (most notably the hatch being closed with only four bolts, and one or more containment penetrations being closed under emergency conditions) will result in higher containment leakage than a typical at-power accident situation. In the end, any of these actions is a viable first priority, and the decision-making will likely be

highly influenced by details of the system status that are below the resolution of the PRA/HRA. Assumptions are made that:

- Restoration of containment sprays is not the action modeled here, because (a) it was not being pursued prior to core damage and (b) it is the subject of subsequent candidate actions; and
- Operators will preferentially pursue RWST refill and subsequent ECCS injection, because the general SAMG training will cultivate a hierarchy where, in the absence of an immediate challenge to either containment or the steam generator tubes, and in the absence of an ongoing elevated fission product release, depressurizing and injecting into the RCS is a high priority.

To complete the scenario description, and as previously recorded, core damage starts at approximately 15 hours, with vessel breach and a large combustion occurring approximately 9 hours later. No other “accident-altering” event occurs until much later (a second large combustion).

So, with this background established, the screening HRA approach can be applied to the three candidate actions (that are taken from Table E-4, considering the additional refinement presented above). This is shown in Table E-9. After identifying these screening HEP values, a more detailed HRA approach is developed for these three candidate actions, as described in Sections E.4.1.1, E.4.1.2, and E.4.1.3.

Table E-9 Scenario #1 Candidate Action Screening HRA

Action	Viability?	Screening HEP	Basis for Screening HEP
Refill RWST and restore RCS injection, to arrest fuel damage	As of this screening step, these actions appear viable	0.1	This is very similar to an EOP action, is the highest perceived priority and not concurrent with an accident-altering event, and there are no obvious habitability or survivability concerns.
Spray within containment, to combat rising containment pressure		0.5	The other criteria (for HEP = 0.1, 0.9 or 1.0) do not apply.
Spray within containment or spray external release point, to reduce fission product releases		0.9	It is never the 1 st or 2 nd priority.

E.4.1.1 Scenario #1, Candidate Action #1, Continued attempts to accomplish pre-core damage injection/recovery

HFE Definition:

Having diagnosed the onset of core damage (i.e., ongoing heatup and degradation of the fuel in the RPV), and in concert with the TSC which was activated well before this time (notionally 10

hours prior), operators fail to refill the RWST and restore RCS injection via charging and/or safety injection pumps.³¹

Viable Actions:

The details of what it means to restore RWST inventory are again dependent on details that are below the resolution of the PRA/HRA. As a representative situation, it will be assumed that the failure to replenish the RWST prior to core damage was hampered by diagnosis (due to the poor procedural basis and other decision-making complexities), rather than any mechanical issues (which is supported notionally by the successful use of the RWST in the PRA sequence, prior to depletion).

The reference plant's SAMG guidance document provides a list of potential RWST makeup sources, as well as a list of conditions necessary to inject using SI or charging pumps. It is invoked here as a reference for understanding the plant capabilities better, as opposed to a procedural basis for taking action (since accidents at-shutdown are specifically out-of-scope for the circa 2012 plant SAMGs). The estimated flow rate for most sources is listed as "based on lineup," but several sources have the needed flow rate to match minimal RCS injection targets during a core damage accident.

Meanwhile, the EDMGs include a strategy to refill the RWST from the Fire Water Storage Tanks or the Demineralized Water Storage Tank, using a trailer-mounted diesel-driven EDMG (a.k.a., Godwin or B.5.b or EMG) pump.

Of the listed potential makeup sources, the fire water storage tank and the EDMG pump appear to provide the best combination of adequate available information, more-than-adequate flow rate, and the best long-term capability. Some consideration would be given to the use of borated versus un-borated sources, but all sources are considered viable in the midst of an ongoing accident. Since the above combination uses on-site equipment, and is contained in the EDMGs, it is the capability selected for modeling in the HRA. Some qualitative consideration could be given to the possibility that any of the available makeup sources may be used as a bridge in the case of delays in deployment.

To accomplish the action (RWST refill using the EDMG pump), plant personnel must:

- Assess the condition of the Fire Protection System and the status of Fire Fighting Actions or Plans
- Retrieve the portable pumps and associated tools from the warehouse location (outside the protected area, on the owner-controlled area, roughly a 5-minute drive from the power block)
- Connect the suction side of the EDMG pump to one of two fire water storage tanks (FWST) or demineralized water storage tank (DWST), located in various locations in the yard within the protected area

³¹ Since the RCS pressure will be low, any of the four relevant trains of SI/charging is capable of delivering the minimum needed flow.

- Install the EDMG pump-to-RWST adapter from the EDMG toolbox
- Connect the injection-side fire hose and adapter to the quick-disconnect connection point downstream of a specified valve, which is located in the RWST moat (this includes cutting the lock/chain since these are locked-closed valves)
- Open additional valves in the RWST piping and the FWST/DWST piping
- Start and operate the EDMG pump
- Monitor RWST level

Once minimum RWST level has been restored, it is assumed that subsequent actions to inject using Safety Injection and Charging can be taken from the main control room.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as defeating security barriers and acquiring the tow vehicle. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~9 hours). Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the "yard" within the Protected Area, or in buildings set off from the power block. As such, no specific radiological habitability issues are anticipated. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is EOP guidance that spells out the same action using other means, there is EDMG support (which is what is being invoked), and there is general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the "Best" branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew is likely aware of the status of the RWST and the injection trains by virtue of their role earlier in the accident sequence. Conversely, there are a number of potential makeup sources that have to be evaluated, and some will require sending out personnel or off-site communications to confirm their availability. This will complicate the decision-making process in the TSC, given the lack of the more structured process when the SAMGs are in play (and the much more structured process when the EOPs/AOPs were being used).

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that injecting into a degraded core has particular complications (e.g., increased hydrogen production, potential of shattering fuel), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP: The resulting HEP for diagnosis from the decision tree is 0.04.

Quantification of Execution:

Entry Conditions: All of the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and the operations support center as needed. The personnel will largely be following the EDMG procedure for RWST refill, and the needed steps for resuming injection can be developed by the TSC using existing procedures for the same action (namely the EOPs).

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-CR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are no particular radiological habitability or hazard-related concerns. Therefore, the “Good” branch is taken.

Execution HEP: The resulting HEP for execution from the decision tree is 0.1.

Total HEP:

Total HEP for Scen. #1, Action #1 = $0.04 + 0.1 - (0.04) \cdot (0.1) = 0.14 - 0.004 = \sim 0.14$.

E.4.1.2 Scenario #1, Candidate Action #2, Spray in containment

HFE Definition:

Having either succeeded or failed at restoring RCS injection, and in concert with the TSC, which was activated well before, operators fail to spray within containment using the EDMG strategy.

Viable Actions:

Here, the prevalent question is whether operators attempt to restore the installed containment spray system or pursue the EDMG strategy (which uses the portable pump previously discussed in the prior modeled action). In reality, staffing and other resources permitting, they would likely pursue both. Here, an assumption must be made as to which of these is more viable, or which receives the focused attention if resources are over-constrained. Restoring the installed containment spray system is clearly the preferred approach for situations where this is readily achievable, because it is both an engineered system and because it has a much higher capacity. However, there are three factors going against its inclusion here:

1. For some situations the amount of ongoing maintenance at the time of the event will prohibit restoration during the event. For instance, one train might be in a test configuration that would require containment entry to address, while the other train may have a support system (such as nuclear service cooling water) in a state of significant disassembly.
2. The installed system relies on the RWST as its injection source, and being a high-capacity system, it will deplete the RWST much more rapidly than the RWST refill sources discussed in the preceding section can support.
3. The at-shutdown Level 2 PRA HRA approach is based around a focus on the EDMGs as the surrogate accident management framework in the absence of SAMG applicability, and the EDMGs are not written so as to consider the installed containment spray systems.

Given this, and as a somewhat subjective assumption, the EDMG strategy will be the containment spray strategy modeled in the HRA. This carries the key assumption that the EDMG pump is not being used for other actions (including the previously-covered RWST refill action), or that the second EDMG pump and a secondary water source (e.g., the DWST) are being used.

In terms of the actions to be taken, the actions are fairly similar to those described previously for RWST refill using the EDMG pump, with one major deviation. In this case, the injection hoses are being routed into the auxiliary building, to a test connection located adjacent to the containment wall. The preferred path, based on walkdowns with plant operational staff, is through one of the main steam valve rooms.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as acquiring the tow vehicle and performing dose surveys. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~9 hours), even given the notional 2-hour timeframe associated with the preceding candidate action. Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the "yard" within the Protected Area, buildings set off from the power block, one of the two main steam valve rooms, the lower elevations of the steam tunnel area, and connection points adjacent to containment. Radiological habitability issues are possible near the connection points, that may require limited stay time and donning of protective gear. Since there is no specific containment impairment, it is expected that these radiological habitability issues may affect the implementation of the strategy, but that they will not prevent it. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is no EOP guidance that spells out the equivalent action, there is EDMG support (which is what is being invoked), and there is no general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the "Fair" branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be "Adequate" (up branch). The crew may be aware of the status of the EDMG equipment from attempting the prior action to refill the RWST. Information about what is taking place within containment, and whether spraying containment may cause a combustion event by de-inerting an inerted containment

atmosphere may be impaired, if the necessary equipment and indication to do this assessment were part of the ongoing maintenance scope.

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that spraying containment has particular complications (e.g., de-inerting containment), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.16.

Quantification of Execution:

Entry Conditions: All the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and the operations support center as needed. The personnel will largely be following the EDMG procedure for spraying the containment atmosphere, and the needed guidance is available in the EDMGs.

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-CR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are general radiological habitability concerns that may require limitation in stay time or donning of protective clothing, but there are no specific habitability concerns. There are no hazard-related concerns. Therefore, the "Good" branch is taken.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1.

Total HEP:

Total HEP for Scen. #1, Action #2 = $0.16 + 0.1 - (0.16)(0.1) = 0.26 - 0.016 = \sim 0.24$.

NOTE: This HEP is lower than that for the corresponding action in the at-power post-core-damage HRA (NRC, 2022a), which may appear counter-intuitive. However, the action only manifested in the at-power HRA as a post-vessel breach action, where radiological habitability concerns would be greater. This led to a more punitive choice for execution decision tree branch #3 (Environmental concerns). More specifically, in the shutdown situation, the general amount of short-lived radionuclides is smaller, and the timing of this action precedes vessel breach. The vented RCS situation (specifically through the removed SRVs) represents an intermediate situation (relative to the prior-to and post- vessel breach situations in the at-power HRA) with regard to the path of fission products exiting the degraded fuel. Overall, the premise of lower environmental concerns was judged appropriate.

E.4.1.3 Scenario #1, Candidate Action #3, Spray in containment or spray the release point

HFE Definition:

Having either succeeded or failed at spraying in containment using the EDMG strategy (for the purpose of controlling containment conditions), and in concert with the TSC, which was activated well before, operators fail to spray an external release point using the EDMG guidance. [Spraying in the containment is not considered here, since it was already considered in the previous candidate action.]

Viable Actions:

This candidate action is not further developed here but rather is left as a placeholder. There are three key reasons for this:

1. In the reference scenario, containment does not fail for almost seven days.
2. Containment failure due to long-term overpressure, is likely to lead to a situation where a specific release point is not identifiable during the accident response.
3. Previous work done under other projects has suggested that the spray flux and fission product in-spray residence time is unlikely to lead to significant fission product capturing, for the spray flow rates and geometries of relevance here.

Feasibility Assessment:

Not applicable (N/A)

Determination of Habitability:

N/A

Quantification of Diagnosis:

N/A

Quantification of Execution:

N/A

Total HEP:

N/A

E.4.2 Scenario #2 Basic Scenario Information

Scenario #2 is a LOCA type accident in POS 10 with the containment closed and isolated at the time of core damage. Table E-10 shows the general accident scenario timeline.

Table E-10 Scenario #2 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	2	
Plant Operating State / Tech Spec Mode	10 / 5	"Cold" mid-loop; SG dams are installed; all 3 pressurizer safety valves are held open or removed
Initiating event	LoRHR	(Loss of residual heat removal)
Level 1 successes/failures	→	Gravity feed from the RWST starts at t=30 minutes until RWST depletion (which is not predicted within 7 days); operators are unable to restore RHR but do isolate containment and restore containment coolers
Containment status	→	Initially closed and un-isolated; isolation occurs prior to core damage
Contain. systems	→	Containment sprays are unavailable; 1 train of coolers is available
AC power status	Avail.	
DC power status	Avail.	
Invocation of Emergency Classification in AOPs	< 1	Loss of residual heat removal AOP directs initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 20	RCS temperature > 200 °F for >20 minutes, containment closed, and RCS integrity not established. Condition will apply intermittently between ~5 to 34 hours; it is assumed that an ALERT is declared at the mid-point of this time window, based on ED discretion
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	n/a	Site area emergency conditions are never met.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	n/a	General emergency conditions are never met.
Time of onset of core damage ²	>168	Core damage is not predicted within 7 days for this calculation.
Time of vessel breach	>168	
Time of combustion event(s)	n/a	
Time of containment overpressure liner tear	n/a	
Time of basemat melt-through	n/a	
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.3 Scenario #3 Basic Scenario Information

Scenario #3 is an SBO type accident in POS 8E with the containment un-isolated at the time of core damage. Though, the containment equipment hatch is successfully closed at the time of core damage. Table E-11 shows the general accident scenario timeline.

Table E-11 Scenario #3 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	3	
Plant Operating State / Tech Spec Mode	8E / 6	Refueling, prior to fuel movement with the transfer tube assumed closed, SG dams installed, vessel head off, refueling cavity flooded
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators successfully close the containment hatch and personnel airlock; gravity feed from the RWST starts when the bulk water temperature approaches 100 °C; operators are unable to restore power and fail to isolate all containment penetrations
Containment status	→	Initially open and un-isolated; closure occurs early, but isolation does not occur
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavail.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of Emergency Classification in AOPs	< 1	Loss of Electrical Systems and Loss of Residual Heat Removal AOPs direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite ac sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	E.g., 26	Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 26 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under Emergency Discretion.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., 33	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE or GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared upon RWST depletion, at 33 hours. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
Time of onset of core damage ²	125	
Time of vessel breach	137	
Time of combustion event(s)	n/a	No significant combustion events occur

Table E-11 Scenario #3 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of containment overpressure liner tear	n/a	The containment isolation failure prevents long-term overpressure failure.
Time of basemat melt-through	n/a	Cavity erosion is occurring very slowly.
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.4 Scenario #4 Basic Scenario Information

Scenario #4 is an SBO type accident in POS 8L with the containment closed and isolated at the time of core damage. Table E-12 shows the general accident scenario timeline.

Table E-12 Scenario #4 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	4	
Plant Operating State / Tech Spec Mode	8L / 6	Refueling, following core re-load with the transfer tube assumed closed, SG dams installed, vessel head off, refueling cavity flooded
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators successfully close the containment hatch, personnel airlock and all containment penetrations; gravity feed from the RWST starts when the bulk water temperature approaches 100C; operators are unable to restore power
Containment status	→	Initially open and un-isolated; closure and isolation occur early
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavail.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of Emergency Classification in AOPs	< 1	Loss of Electrical Systems and Loss of Residual Heat Removal AOPs direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite ac sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	E.g., 91	Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 91 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100 °C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under Emergency Discretion.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., >168	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE or GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is not declared prior to the end of the simulation.
Time of onset of core damage ²	>168	Core damage is not predicted within 7 days for this calculation.
Time of vessel breach	>168	
Time of combustion event(s)	n/a	
Time of containment overpressure liner tear	n/a	Containment pressure is rising very slowly throughout the calculation.
Time of basemat melt-through	n/a	Core damage and vessel breach have not occurred as of the end of the simulation

Table E-12 Scenario #4 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix.		
² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.		

E.4.5 Scenario #5 Basic Scenario Information

Scenario #5 is an SBO type accident in POS 8E with the containment hatch open at the time of core damage. Table E-13 shows the general accident scenario timeline.

Table E-13 Scenario #5 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	5	
Plant Operating State / Tech Spec Mode	8E / 6	Refueling, prior to fuel movement with the transfer tube assumed closed, SG dams installed, vessel head off, refueling cavity flooded
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators fail to close the containment hatch prior to hazardous environmental conditions, and to inject into the RCS
Containment status	→	Initially open and un-isolated; and operators fail to successfully close containment
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavail.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of Emergency Classification in AOPs	< 1	Loss of Electrical Systems and Loss of Residual Heat Removal AOPs direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite ac sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	E.g., 6	Declaration of Site Area Emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at 6 hours, under Emergency Discretion, when significant steam would be exiting the open hatch.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., 48	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE or GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via Emergency Director discretion based on extended boiling with (i) no compensating injection and (ii) containment open. A time of 48 hours is somewhat arbitrarily chosen. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
Time of onset of core damage ²	100	
Time of vessel breach	110	
Time of combustion event(s)	n/a	No significant combustion events occur
Time of containment overpressure liner tear	n/a	The containment opening prevents long-term overpressure failure.

Table E-13 Scenario #5 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of basemat melt-through	n/a	Cavity erosion is occurring very slowly.
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.6 Scenario #6 Basic Scenario Information

Scenario #6 is a loss of heat removal transient type accident in POS 11 with the containment hatch open at the time of core damage. Table E-14 shows the general accident scenario timeline.

Table E-14 Scenario #6 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	6	
Plant Operating State / Tech Spec Mode	11 / 5	Vacuum fill of the RCS post-refueling and post-cold-mid-loop; SG dams have been removed; RCS boundary is intact but initial pressure is low; primary-side initial water level is just above the pressurizer inlet and the SG tube sheet
Initiating event	LoRHR	(Loss of Residual Heat Removal)
Level 1 successes/failures	→	SI and charging inject from t=4 hours until RWST depletion at ~38 hours; operators are unable to restore RHR and containment coolers; operators fail to close the containment hatch prior to hazardous environmental conditions
Containment status	→	The containment hatch is open at all times; the personnel airlock and all penetrations are closed
Contain. systems	→	Containment sprays and fan coolers are unavailable
AC power status	Avail.	
DC power status	Avail.	
Invocation of Emergency Classification in AOPs	< 1	Loss of Residual Heat Removal AOP directs initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 8	RCS pressure increasing greater than 10 psig
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	n/a	Site area emergency conditions are never met.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	n/a	General emergency conditions are never met.
Time of onset of core damage ²	>168	Core damage is not predicted within 7 days for this calculation.
Time of vessel breach	>168	
Time of combustion event(s)	n/a	
Time of containment overpressure liner tear	n/a	
Time of basemat melt-through	n/a	
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.7 Scenario #7 Basic Scenario Information

Scenario #7 is an SBO type accident in POS 5A with the containment hatch open at the time of core damage. Table E-15 shows the general accident scenario timeline.

Table E-15 Scenario #7 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	7	
Plant Operating State / Tech Spec Mode	5A / 5	Taking the system solid for de-gassing; initial RCS pressure is 350 psig with RCS temperature of 180 °F
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators are unable to restore AC power and fail to close the containment hatch prior to hazardous environmental conditions
Containment status	→	Initially open and un-isolated; isolation occurs early, but closure does not occur
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavail.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of Emergency Classification in AOPs	< 1	Loss of Electrical Systems and Loss of Residual Heat Removal AOPs direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite AC sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	E.g., 10	Declaration of site area emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at 10 hours, under Emergency Discretion, when significant steam would be exiting the open hatch.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., 15	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE or GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via Emergency Director discretion based on extended boiling with (i) no compensating injection and (ii) containment open. At this time, core damage is starting, which will also lead to increasing radiation doses near the open hatch. A time of 15 hours is somewhat arbitrarily chosen. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
Time of onset of core damage ²	15	
Time of vessel breach	21	
Time of combustion event(s)	n/a	No significant combustion events occur
Time of containment overpressure liner tear	n/a	The containment opening prevents long-term overpressure failure.

Table E-15 Scenario #7 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of basemat melt-through	n/a	Cavity erosion is occurring very slowly.
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.8 Scenario #8 Basic Scenario Information

Scenario #8 is a loss of heat removal transient type accident in POS 3 with the containment closed and isolated at the time of core damage. Table E-16 shows the general accident scenario timeline. Table E-17 gives the general survivability and habitability considerations.

Table E-16 Scenario #8 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	8	
Plant Operating State / Tech Spec Mode	3 / 4	Cooldown using RHR with the RCS at 270 °F; SGs are available w/ level at 95% wide range indication, but feedwater is not; RCS level is at 80% in the pressurizer; no reactor coolant pumps are operating in the simulation, but they would be in actuality
Initiating event	LoRHR	(Loss of residual heat removal)
Level 1 successes/failures	→	Operators fail to recover the failed train of RHR, fail to align the standby train, and fail to control SG level
Containment status	→	Initially closed and isolated
Contain. systems	→	Containment sprays and fan coolers are available (and operate)
AC power status	Avail.	
DC power status	Avail.	
Invocation of Emergency Classification in AOPs	< 1	Loss of Residual Heat Removal AOP directs initiation of emergency classification upon being unable to restore RHR in a timely manner.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 20	Potential Loss of the Fuel Clad Barrier based on Heat Sink due to SG narrow range level < 10%
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	~ 20	Potential Loss of the Fuel Clad Barrier and the RCS Barrier based on Heat Sink due to SG narrow range level < 10%
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	~ 28	Loss of Fuel Clad Barrier based on Core Cooling due to reactor vessel level < 41% and CETC > 711 °F; Loss of RCS Barrier based on high containment radiation readings due to ongoing fuel damage and cycling PORV; Potential loss of Containment Barrier based on Core Cooling for > 15 minutes with RPV level < 63%]
Time of onset of core damage ²	28	
Time of hot leg nozzle creep failure	30	
Time of vessel breach	33	
Time of combustion event(s)	Multiple	A series of 6 burns is predicted in the simulation during the time period between 30 to ~55 hours
Time of containment overpressure liner tear	Never	Containment fan coolers keep containment pressure low
Time of basemat melt-through	>168	
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.		

Table E-17 Scenario #8 General Survivability and Habitability Perspective

Timing (Top of the Hour)	General I&C and Equip. Survivability Concerns	Habitability Concerns	Accident-Altering Event ¹
Onset of core damage			
29	The intermittent combustions inside containment have the potential to damage equipment or instrumentation.	Only inside containment.	
30			Hot leg nozzle creep rupture; combustion
31			
32			
33	With containment pressure being retained low, and containment being intact, radiation levels outside containment would be expected to be reasonably low. The intermittent combustions inside containment have the potential to damage equipment.	With containment pressure being retained low, and containment being intact, radiation levels outside containment would be expected to be reasonably low. No known habitability conditions exist outside of containment.	Vessel breach
34			
35			
36			Combustion
37			
38			
39			
40			
41			
42			
43			Combustion
¹ This refers to a combustion event, hot leg creep rupture, vessel breach, containment failure, or failure of a surrounding structure.			

Scenario description:

The accident is initiated by a loss of residual heat removal while in POS 3 (RCS cooldown to 200 °F using RHR) of a re-fueling outage. The initiating event nominally occurs 5 hours after the reactor is tripped following the coast down to the refueling outage. At this point in time decay heat will be relatively high (21.2 MW) relative to other reactor at-shutdown accident scenarios. The secondary-side of the steam generators will be flooded (wide range level of 95%), but in this scenario there is no feedwater available. The RCS is at 270 °F (which is an intermediate value relative to the range of POS 3 temperatures). In reality, two RCPs would be running, but this is not modeled in the MELCOR analysis. In addition, the relief valves are allowed to cycle without the possibility of failure, which they are predicted to do an extremely large number of times. Both of these modeling assumptions could affect accident timing. In particular, if a PORV failed open, this might prevent the hot leg creep rupture that is predicted, which would in turn have some impact on subsequent event timings. Otherwise, the relative timing of events following the onset of core damage would be expected to be similar. Containment is closed and isolated, having nominally not been yet entered for outage activities. Containment fan coolers and containment sprays are available, as is the containment sump, as none of these have yet been taken out-of-service.

Upon transient conditions occurring, operators diagnose the loss of RHR and enter the AOP associated with this failure. In this PRA sequence, operators are unable to restore RHR to service (either by addressing the problem that stopped the running train or returning the opposite train to operation). Further, in this PRA sequence, the operators fail to control SG level (i.e., no feedwater is initiated) and are never procedurally directed to establish a bleed path for feed and bleed cooling. Sources of feedwater called out in the AOP are main feedwater, auxiliary feedwater, and the condensate system.

Given these failures, core damage ensues at ~28 hours. Due to the relatively high decay heat and the intact RCS, the system re-pressurizes, and hot leg creep rupture is predicted to occur around 30 hours. Un-arrested core damage and relocation to the lower head ultimately leads to failure of the vessel lower head. Following relocation of the damaged core to the reactor cavity, containment pressure continues to be controlled by containment fan coolers. Several combustions are predicted to occur, with the first occurring at the time of hot leg nozzle creep rupture. In the baseline Scenario #8, these combustion events are assumed not to impact long-term containment cooler performance.

The types of actions envisioned by the AOP are restoration of RHR in shutdown cooling mode, depressurizing the RCS if pressure has risen to the point of dead-heading RHR, using charging and letdown to control pressurizer level, and re-filling of the SGs. Restoration of RHR in shutdown cooling mode is an action that will have been pursued since the start of the accident and will be moot by the time of core damage due to voiding of the RCS loops. Given the inability to restore RHR, the fact that RCS pressure is above the RHR dead-head pressure is of less relevance. Meanwhile, charging and letdown as a means of controlling RCS level are (implicitly) unsuccessful in the Level 1 PRA sequence. The cue for SG feed will occur when SG narrow range level drops below 10 percent in more than two SGs, which occurs approximately 8 hours prior to core damage in the simulation. The AOP also includes a response-not-obtained (upon continued inability to establish RHR shutdown cooling) to “Consult TSC for appropriate contingency actions if applicable.” Thus, there are a number of actions that could be in play at the time of core damage, and they depend highly on the specifics of the hardware failures and operator accident response. No clear preferential action could be identified in the context of the Level 2 HRA’s first priority candidate action, “continued attempts to accomplish pre-core damage injection/recovery.” Attempts to feed SGs will already be addressed by the Level 2 PRA HRA’s second priority for this scenario.

The screening HRA approach is applied to the candidate actions (that are taken from Table E-5, considering the additional refinement presented above). This is shown in Table E-18.

Table E-18 Scenario #8 Candidate Action Screening HRA

Action	Viability?	Screening HEP	Basis for Screening HEP
No clear preferential action identified for continued attempts to accomplish pre-core damage injection/recovery	n/a	n/a	n/a
Inject into SGs		0.9	An accident-altering event (hot leg nozzle creep rupture and associated combustion) occur during the implementation period
Open head vents		0.9	It is never the 1 st or 2 nd priority.

E.4.8.1 Scenario #8, Candidate Action #1, No action identified

No viable candidate action is identified based on the prior failures to establish injection in this scenario.

E.4.8.2 Scenario #8, Candidate Action #2, Inject into SGs

HFE Definition:

In concert with the TSC, which was activated well before, operators fail to inject into the steam generators using the EDMG strategy.

NOTE: EDMG guidance envisions that temporary instrumentation will be installed to monitor SG level, pressure, and feed flow, and that the steam generator ARVs will be locally and manually operated. Here, the normal means of valve control and indication for these parameters is expected to be available (the event here does not involve a loss of large area of the plant due to fire or explosion, nor does it involve a station blackout event causing eventual complete loss of instrumentation). Loss of one train of control or instrumentation due to the initiating event, coincident with independent failure or unavailability of the other train of control or instrumentation, is not considered further. Such a situation would complicate this human action and is only neglected here due to the expectation that it would not significantly contribute to the frequency of the events under consideration. This is all in contrast to the use of the EDMG guidance for SG feed, rather than other installed equipment. This is deliberate and done on the assumption that the installed main feedwater, auxiliary feedwater, and condensate systems are unavailable by virtue of their failure in the Level 1 PRA sequence.

Viable Actions:

The relevant action in question here is to use the EDMG trailer-mounted pump to inject into the steam generators, and if steam generator pressure necessitates, to depressurize the steam generators by opening the steam generator ARVs. This carries the key assumption that the EDMG pump is not being used for other actions, or that the second EDMG pump and a secondary water source (e.g., the demineralized water tank) are being utilized. Along with the other scenario characteristics associated with this candidate action (transient accident type, RPV not vented, containment closed and isolated), this action is only relevant prior to installation of the SG dams (which occurs during POS 6) or after removal of the SG dams (which occurs during POS 10).

In the MELCOR simulation of Scenario #8, the steam generators boil dry at ~22 hours. At this point, and in the absence of further steaming, the secondary-side of the steam generators begin to slowly depressurize due to not being leak-tight. This process is gradual, and the amount of leakage in such situations is very uncertain, as discussed further in a sub-section of the entry entitled, "Consequential Steam Generator Tube Rupture," in (NRC, 2022a). In the MELCOR simulation, the steam generator pressure drops below the target cited in EDMG guidance at ~26 hours, which coincides with the onset of significant fuel heatup. At the time of the onset of core damage at ~28 hours, steam generator pressure is ~180 psig (12 bar), and it continues to drop thereafter. Based on this, depressurizing all four steam generators is not required, though ARV manipulation will be required for other reasons (see below).

More specifically, EDMG guidance (as applied to the scenario in question, and given the aforementioned assumptions) directs plant personnel to:

Up-Front Cautions:

- Ensure SG level, pressure, and feed instrumentation is available (in this case, being provided by the normal MCR means)
- Determine which SG(s) is to be fed, stating that SG 2 provides the easiest physical access
- Plan to maintain pressure on the non-fed SGs to 200–300 psi, and no less than 150 psi. The latter constraint is to prevent nitrogen fill gas from the cold leg accumulators from creating a void in the RCS that can disrupt natural circulation
- Perform core monitoring and local radioactivity monitoring prior to, and during, execution of the actions

Action Guidance:

- Refer to EDMG guidance and figures to develop the approach to acquiring the needed equipment, establish the deployment path, and identify target flow rates
- Depressurize all SGs to 200–300 psi and ensure non-fed SG pressure does not go below 150 psi
- Open the ARV on the SG to be fed
- When pressure on the to-be-fed SG drops below 100 psi, initiate feed by:
 - Retrieving the portable pumps and associated tools from the warehouse (outside the protected area, on the owner-controlled area, roughly a 5-minute drive from the power block)
 - Obtaining tools for feedwater flange removal from the Maintenance Building
 - Connecting the suction side of the EDMG pump to a fire water storage tank (FWST), or demineralized water storage tank (DWST), located in various locations in the yard within the protected area
 - Installing the EDMG pump-to-RWST adapter from the EDMG toolbox
 - Connecting the injection-side fire hose and adapter to the blind flange connection point for the desired SG, which is located in the corresponding Feed Reg Valve rooms (this includes removing the blind flange, installing an adaptor, and re-using the existing flange gasket)
 - Opening the applicable auxiliary feedwater recirculation valve in the same general vicinity (which includes cutting the associated lock)
 - Opening the FWST or DWST isolation valve

- Starting and operating the EDMG pump
- Monitoring SG level and throttling flow as necessary to maintain SG level

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours (30 min. action identification + 30 min. development + 1 hr. implementation). However, an additional 30 minutes is assumed needed to determine that no continuation of pre-core damage actions is preferable to proceeding with the EDMG-based SG feed. This is consistent with the overall scheme discussed in Section E.2, wherein the initial actions upon onset of core damage are to continue ongoing restoration and recovery actions given the lack of applicable SAMG guidance, as well as the generic timing estimate of 30 minutes to address the diagnosis phase of a candidate action. Thus, the overall estimated time for candidate action #2 is 2.5 hours. Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate.

However, it was acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as acquiring the tow vehicle and performing dose surveys. In addition, hot leg nozzle creep rupture and an associated combustion event are predicted to occur during the implementation phase of the action. These events could prompt a hold in the implementation while plant staff assess the situation, though these events do not directly de-motivate the feeding of SGs. Were this delay to persist, vessel breach is predicted to occur a few hours later, which similarly has the potential to cause a re-visitation of plant status. Nevertheless, given the estimated needed time of 2.5 hours as compared to the 5 hours between the onset of core damage and the occurrence of vessel breach, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the "yard" within the Protected Area, buildings set off from the power block, one of the two main steam valve rooms, the lower elevations of the steam tunnel area, and connection points therein. Radiological habitability issues are not anticipated in these areas since there is no specific containment impairment and there is no elevated containment pressure driving fission product leakage. The combustion that occurs within containment will not be sufficient to compromise containment integrity and thus is not expected to have any direct influence on the accessibility to the relevant plant areas. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is no EOP guidance that spells out the equivalent action (EOPs focus on the use of installed equipment for this purpose). There is EDMG support (which is what is being invoked), and there is no general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the “Fair” branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew may not be aware of the status of the EDMG equipment, and the plant may have begun to take some equipment (including indication) out-of-service as part of the outage.

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that injecting water into dry SGs has the potential to thermally shock tubes and potentially cause a steam generator tube rupture. EDMG guidance includes cautions about monitoring core conditions and radiation levels on the secondary-side system due to this concern.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.16.

Quantification of Execution:

Entry Conditions: All of the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and the operations support center as needed. The personnel will largely be following the EDMG procedure for injecting into the SGs, and the needed guidance is available in the EDMGs. As discussed above in the diagnosis portion, changing plant conditions associated with hot leg nozzle creep rupture and a corresponding combustion event could delay implementation, but are not expected to prevent it.

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-CR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are no radiological habitability concerns or hazard-related concerns. Therefore, the “Good” branch is taken.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1.

Total HEP:

Total HEP for Scen. #8, Action #2 = $0.16 + 0.1 - (0.16)(0.1) = 0.26 - 0.016 = \sim 0.24$.

E.4.8.3 Scenario #8, Candidate Action #3, Open head vents

NOTE: This action would only be relevant if hot leg nozzle creep rupture has not occurred by the time of diagnosing and executing this action (which is not the case for Scenario #8). This is not the expected outcome for early outage accidents, and there is insufficient information in the MELCOR simulations to estimate its likelihood in late-outage accidents. Note that a sensitivity to Scenario #8 described in Section 2.3.3 does provide limited information about the early-outage situation where hot leg nozzle creep rupture is deliberately suppressed. Since modeling this action in the PRA model is plausible (for situations where hot leg nozzle creep rupture is assumed not to occur), and since sufficient information is available, it is pursued here. Keep in mind that pressurized-water reactor head vents generally provide less pressure relief capacity than PORVs, so they may have limited effect under high decay heat or high oxidation energy situations.

NOTE: Under most circumstances, the operators would use the pressurizer PORVs or dump steam to accomplish this action. Even if the head vents were the desired vent path, these would normally be operated remotely. However, in the unique case here where EDMG-guided actions are the focus, the EDMGs do not consider opening PORVs (on the presumption that they are unavailable due to the event that prompted EDMG entry), while feeding SGs and dumping steam has already been investigated in the prior candidate action.

HFE Definition:

Having either succeeded or failed at injecting into the SGs using the EDMG pump, and in concert with the TSC which was activated well before, operators fail to open both trains of reactor head vents using EDMG guidance.

Viable Actions:

The relevant action in question here is to use a portable gasoline powered generator to locally open valves in the reactor head vent piping. The portable gasoline powered generator and the also-required 125V AC-to-125V DC power supply are understood to be in the maintenance building located on the Protected Area. The electrical connections are made (three per head vent train) in locations of the Control Building. Credit for this equipment of course carries the key assumption that the generator and power supply are not being used for other accident response purposes.

More specifically, EDMG guidance directs plant personnel to:

- Obtain the portable gasoline powered generator and power supply
- Locate the desired penetration within the control building
- Open three different valves by applying 125V DC power to the designated terminal

The guidance for this action is notably more terse than the other EDMG strategies. While it specifies the needed equipment and connection location information, it does not address such issues as ensuring the generator has sufficient gas, refilling the gas, or confirming that there is sufficient cabling to energize multiple terminals (or provide any other cautions/notes). Regarding the issue of gasoline for the portable generator, this would affect the period of time that the head vents remained open. Unless this HFE was ultimately credited in the PRA as part of preventing vessel breach entirely, then the potential relevance of this concern would be decreased by RCS depressurization and the existence of a larger RCS leak path following vessel breach.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 1.5 hours (30 min. action identification + 30 min. development + 30 min. implementation). Referring back to the timing discussion for Candidate Action #2, this would nominally result in completed implementation 4 hours after the onset of core damage. In both the baseline and sensitivity MELCOR simulation, vessel breach occurs approximately 4.6 hours after the onset of core damage. Vessel breach (for the situation here where hot leg nozzle creep rupture has not already occurred) makes this action moot as a pressure control measure (since vessel breach will equilibrate RCS pressure with containment pressure). Otherwise, the main effect of this action will be to create a fission product path from the RCS to the pressure relief tank. Thus, there is sufficient time available to credit the action, but not much margin.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the “yard” within the Protected Area, the maintenance building, and the control building. Radiological habitability issues are not anticipated in these areas since there is no specific containment impairment and there is no elevated containment pressure driving fission product leakage. The combustion that occurs within containment will not be sufficient to compromise containment integrity and thus is not expected to have any direct influence on the accessibility to the relevant plant areas. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is EOP guidance focused on the equivalent action (albeit remotely rather than locally), and there is EDMG support (which is what is being invoked). Operation of the head vents as a means of assisting in RCS depressurization is covered in general accident management (SAMG) training, in the sense that it is invoked for this purpose. Because the local and manual aspects of the EDMG actions differ from the EOP invocation, and because the level-of-support of this particular EDMG strategy is notably less than that of other EDMG strategies (as previously discussed), it was judged that the “Best” branch would be optimistic, and so the “Good” branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew may not be aware of the status of the EDMG equipment, and the plant may have begun to take some equipment (including indication) out-of-service as part of the outage.

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that opening reactor head vents has the potential to provide a pathway for hydrogen and fission products to the pressurizer relief tank, and subsequently to the containment.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.08.

Quantification of Execution:

Entry Conditions: All the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and OSC, as needed. The personnel will largely be following the EDMG procedure for locally and manually opening the head vents.

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-CR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The down branch (low complexity) is taken because the action requires two locations to be visited, involves a small number of steps, can potentially be completed by a single equipment operator, and requires minimal coordination activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are no radiological habitability concerns or hazard-related concerns. However, extended operation of a gasoline-powered generator in an enclosed portion of the control building not designed with this situation in mind, may contribute to some environmental complexities. Therefore, the “Poor” branch is taken.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1.

Total HEP:

Total HEP for Scen. #8, Action #3 = $0.08 + 0.1 - (0.08)(0.1) = 0.18 - 0.008 = \sim 0.17$.

E.4.9 Scenario #9 Basic Scenario Information

Scenario #9 is an SBO type accident in POS 8E with the containment closed and isolated at the time of core damage. Table E-19 shows the general accident scenario timeline.

Table E-19 Scenario #9 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	9	
Plant Operating State / Tech Spec Mode	8E / 6	Refueling, prior to fuel movement with the transfer tube assumed closed, SG dams installed, vessel head off, refueling cavity flooded
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators successfully close the containment hatch, personnel airlock, and any open penetrations; gravity feed from the RWST starts when the bulk water temperature approaches 100 °C (but becomes dead-headed due to rising containment pressure); operators are unable to restore power
Containment status	→	Initially open and un-isolated; closure and isolation occur early; however, containment overpressure failure occurs prior to core damage
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavai.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of emergency classification in AOPs	< 1	Loss of Electrical Systems and Loss of Residual Heat Removal AOPs direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite ac sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	E.g., 23	Declaration of site area emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (containment is closed early). Gravity drain is assumed to start at 23 hours, under the assumption that via calculations or limited instrumentation restoration, operators are aware that the bulk refueling cavity water temperature has reached 100 °C, and significant water loss from evaporation has been occurring. Here, it is assumed that SAE is declared at this time, under Emergency Discretion.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., 24	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE or GE have a criterion during shutdown equivalent to the alert criterion cited above). Here, it is assumed that GE is declared when the RWST gravity-feed becomes dead-headed and under the presumption that RWST level can be obtained, at 24 hours. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
Time of onset of core damage ²	125	
Time of vessel breach	135	
Time of combustion event(s)	n/a	Three relatively small combustion events can be observed from the figures, occurring between 125 to 135 hours.

Table E-19 Scenario #9 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of containment overpressure liner tear	118	Prior to core damage and due to extended steaming of the refueling cavity
Time of basemat melt-through	n/a	Cavity erosion is occurring very slowly.
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

E.4.10 Scenario #10 Basic Scenario Information

Scenario #10 is a LOCA type accident in POS 8E with the containment closed and isolated at the time of core damage. Table E-20 shows the general accident scenario timeline. Table E-21 gives the general survivability and habitability considerations.

Table E-20 Scenario #10 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	10	
Plant Operating State / Tech Spec Mode	8E / 6	Refueling, prior to movement of fuel and prior to opening of the fuel transfer tube
Initiating event	LoINV-LG	Loss of Inventory – Large diversion of water from the operating train of RHR, with the train continuing to pump until the leakage path is isolated
Level 1 successes/failures	→	Operators successfully diagnose the loss of inventory and isolate the leakage path at t~1.8 hours; SI and charging inject from t=30 minutes until RWST low level at ~1.1 hours and injection is assumed to end at this point; operators are unable to restore RHR and containment coolers, and fail to refill the RWST
Containment status	→	The containment hatch and personnel airlock are initially open, and are closed at t=55 minutes
Contain. systems	→	Containment sprays and fan coolers are unavailable
AC power status	Avail.	
DC power status	Avail.	
Invocation of emergency classification in AOPs	< 1	Loss of Residual Heat Removal AOP directs initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 2.5	RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	8	RPV level at top of active fuel with containment closure established
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	8.5	Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig NOTE: The containment level rise is interpreted to be “unexplained” in the context of this condition; separately, containment closure is expected to be with four bolts (thus the containment closure cited above, as opposed to the alternate Tech Spec intact state).
Time of onset of core damage ²	9.5	
Time of vessel breach	20	
Time of combustion event(s)	Multiple	Large combustions occur at vessel breach (20 hours) and again at ~30, 47, and 58 hours
Time of containment overpressure liner tear	> 168	
Time of basemat melt-through	>168	

Table E-20 Scenario #10 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific Emergency Action Level Matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

Table E-21 Scenario #10 General Survivability and Habitability Perspective

Timing (Top of the Hour)	General I&C and Equip. Survivability Concerns	Habitability Concerns	Accident-Altering Event ¹
Onset of core damage			
10	With the vessel head removed, and containment closure (as opposed to Tech Spec leakage conditions), and with containment pressure steadily rising, areas adjacent to containment will experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion	For the same reasons as described to the left, habitability near containment penetrations may be impeded; no wide-spread habitability concern is expected, though cumulative noble gas releases to the environment are approaching 0.1% during this timeframe, so there may be a need for recurring dose surveys associated with onsite actions.	
11			
12			
13			
14			
15			
16			
17			
18			
19			
20	In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach		Vessel breach, combustion
21			
22			
23			
24			
NOTE: If the original RHR failure and ensuing nature of isolation result in a pathway outside of containment, then radiological habitability in the auxiliary building and surrounding areas would be a <u>significant concern</u> .			
¹ This refers to a combustion event, hot leg creep rupture, vessel breach, containment failure, or failure of a surrounding structure.			

Scenario description:

The accident is initiated by a large loss of inventory in the operating train of RHR in POS 8E, with fuel still in the RPV and the fuel transfer tube closed, roughly 6 days into the outage. The RPV head is removed, and the lower and upper refueling cavities are flooded. Initial water level is ~38 ft above the top of the fuel in the RPV.

Upon loss of inventory and the related off-normal conditions for RHR, operators enter the AOP associated with loss of RHR. They may also enter the AOP associated with RCS leakage, but it does not strictly apply to this plant operating mode per the procedure itself. In this PRA sequence operators are able to stop the operating train of RHR and isolate the break, after some delay which results in significant water loss to either lower containment or the auxiliary building (depending on the cause of the loss of inventory). Operators also succeed in closing the containment hatch and personnel airlock (if not already closed). Operators are not able to

restore RHR or containment coolers. They successfully initiate RCS injection using the charging and SI pumps but are unable to refill the RWST once it is depleted.

The failure to refill the RWST is currently an assumed failure in the Level 1 PRA model. The action is modeled in the Level 1 PRA event trees but is set to fail (TRUE) due to broad concerns about makeup flow rate capacity limitations and lack of proceduralized direction. Separately, containment sprays are unavailable (out-of-service), and the relevant AOPs will not direct their restoration or use. Thus, at the onset of core damage, the systems that may warrant re-consideration for restoration/recovery in the Level 2 PRA HRA are the RWST (for ECCS injection), containment coolers, and containment sprays. The situation in this regard is very analogous to that discussed earlier for Scenario #1 (see Table E-8 and the discussion that immediately follows it). For the same reasons as outlined there, RWST refill and subsequent resumption of ECCS injection is the candidate action considered here.

To complete the scenario description, and as previously recorded, core damage starts at approximately 9.5 hours, with vessel breach and a large combustion occurring approximately 10 hours later. No other “accident-altering” event occurs until much later (a second large combustion).

So, with this background established, the screening HRA approach can be applied to the three candidate actions (that are taken from Table E-5). This is shown in Table E-22.

Table E-22 Scenario #10 Candidate Action Screening HRA

Action	Viability?	Screening HEP	Basis for Screening HEP
Refill RWST and restore RCS injection, to arrest fuel damage	As of this screening step, these actions appear viable	0.1	This is very similar to an EOP action, is the highest perceived priority and not concurrent with an accident-altering event, and there are no obvious habitability or survivability concerns.
Spray within containment, to combat rising containment pressure		0.5	The other criteria (for HEP = 0.1, 0.9 or 1.0) do not apply.
Spray within containment or spray external release point, to reduce fission product releases		0.9	It is never the 1 st or 2 nd priority.

E.4.10.1 Scenario #10, Candidate Action #1, Continued attempts to accomplish pre-core damage injection/recovery

HFE Definition:

Having diagnosed the onset of core damage (i.e., ongoing heatup and degradation of the fuel in the RPV), and in concert with the TSC which was activated well before this time (notionally

seven hours prior), operators fail to refill the RWST and restore RCS injection via charging and/or safety injection pumps.³²

Viable Actions:

The details of what it means to restore RWST inventory are again dependent on details that are below the resolution of the PRA/HRA. As a representative situation, it will be assumed that the failure to replenish the RWST prior to core damage was hampered by diagnosis (due to the poor procedural basis and other decision-making complexities), rather than any mechanical issues (which is supported notionally by the successful use of the RWST in the PRA sequence, prior to depletion).

Section E.4.1.1 presents discussion about the potential modes of RWST refill and is directly applicable to the current HFE as well. It was concluded there that refill using the trailer-mounted pump was the most likely choice.

As presented earlier, to accomplish the action (RWST refill using the EDMG pump), plant personnel must:

- Assess the condition of the Fire Protection System and the status of Fire Fighting Actions or Plans
- Retrieve the portable pumps and associated tools from the warehouse location (outside the protected area, on the owner-controlled area, roughly a 5-minute drive from the power block)
- Connect the suction side of the EDMG pump to one of two fire water storage tanks (FWST) or demineralized water storage tank (DWST), located in various locations in the yard within the protected area
- Install the EDMG pump-to-RWST adapter from the EDMG toolbox
- Connect the injection-side fire hose and adapter to the quick-disconnect connection point downstream of a specified valve, which is located in the RWST moat (this includes cutting the lock/chain since these are locked-closed valves)
- Opening additional valves in the RWST piping and the FWST/DWST piping
- Starting and operating the EDMG pump
- Monitoring RWST level

Once minimum RWST level has been restored, it is assumed that subsequent actions to inject using safety injection and charging can be taken from the MCR.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was

³² Since the RCS pressure will be low, any of the four relevant trains of SI/charging is capable of delivering the minimum needed flow.

acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as defeating security barriers and acquiring the tow vehicle. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~10 hours). Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the “yard” within the Protected Area, or in buildings set off from the power block. As such, no specific radiological habitability issues are anticipated. However, the RWST is adjacent to the auxiliary building. Therefore, any usages where the Level 1 failures have led to an ISLOCA, the ensuing release of radioactive material out of the piping penetration area filtration and exhaust system (PPAFES) or through auxiliary building failure may make relevant areas significantly contaminated. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is EOP guidance that spells out the same action using other means, there is EDMG support (which is what is being invoked), and there is general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the “Best” branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew is likely aware of the status of the RWST and the injection trains by virtue of their role earlier in the accident sequence. Conversely, there are a number of potential makeup sources that have to be evaluated, and some will require sending out personnel or off-site communications to confirm their availability. This will complicate the decision-making process in the TSC, given the lack of the more structured process when the SAMGs are in play (and the much more structured process when the EOPs/AOPs were being used).

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that injecting into a degraded core has particular complications (e.g., increased hydrogen production, potential of shattering fuel), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.04.

Quantification of Execution:

Entry Conditions: All of the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and the operations support center as needed. The personnel will largely be following the EDMG procedure for RWST refill, and the needed steps for resuming injection can be developed by the TSC using existing procedures for the same action (namely the EOPs).

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-MCR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed, and the action will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are no particular radiological habitability or hazard-related concerns. Therefore, the "Good" branch is taken. If this HFE/HEP were used to address an analogous ISLOCA scenario, radiological habitability in the vicinity of the auxiliary building (which would include the RWST connection point), would warrant selection of the "Poor" branch.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1. It would be 0.5 for the analogous ISLOCA scenario.

Total HEP:

Total HEP for Scen. #10, Action #1 = $0.04 + 0.1 - (0.04) \cdot (0.1) = 0.14 - 0.004 = \sim 0.14$.

Total HEP for ISLOCA-equivalent = $0.04 + 0.5 - (0.04) \cdot (0.5) = 0.54 - 0.02 = \sim 0.52$.

E.4.10.2 Scenario #10, Candidate Action #2, Spray in containment

HFE Definition:

Having either succeeded or failed at restoring RCS injection, and in concert with the TSC, which was activated well before, operators fail to spray within containment using the EDMG strategy.

Viable Actions:

Here, the prevalent question is whether operators attempt to restore the installed containment spray system or pursue the EDMG strategy (which uses the portable pump previously discussed in the prior modeled action). In reality, staffing and other resources permitting, they would likely pursue both. Here, an assumption must be made, and for the same reasons as previously discussed in Section E.4.1.2, the EDMG-based strategy is selected.

This carries the key assumption that the EDMG pump is not being used for other actions (including the previously-covered RWST refill action), or that the second EDMG pump and a secondary water source (e.g., the DWST) are being used.

In terms of the actions to be taken, the actions are fairly similar to those described previously for RWST refill using the EDMG pump, with one major deviation. In this case, the injection hoses are being routed into the auxiliary building, to a test connection located adjacent to the containment wall. The preferred path, based on walkdowns with plant operational staff, is through one of the main steam valve rooms.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as acquiring the tow vehicle and performing dose surveys. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~10 hours), even given the notional 2-hour timeframe associated with the preceding candidate action. Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the "yard" within the Protected Area, buildings set off from the power block, one of the two main steam valve rooms, the lower elevations of the steam tunnel area, and connection points adjacent to containment.

For the baseline scenario, radiological habitability issues are possible near the connection points that may require limited stay time and donning of protective gear. Since there is no specific containment impairment, it's expected that these radiological habitability issues may affect the implementation of the strategy, but that they will not prevent it. Of particular note here,

relative to other situations, is that there has been a direct path for fission products to enter containment throughout core damage (RPV head removed versus a situation where the RCS is intact or the RCS is vented through the pressurizer relief valves). A review of the radiological release results from MELCOR calculations demonstrates that retention of volatile radionuclides within the RCS is significantly (roughly an order-of-magnitude) lower for cases with the vessel head removed. This would lead to higher ex-containment doses, relative to situations with the RCS vented or intact, all other things being equal.

Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

For an ISLOCA variant of this scenario, entry into the areas of the auxiliary building adjacent to containment may be precluded by ongoing radiological releases into this area. This would depend on precisely where the release pathway into the auxiliary building is, but given the proximity of the connection point to ISLOCA piping (same general spatial area, and both sets of piping in the subterranean portion of the building) it is appropriate to preclude credit.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is no EOP guidance that spells out the equivalent action, there is EDMG support (which is what is being invoked), and there is no general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the "Fair" branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be "Adequate" (up branch). The crew may be aware of the status of the EDMG equipment from attempting the prior action to refill the RWST. Information about what is taking place within containment, and whether spraying containment may cause a combustion event by de-inerting an inerted containment atmosphere may be impaired, if the necessary equipment and indication to do this assessment were part of the ongoing maintenance scope.

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that spraying containment has particular complications (e.g., de-inerting containment), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.16.

Quantification of Execution:

Entry Conditions: All the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and OSC, as needed. The personnel will largely be following the EDMG procedure for spraying the containment atmosphere, and the needed guidance is available in the EDMGs.

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-MCR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lighting will be available. As discussed earlier, there are general radiological habitability concerns that are exacerbated by the direct path between the vessel and containment. This will further impact limitation in stay time or donning of protective clothing. There are no hazard-related concerns. Based on the totality of these considerations, the "Poor" branch is taken.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.5.

Total HEP:

Total HEP for Scen. #10, Action #2 = $0.16 + 0.5 - (0.16)(0.5) = 0.66 - 0.08 = \sim 0.58$.

Total HEP for ISLOCA-equivalent = 1.0 (screened out based on habitability).

E.4.10.3 Scenario #10, Candidate Action #3, Spray in containment or spray the release point

HFE Definition:

Having either succeeded or failed at spraying in containment using the EDMG strategy (for the purpose of controlling containment conditions), and in concert with the TSC, which was activated well before, operators fail to spray an external release point using the EDMG

guidance. [Spraying in the containment is not considered here, since it was already considered in the previous candidate action.]

Viable Actions:

This candidate action is not further developed here but rather is left as a placeholder. There are three key reasons for this:

1. In the reference scenario, containment does not fail for almost 7 days.
2. Containment failure due to long-term overpressure is likely to lead to a situation where a specific release point is not identifiable during the accident response.
3. Previous work done under other projects has suggested that the spray flux and fission product in-spray residence time is unlikely to lead to significant fission product capturing, for the spray flow rates and geometries of relevance here.

Feasibility Assessment:

Not applicable (N/A)

Determination of Habitability:

N/A

Quantification of Diagnosis:

N/A

Quantification of Execution:

N/A

Total HEP:

N/A

E.4.11 Scenario #11 Basic Scenario Information

Scenario #11 is an LOCA type accident in POS 5A with the containment closed and isolated at the time of core damage. Table E-23 shows the general accident scenario timeline. Table E-24 gives the general survivability and habitability considerations.

Table E-23 Scenario #11 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	11	
Plant Operating State / Tech Spec Mode	5A / 5	Taking the system solid for de-gassing; initial RCS pressure is 350 psig with RCS temperature of 180 °F
Initiating event	LoINV-OP	Loss of inventory due to overpressure in the operating train of RHR leading to a failed-open relief valve and trip of the operating RHR train
Level 1 successes/failures	→	Operators successfully diagnose the loss of inventory, but fail to isolate the leak path and initiate the standby train of RHR prior to the need for makeup; operators close the containment hatch and personnel airlock within 35 minutes, but fail to restore a train of containment cooling; SI and charging inject from ~1.6 hours until RWST low level at ~6 hours and injection is assumed to end at this point; following injection, operators are unable to isolate the leak path and re-establish RHR, or to refill the RWST.
Containment status	→	The containment hatch and personnel airlock are initially open, and are closed at t=35 minutes.
Contain. systems	→	Containment sprays and fan coolers are unavailable.
AC power status	Avail.	
DC power status	Avail.	
Invocation of emergency classification in AOPs	< 1	AOP for RCS leakage directs initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	~ 6	RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established—earlier declaration might occur due to fluctuating temperatures around 200 °F between 2 to 6 hours
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	17	RPV level at top of active fuel with containment closure established
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	18	Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes, and containment closure established with containment pressure greater than 3 psig NOTE: The containment level rise is interpreted to be “unexplained” in the context of this condition; separately, containment closure is expected to be with four bolts (thus the containment closure cited above, as opposed to the alternate Tech Spec intact state).
Time of onset of core damage ²	21	
Time of vessel breach	28	
Time of combustion event(s)	~28, 38, and 55	Large combustions occur at vessel breach (28 hours) and again at ~38 and 55 hours
Time of containment overpressure liner tear	166	
Time of basemat melt-through	>168	

Table E-23 Scenario #11 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

Table E-24 Scenario #11 General Survivability and Habitability Perspective

Timing (Top of the Hour)	General I&C and Equip. Survivability Concerns	Habitability Concerns	Accident-Altering Event ¹
Onset of core damage			
22	With an RCS leak path, and containment closure (as opposed to Tech Spec leakage conditions), and with containment pressure steadily rising, areas adjacent to containment could experience increasing temperatures and dose fields during this time period in a potentially unpredictable fashion	For the same reasons as described to the left, habitability near containment penetrations may be impeded; no wide-spread habitability concern is expected, though cumulative noble gas releases to the environment are approaching 0.1% during this timeframe, so there may be a need for recurring dose surveys associated with onsite actions.	
23			
24			
25			
26			
27			
28	In addition to the above, some damage within-containment is possible from the large combustion at the time of vessel breach		Vessel breach, combustion
29			
30			
31			
32			
33			
34			
35			
36			
NOTE: If the original RHR failure and ensuing nature of isolation result in a pathway outside of containment, then radiological habitability in the auxiliary building and surrounding areas would be a <u>significant concern</u> .			
¹ This refers to a combustion event, hot leg creep rupture, vessel breach, containment failure, or failure of a surrounding structure.			

Scenario description:

The accident is initiated by a loss of inventory in the operating train of RHR in POS 5A, associated with an overpressure transient causing a relief valve failure in the operating train of RHR. The transient is presumed to result in tripping of the operating train. The RPV head is in place, the RCS has yet to be vented, and the plant is in a near-solid configuration.

Upon loss of inventory and the related off-normal conditions for RHR, operators enter the AOPs associated with RCS leakage and loss of RHR. In this PRA sequence operators succeed in diagnosing the loss of inventory but fail to isolate the leak path and initiate the standby train of RHR prior to the need for makeup. Upon the need for makeup, operators successfully start

ECCS injection, which keeps RCS water level high, and causes RCS temperature to oscillate around 200 °F. Operators also succeed in closing the containment hatch, any open containment penetrations, and the personnel airlock. Operators are not able to restore RHR or containment coolers, nor to refill the RWST once it is depleted.

The failure to refill the RWST is currently an assumed failure in the Level 1 PRA model. The action is modeled in the Level 1 PRA event trees but is set to fail (TRUE) due to broad concerns about makeup flow rate capacity limitations and lack of proceduralized direction. Separately, containment sprays are unavailable (out-of-service), and the relevant AOPs will not direct their restoration or use. Thus, at the onset of core damage, the systems that may warrant re-consideration for restoration/recovery in the Level 2 PRA HRA are the RWST (for ECCS injection), containment coolers, and containment sprays. The situation in this regard is very analogous to that discussed earlier for Scenario #1 (see Table E-8 and the discussion that immediately follows it). For the same reasons as outlined there, RWST refill and subsequent resumption of ECCS injection is the candidate action considered here.

To complete the scenario description, and as previously recorded, core damage starts at approximately 21 hours, with vessel breach and a large combustion occurring approximately 7 hours later. No other “accident-altering” event occurs until much later (a 2nd large combustion).

So, with this background established, the screening HRA approach can be applied to the three candidate actions (that are taken from Table E-5). This is shown in Table E-25.

Table E-25 Scenario #11 Candidate Action Screening HRA

Action	Viability?	Screening HEP	Basis for Screening HEP
Refill RWST and restore RCS injection, to arrest fuel damage	As of this screening step, these actions appear viable	0.1	This is very similar to an EOP action, is the highest perceived priority and not concurrent with an accident-altering event, and there are no obvious habitability or survivability concerns.
Spray within containment, to combat rising containment pressure		0.5	The other criteria (for HEP = 0.1, 0.9 or 1.0) do not apply.
Spray within containment or spray external release point, to reduce fission product releases		0.9	It is never the 1 st or 2 nd priority.

E.4.11.1 Scenario #11, Candidate Action #1, Continued attempts to accomplish pre-core damage injection/recovery

HFE Definition:

Having diagnosed the onset of core damage (i.e., ongoing heatup and degradation of the fuel in the RPV), and in concert with the TSC which was activated well before this time (notionally

7 hours prior), operators fail to refill the RWST and restore RCS injection via charging and/or safety injection pumps.³³

Viable Actions:

The details of what it means to restore RWST inventory are again dependent on details that are below the resolution of the PRA/HRA. As a representative situation, it will be assumed that the failure to replenish the RWST prior to core damage was hampered by diagnosis (due to the poor procedural basis and other decision-making complexities), rather than any mechanical issues (which is supported notionally by the successful use of the RWST in the PRA sequence, prior to depletion).

Section E.4.1.1 presents discussion about the potential modes of RWST refill and is directly applicable to the current HFE as well. It was concluded there that refill using the trailer-mounted pump was the most likely choice.

As presented earlier, to accomplish the action (RWST refill using the EDMG pump), plant personnel must:

- Assess the condition of the fire protection system and the status of fire fighting actions or plans
- Retrieve the portable pumps and associated tools from the warehouse location (outside the protected area, on the owner-controlled area, roughly a 5-minute drive from the power block)
- Connect the suction side of the EDMG pump to one of two fire water storage tanks (FWSTs) or demineralized water storage tank (DWST), located in various locations in the yard within the protected area
- Install the EDMG pump-to-RWST adapter from the EDMG toolbox
- Connect the injection-side fire hose and adapter to the quick-disconnect connection point downstream of a specified valve, which is located in the RWST moat (this includes cutting the lock/chain since these are locked-closed valves)
- Open additional valves in the RWST piping and the FWST/DWST piping.
- Start and operate the EDMG pump
- Monitor RWST level

Once minimum RWST level has been restored, it is assumed that subsequent actions to inject using safety injection and charging can be taken from the main control room.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was

³³ Since the RCS pressure will be low, any of the four relevant trains of SI/charging is capable of delivering the minimum needed flow.

acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as defeating security barriers and acquiring the tow vehicle. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~7 hours). Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the “yard” within the protected area, or in buildings set off from the power block. As such, no specific radiological habitability issues are anticipated. However, the RWST is adjacent to the auxiliary building. Therefore, for any usages where the Level 1 failures have led to an ISLOCA, the ensuing release of radioactive material out of the piping penetration area filtration and exhaust system (PPAFES) or through auxiliary building failure may make relevant areas significantly contaminated. Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is EOP guidance that spells out the same action using other means, there is EDMG support (which is what is being invoked), and there is general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the “Best” branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew is likely aware of the status of the RWST and the injection trains by virtue of their role earlier in the accident sequence. Conversely, there are several potential makeup sources that have to be evaluated, and some will require sending out personnel or relying on offsite communications to confirm their availability. This will complicate the decision-making process in the TSC, given the lack of the more structured process when the SAMGs are in play (and the much more structured process when the EOPs/AOPs were being used).

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that injecting into a degraded core has particular complications (e.g., increased hydrogen production, potential of shattering fuel), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.04.

Quantification of Execution:

Entry Conditions: All the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and OSC, as needed. The personnel will largely be following the EDMG procedure for RWST refill, and the needed steps for resuming injection can be developed by the TSC using existing procedures for the same action (namely the EOPs).

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-MCR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lightning will be available. As discussed earlier, there are no particular radiological habitability or hazard-related concerns. Therefore, the "Good" branch is taken. If this HFE/HEP were used to address an analogous ISLOCA scenario, radiological habitability in the vicinity of the auxiliary building (which would include the RWST connection point), would warrant selection of the "Poor" branch.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1. It would be 0.5 for the analogous ISLOCA scenario.

Total HEP:

Total HEP for scenario #11, Action #1 = $0.04 + 0.1 - (0.04) \cdot (0.1) = 0.14 - 0.004 = \sim 0.14$.

Total HEP for ISLOCA-equivalent = $0.04 + 0.5 - (0.04) \cdot (0.5) = 0.54 - 0.02 = \sim 0.52$.

E.4.11.2 Scenario #11, Candidate Action #2, Spray in containment

HFE Definition:

Having either succeeded or failed at restoring RCS injection, and in concert with the TSC, which was activated well before, operators fail to spray within containment using the EDMG strategy.

Viable Actions:

Here, the prevalent question is whether operators attempt to restore the installed containment spray system or pursue the EDMG strategy (which uses the portable pump previously discussed in the prior modeled action). In reality, staffing and other resources permitting, they would likely pursue both. Here, an assumption must be made, and for the same reasons as previously discussed in Section E.4.1.2, the EDMG-based strategy is selected.

This carries the key assumption that the EDMG pump is not being used for other actions (including the previously-covered RWST refill action), or that the second EDMG pump and a secondary water source (e.g., the DWST) are being used.

In terms of the actions to be taken, the actions are fairly similar to those described previously for RWST refill using the EDMG pump, with one major deviation. In this case, the injection hoses are being routed into the auxiliary building, to a test connection located adjacent to the containment wall. The preferred path, based on walkdowns with plant operational staff, is through one of the main steam valve rooms.

Feasibility Assessment:

Based on the generic timing estimates discussed earlier in this report, the estimated time required to identify the candidate action, work out the details of the approach, and execute that plan would be about 2 hours after the onset of core damage (30 min. action identification + 30 min. development + 1 hr. implementation). Based on the considerations discussed above under, "Viable Actions," this was judged to be a reasonable estimate. However, it was acknowledged that there could be quite a bit of variability in how long some things could take given the conditions, such as acquiring the tow vehicle and performing dose surveys. Nevertheless, there is still plenty of additional time to accomplish the actions given the long time frame between the onset of core damage and the prediction of vessel breach (~7 hours), even given the notional 2-hour timeframe associated with the preceding candidate action. Thus, the actions were determined to be feasible.

Determination of Habitability:

All relevant deliberations and actions take place in either the main control room, the TSC, the "yard" within the protected area, buildings set off from the power block, one of the two main steam valve rooms, the lower elevations of the steam tunnel area, and connection points adjacent to containment.

For the baseline scenario, radiological habitability issues are possible near the connection points that may require limited stay time and donning of protective gear. Since there is no specific containment impairment, it's expected that these radiological habitability issues may affect the implementation of the strategy, but that they will not prevent it. Since the leak path in the RCS is via an RHR relief valve, fission products must transit through the primary-system piping, which will result in additional retention relative to the case with the RPV head removed (Scenario #10).

Since the initiating event is an internal event, as defined by the scope of the at-shutdown Level 1 PRA, no hazard-related habitability issues are anticipated either.

For an ISLOCA variant of this scenario, entry into the areas of the auxiliary building adjacent to containment may be precluded by ongoing radiological releases into this area. This would depend on precisely where the release pathway into the auxiliary building is, but given the proximity of the connection point to ISLOCA piping (same general spatial area, and both sets of piping in the subterranean portion of the building) it is appropriate to preclude credit.

Quantification of Diagnosis:

Entry Conditions: Instrumentation is nominally available and there are no known negative impacts that would preclude pursuing the action. Thus, the entry conditions for performing detailed analysis of the diagnosis portion of the response were met.

Answers to decision tree questions:

1) Branch Point 1. Type of underlying or supporting procedural guidance and/or knowledge

There is no EOP guidance that spells out the equivalent action, there is EDMG support (which is what is being invoked), and there is no general accident management (SAMG) training stemming from the existence of a similar action within the SAMGs. Therefore, the “Fair” branch was chosen.

2) Branch Point 2. Information availability

Information availability was decided to be “Adequate” (up branch). The crew may be aware of the status of the EDMG equipment from attempting the prior action to refill the RWST. Information about what is taking place within containment, and whether spraying containment may cause a combustion event by de-inerting an inerted containment atmosphere may be impaired, if the necessary equipment and indication to do this assessment were part of the ongoing maintenance scope.

3) Branch Point 3. Potential negative Impacts

The up branch (negative impacts that may be mitigated) was chosen. From general accident management (SAMG) training, the TSC will know that spraying containment has particular complications (e.g., de-inerting containment), and will not have the same guidance construct for evaluating these potential negative impacts that is available when the SAMGs are the primary resource for decision-making.

Diagnosis HEP:

The resulting HEP for diagnosis from the decision tree is 0.16.

Quantification of Execution:

Entry Conditions: All the entry conditions for modeling execution of the actions are met. The area is accessible. The needed equipment and information are available. Finally, there will be adequate staff available from the MCR and OSC, as needed. The personnel will largely be following the EDMG procedure for spraying the containment atmosphere, and the needed guidance is available in the EDMGs.

Answers to decision tree questions:

1) Branch Point 1. Location of action

The actions are to be performed in various ex-MCR locations; therefore, the up branch is taken.

2) Branch Point 2. Complexity of response execution

The up branch (high complexity) is taken because the action requires multiple locations to be visited, multiple steps need to be completed and will likely involve multiple crew members coordinating activities.

3) Branch Point 3. Environmental concerns

The area and any necessary equipment and tools are accessible. Adequate lightning will be available. As discussed earlier, there are general radiological habitability concerns that may require limitation in stay time or donning of protective clothing, but there are no specific habitability concerns. There are no hazard-related concerns. Therefore, the "Good" branch is taken.

Execution HEP:

The resulting HEP for execution from the decision tree is 0.1.

Total HEP:

Total HEP for scenario #11, Action #2 = $0.16 + 0.1 - (0.16) \cdot (0.1) = 0.26 - 0.016 = \sim 0.24$.

Total HEP for ISLOCA-equivalent = 1.0 (screened out based on habitability).

E.4.11.3 Scenario #11, Candidate Action #3, Spray in containment or spray the release point

HFE Definition:

Having either succeeded or failed at spraying in containment using the EDMG strategy (for the purpose of controlling containment conditions), and in concert with the TSC, which was activated well before, operators fail to spray an external release point using the EDMG guidance. [Spraying in the containment is not considered here, since it was already considered in the previous candidate action.]

Viable Actions:

This candidate action is not further developed here but rather is left as a placeholder. There are three key reasons for this:

1. In the reference scenario, containment does not fail for nearly 7 days.
2. Containment failure due to long-term overpressure is likely to lead to a situation where a specific release point is not identifiable during the accident response.

3. Previous work done under other projects has suggested that the spray flux and fission product in-spray residence time is unlikely to lead to significant fission product capturing, for the spray flow rates and geometries of relevance here.

Feasibility Assessment:

N/A

Determination of Habitability:

N/A

Quantification of Diagnosis:

N/A

Quantification of Execution:

N/A

Total HEP:

N/A

E.4.12 Scenario #12 Basic Scenario Information

Scenario #12 is a LOCA type accident in POS 10 with the containment closed and isolated at the time of core damage. Table E-26 shows the general accident scenario timeline. Table E-27 gives the general survivability and habitability considerations.

Table E-26 Scenario #12 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	12	
Plant Operating State / Tech Spec Mode	10 / 5	"Cold" mid-loop; SG dams are installed; all three pressurizer safety valves are held open or removed
Initiating event	LOINV	Loss of inventory due to inadvertent diversion of water, with an initial leak rate of 300 gpm
Level 1 successes/failures	→	Operators successfully diagnose the loss of inventory, but fail to isolate the leak path and initiate the standby train of RHR prior to the need for makeup; operators initiate SI and charging injection 15 minutes after the initiating event; operators further close all containment penetrations and restore containment cooling soon thereafter; RWST low level occurs at ~7 hours and injection is assumed to end at this point; following loss of injection, operators are unable to isolate the leak path and re-establish RHR, or to refill the RWST
Containment status	→	The containment hatch and personnel airlock are initially closed, and containment penetrations are closed at t=35 minutes
Contain. systems	→	Containment sprays are unavailable, but one train of containment fan coolers are restored
AC power status	Avail.	
DC power status	Avail.	
Invocation of emergency classification in AOPs	< 1	AOP for RCS leakage directs initiation of emergency classification
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	9	RCS temperature greater than 200 °F for greater than 20 minutes, with containment closure established and RCS integrity not established
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	17	RPV level at top of active fuel with containment closure established
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	E.g., 30 or 49	Likely via a combination of conditions with significant uncertainty: Containment sump level rise, RPV level less than top of active fuel for greater than 30 minutes are met by 18 hours, but a containment challenge is also required; containment pressure hovers around the 3 psig threshold after ~30 hours; meanwhile, the H ₂ concentration limit is probably met soon before the 2 nd large deflagration at ~49 hours. Emergency Director discretion may also be invoked.
Time of onset of core damage ²	25	
Time of vessel breach	56	
Time of combustion event(s)	~30, 49, 90, and 135	Large combustions occur periodically as combustible gases reach MELCOR auto-ignition limits in concert with containment coolers keeping containment de-inerted; the timing of the later events should not be viewed to have any real precision.

Table E-26 Scenario #12 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of containment overpressure liner tear	n/a	Containment fan coolers prevent significant pressurization
Time of basemat melt-through	>>168	
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with "onset of core oxidation" and "spatially maximized clad temperature exceeds 1478K" are used for establishing this time.		

Table E-27 Scenario #12 General Survivability and Habitability Perspective

Timing (Top of the Hour)	General I&C and Equip. Survivability Concerns	Habitability Concerns	Accident-Altering Event ¹		
Onset of core damage					
26	There is an RCS leak path and containment closure (as opposed to Tech Spec leakage conditions), but containment pressure is being kept near atmospheric conditions, so ex-containment areas should be relatively unaffected relative to the situation in other cases analyzed in this report. However, some damage within-containment is possible from the combustions that are predicted to occur.	For the same reasons as described to the left, habitability near containment penetrations is of less concern than relative to other cases herein; no wide-spread habitability concern is expected, and cumulative noble gas releases to the environment are lower during this timeframe than for other cases			
27					
28					
29					
30			Combustion		
31					
32					
33					
34					
35					
36					
37					
38					
39					
40					
NOTE: If the original RHR failure and ensuing nature of isolation result in a pathway outside of containment, then radiological habitability in the auxiliary building and surrounding areas would be a <u>significant concern</u> .					
¹ This refers to a combustion event, hot leg creep rupture, vessel breach, containment failure, or failure of a surrounding structure.					

Scenario description:

The accident is initiated by a loss of inventory in the operating train of RHR in POS 10, associated with an unspecified diversion of water (e.g., due to a valve misalignment). The transient results in an early trip of the operating train of RHR. The RPV head is in place, the RCS is vented, and the plant is in a reduced inventory configuration.

Upon loss of inventory and the related off-normal conditions for RHR, operators enter the AOPs associated with RCS leakage and loss of RHR. In this PRA sequence operators succeed in diagnosing the loss of inventory but fail to isolate the leak path and initiate the standby train of

RHR prior to the need for makeup. Upon the need for makeup, operators successfully start ECCS injection, which keeps RCS water level high. Operators also succeed in closing any open containment penetrations and restoring one train of containment fan coolers. Operators are not able to restore RHR, align the standby train of RHR, or to refill the RWST once it is depleted.

As discussed earlier in this report, the failure to refill the RWST is currently an assumed failure in the Level 1 PRA model. The action is modeled in the Level 1 PRA event trees but is set to fail (TRUE) due to broad concerns about makeup flow rate capacity limitations and lack of proceduralized direction. Separately, containment sprays are unavailable (out-of-service), and the relevant AOPs will not direct their restoration or use. Meanwhile, one train of containment coolers has already been restored prior to core damage. Thus, at the onset of core damage, the systems that may warrant re-consideration for restoration/recovery in the Level 2 PRA HRA are the RWST (for ECCS injection), containment sprays, and/or the other train of containment fan coolers. The situation in this regard is very analogous to that discussed earlier for Scenario #1 (see Table E-8 and the discussion that immediately follows it). For the same reasons as outlined there, RWST refill and subsequent resumption of ECCS injection is the candidate action considered here.

To complete the scenario description, and as previously recorded, core damage starts at approximately 25 hours, with vessel breach occurring much later (~30 hours later) due to the low decay heat associated with this scenario's timing. Two deflagrations are predicted during the time between the onset of core damage and the time of vessel breach.

So, with this background established, the screening HRA approach can be applied to the three candidate actions (that are taken from Table E-5). This is shown in Table E-28.

Table E-28 Scenario #12 Candidate Action Screening HRA

Action	Viability?	Screening HEP	Basis for Screening HEP
Refill RWST and restore RCS injection, to arrest fuel damage	As of this screening step, these actions appear viable	0.1	This is very similar to an EOP action, is the highest perceived priority and not concurrent with an accident-altering event, and there are no obvious habitability or survivability concerns.
Spray within containment, to combat rising containment pressure		0.5	The other criteria (for HEP = 0.1, 0.9 or 1.0) do not apply.
Spray within containment or spray external release point, to reduce fission product releases		0.9	It is never the 1 st or 2 nd priority.

A detailed HRA is not documented for this case, because all the responses to the qualitative and quantitative HRA steps would be equivalent to those from Scenario #11.

E.4.13 Scenario #13 Basic Scenario Information

Scenario #5 is an SBO type accident in POS 5B with the containment hatch open at the time of core damage. Table E-29 shows the general accident scenario timeline.

Table E-29 Scenario #13 General Accident Timeline and Provisional EALs

Item	Value / Time (hr)	Cause / Notes
Representative Scenario #	13	
Plant Operating State / Tech Spec Mode	5B / 5	RCS level reduced near reactor head flange; all three pressurizer safety valves are held open or removed
Initiating event	SBO	(Station blackout)
Level 1 successes/failures	→	Operators fail to close the containment hatch prior to hazardous environmental conditions, and to inject into the RCS
Containment status	→	Initially open and un-isolated; and operators fail to successfully close containment
Contain. systems	→	Containment sprays and coolers are unavailable due to SBO conditions
AC power status	Unavail.	
DC power status	→	Available early, but unavailable due to battery depletion well in advance of core damage; safety-related batteries are assumed to deplete at 4 hours, consistent with the at-power PRA
Invocation of Emergency Classification in AOPs	< 1	AOPs for loss of electrical systems and loss of RHR direct initiation of emergency classification.
ALERT ¹ conditions met (activate TSC, OSC, and EOF) - <i>Preliminary estimate</i>	< 1	Loss of all offsite and onsite AC sources for greater than 15 minutes
SITE AREA EMER. ¹ conditions met - <i>Preliminary estimate</i>	1	Declaration of site area emergency (SAE) is difficult to predict, in that all indication is lost after 4 hours, preventing further assessment of plant conditions (though the open containment hatch will make it obvious that significant inventory loss is occurring). Here, it is assumed that SAE is declared at 1 hour, under Emergency Director discretion, when steam would be exiting the open hatch.
GENERAL EMERGENCY ¹ conditions met - <i>Preliminary estimate</i>	5	Similar to the above, there is a high degree of uncertainty upon when the declaration would be made under blackout conditions (neither SAE nor GE have a criterion during shutdown equivalent to the Alert criterion cited above). Here, it is assumed that GE is declared via Emergency Director discretion based on extended boiling with (i) no compensating injection and (ii) containment open. At around five hours, indications of core damage are starting, which will also lead to increasing radiation doses near the open hatch. Earlier or later declaration is certainly plausible given the extended lack of all instrumentation.
Time of onset of core damage ²	7	
Time of vessel breach	14	
Time of combustion event(s)	n/a	No significant combustion events occur.
Time of containment overpressure liner tear	n/a	The containment opening prevents long-term overpressure failure.

Table E-29 Scenario #13 General Accident Timeline and Provisional EALs (cont.)

Item	Value / Time (hr)	Cause / Notes
Time of basemat melt-through	n/a	Cavity erosion is occurring very slowly.
¹ Note that this accident occurs in Tech Spec Mode 5 or 6, so the shutdown-specific emergency action level matrix is the applicable matrix. ² As discussed in Section E.3.4, core-exit thermocouples will not be available for all shutdown states, but it is assumed that in these cases an alternate measurement of temperature or core damage assessment based on RPV level time history is used when needed. The MELCOR timings associated with “onset of core oxidation” and “spatially maximized clad temperature exceeds 1478K” are used for establishing this time.		

E.5 Summary of Post-Core-Damage HRA Approach and Application

This section summarizes the post-core-damage HRA approach and results for the shutdown Level 2 PRA. Section E.1 gives an introduction and background on the approach development. Section E.2 explains the operational situation post-core-damage for reactor at-shutdown events and develops an approach to identifying the candidate actions that could be pursued by the plant staff in their attempts to halt the accident. Section E.3 addresses modifications to the at-power Level 2 HRA approach needed for its application in the reactor at-shutdown situation. Section E.4 documents the application of a screening and detailed HRA approach for each candidate action, for each scenario. The final human error probabilities stemming from this work are catalogued in Table E-30.

With this set of analyzed HEPs established, one can re-visit the categorization provided in Table E-3, in concert with the mapping between categories discussed at the beginning of Section E.4, to arrive at a more generic set of HEPs for use in the shutdown Level 2 PRA model. These are captured in Table E-31.

Table E-30 Summary of Developed HEPs

Scenario	Candidate Action	Screening HEP	Detailed HEP
#1: LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection, to arrest fuel damage	0.1	0.14
	Spray within containment, to combat rising containment pressure	0.5	0.24
	Spray within containment or spray external release point, to reduce fission product releases	0.9	n/a
#2: LOCA accident type, containment closed & isolated	Core damage was not predicted within 7 days in this scenario		
#3: Station blackout, containment un-isolated	No credit for post-core-damage human actions is considered		
#4: Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
#5: Station blackout, containment open	No credit for post-core-damage human actions is considered		
#6: Transient accident type, RPV not vented, containment open	Core damage was not predicted within 7 days in this scenario		
#7: Station blackout, containment open	No credit for post-core-damage human actions is considered		
#8: Transient accident type, RPV not vented, containment closed & isolated	No clear preferential action identified for continued attempts to accomplish pre-core damage injection/recovery	n/a	n/a
	Inject into SGs ¹	0.9	0.24
	Open head vents ²	0.9	0.17
#9: Station blackout, containment closed & isolated	No credit for post-core-damage human actions is considered		
#10: LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection, to arrest fuel damage	0.1	0.14 ³
	Spray within containment, to combat rising containment pressure	0.5	0.58 ⁴
	Spray within containment or spray external release point, to reduce fission product releases	0.9	n/a
#11: LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection, to arrest fuel damage	0.1	0.14 ³
	Spray within containment, to combat rising containment pressure	0.5	0.24 ⁴
	Spray within containment or spray external release point, to reduce fission product releases	0.9	n/a

Table E-30 Summary of Developed HEPs (cont.)

Scenario	Candidate Action	Screening HEP	Detailed HEP
#12: LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection, to arrest fuel damage	0.1	0.14 ³
	Spray within containment, to combat rising containment pressure	0.5	0.24 ⁴
	Spray within containment or spray external release point, to reduce fission product releases	0.9	n/a
#13: Station blackout, containment open	No credit for post-core-damage human actions is considered		

¹ This action is only relevant prior to installation of the SG dams (which occurs during POS 6) or after removal of the SG dams (which occurs during POS 10).

² This action is only relevant for situations where hot leg nozzle creep rupture has not occurred.

³ This would increase to 0.52 for the corresponding ISLOCA variant.

⁴ This would increase to 1.0 for the corresponding ISLOCA variant.

Table E-31 Final Set of HFEs and HEPs for Scenario Categories

Cat.	Description	HFE & HEP #1	HFE & HEP #2	HFE & HEP #3
I	Transient accident type, RPV not vented, containment closed & isolated	n/a	Inject into SGs ¹ – 0.24	Open head vents ² – 0.17
II	Transient accident type, RPV not vented, containment open	None of the MELCOR scenarios that proceeded to core damage covered this situation		
III	Transient accident type, RPV vented or flooded, containment closed and isolated	The Category VI results can be directly applied here.		
IV	Transient accident type, RPV vented or flooded, containment open or un-isolated	The Category VI results can be extended to cover this grouping, using the ISLOCA outcomes (similar radiological habitability concerns): • RWST refill and ECCS injection: HEP of 0.52 • Containment spray: HEP of 1.0		
V	SBO	n/a	n/a	n/a
VI	LOCA accident type, containment closed & isolated	Refill RWST and restore RCS injection – 0.14 ³	Spray in containment - 0.24 to 0.58 ⁴	Spray int/ext containment – n/a
VII	LOCA accident type, containment open or un-isolated	The Category VI results can be extended to cover this grouping, using the ISLOCA outcomes (similar radiological habitability concerns): • RWST refill and ECCS injection: HEP of 0.52 • Containment spray: HEP of 1.0		

¹ This action is only relevant prior to installation of the SG dams (which occurs during POS 6) or after removal of the SG dams (which occurs during POS 10).

² This action is only relevant for situations where hot leg nozzle creep rupture has not occurred.

³ This would increase to 0.52 for the corresponding ISLOCA variant.

⁴ The lower end of this range represents head-on conditions, and the upper end represents head-off conditions, with the distinction being the amount of ongoing containment leakage that might affect habitability near the connection point. This would increase to 1.0 for the corresponding ISLOCA variant.

APPENDIX F

COMBUSTION MODELING SUPPLEMENT TO THE LEVEL 3 PROBABILISTIC RISK ASSESSMENT PROJECT

F.1 Background, Objectives, and Outline

This appendix documents the approach for combustion modeling in the shutdown Level 2 PRA. Energy Research, Inc. (ERI) has supported the NRC in the development and quantification of the Level 2 PRA logic models. This consisted, in part, of preparing logic models, phenomenological analysis, and event quantification of various issues including hydrogen and carbon monoxide generation and combustion in the containment. The core of ERI's modeling approach consisted of the use of the ERPRA-BURN probabilistic combustion simulation code to analyze various combustion scenarios and quantify case logic and basic event failure probabilities for the Level 2 PRA containment event tree (CET) for at-power internal events and floods. This approach is described in the at-power Level 2 PRA report (NRC, 2022a) and its appendices. The same approach was later used, with different scenarios, for the low-power and shutdown (LPSP) portion of the PRA.

Later, an uncertainty analysis was performed on the LPSP PRA, which included attaching uncertainty distributions to many of the basic event probabilities in the model which had previously been defined only using point values. In addition, it was realized at this juncture that the LPSP combustion analysis included some corrections and enhancements that were missing from the at-power analysis, and it was decided that the at-power basic event quantification would be updated to be consistent with the methodology used for the LPSP. This appendix was developed to supplement the documentation and provide an overview of the combustion modeling approach that was used in both the at-power PRA and the LPSP PRA.

The objectives of the appendix are to compile the fundamental information and results regarding the combustion analysis (e.g., all basic event probabilities as quantified) in one document, provide clarity on some aspects of the methodology, and provide some high-level overview of insights and findings discovered during the various phases of the combustion study.

Section F.2 of this appendix provides an overview of how the combustion aspects of the Level 2 PRA event progression are decomposed and modeled. Section F.3 discusses how combustion may be modeled in general for Level 2 PRA applications and gives some rationale for why ERPRA-BURN was considered appropriate for this task. Section F.4 briefly summarizes the approach used by ERPRA-BURN in analyzing combustion event progression and failure probabilities. Section F.5 is a compendium of all scenarios and input parameters used in ERPRA-BURN, with sources for the parameter values used, all the ERPRA-BURN calculation results, and a discussion of how the point values and uncertainty distributions for all basic events related to combustion were treated and quantified in the Level 2 PRA. Finally, some insights and conclusions are given in Section F.6.

F.2 Combustion Modeling in the Level 2 PRA

Oxidation of metals in the core, as well as molten core-concrete interactions following vessel breach, can result in the generation and transmission into the containment of combustible gases (hydrogen and carbon monoxide) during a severe accident. The potential burning of these gases can pose a challenge to the integrity of the containment, resulting in its failure; or, in

some studies, can be considered to have other effects in the post-core-damage event progression (e.g., combustion events damaging filters or other containment systems).

In Level 2 PRAs, the combustion event progression is generally 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, molten core-concrete interaction (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 CET and its decomposition event trees decompose the question of combustion into three to four top events for each of the three time frames. Figure F-1 shows a simplified representation of how combustion is resolved for the very early or early time frames:

- 1-L2-H2IGNSRC-*time* (where time is either “VE”, “E”, or “L” to denote the time phase of the accident, e.g., 1-L2-H2IGNSRC-VE): Asks whether or not there is a source of ignition in the containment in a given time phase, whether from sparks generated by machinery or moving metal parts or random ignition. The success branch here indicates no ignition source is present, and therefore downstream questions related to combustion in this time phase are bypassed (i.e., no ignition or burn-induced containment failure can happen if there is no cause for ignition in the first place). The failure branch indicates that there is a source of ignition in containment in this time phase.
- 1-L2-H2IGN-*time* (where time is either “VE”, “E”, or “L” to denote the time phase of the accident): Asks whether or not ignition and combustion actually occur, conditional on there being a source of ignition in this time phase of the accident. It is bypasses if there is no potential source of ignition in the same time phase. Even if there is an ignition source, ignition might not occur, whether because there is insufficient combustible gas to

burn; or there is insufficient oxygen to render the mixture flammable; or the mixture is inert to combustion due to high concentrations of steam and/or carbon dioxide. Success at this top event means that there is no ignition; in that branch, the downstream question of burn-induced containment failure is obviously bypassed. The failure branch means that one or more burns take place in this time phase.

- 1-L2-H2CF-time (where time is either “VE”, “E”, or “L” to denote the time phase of the accident): Asks whether or not containment fails as the result of combustion in a particular time frame, assuming one or more burns actually occur. The top event is bypassed in case there is either no ignition source in the containment, or there is an ignition source, but no actual ignition takes place. 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.

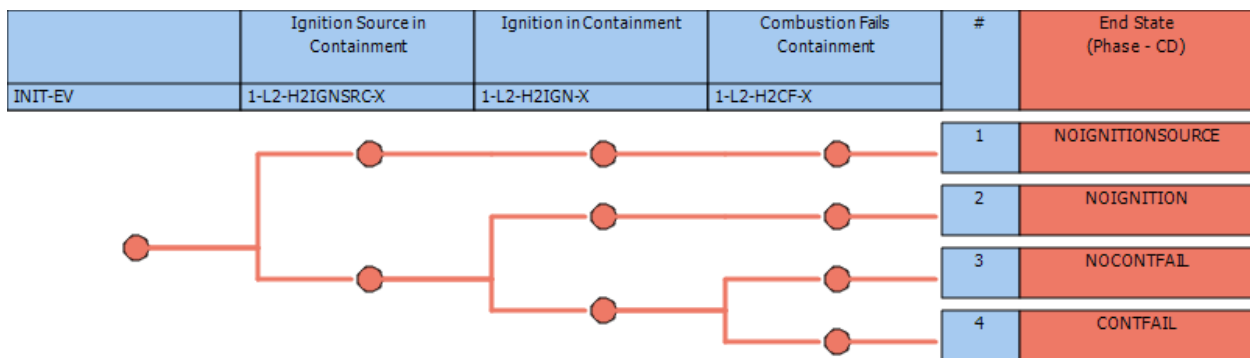


Figure F-1 Simplified Decomposition of Events for Very Early and Early Combustion Modeling

The decomposition for the late time frame adds one extra top event in between the questions of ignition and containment failure Figure F-2:

- 1-L2-H2DET-L: Asks, given ignition in the late time frame, whether or not the burn undergoes a deflagration-to-detonation transition (DDT). 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. Success at this branch indicates that there is no DDT; however, the question of global deflagrations remains, and therefore top event 1-L2-H2CF-L is still asked. Failure at this branch means that there is a detonation event, which the model currently assumes results in automatic containment failure. The failure branch here bypasses 1-L2-H2CF-L, because it is routed directly to a DET end state; this aids the analyst if they wish later to determine which sequences are containment failures as a result of detonation versus deflagration.

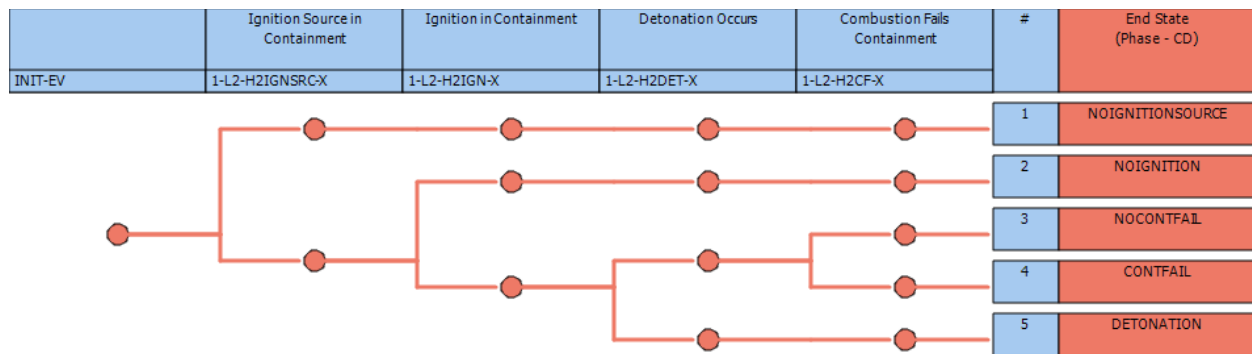


Figure F-2 Simplified Decomposition of Events for Late Combustion Modeling

Linkage rules in the decomposition event trees of the Level 2 PRA model serve to assign a basic probability to these top events depending on accident initial conditions or other prior events, or in some cases to yield either automatic success or failure.

- Basic events assigned to 1-L2-H2IGNSRC-*time* are assigned a name of the form 1-L2-BE-H2IGNSRC-*time-condition-model*. (In some cases, the portion “H2IGNSRC” is abbreviated to “H2IGSRC” or “H2IGSC” due to SAPHIRE-imposed limits on the length of a basic event name). Here, time is either “VE”, “E”, or “L” to denote the time phase of the accident; condition is an abbreviation for a particular condition in the case logic of the top event’s linkage rules (e.g., “CHRPB” for sequences with containment heat removal and one or more prior burns); and model is either omitted (for the at-power internal events PRA model), “SD” (for the shutdown model), or “EXT” (for the external events model). For example: 1-L2-H2IGNSRC-VE-NAC-SD is the basic event assigned to the probability of a very early ignition source in containment in the shutdown portion of the PRA model for the case with no AC power (“NAC”).
- Basic events assigned to 1-L2-H2IGN-*time* are assigned a name of the form 1-L2-BE-H2IGN-*time-condition-model*, otherwise with the same naming scheme described above.
- Basic events assigned to 1-L2-H2DET-L are assigned a name of the form 1-L2-BE-H2DET-L-*condition-model*.
- Basic events assigned to 1-L2-H2CF-*time* are assigned a name of the form 1-L2-BE-H2CF-*time-condition-model*.

The different conditions used in determining the case logic for a particular top event varies from time to time and event to event, but currently includes such elements as the accident initiator or type; whether or not there is AC power available; whether or not active containment heat removal (CHR) systems are operating; and whether or not any prior combustion events have occurred in earlier time phases of the accident for this sequence (i.e., a failure of 1-L2-H2IGN-*time* for an earlier time phase). For a detailed list of all basic events used in the combustion portion of the PRA model, see Table F-5 through Table F-8 of this appendix.

F.3 Methods of Combustion Quantification

The MELCOR system-level severe accident simulation code (Humphries, 2015a-c) is often used to aid in quantifying various aspects of severe accident progression and phenomenological

impacts for Level 2 PRAs, and the code includes models for core oxidation and MCCI generating combustible gases, gas transport, and deflagration. However, MELCOR predictions cannot generally be used directly in quantifying combustion issues for a Level 2 PRA, since the code by nature is deterministic; it uses fixed values for various parameters governing inherently uncertain issues such as the extent of in-vessel core oxidation or the structural fragility of the containment. Furthermore, ignition is generally assumed to occur at a fixed, specified combustible gas generation (10 percent by default), even though random ignition is a highly stochastic phenomenon, and one can certainly envision random ignition occurring at lower or higher values. In the case of many large dry pressurized-water reactor (PWR) containments, a global deflagration at 10 percent 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 impossible for ignition to be delayed to slightly later in time when combustible gas concentrations are higher. Some PRAs base their combustion analysis on MELCOR-predicted pressure loads from combustion events and then simply assign an arbitrary uncertainty distribution that allows the load to be somewhat lower or higher; however, such arbitrary assignment lacks much basis.

The probabilistic ERPRA-BURN code was 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 percent. 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 percent, the results are less optimistic than one would obtain from a deterministic MELCOR calculation. For example, if a burn at 12 percent 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 percent or greater.

F.4 ERPRA-BURN Probabilistic Framework

ERPRA-BURN is a code developed by ERI to analyze the likelihood and consequences of combustion in a large dry PWR containment within a probabilistic framework. Important features of the code include:

- Combustion is analyzed in a simple Monte Carlo fashion through the use of thousands of “combustion histories”, each of which represents one possible course of the accident

progression from the standpoint of time-dependent sources of gas generation and random timing of ignition sources and ignition consequences.

- Each history is divided into three time phases: Phase I (very early – before vessel breach), Phase II (early – at or around the time of vessel breach), and Phase III (late – the long term after vessel breach until the end of the calculation). Phase I is characterized by an assumed linearly increasing hydrogen mass in containment due to in-vessel Zr and/or steel oxidation during core damage. Phase II is characterized by a sudden ignition source at vessel breach, potentially accompanied by a higher base pressure due to the containment transient. Phase III is characterized by a source of hydrogen, carbon monoxide, and carbon dioxide due to ex-vessel Zr oxidation and MCCI, and also potentially time-dependent containment conditions (e.g., rising containment base pressure).
- A separate parallel “history set” is run for each relevant combination of assumptions regarding existence of an ignition source. These include: set 1 (ignition source present very early and late), set 2 (ignition source present early and late), set 3 (ignition source present in all time phases), and set 4 (ignition source present late only). A separate set of combustion histories is calculated independently for each history set. The use of history sets is required as a form of Monte Carlo weighting in case of very low or high ignition source probabilities; for example, if the ignition source probability is 0.99 in Phase I, simple Monte Carlo results would include very few samples without ignition in Phase I, which would make it difficult to accurately quantify CET results for cases without such prior combustion. Combustion history set results are weighted in the final ERPRA-BURN results to preserve their relative likelihoods.
- Ignition sources are randomly simulated in each history, under the assumption that the rate of such sources per unit time is constant and consistent with the user-specified cumulative source probability. This implies an exponential probability distribution in the duration of time from one ignition source to the next, which varies randomly from burn to burn and from history to history. Multiple burns are possible in a given time phase, depending on the random number generation. Time-to-next-source is weighted to be consistent with the history set assumptions (e.g., in history set 4 (late sources only), there must be at least one ignition source event in Phase III, but no sources are allowed in Phases I and II).
- Cumulative gas generation from in- and ex-vessel oxidation and MCCI can alter the containment mixture in between ignition source events, as can evolving base conditions (e.g., time-dependent rising pressure during Phase III) and the reaction products of previous burns (e.g., depleted carbon monoxide and oxygen and increased carbon dioxide).
- When an ignition source is randomly indicated, conditions in the containment at that point in time are evaluated to see whether the mixture is flammable (i.e., sufficiently high concentration of oxygen and combustible gases, and sufficiently low concentration of inertants).
- If the mixture is flammable at the instant of an ignition source, ignition is predicted, and the consequences are predicted in terms of peak burn pressure load through use of an adiabatic isochoric complete combustion (AICC) calculation. (Note: Although this is

termed AICC by convention, ERPRA-BURN can model incomplete combustion at low combustible gas concentrations. Also, various input parameters can alter the “adiabatic” assumption, although all base case calculations performed for this study assume no heat loss.)

- Peak burn pressures are compared against a containment fragility curve to obtain the individual containment failure probabilities of each burn. Cumulative containment failure probabilities (CCFPs) for each history are obtained through multiplication over successive burns, if applicable (i.e., the cumulative probability of containment not failing is equal to the product of it not failing at each burn).
- 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.
- ERPRA-BURN results can be collated conditional on whether or not a previous burn was indicated (e.g., what is the probability of a burn occurring in Phase II, given that at least one burn already occurred in Phase II [thus depleting some of the available hydrogen]?).

Later sections in this appendix discuss the assignment of input parameters to the ERPRA-BURN code; the results generated by the code; and how these results were used in quantifying linkage rules and basic event failure probabilities in the Level 2 PRA model.

F.5 ERPRA-BURN Results and CET Assignments

F.5.1 ERPRA-BURN Scenario Formulation

Sets of scenarios were developed for combustion analysis by ERPRA-BURN, for accidents at full power and at shutdown. The seven scenarios arrived at are listed and described in Table F-1. Generally, each ERPRA-BURN scenario has an accompanying MELCOR scenario from the PRA deterministic analysis that was used to aid in formulating input values to the code.

In establishing a set of scenarios for combustion analysis, it is as important not to have too many as not to have too few. An overly wide range of scenarios may give a false impression of variance in combustion behavior that is actually the result of incidental differences between the outputs of the MELCOR calculations. From past experience in combustion modeling for Level 2 PRAs, the following two factors were found to be most important:

- 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, so often a single MELCOR scenario can be used as the basis for two ERPRA-BURN scenarios with and without AC power assumed. (In fact, this is preferable, since the results will then show the influence of AC power alone without any incidental factors.)

Table F-1 ERPRA-BURN Scenario Descriptions

Scenario Label	Description
S1A1	A short-term station blackout (SBO) at full power, simulated in MELCOR as the scenario of the same name in the deterministic analysis for the full-power PRA. This scenario is used to quantify combustion for all sequences with AC power unavailable.
S1A1-S1	A variant of SBO scenario S1A1 in which AC power is assumed to be available (i.e., a transient with AC power available but all relevant safety systems failed by other causes). This scenario was not simulated in MELCOR; the ERPRA-BURN scenario simply consists of assumptions identical to that for S1A1 but with the ignition source probabilities changed to reflect that AC power is available to generate sparks in the containment. This scenario is used to quantify combustion for all sequences with AC power available but late containment heat removal (CHR) unavailable.
S6B	A transient scenario with AC power available at full power, in which containment heat removal systems (both sprays and fan coolers) operate in the late time frame to keep pressure and steam concentration low. This was simulated in MELCOR as the scenario of the same name in the deterministic analysis for the full-power PRA. It is used to quantify combustion for all sequences with late CHR available.
LPSD1	A transient scenario with AC power available at shutdown, denoted MELCOR scenario SD01.1 in the deterministic analysis for the LPSD PRA (except that this ERPRA-BURN scenario assumed AC power available, which is reflected in the ignition source probabilities input to the calculation). This ERPRA-BURN scenario is used to quantify combustion behavior for all shutdown sequences which have AC power available but no CHR, and which are not initiated by a LOINV.
LPSD2	A short-term SBO scenario at shutdown, denoted MELCOR scenario SD10.1 in the deterministic analysis. It is used to quantify the combustion behavior for all shutdown sequences without AC power.
LPSD3	A loss-of-inventory (LOINV) accident at shutdown with AC power available but no active late CHR, denoted MELCOR scenario SD10.1 in the deterministic analysis. It is used to quantify the combustion behavior of all LOINV sequences without late CHR. Due to the large quantity of water dumped into the containment as a result of the LOINV, steam generation and pressure rise in the containment over the long term are significantly slowed, resulting in containment conditions somewhat intermediate between those of a CHR scenario and a typical non-CHR scenario (such as the SBO).
LPSD4	A transient accident at shutdown with containment coolers operating successfully in the late time frame to keep pressure and steam concentration low in the containment. This scenario is denoted SD08 in the MELCOR-based deterministic analysis for the LPSD PRA. It is used to quantify the combustion behavior for all sequences with long-term CHR.

- 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 such as the ones at shutdown. There is often a sharp difference in behavior between scenarios with late 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 (sprays and/or fan coolers) and one without. In the case of the shutdown model, it was found that loss-of-inventory (LOINV) accidents can have an intermediate effect even without CHR simply due to the sheer mass of coolant dumped into the containment with its large thermal inertia, and therefore a separate LOINV scenario was added for the shutdown analysis.

F.5.2 ERPRA-BURN Input Parameter Assignments

For reference, Table F-2 lists the values of all ERPRA-BURN input parameters that have common values shared by all scenarios considered in the PRA study, while Table F-3 lists the input parameters that are scenario-dependent. Generally, the scenario-dependent parameter values are set by extracting results from a corresponding MELCOR calculation. The input parameters to ERPRA-BURN are listed below along with a discussion of the sources used in formulating their input values.

Number of combustion histories per history set: Number of Monte Carlo samples used in simulating combustion histories in each history set. Larger values reduce the uncertainty in the results, with correspondingly higher memory and CPU requirements. It was found that 3000 samples were an adequate balance for most purposes, giving adequate accuracy down to CCFPs of 0.001 or less. (Note: From the ERPRA-BURN results, all significant results were found to be much larger than 0.001, with the exception of early time phase CCFPs which were sometimes less than 1×10^{-4} and drastically increasing the number of samples in an attempt to get better resolution of such small CCFPs was not thought to be useful.)

Spray model: This is used in ERPRA-BURN solely to pick which correlation to use for combustion completeness. In the current set of scenarios, only LPSD4 uses the “sprays on” option. (S6B consists of heat removal only by fan coolers, not sprays.)

Accident timing: Times and durations from the start of the accident are extracted from MELCOR and input on a scenario-specific basis. Note that the combustion history simulation process is dimensionless with respect to time; none of the probabilities output by the code are affected by these inputs. They are present only to output some reference times for purposes of further understanding of the results. Of these, only the values “Average Time of Phase III Burn (hr)” and “Latest Time of Phase III Burn (hr)” are exposed in the ERPRA-BURN graphical user interface, although others can be revealed in debug mode if required. For example, it may be useful to know that most late burns in the case of SBO scenarios occur in a narrow window of a few hours following vessel breach.

Pre-accident containment conditions: Pressure, temperature, and humidity of the containment atmosphere are obtained from the MELCOR base input deck initial conditions.

Table F-2 ERPRA-BURN Input Assignment – Common Parameters

Input Parameter		Value
Number of combustion histories per history set		3000
Pre-accident containment pressure		1.034 bar-abs
Pre-accident containment temperature		322.0 K
Pre-accident containment relative humidity		1.0
In-vessel Zr oxidation fraction uncertainty distribution	Cumulative probability	Oxidation fraction
	0.00	0.20
	0.05	0.35
	0.10	0.42
	0.50	0.52
	0.80	0.60
	0.99	0.70
		0.75
MCCI rate inflection exponent		1.0
Hydrogen mass from 100% Zr oxidation		986.9 kg
Hydrogen transmission fraction to containment		1.0
Containment free volume		77871.0 m ³
Containment fragility uncertainty distribution	Cumulative probability	Containment failure pressure (bar-abs)
	0.00	7.24
	0.01	8.21
	0.10	9.17
	0.50	10.55
	0.95	11.65
	0.99	12.07
		12.48
Concentration for 0% burn completeness, with sprays		0.03639
Concentration for 100% burn completeness, with sprays		0.07573
Concentration for 0% burn completeness, without sprays		0.04295
Concentration for 100% burn completeness, without sprays		0.0831
Minimum flammable O ₂ concentration		0.05
Inerting concentration for H ₂ O + CO ₂		0.55
Hydrogen heat of combustion		121000 kJ/kg
CO heat of combustion		10107.14 kJ/kg
Burn efficiency		1.0

Table F-2 ERPRA-BURN Input Assignment – Common Parameters (cont.)

Input Parameter	Value
Coefficient for CO in combustible gas concentration	0.541
Minimum detonable concentration	1.0

Table F-3 ERPRA-BURN Input Assignment – Scenario-Dependent Parameters

Input Parameter	Scenario						
	S1A1	S1A1-S1	S6B	LPSD1	LPSD2	LPSD3	LPSD4
Spray model	Off	Off	Off	Off	Off	Off	On
Time of start of H ₂ generation (hr)	16.7	16.7	15.0	14.86	14.86	9.33	27.25
Time of end of H ₂ generation (hr)	19.8	19.8	17.8	18.00	18.00	12.98	31.36
Time of vessel failure (hr)	19.8	19.8	17.8	23.83	23.83	19.74	32.66
Time of start of MCCI (hr)	21.1	21.1	18.3	23.83	23.83	19.74	32.66
Duration of ex-vessel Zr oxidation (hr)	31.3	31.3	29.96	89.0	89.0	128.0	75.0
Time of end of calculation (hr)	82.4	82.4	48.25	168.0	168.0	168.0	168.0
Pre-burn containment pressure, Phase I (bar-abs)	1.9	1.9	1.23	1.503	1.503	1.499	1.034
Reference mass of H ₂ for Phase I (kg)	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pre-burn containment pressure, Phase II (bar-abs)	2.69	2.69	1.63	1.606	1.606	1.551	1.034
Reference mass of H ₂ for Phase II (kg)	822.0	822.0	856.0	643.0	643.0	509.0	0.0
Pre-burn containment pressure, start of Phase III (bar-abs)	2.74	2.74	1.42	1.606	1.606	1.551	1.034
Reference mass of H ₂ for start of Phase III (kg)	822.0	822.0	856.0	643.0	643.0	509.0	0.0
Pre-burn containment pressure, end of Phase III (bar-abs)	8.55	8.55	2.03	9.14	9.14	6.658	1.034
Reference mass of H ₂ for end of Phase III (kg)	4442.2	4442.2	3582.0	3732.0	3732.0	4202.0	0.0
Reference mass of CO for end of Phase III (kg)	5726.3	5726.3	4096.0	4750.0	4750.0	5599.0	0.0
Reference mass of CO ₂ for end of Phase III (kg)	4359.4	4359.4	155.5	3000.0	3000.0	1300.0	0.0
Pre-burn containment relative humidity, Phase I	0.4	0.4	0.15	1.0	1.0	0.3	0.3
Pre-burn containment relative humidity, Phase II	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Pre-burn containment relative humidity, Phase III	0.3	0.3	0.01	0.45	0.45	0.1	0.1
Total H ₂ gas generation (kg)	4442.2	4442.2	3582.0	3732.0	3732.0	4202.0	4630.0
Total CO gas generation (kg)	5726.3	5726.3	4096.0	4750.0	4750.0	5599.0	6451.0
Total CO ₂ gas generation (kg)	4359.4	4359.4	155.5	3010.0	3010.0	1250.0	1300.0

Table F-3 ERPRA-BURN Input Assignment – Scenario-Dependent Parameters (cont.)

Input Parameter	Scenario						
	S1A1	S1A1-S1	S6B	LPSD1	LPSD2	LPSD3	LPSD4
Reference mass of H ₂ from in-vessel Zr oxidation (kg)	0.0	0.0	0.0	518.0	518.0	432.0	441.0
Reference mass of H ₂ from in-vessel Zr and steel oxidation (kg)	0.0	0.0	0.0	643.0	643.0	509.0	459.0
Ignition source probability, Phase I	0.1	0.99	0.99	0.99	0.1	0.99	0.99
Ignition source probability, Phase II	0.5	0.99	0.99	0.99	0.5	0.99	0.99
Ignition source probability, Phase III	0.3	0.99	0.99	0.99	0.3	0.99	0.99

Reference pre-burn containment conditions: Containment pressure (air and steam only) is simplistically assumed in ERPRA-BURN to be constant throughout Phase I; constant (generally at a different value) throughout Phase II; and then linearly varying with time in Phase III between a starting and ending value. The base pressure prior to combustion at any time can be obtained as the air and steam pressure from this input, plus the partial pressure of hydrogen, carbon monoxide, and carbon dioxide tracked as the result of gas generation and combustion for a particular history. For these input parameters, the user inputs the containment pressure in Phase I, Phase II, and the start and end of Phase III. For each pressure, one also inputs the reference masses of hydrogen, carbon monoxide, and carbon dioxide included (so that the ERPRA-BURN input processor can subtract those partial pressures to back-calculate the pressures of air and steam only). When extracting the data from MELCOR, one simply picks these four points in time and provides the calculation's pressure and gas component masses. In some cases, however, alternate formulation is more convenient. For example, in LPSD3 (shutdown scenario with late CHR), it is observed that the late containment temperature is constant, as regulated by the fan coolers. Therefore, we know that the containment pressure of air and steam should be constant at about atmospheric value in all time phases, excluding masses of non-condensable gases generated (i.e., all reference masses zero, and all pressures 1.034 bar-abs).

Pre-burn relative humidity: The relative humidity of the containment atmosphere is assumed by ERPRA-BURN to be constant throughout each time phase (although generally different from one phase to the next). These values are also extracted from MELCOR results and input to ERPRA-BURN, usually as approximate averages truncated to one significant digit. Older sets of ERPRA-BURN results used the original code assumption that relative humidity is equal to 1.0 at all times. This assumption was found to be a poor one as a result of the LPSD combustion analysis, and the full-power combustion analysis was retroactively revised at the time of the PRA uncertainty analysis.

In-vessel Zr oxidation uncertainty distribution: These data points represent a cumulative uncertainty distribution with respect to the fraction of core Zr oxidized before vessel breach, which is used in the Monte Carlo process to randomly sample the amount of in-vessel hydrogen generated in Phase I of a combustion history. The generic uncertainty distribution previously developed by ERI for use in other hydrogen studies and was verified by comparison to the reference plant-specific MELCOR results to be generally representative.

Total gas generation: Here are input the cumulative amounts of hydrogen, carbon monoxide, and carbon dioxide generated (in- and ex-vessel) through the end of the accident. They are obtained directly from the results of the corresponding MELCOR scenario. The option exists to use a uniform uncertainty distribution, between specified minimum and maximum values for each; but this feature was not used for the base case ERPRA-BURN scenarios.

MCCI inflection exponent: This is an optional parameter to control the shape of the time-dependent rate of ex-vessel Zr oxidation. By default, the value is 1.0, which implies that ex-vessel Zr oxidation occurs at a constant rate over the course of the previously user-specified duration. (A value of 0.5 implies parabolic shape, etc.)

Hydrogen mass from 100% Zr oxidation: This is related by simple stoichiometry to the total mass of Zr in the core (i.e., total mass of hydrogen equals total Zr mass multiplied by (4/91.2)).

Transmission fraction to containment: By default (parameter value 1.0) all hydrogen generated in-vessel in ERPRA-BURN is immediately transported to the containment. If the value of the parameter is less than 1.0, then the remainder will be held up in the RCS and instantaneously released to the containment at the start of Phase II (early time phase). The effect of this parameter on the results is generally found to be slight, so it was not thought worth performing sensitivity analysis on.

Reference mass of hydrogen from in-vessel Zr and Zr + steel oxidation: These are optional parameters to allow for some hydrogen to be generated through in-vessel oxidation of steel in addition to Zr. By default (both parameters equal to zero), no steel oxidation is allowed. If data from a particular MELCOR calculation supports that there is significant in-vessel oxidation of steel, those results can be used to input the cumulative amount of hydrogen generation from Zr alone and from Zr plus steel combined; the code will then add an amount of hydrogen from steel oxidation in the same proportion, based on the sample-dependent amount of hydrogen generated in-vessel from Zr. In general, a significant amount of steel oxidation was found to occur only in the MELCOR calculations for accidents at shutdown, so it is employed there.

Containment total free volume: This value is chosen to be consistent with the total free volume of all containment CVs in the MELCOR model.

Containment fragility uncertainty distribution: The points in this table represent a cumulative uncertainty distribution for failure of the containment at static pressures at or below a certain value. The source is the fragility distribution that was prepared for the reference plant Individual Plant Examination submittal to the NRC.

Ignition source probabilities: These are input by assumption, based on the discussion in Section F.5.4 .

Combustion completeness parameters: For each of the cases with and without operating containment sprays, combustion is assumed to be 0 percent complete (i.e., not flammable at all) at and below a certain combustible gas concentration, 100 percent complete above another gas concentration, and linearly varying in between.

Minimum oxygen concentration: The gas mixture in containment is assumed to be no longer flammable at oxygen concentrations below this limit. The default value (0.05) is consistent with MELCOR's combustion models.

Maximum inertants concentration: The gas mixture in containment is assumed to be rendered inert to combustion at a combined steam and carbon dioxide concentration at or above this value. The default value of the parameter (0.55) is consistent with MELCOR's combustion models.

Heats of combustion: Basic physical parameters for the heats of reaction of hydrogen or carbon monoxide with oxygen. The exact values were drawn from MELCOR defaults.

Burn efficiency: A default value of 1.0, as used in these calculations, implies that all the heat generated by combustion is added directly to the containment. Values less than 1.0 might be used, for example, to examine possible heat absorption by solid structures during combustion, but that option was not explored in this study.

Coefficient of carbon monoxide concentration: A Le Chatelier coefficient, with default value of 0.541, which is used in calculating the effective concentration of combustible gases (i.e., effective combustible concentration equals concentration of hydrogen plus 0.541 times the concentration of carbon monoxide).

Minimum detonable concentration: ERPRA-BURN contains no detailed detonation models, being mainly a code to calculate the probabilistic consequences of slow global deflagrations. However, the option exists to assume that burns at an effective combustible gas concentration above a certain value undergo a deflagration-to-detonation transition (DDT), in which case ERPRA-BURN assumes automatic containment failure. The default value is 1.0, meaning that no detonation is ever considered. For the purposes of some sensitivity cases discussed later, the value was lowered to 0.15 (i.e., burns at 15% or higher combustible gas concentration are assumed to result in DDT).

F.5.3 ERPRA-BURN Summary of Results

Table F-4 shows the relevant ERPRA-BURN results for each of the seven combustion scenarios formulated in Section F.5.1. Rows in the table are labeled for ease of reference in later discussion of the sources of various basic event quantifications (e.g., “Source: Table F-4, Scenario S1A1, row 12”). The main outputs used for PRA purposes are:

- Probabilities of an ignition source in containment in each time phase. (Note that these are not calculated by the code but are just copies of the input values specified by the user.)
- Conditional probabilities of ignition occurring in the containment in each time phase, conditional on an ignition source existing. In Phases II and III, these probabilities are further broken down to be conditional on there being a prior burn or not being a prior burn. (For example, the value in Phase III with prior burn considers only those histories in which one or more burns were predicted to occur in Phases I and/or II.)
- Conditional probabilities of containment failure in each time phase, given that ignition occurs. These probabilities are also further broken down into those conditional with or without prior combustion.

Some other values, such as the time of latest predicted combustion and the net containment failure probabilities, are present for perspective. For example, although the SBO scenario S1A1 includes some fairly high CCFPs in the late time frame, the net probability of late failure is only about 0.205 because the probability of a late ignition source is low (0.3).

For each scenario, a sensitivity case was also run in which the “minimum detonable concentration” input parameter was reduced from 1.0 to 0.15 (i.e., it was assumed that any burn at an effective combustible gas concentration at or above 15 percent resulted in DDT and unconditional containment failure). The late CCFPs thus predicted by ERPRA-BURN are listed in the table and used in Section F.5.6 to help infer conditional probabilities of detonation for some cases in the CET, as discussed in more detail there.

Table F-4 Summary of ERPRA-BURN Results

Result		S1A1	S1A1-S1	S6B	LPSD1	LPSD2	LPSD3	LPSD4
Before VB (Phase I):								
1	Ignition Source Present	0.1	0.99	0.99	0.99	0.1	0.99	0.99
2	Ignition Occurs, Given a Source	0.1615	0.5242	0.5253	0.8327	0.3920	0.9163	0.5430
3	Containment Fails, Given Ignition	0	0	0	0	0	0	0
At or Around VB (Phase II):								
4	Ignition Source Present	0.5	0.99	0.99	0.99	0.5	0.99	0.99
5	Ignition Occurs, Given a Source (with prior burn)	0.7367	0.6773	1	1	1	1	1
6	Ignition Occurs, Given a Source (without prior burn)	0.9439	1	1	1	1	1	1
7	Containment Fails, Given Ignition (with prior burn)	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4
8	Containment Fails, Given Ignition (without prior burn)	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4	< 1E-4
Late Time Frame (Phase III):								
9	Ignition Source Present	0.3	0.99	0.99	0.99	0.3	0.99	0.99
10	Ignition Occurs, Given a Source (with prior burn)	1	0.9999	1	0.9789	0.8042	1	1
11	Ignition Occurs, Given a Source (without prior burn)	1	1	1	0.9980	0.9886	1	1
12	Containment Fails, Given Ignition (with prior burn)	0.6591	0.2785	0.0531	0.1229	0.3338	0.0504	0.0660
13	Containment Fails, Given Ignition (without prior burn)	0.6728	0.2915	0.1041	0.2276	0.6218	0.1796	0.2003

Result		S1A1	S1A1-S1	S6B	LPSD1	LPSD2	LPSD3	LPSD4
14	Containment Fails, Given Ignition (with prior burn; detonable case)	0.7634	0.4589	0.7802	0.1276	0.3475	0.6647	0.8512
15	Containment Fails, Given Ignition (without prior burn; detonable case)	0.7657	0.4724	0.8485	0.2634	0.6623	0.8266	0.9171
Other Results:								
Time of Latest Burn (hr)		82.4	82.3	27.7	55.7	83.6	59.0	67.8
Net CCFP, Phase I		0	0	0	0	0	0	0
Net CCFP, Phase II		0	0	0	0	0	0	0
Net CCFP, Phase III		0.2050	0.2777	0.0528	0.1193	0.1368	0.0500	0.0659

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 (lower bound of 7.24 bar-abs and median failure pressure of 10.55 bar-abs).
- 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).
- 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 0.14 to 0.21. (For example, at power, the CCFP is about 0.66 or 0.67, depending on whether a prior burn occurred, which is close enough not to matter much even though these two eventualities are about equally likely. But the probability of an ignition source late is 0.3, and with a conditional ignition probability of 1, this approximately yields the net containment failure probability of $(0.3) (1.0) (0.67) = 0.2$. For SBOs at shutdown, the two CCFPs are 0.33 and 0.62, which are still about equally likely, so the approximate net containment failure probability is $(0.3) (1.0) (0.33 + 0.62)/2 = 0.14$, which is what ERPRA-BURN outputs.)
- 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, and for shutdown transients it is slightly lower). (Example sequences: At power, the late CCFP with prior burn is about 0.28. Prior burns are nearly assured (0.99 each in the very early and early phases, so the "with prior burn" value dominates. In the late phase as well, the probability of an ignition source is 0.99 and the conditional probability of ignition is about 1, so the net containment failure probability is $(0.99) (1.0) (0.28) = 0.28$. The shutdown transient scenario LPSD1 works out in much the same way, with a CCFP with prior burn of 0.12 yielding a net containment failure probability of 0.12 also.)
- Earlier iterations of the combustion analysis for full-power SBOs indicated that there was a very narrow window of a few hours following vessel breach during which combustion could potentially take place, before rising steam concentration inerted the containment. However, upon subsequent examination, it was noted that the relative humidity in the containment was surprisingly low in the MELCOR results for these scenarios, contrary to ERPRA-BURN's assumption that the atmosphere is steam-saturated at all times. The code was subsequently modified to permit relative humidity values less than 1; results now show that consequential burns are more evenly spread out over the whole duration of the late time frame.

- 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 concentrations of oxygen and combustible gas relatively speaking, 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 percent 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 percent or very slightly less; therefore, almost all containment failures for CHR scenarios are detonations, with very few deflagrations challenging the containment.
- In the CHR scenario at full power, for example (S6B), 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 percent, the combustion of which would require about 20 percent concentration of oxygen. Note that the initial concentration of oxygen is 21 percent, and combustion in the model is assumed to stop at 5 percent oxygen as the mixture becomes no longer flammable, so only about 16 percent 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 percent or greater combustible gas concentration will result in a detonation, the late time frame as simulated in ERPRA-BURN plays out as a game of 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. The ERPRA-BURN's prediction of a conditional detonation probability is in the range of 0.6 to 0.8 for CHR scenarios. (Note that these scenarios entail AC power availability and a cumulative ignition source probability of 0.99; under ERPRA-BURN's assumptions, this means on average that there will be 6 to 7 times at which an ignition source exists in the containment and therefore a potential burn.)
- Loss-of-inventory scenarios for accidents at shutdown are somewhat intermediate between transients and CHR scenarios from a late combustion perspective, in that the potentially large inventory of water dumped into the containment (from the RPV and flooded reactor cavity) have a very large thermal inertia that delays heatup of the containment even in the absence of active heat removal. The conditional probability of detonations for such scenarios is only slightly less than for CHR scenarios (i.e., about 0.5 as compared with 0.8), and the net containment failure probability in the late time frame is similar. LOINV scenarios at shutdown, like CHR scenarios, are dominated by detonations, and the discussion in the above bullet point applies qualitatively here, also.

F.5.4 Ignition Source Probability Assignments

The probability that an ignition source exists in the containment is quantified separately in the CET for the time frames very early (i.e., before vessel breach), early (i.e., at and around the

time of vessel breach), and late (i.e., after the rapid containment transient accompanying vessel breach, somewhat imprecisely defined). In the decomposition event trees called by the CET, these are defined by top events 1-L2-H2IGNSRC-VE, 1-L2-H2IGNSRC-E, and 1-L2-H2IGNSRC-L.

The quantification of these top events in the CET are not the result of any calculation but are simply a reproduction of the assumptions discussed in Section F.5.2 and used as input parameters to ERPRA-BURN. Assignment of the probabilities of an ignition source was performed in a very qualitative fashion based on engineering judgement. Separate sets of basic events were defined for sequences with AC power and sequences without AC power (i.e., non-SBO and SBO accident types as binned in the PDS event trees), since this is viewed as the single biggest factor determining whether ignition could occur in an otherwise flammable gas mixture. The existence of an ignition source is viewed as highly likely (i.e., approximately 0.99) in a non-SBO scenario, since AC power operated machinery may serve as a potential ignition source. In an SBO scenario, combustion can be initiated only by random ignition or via sparks generated passively by metal objects (e.g., unsecured hatches); in such a scenario, existence of an ignition source is viewed as highly uncertain, and a probability between about 0.1 and 0.5 is assigned.

ERPRA-BURN models potential ignition events in the containment in a stochastic manner over a set of Monte Carlo simulations of combustion history. Within a particular time frame (e.g., the late time frame following vessel breach), the code assumes random ignition sources are distributed with a uniform probability with respect to time. The functional behavior is similar to the decay of radioisotopes, and occurrences can be characterized in terms of the frequency of potential ignition events (i.e., rate per unit time) or characteristic time in between such events.

The input parameter dictating ignition frequency in ERPRA-BURN is the ignition probability. This is interpreted as the cumulative probability of having at least one potential ignition event over the entire time frame under consideration. That is:

$$P = 1 - e^{-T/\tau}$$

Where:

P = the (cumulative) probability of having at least one ignition source event over the entire time frame;

T = the total duration of the time frame; and

τ = the characteristic time in between potential ignition events.

Furthermore, the mean number of such ignition events over the entire time frame can be derived as T/τ . For example, if the cumulative probability of ignition in the late time frame is set to 0.99, and the total duration of the late time frame for a particular scenario is 6 days (144 hours), then the characteristic time in between ignition source events can be derived from the above equation as 31.3 hours, and the mean number of ignition source events would be about 4.6.

Table F-5 summarizes the basic event probabilities assigned to the existence of an ignition source in the CET. The table lists the “point value” probability that an ignition source exists, as

the result of qualitative assignment as discussed above. Based on these point value assignments, the authors later developed probability distributions in the form of histograms for purposes of the PRA uncertainty analysis:

- In the case of the random ignition source probabilities (no AC power), histograms were created representing the uncertainty in the basic event probability based on qualitative engineering arguments. For details of these histograms, see Table C-13 through Table C-15 in Appendix C.
- In the case of the ignition probability in the presence of AC power (i.e., point value of 0.99), a more logarithmic type of distribution was viewed as more appropriate. Here the probability of no ignition source (i.e., one minus the basic event probability) is assumed to be distributed around a value of 0.01 with an error factor of 10. Based on this assumption, a histogram was created which represented in discretized form the resulting analytical one-minus-lognormal distribution. (Note that SAPHIRE can represent uncertainties in basic events with a lognormal distribution, but not ones with a one-minus-lognormal distribution.) Details on this histogram can be found accompanying Table C-12 in Appendix C.

For more detailed representation and discussion of formulation of these histograms, see Appendix C. Finally, Table F-5 shows the as-quantified mean value of the basic event probability as calculated by SAPHIRE using the input histograms (i.e., the “Calculated” probability from SAPHIRE). Note that the calculated value does not necessarily match the initially determined point value. In the case of the probabilities given AC power, this is because the qualitatively developed uncertainty distribution (one-minus lognormal) has a long tail in the direction of lower probabilities and thus does not exactly preserve a mean value equal to the mode; also, some inaccuracy is introduced as a result of the discretization process used to prepare the histogram. Other calculated values also differ slightly, mainly due to the randomness of the sampling process in SAPHIRE (e.g., 0.1005 instead of 0.1).

Note that the internal events and external events models share the same basic events and quantification in all time frames and scenarios. In the case of the shutdown PRA model, it was felt better to define a parallel set of basic events (even if they were quantified with the same exact values and distributions) since the underlying uncertainty issues may not be completely correlated given the differences in plant initial conditions while at shutdown.

Table F-5 Assignment of Ignition Source Probabilities in the CET

Model	Time Phase	Case	Basic Event	Source in Table F.4	Failure Probability		
					Point-Value	Uncertainty	Calculated
At-power	Very Early	Without AC	1-L2-BE-H2IGNSRC-VE-NAC	S1A1, row 1	0.1	Histogram (qualitatively developed by NRC)	0.1005
		With AC	1-L2-BE-H2IGNSRC-VE-AC	S1A1-S1, row 1	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563
	Early	Without AC	1-L2-BE-H2IGNSRC-E-NAC	S1A1, row 4	0.5	Histogram (qualitatively developed by NRC)	0.5000
		With AC	1-L2-BE-H2IGNSRC-E-AC	S1A1-S1, row 4	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563
	Late	Without AC	1-L2-BE-H2IGNSRC-L-NAC	S1A1, row 9	0.3	Histogram (qualitatively developed by NRC)	0.3001
		With AC	1-L2-BE-H2IGNSRC-L-AC	S1A1-S1, row 9	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563

(Continued on next page)

Table F-5 Assignment of Ignition Source Probabilities in the CET (cont.)

Model	Time Phase	Case	Basic Event	Source in Table F.4	Failure Probability		
					Point-Value	Uncertainty	Calculated
SD	Very Early	Without AC	1-L2-BE-H2IGSC-VE-NAC-SD	LPSD2, row 1	0.1	Histogram (qualitatively developed by NRC)	0.1005
		With AC	1-L2-BE-H2IGSRC-VE-AC-SD	LPSD4, row 1	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563
	Early	Without AC	1-L2-BE-H2IGSRC-E-NAC-SD	LPSD2, row 4	0.5	Histogram (qualitatively developed by NRC)	0.5000
		With AC	1-L2-BE-H2IGSRC-E-AC-SD	LPSD4, row 4	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563
	Late	Without AC	1-L2-BE-H2IGSRC-L-NAC-SD	LPSD2, row 9	0.3	Histogram (qualitatively developed by NRC)	0.3001
		With AC	1-L2-BE-H2IGSRC-L-AC-SD	LPSD4, row 9	0.99	Histogram (developed by ERI to discretely approximate a one-minus-lognormal distribution centered on 0.01 with error factor of 10)	0.9563

F.5.5 Conditional Ignition Probability Assignments

For each of the three time phases (very early, early, and late) in one combustion history of a given scenario, ERPRA-BURN either predicts that one or more burns take place or that no burns take place. Whether or not any burns take place is a consequence of the random conditions of a particular history, including whether there is an ignition source in containment; whether there are sufficient concentrations of oxygen and combustible gas to constitute a flammable mixture; whether the atmosphere is inerted by steam and/or carbon dioxide; and whether the randomly generated timing of ignition occurs at such a time for these conditions to be flammable. The conditional ignition probability is defined as the probability that a burn occurs in a particular time phase given the existence of an ignition source, and this is derived in ERPRA-BURN as follows:

$$p_{ig,i} = \frac{\sum_j \delta_{ij} w_j (N_{burn,ij} / N)}{\sum_j \delta_{ij} w_j}$$

Where:

$p_{ig,i}$	=	point-value of conditional probability of ignition in time phase i ;
i	=	index for time phase (1 = very early, 2 = early, 3 = late);
j	=	index for combustion history set (1 = ignition source present very early and late; 2 = ignition source present early and late; 3 = ignition source present in all time phases; 4 = ignition source present late);
$N_{burn,ij}$	=	number of histories in history set j which predict at least one burn in time phase i ;
N	=	the total number of histories used in each history set;
δ_{ij}	=	delta function equal to 1 if history set j includes the assumption of ignition in time phase i , and equal to 0 if it does not; and
w_j	=	weighting factor for history set j .

The weighting factor w_j is the relative likelihood of the ignition source assumptions in a particular history set and calculated as the product of the three ignition source probabilities defining it. For example, in an SBO scenario, the probabilities assumed for the three time frames are 0.1, 0.5, and 0.3, respectively; so for history set 1, which assumes an ignition source is present only very early and late, the weighting factor is $(0.1) (1 - 0.5) (0.3) = 0.015$.

Since each combustion history is binary with respect to whether a burn occurs or not in a particular time phase, ERPRA-BURN is unable to supply information with which to develop a smooth uncertainty distribution as to whether combustion occurs given an ignition source. That is, ERPRA-BURN by assumption deterministically predicts whether a burn occurs given a particular set of containment conditions (i.e., flammable or not, inerted or not), and there is no meaning to a supposition that a burn “might” occur; uncertainty is only present as to the conditions that lead up to a possible burn. Therefore, for purposes of defining uncertainty distributions in SAPHIRE, the conditional ignition probabilities are all assumed to be bimodal histograms with potential values of 1.0 (probability $p_{ig,i}$) and 0.0 (probability $1 - p_{ig,i}$). (Note that, in practice, the zero bin of the histogram was instead set to 1.0E-6, since SAPHIRE cannot handle values on the histogram that are identically zero.)

Table F-6 summarizes the conditional probabilities of ignition obtained from ERPRA-BURN and used to quantify various basic events. Note that, in some cases, ignition was found to be certain

given the presence of an ignition source (i.e., $p_{ig,i}$ was calculated to be identically 1, and every combustion history with an ignition source present predicted a burn to occur). In such cases, the event tree logic was simplified by using SKIP/TRUE logic for the success and failure branches (routing all sequences automatically to the failure branch) rather than assigning a basic event. In words, the ignition results from ERPRA-BURN can be summarized as follows:

- Burns before vessel breach are possible but not assured, and they are somewhat less likely in scenarios with less containment heat removal (since the steam concentration is lower, the relative concentration of hydrogen is larger and therefore more likely to be flammable).
- Burns at or around the time of vessel breach are certain to occur, given that a source of ignition is present in the containment; and this was found to be true whether or not an earlier burn consumed some of the available in-vessel hydrogen.
- Burns in the late time frame after vessel breach are nearly certain to occur given an ignition source, with conditional probabilities in the range of 0.8 to 1.0. The lower end of this range occurs for scenarios without containment heat removal, in which there may be a narrow time window before the containment becomes inerted by rising steam concentration.

Table F-6 Assignment of Conditional Ignition Probabilities in the CET

Model	Time Phase	Case	Basic Event	Source in Table F-4	Failure Probability		
					Point Value	Uncertainty	Calculated
At-power	Very Early	SBO	1-L2-BE-H2IGN-VE-SBO	S1A1, row 2	0.16	Bimodal histogram with values of 1E-6 and 1	0.16
		Non-SBO	1-L2-BE-H2IGN-VE-GEN	S1A1-S1, row 2	0.52	Bimodal histogram with values of 1E-6 and 1	0.52
	Early	All	n/a	S1A1, S1A1-S1, S6B, rows 5-6 ¹	1	none	n/a
	Late	All	n/a	S1A1, S1A1-S1, S6B, rows 10-11 ¹	1	none	n/a
SD	Very Early	With CHR	1-L2-BE-H2IGN-VE-CHR-SD	LPSD4, row 2	0.54	Bimodal histogram with values of 1E-6 and 1	0.54
		SBO	1-L2-BE-H2IGN-VE-SBO-SD	LPSD2, row 2	0.39	Bimodal histogram with values of 1E-6 and 1	0.39
		LOINV	1-L2-BE-H2IGN-VE-LOCA-SD	LPSD3, row 2	0.92	Bimodal histogram with values of 1E-6 and 1	0.92
		Other accident type, no CHR	1-L2-BE-H2IGN-VE-TRAN-SD	LPSD1, row 2	0.83	Bimodal histogram with values of 1E-6 and 1	0.83
	Early	All	n/a	LPSD1-4, rows 5-6	1	none	n/a
	Late	SBO with prior burn	1-L2-BE-H2IGN-L-NACPB-SD	LPSD2, row 10	0.80	Bimodal histogram with values of 1E-6 and 1	0.80
		SBO without prior burn	1-L2-BE-H2IGN-L-NACNP-SD	LPSD2, row 11	0.99	Bimodal histogram with values of 1E-6 and 1	0.99
		LOCA or CHR with prior burn	n/a	LPSD3-4, row 10	1	none	n/a
		LOCA or CHR without prior burn	n/a	LPSD 3-4, row 11	1	none	n/a

Table F-6 Assignment of Conditional Ignition Probabilities in the CET (cont.)

Model	Time Phase	Case	Basic Event	Source in Table F-4	Failure Probability		
					Point Value	Uncertainty	Calculated
SD	Late	Transient with prior burn	1-L2-BE-H2IGN-L-NCHPB-SD	LPSD1, row 10	0.98	Bimodal histogram with values of 1E-6 and 1	0.98
		Transient without prior burn	n/a	LPSD1, row 11	1	none	n/a
¹ The calculated probabilities for these cases are mostly equal to 1, with a few relatively high but less than 1 (i.e., down to about 0.7). It was not thought useful to define many cases with probabilities slightly less than 1 here, so a uniform probability of 1 was used.							

F.5.6 Detonation Probability Assignments

The derivations below are labeled with paragraph markings in braces for purposes of reference by Table F-7.

{*Paragraph 1*} In the course of performing the ERPRA-BURN calculations, it was observed that there was a significant probability of building up to very high concentrations of combustible gas (around 0.15 to 0.30) in the containment in the late time frame of the accident. Such concentrations are potentially combustible; however, ERPRA-BURN has no detailed detonation model, and the containment loads for such burns would simply be calculated using AICC calculations assuming global deflagration. This potentially underestimates the containment failure probability.

{*Paragraph 2*} In order to explore the possible impact of detonation on containment failure within the ERPRA-BURN probabilistic framework, sensitivity cases were run using the code's "Minimum Detonable Concentration" feature. This input option instructs the program to unconditionally fail the containment any time a burn occurs at a combustible gas concentration equal to or greater than the input value. Normally 1.0 by default, this parameter was changed to 0.15 for purposes of the sensitivity cases (i.e., ERPRA-BURN will assume any burn at 15 percent or higher combustible gas concentration will in fact be a detonation and will fail the containment with a conditional probability of 1). The results of these sensitivity cases on late conditional containment failure probabilities are listed in Table F-4, rows 14 and 15.

{*Paragraph 3*} For SBOs at full power (case S1A1), the CCFPs calculated using the Minimum Detonable Concentration option are not significantly higher than without the option (i.e., about 0.76 versus 0.66) (Table F-4, rows 12–15). Similarly, for transients with AC power (case S1A1-S1), transients with AC power at shutdown (case LPSD1), and SBOs at shutdown (case LPSD2), the CCFPs are also not drastically affected by the Detonable option. From this, we infer that detonation is fairly unlikely, and the CET will assume the conditional probability is zero.

{*Paragraph 4*} For scenarios with late CHR at power (case S6B), the effective CCFPs are considerably increased through use of the Minimum Detonable Concentration option, from 0.0531 to 0.7802 (with prior burn) or from 0.1041 to 0.8485 (without prior burn) (Table F-4, rows 12–15). A detailed examination of the history-by-history ERPRA-BURN results for this case showed that all instances of late containment failure due to global deflagration (base case S6B) are ones which occurred at 15 percent or higher concentration of combustible gases. In other words, under the assumption that burns at 15 percent or higher concentration detonate, the CCFP due to deflagrations (i.e., burns at concentrations less than 15 percent) is zero, and all containment failures are the result of detonation. Therefore, using the data from the ERPRA-BURN results with the detonable option, the conditional probability of detonation would simply be 0.7802 (with prior burn) or 0.8485 (without prior burn). However, it was not thought useful to differentiate these very similar numbers, especially since almost all cases of late combustion with CHR occur with a prior burn (AC power available implying an ignition source probability of 0.99 in earlier time frames), so a flat value of 0.78 is used for conditional probability of detonation given ignition in the late time frame for full-power sequences with late CHR.

Table F-7 Assignment of Conditional Detonation Probabilities in the CET

Model	Time Phase	Case	Basic Event	Source	Failure Probability		
					Point Value	Uncertainty	Calculated
At-power	Late	With CHR	1-L2-BE-H2DET-L-CHR	See Section F.5.6 { <i>Paragraph 4</i> }	0.78	Bimodal histogram with values of 1E-6 and 1	0.78
		Without CHR	n/a	See Section F.5.6 { <i>Paragraph 3</i> }	0	None	n/a
SD	Late	LOINV without prior burn, or any case with CHR	1-L2-BE-H2DET-L-CHR-SD	See Section F.5.6 { <i>Paragraph 5</i> }	0.79	Bimodal histogram with values of 1E-6 and 1	0.79
		LOINV with prior burn	1-L2-BE-H2DET-L-LOIPB-SD	See Section F.5.6 { <i>Paragraph 5</i> }	0.61	Bimodal histogram with values of 1E-6 and 1	0.61
		Otherwise	n/a	See Section F.5.6 { <i>Paragraph 3</i> }	0	none	n/a

{Paragraph 5} For scenarios at shutdown with LOINV or late CHR, contrary to the corresponding full-power case with CHR, it was found that some instances of combustion at less than 15 percent gas concentration would be capable of failing the containment. Therefore, it was thought that a better approximation to the conditional detonation probability and CCFPs would be obtained by (1) subtracting the CCFP without detonations from the CCFP with detonations to get the conditional probability of detonation, and (2) assuming that the original CCFP without detonations continues to apply to burns at less than the detonable concentration. The detonation probabilities were calculated from the LPSP ERPRA-BURN results as shown below. Note that these values were based on an earlier iteration of the LPSP ERPRA-BURN results, and there may be small differences from the results shown in Table F-4.

- CHR with prior burn: $0.85 - 0.06 = 0.79$ (Table F-4, LPSP4, rows 14 and 12).
- CHR without prior burn: $0.92 - 0.20 = 0.72$ (Table F-4, LPSP4, rows 15 and 13).
- LOINV with prior burn: $0.66 - 0.05 = 0.61$ (Table F-4, LPSP3, rows 14 and 12).
- LOINV without prior burn: $0.83 - 0.18 = 0.65$ (Table F-4, LPSP3, rows 15 and 13).

Here again, the actual probabilities as used are heavily weighted towards sequences with prior burn (since most of these have AC power and therefore 0.99 probability of ignition source in earlier time frames). Therefore, it was not thought useful to distinguish between cases with and without prior burn. The 0.61 probability was then applied to LOINV cases with prior burn, and 0.79 to all CHR cases and to LOINV cases without prior burn.

Note that, as with the conditional ignition probabilities discussed in Section F.5.5, ERPRA-BURN's prediction of detonation in the cases using that option is a binary one – each history either includes a detonation or it does not. Therefore, ERPRA-BURN cannot give meaningful “smooth” uncertainty distributions with respect to the conditional detonation probability. Therefore, as with the conditional ignition probabilities, uncertainty distributions were input for the non-zero basic events using bimodal histograms with possible values of 1 (with probability p_{det}) and 1E-6 (with probability $1 - p_{det}$).

F.5.7 Containment Failure Probability Assignments

ERPRA-BURN simulates thousands of combustion histories for each set of assumptions regarding existence of an ignition source, and for each of these histories a containment failure probability is calculated in each of the three time frames of the accident (very early, early, and late). Since there may be multiple burns in one time frame, the probability of failure for a particular history can be defined as:

$$p_{CF,ijH} = 1 - \prod_k \left(1 - f_{CF}(P_{ijkH}) \right),$$

Where:

i	=	index for time phase of the accident (1 = very early, 2 = early, 3 = late);
j	=	index for combustion history set (1 = ignition source present very early and late; 2 = ignition source present early and late; 3 = ignition source present in all time phases; 4 = ignition source present late);
k	=	index for successive burns within one time frame and history set;
$p_{CF,ijH}$	=	cumulative probability of containment failure in the i th time frame of the j th combustion history set of simulated history H ;

P_{ijkH} = peak pressure resulting from AICC calculation of the k th burn in the i th time frame of the j th combustion history set of simulated history H ; and
 $f_{CF}(P)$ = containment fragility function, in the form of an integrated failure probability at static pressures less than or equal to P .

Then, given a containment failure probability $p_{CF,ijH}$ for a single combustion history, the integrated conditional containment failure probability is equal to:

$$CCF_i = \frac{\sum_j \sum_H \phi_{jH} w_j p_{CF,ijH}}{\sum_j \sum_H \phi_{jH} w_j},$$

where ϕ_{jH} is a delta function equal to zero or one depending on whether the H th history in the j th history set has features which are part of the assumptions one is using to establish a particular CCFP. For example, one may want to evaluate the CCFP in the late time frame with prior combustion, in which case ϕ_{jH} is equal to 1 if and only if the H th history in the j th history set includes at least one combustion event in either the very early or early time phase.

Typically, what one wishes out of ERPRA-BURN for purposes of CET quantification is the CCFP in each of the three time frames, separately with and without prior burns for each (except for the very early time phase, since no prior burns are possible as it is the first time frame considered).

Table F-4 lists the ERPRA-BURN results for CCFP in each of the three time phases, with and without prior burns, for each of the seven scenarios analyzed. These were assigned as the point-value basic event probabilities used to quantify conditional containment failure from combustion in the CET, as described in Table F-8. A few special cases are present:

- In some cases, the CCFP is identically zero. Rather than define an unnecessary basic event, the linkage rules in SAPHIRE are written as FALSE/SKIP (i.e., the success branch – no containment failure – is taken automatically).
- As part of the identically zero CCFPs, the burn instances with late CHR at full power and resulting in containment failure were found on detailed analysis to always accompany detonable mixtures (see Section F.5.6). Therefore, all containment failures are considered to be captured already by the conditional detonation top event and branch, and the residual CCFP for deflagrations is considered to be zero.
- In some cases, the CCFP is identically one. Rather than define an unnecessary basic event, the linkage rules in SAPHIRE are written as SKIP/TRUE (i.e., the failure branch – containment failure occurs – is taken automatically).
- In the case of some CCFPs in the early time frame, very small non-zero values were calculated by ERPRA-BURN. The precise quantification of such small values or probability is unreliable in ERPRA-BURN given the nature of the Monte Carlo sampling process with a few thousand samples, and therefore 1×10^{-4} was assigned broadly for such cases as a bounding value.

Table F-8 Assignment of Conditional Containment Failure Probabilities in the CET

Model	Time Phase	Case	Basic Event	Source in Table F-4	Failure Probability		
					Point Value	Uncertainty	Calculated
At-power	Very Early	All	none	S1A1, S1A1-S1, S6B, row 3	0	none	n/a
	Early	All	1-L2-BE-H2CF-E-GEN	S1A1, S1A1-S1, S6B, rows 7-8	1E-4	Lognormal, error factor (EF) = 76	1E-4
	Late	Detonation	none	assumption	1	none	n/a
		No AC, prior burn	1-L2-BE-H2CF-L-NACPB	S1A1, row 12	0.6591	Histogram from ERPRA-BURN results	0.6588
		No AC, no prior burn	1-L2-BE-H2CF-L-NACNPB	S1A1, row 13	0.6728	Histogram from ERPRA-BURN results	0.6725
		AC, no CHR, prior burn	1-L2-BE-H2CF-L-NCHRPB	S1A1-S1, row 12	0.2785	Histogram from ERPRA-BURN results	0.2814
		AC, no CHR, no prior burn	1-L2-BE-H2CF-L-NCHRNPB	S1A1-S1, row 13	0.2915	Histogram from ERPRA-BURN results	0.2946
		CHR, prior burn	none	Section F.5.6 {Paragraph 4} (only detonations are capable of failing containment)	0	None	n/a
		CHR, no prior burn	none	Section F.5.6 {Paragraph 4} (only detonations are capable of failing containment)	0	None	n/a
SD	Very Early	All	none	LPSD1-4, row 3	0	None	n/a
	Early	All	1-L2-BE-H2CF-E-GEN	LPSD1-4, rows 7-8	1E-4	Lognormal, EF = 76	
	Late	Detonation	none	assumption	1	None	n/a
		No AC, prior burn	1-L2-BE-H2CF-L-NACPB-SD	LPSD2, row 12	0.3338	Histogram from ERPRA-BURN results	0.3343
		No AC, no prior burn	1-L2-BE-H2CF-L-NACNPB-SD	LPSD2, row 13	0.6218	Histogram from ERPRA-BURN results	0.6216
		CHR, prior burn	1-L2-BE-H2CF-L-CHRPB-SD	LPSD4, row 12	0.0660	Histogram from ERPRA-BURN results	0.0686

Table F-8 Assignment of Conditional Containment Failure Probabilities in the CET (cont.)

Model	Time Phase	Case	Basic Event	Source in Table F.4	Failure Probability		
					Point Value	Uncertainty	Calculated
SD	Late	CHR, no prior burn	1-L2-BE-H2CF-L-CHRNPB-SD	LPSD4, row 13	0.2003	Histogram from ERPRA-BURN results	0.2036
		LOINV, no CHR, prior burn	1-L2-BE-H2CF-L-LOIPB-SD	LPSD3, row 12	0.0504	Histogram from ERPRA-BURN results	0.0533
		LOINV, no CHR, no prior burn	1-L2-BE-H2CF-L-LOINPB-SD	LPSD3, row 13	0.1796	Histogram from ERPRA-BURN results	0.1829
		AC, no CHR, prior burn	1-L2-BE-H2CF-L-NCHRPB-SD	LPSD1, row 12	0.1229	Histogram from ERPRA-BURN results	0.1251
		AC, no CHR, no prior burn	1-L2-BE-H2CF-L-NCHRNPD-SD	LPSD1, row 13	0.2276	Histogram from ERPRA-BURN results	0.2295

In the case of each CCFP that was quantified directly using ERPRA-BURN, uncertainty in the CCFP was defined in the form of a histogram in SAPHIRE. Such histograms are easily created by ERPRA-BURN, since one only needs to export the individual failure probabilities for each history ($p_{CF,ijH}$) along with the history weights (w_j), sort them in order of ascending $p_{CF,ijH}$. The values of $p_{CF,ijH}$ then form the x-axis of a cumulative distribution function, with the normalized and integrated weights w_j as the y-axis. This can be discretized as needed for input to SAPHIRE. Note that the discretization process introduces some small measure of imprecision, which is why the calculated probabilities in the last column of Table F.8 are slightly different from the point values output by ERPRA-BURN (generally by less than 1–2 percent). Some of the histograms resulting from the ERPRA-BURN analysis for late CCFP are reproduced in Appendix C as Table C-20 through Table C-27.

F.5.8 Sensitivity Analysis

In Section 2.6.3.7, sensitivity case MU-7.1, several parameter variations were performed with respect to ERPRA-BURN scenarios LPD2 (SBO at shutdown) and LPD3 (LOINV at shutdown), in an attempt to determine whether the combustion results were highly sensitive to any of the parameter assumptions. The results of these sensitivity studies can be summarized as follows:

- CCFP for the SBO case dropped substantially with either significantly increased temperature (by 20 to 50 °C) or significantly increased relative humidity (from 45 percent to 60 percent). Both changes had a similar effect of significantly narrowing the time window during which combustion could take place due to late inerting of the containment.
- Other parameter variations for the SBO case, including a range of MCCI rate factors or a decrease in the relative humidity, showed no strong effect on CCFP.
- The LOINV case (with a cooler containment) did not show any significant sensitivity to any of the examined parameter variations.

In addition to the sensitivity analyses briefly summarized above, three additional sensitivity analyses were performed in with respect to the following issues or input parameters to ERPRA-BURN:

- Assumed termination of ex-vessel gas generation due to spill-over into the reactor cavity of water injected to the containment (e.g., by means of containment sprays), and the postulated arrest of MCCI. From the results of one MELCOR calculation, spillover (although not subsequent MCCI arrest) was predicted at a time when only about 25 percent of the 7-day cumulative ex-vessel gas production had occurred, and therefore the ERPRA-BURN sensitivity case assumed ex-vessel gas generation was reduced uniformly by 75 percent.
- When the detonation option in ERPRA-BURN is activated, by default DDT is assumed for all burns occurring at 15 percent initial combustible gas concentration or higher. This detonability threshold was increased in two sensitivity calculations to 20 percent and 25 percent.

- According to the discussion in Section F.5.4, a late ignition source probability of 0.99 was assumed for non-SBO scenarios, which is equivalent to a characteristic time in between ignition source events of about 30 hours. Two sensitivity cases were performed in which the assumed late ignition source frequency was doubled or tripled (corresponding to cumulative ignition source probabilities of 0.9999 and 0.999999, respectively).

Some of these three additional sensitivity analyses resulted in significant changes to the conditional probabilities of late detonation or late containment failure given deflagration.

F.6 Insights and Conclusions

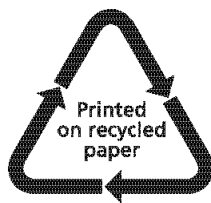
The following observations are made in view of the combustion analysis performed for the various parts of the Level 2 PRA project:

- While combustion can occur in the containment in the time frames before or at and around the time of vessel breach, the conditional probability of containment failure from such combustion is very low (less than 1×10^{-4}). This is largely in line with the state of knowledge of combustion for other large dry PWRs. Note that, although the probability of early containment failure from combustion is very low, it shows up in the PRA results as a significant contributor to overall early containment failures in a relative sense, simply because all other causes of early containment failure (e.g., in-vessel steam explosions) have a similarly low CCFP.
- Under typical late SBO conditions (e.g., scenario S1A1), it requires combustion of approximately 12 percent concentration of hydrogen to exceed the lower-bound failure pressure of the containment, and approximately 18 percent concentration to reach the median containment failure pressure. Therefore, a conventional combustion analysis such as by means of deterministic MELCOR calculations—which assume by default combustion always at 10 percent concentration of hydrogen—would indicate that there is zero chance of late containment failure due to combustion. However, given the randomness of ignition, it is falsely optimistic to assume that burns always happen at such relatively low concentrations of hydrogen. The ERPRA-BURN code was designed to overcome this limitation by modeling random combustion histories in a Monte Carlo framework, where individual burns could happen at low or high concentrations according to the random sampling process. Therefore, ERPRA-BURN predicts a finite probability that some burns occur that are sufficiently large in magnitude to challenge the containment (net probability 0.12 to 0.28 for the scenarios examined with SBO or no late CHR), where a conventional deterministic analysis would not identify combustion events that would challenge the containment.
- Furthermore, because ERPRA-BURN allows for the possibility of burns at quite high hydrogen concentrations, it can be envisioned that some are in the detonable range. In the context of the current PRA, this becomes most prominent in the case of scenarios with late CHR; the low partial pressure of steam amplifies the relative concentration of hydrogen and renders conditions more detonable. Given the large quantity of hydrogen and carbon monoxide available in the CHR scenarios examined, it is more likely than not in ERPRA-BURN's results that at least one late combustion event in the detonable range would occur. This is the reason why the CCFP for scenarios with late CHR is so high in the current analysis (0.6 to 0.8). In fact, given the low base pressure in this case, all or

almost all combustion events capable of challenging the containment are detonable, and global deflagration contributes little or not at all to containment failure.

- The intermediate-time containment failure release category (ICF) in the Level 2 PRA results constitutes a significant contributor in terms of relative frequency and overall risk. The majority of this release category's frequency owes to detonations in scenarios with late CHR or LOINV scenarios, as discussed above, although other scenarios with global deflagrations also contribute.
- LOI accidents at shutdown behave qualitatively similarly to late CHR scenarios, although sometimes with a slightly lower CCFP. This is because, even in the absence of active heat removal systems, the sheer quantity of water dumped into the containment and reactor cavity from the RPV constitutes a huge amount of thermal inertia which delays the heatup of the containment in the late time frame.

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